

CP-2023-06 Initial Exam Answer
Key

1		C
2		C
3		D
4		D
5		B
6		C
7		A
8		B
9		D
10		B
11		D
12		D
13		A
14		C
15		C
16		C
17		B
18		D
19		B
20		A
21		C
22		A
23		A
24		B
25		B
26		C
27		B
28		C
29		C
30		D
31		C
32		B
33		C
34		C
35		A
36		D
37		B
38		D
39		D

40		A
41		A
42		A
43		C
44		B
45		C
46		C
47		D
48		A
49		C
50		A
51		A
52		C
53		C
54		A
55		B
56		A
57		D
58		B
59		A
60		B
61		A
62		C
63		C
64		C
65		B
66		C
67		A
68		C
69		B
70		B
71		A
72		B
73		B
74		B
75		C

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

Examination Outline Cross-Reference	Level	RO
003 Reactor Coolant Pump System	Tier #	2
	Group #	1
Ability to monitor automatic features of the Reactor Coolant Pump System, including RCP lube oil and bearing lift pumps	K/A #	A3.05
	Rating	3.0

Question 1

After a Unit 1 outage, you are preparing to start 1/1-PCPX3 RCP 3.

After completing prerequisites and starting 1/1-PCPX3-LP RCP 3 OIL LIFT PUMP, you observe a blue OIL PRESS light on the Oil Lift Pump handswitch is lit.

What would cause this indication?

Oil Lift Pump discharge pressure:

- A. greater than 325 psig and one-minute timer interlock met
- B. greater than 325 psig with oil lift pump motor running
- C. greater than 600 psig with oil lift pump motor running
- D. greater than 600 psig and one-minute timer interlock met

Answer: C.

Explanation:

A Incorrect – Plausible since RCS pressure is maintained at approximately 325 psig prior to starting a RCP; however the oil pressure switch setpoint is 600 psig. Second part is plausible but incorrect since the lift oil pump is stopped one minute after the RCP has been running and no timed interlock is associated with this circuit.

B Incorrect – Plausible since RCS pressure is maintained at approximately 325 psig prior to starting a RCP; however the oil pressure switch setpoint is 600 psig. Second part is correct since the oil lift pump motor must be running (42a contact closed) for the OIL PRESS light to turn on.

C Correct – Oil lift pump discharge pressure must be greater than 600 psig with the oil lift pump motor running (42a contact closed) for the blue OIL PRESS light to turn on.

D Incorrect – Plausible since oil lift pump discharge pressure must be greater than 600 psig for the OIL PRESS light to turn on. Second part is plausible since the lift oil pump is stopped one minute after the RCP has been running but there are no timed interlocks associated with this circuit.

Technical References:

SOP-108A pg. 19 (rev. 15)

References to be provided to applicants during exam: None

Learning Objective:

LO21.SST.RC1.OB01 - **DISCUSS** applicability, precautions, limitations, and notes contained in SOP-108, "Reactor Coolant Pump".

LO21.SST.RC1.OB02 - **DISCUSS** the actions necessary to Start and Stop the Reactor Coolant Pumps in accordance with SOP-108, "Reactor Coolant Pump".

Question Source:	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41 (b) (7)	

Examination Outline Cross-Reference	Level	RO
003 Reactor Coolant Pump System	Tier #	2
	Group #	1
Knowledge of electrical power supplies to the	K/A #	K2.01
Reactor Coolant Pump System RCPs	Rating	3.7

Question 2

Unit 1 experiences a loss of Non-Safeguards Bus 1A3 while operating at 700 MWe load.

Which of the following automatic actions will occur and why?

- A. Turbine Runback due to loss of power to 1-01 Condensate Pump.
- B. Reactor Trip due to loss of 1-01 Rod Drive Motor Generator Set.
- C. Reactor Trip due to loss of 1-03 Reactor Coolant Pump.
- D. Turbine Trip due to loss of 1-01 Turbine Plant Cooling Water Pump.

Answer: C

Explanation:

- A. Incorrect. Plausible because power is lost to Condensate Pump 1-01 and a Turbine Runback would initiate to 60% power, however, the Unit is already < 60%.
- B. Incorrect. Plausible because the Rod Drive Motor Generator Set 1-01 is powered from 1B3, however, 1B4 remains energized which powers Rod Drive Motor Generator Set 1-02.
- C. Correct. Unit 1 is greater than 48% power. In this condition a Reactor Trip will occur when one RCP is lost.
- D. Incorrect. Plausible because power is lost to TPCW Pump 1-01, however, the standby pump will AUTO start and prevent the Turbine from tripping.

Technical References:

ALM-0064A, 1-ALB-6D, Window 1.9, pg. 31, Rev. 6
 ALM-0063A, 1-ALB-6C, Window 4.2, pg. 57, Rev. 5
 Bus 1A3 & 1B3 Loads

References to be provided to applicants during exam: None

Learning Objective:

DEMONSTRATE an understanding of the components of the Reactor Coolant System including interrelations with other systems to include interlocks and control loops.

Question Source:

Bank #	X
Modified Bank #	
New	

Question History:

Last NRC Exam	NRC 2012-06, Q1
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Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.41(b) (3,5,7)

Examination Outline Cross-Reference	Level	RO
004 (SF2) Chemical and Volume Control System	Tier #	2
	Group #	1
	K/A #	K1.25
Knowledge of the physical connections and/or cause and effect relationships between the Chemical and Volume Control System and the Interface between HPI flowpath and excess letdown flowpath	Rating	3.5

Question 3

Unit 2 is at NOP, NOT in Mode 3 with startup preparations in progress.

Excess Letdown is in service and aligned to the VCT.

Several alarms annunciate in the Unit 2 Control Room and the following parameters are noted:

- Indicated Pressurizer level 20%
- Pressurizer pressure 1790 psig
- Tavg 540°F
- Containment pressure 0.8 psig

At the time the alarms annunciate, what happens to the following valves?

- 1/2-8153 AND 1/2-8154, XS LTDN ISOL VLVs (1) AND
 - 1/2-8143, XS LTDN DIVERT VLV (2)
- A. (1) receive a close signal
(2) realigns to the RCDT
- B. (1) receive a close signal
(2) remains aligned to the VCT
- C. (1) do not receive a close signal
(2) realigns to the RCDT
- D. (1) do not receive a close signal
(2) remains aligned to the VCT

Answer: D.

Explanation:

The alarms that annunciate in the control room in the above scenario are due to an SI actuation that has been triggered by a low Pressurizer pressure below 1820 psig. Excess Letdown valves do not reposition automatically on an Containment Isolation signal. They must be manually closed IAW EOP - 0.0B Attachment 2. It should be noted that the Excess Letdown isolation valves are fail-close, air-operated valves, and would eventually close due to a loss of instrument air to containment caused by the Containment Isolation. The above question, however, specifically asks whether or not the valves receive a containment isolation reposition signal as a result of the SI, and not their final, eventual position.

A is wrong. See above. Plausible because all valves may be thought to reposition when isolation signal is received.

B is wrong. See above. Plausible because the isolation valves only may be thought to reposition when isolation signal is received.

C is wrong. See above. Plausible because the three-way valve may be thought to have an automatic function when an isolation signal is received.

D is correct. See above.

Technical References:

EOP - 0.0B (rev. 9) pg. 29

LO21.SYS.CS1 – Chemical and Volume Control System (Rev. 0) pp. 33-34 (See below)

References to be provided to applicants during exam: None

Learning Objective:

DESCRIBE the components of the Chemical and Volume Control system including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-255, Chemical and Volume Control System. (LO21.SYS.CS1.OB03)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (2-9)	

Examination Outline Cross-Reference	Level	RO
004 Chemical and Volume Control System	Tier #	2
	Group #	1
Knowledge of the effect that a loss or malfunction of the Chemical and Volume Control System will have on the following systems or system parameters: RCS	K/A #	K3.06
	Rating	3.9

Question 4

Unit 1 is at full power.

1-01 CCP is running.

Letdown Isolation Valve LCV-459 drifts off FULL OPEN position and mechanically binds in a throttled position. Without operator action, this will cause (1).

In response to this failure, the operator should (2).

- A. (1) VCT LVL HI
(2) Reposition VCT level control valve LCV-112A to HUT
- B. (1) PRZR LVL HI
(2) Reposition VCT level control valve LCV-112A to HUT
- C. (1) VCT LVL HI
(2) Lower charging to minimum while maintaining seal injection
- D. (1) PRZR LVL HI
(2) Lower charging to minimum while maintaining seal injection

Answer: D

Explanation:

The applicant must know that if the letdown control valve does not read full open, the orifice isolation valves (all three) will automatically shut securing letdown flow. Charging flow is still ongoing therefore will raise pressurizer level. The procedure for loss of letdown directs lowering charging flow to 32 gpm while maintaining seal injection to RCPs.

A is wrong because the applicant must know a loss of letdown has occurred. As a result, VCT level should be going down. Additionally, the applicant may believe re-directing letdown away from the VCT to the HUT will prevent exacerbating the VCT LVL HI condition but re-directing letdown from VCT to the HUT will not do anything.

B is wrong because repositioning LVC-112A will not do anything since letdown has been secured due to the interlock between the letdown isolation valve and the orifice isolation valves. However, the pressurizer level going up due to charging flow is correct.

C is wrong because VCT level should decrease since the charging pumps are still running and letdown has been isolated due to the above-described interlock. However, the second part is correct since lowering charging flow to minimum while maintaining seal injection flow is directed by the loss of letdown ABN-105.

D is correct because of answer explanation above.

Technical References:

LO21SYSCS1 Power Point Presentation

Training Material: Chemical and Volume Control System

ABN-105, CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION

References to be provided to applicants during exam: None.

Learning Objective: Describe the components of the CVCS system including interrelations with other systems to include interlocks, and control loops.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)(7) 55.43	

LO21.SYS.CS1 (slide 42)

Training Manual: CVCS page 11

Examination Outline Cross-Reference	Level	RO
005 RESIDUAL HEAT REMOVAL SYSTEM	Tier #	2
(005K2.02) Knowledge of electrical power supplies to the following: RESIDUAL HEAT REMOVAL SYSTEM Containment isolation valves	Group #	1
	K/A #	K2.02.
	Rating	3.3

Question 5

Per SOP-102A, Residual Heat Removal System, what is the NORMAL and ALTERNATE power supply (if any) for the following RHR Isolation Valve?

- 1/1-8701A, RHRP 1 HL RECIRC ISOL VLV

Normal MOV power supply is 1

Alternate MOV power supply is 2

- A. (1) 1EB3-2
(2) 1EB4-2
- B. (1) 1EB3-2
(2) No alternate power supply
- C. (1) 1EB4-2
(2) 1EB3-2
- D. (1) 1EB4-2
(2) No alternate power supply

Answer: B

Explanation:

- A. INCORRECT: (1) correct. (2) There is no alternate supply for 1/1-8701A. This distractor is plausible because this the normal and alternate supply for 1/1-8701B
- B. CORRECT: (1) The normal power supply for 1/1-8701A is 1-EB3-2. (2) There is no alternate power supply for 1/1-8701A.
- C. INCORRECT: (1) The normal power supply for 1/1-8701A is 1-EB3-2. (2) There is no alternate power supply for 1/1-8701A. This distractor is plausible because this the normal and alternate supply for 1/1-8702A
- D. INCORRECT: (1) The normal power supply for 1/1-8701A is 1-EB3-2. (2) Correct. This distractor is plausible because this the normal and alternate supply for 1/1-8702B

Technical References:

SOP-102A RHR System, Revision 24, pages 25 normal and 95 alternate

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.RH1.OB03

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)7	

Examination Outline Cross-Reference	Level	RO
005 Residual Heat Removal System	Tier #	2
	Group #	1
Knowledge of the effect that a loss or malfunction of the Residual Heat Removal System will have on the following systems or system parameters: ECCS	K/A #	K3.05
	Rating	4.3

Question 6

A large break loss of coolant accident (LB LOCA) occurred on Unit 1.

- 1-01 Residual Heat Removal (RHR) pump tripped shortly after safety injection actuated and could not be recovered.
- 1/1-8804A, RHRP 1 TO CCP SUCT VLV will not open.
- All remaining Emergency Core Cooling System (ECCS) pumps and valves have been aligned per Step 3 of EOS-1.3A, Transfer to Cold Leg Recirculation.

After alignment for cold leg injection has been completed, 1-02 RHR pump is supplying which of the following pumps?

- A. 1-02 SIP ONLY
- B. 1-02 CCP and 1-02 SIP ONLY
- C. Both CCPs and both SIPs
- D. Both SIPs ONLY

Answer: C

Explanation:

There was no way to get at this KA without going to EOPs due to the nature of the KA and systems configuration at Comanche Peak.

Even with one RHP pump tripped, EOS-1.3A, Transfer to Cold Leg Recirculation Step 3 directs opens 1/1-8804A, 1/1-8804B, 1/1-8807A, and 1/1-8807B to allow one train of RHR to supply both CCPs and both SIPs and directly inject into two cold legs. Even with 1/1-8804A failing to open, 1-02 RHR pump discharge is aligned to the suction header for both CCPs and both SIPs.

A. is incorrect but plausible since 1-02 RHR pump directly supplies 1-02 SIP suction line. Applicant may assume that with 1/1-8104A failing to open, the flow path to the remaining SIP and both CCPs is unavailable. However, the procedure directs aligning the suction cross connect header through 1/1-8807A and 1/1-8807B and opens 1/1-8804B such that both CCPs and both SIPs may be supplied from 1-02 RHR pump.

B. is incorrect but plausible since candidate may assume that with 1/1-8804A failing to open, the only flowpath that remains available is to the Train B pumps.

C. is correct. Even with 1/1-8804A failing to open, the suction crossconnect flowpath is aligned to permit one RHR pump to supply both CCPs and both SIPs.
D. is incorrect but plausible since 1-02 RHR injects into the suction lines for the SIPs. Applicant may assume that with 1/1-8804A failing to open, the CCP suction flowpath is unavailable. However, EOS-1.2A opens 1/1-8807A and 1/1-8807B to permit flow from either RHR pump discharge to the CCP suction line.

Technical References:

EOS_1.3A, Transfer to Cold Leg Recirculation, Revision 9, pages 4 and 5.

References to be provided to applicants during exam: None.

Learning Objective: LO21SYSRH1, I.E.2.e

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)(7)	

Examination Outline Cross-Reference	Level	RO
006 Emergency Core Cooling	Tier #	2
	Group #	1
Ability to predict and/or monitor changes in parameters associated with operation of the Emergency Core Cooling System, including: Boron concentration in safety injection tank and BAT	K/A #	A1.02
	Rating	3.4

Question 7

Unit 1 is at full power.

It is desired to raise level in 1-02 SI Accumulator per SOP-202A, Safety Injection Accumulators Section 5.4.1 due to level falling to 39%.

All normal makeup sources are available.

After the evolution is completed, 1-02 SI Accumulator level is 50%.

Assuming the RWST **has not recently** been diluted below its minimum Boron concentration, within 6 hours of completing the fill operation, which actions are required per SOP-202A?

- A. Verify **ONLY** 1-02 Accumulator Boron Concentration is between 2300 and 2600 ppm.
- B. Verify **ALL** Accumulator Boron Concentrations are between 2300 and 2600 ppm.
- C. Verify **ONLY** 1-02 Accumulator Boron Concentration is between 2500 and 2600 ppm.
- D. Verify **ALL** Accumulator Boron Concentrations are between 2500 and 2600 ppm.

Answer: A

Explanation:

A is correct. The note in SOP-202A Section 5.4.1 states Within 6 hours of a volume increase of greater than 101 gallons (12% of span), proper accumulator Boron concentration shall be verified to be between 2300 and 2600 ppm unless the only makeup source is the RWST **AND** the RWST has NOT been diluted below its minimum Boron concentration. Since the RWST has not recently been diluted and the RWST is NOT the only source of makeup, only the recently filled accumulator shall be verified.

B is incorrect but plausible because all accumulator Boron concentrations are verified at various points in the procedure.

C is incorrect. It is plausible because a note in Section 5.1 requires establishing Accumulator #2 Boron Concentration between 2500 and 2600 ppm due to higher in-leakage.

D is incorrect. It is plausible because of the reasons listed in both C and D.

Technical References:

SOP-202A, revision 18, page 5, 26.

References to be provided to applicants during exam: None.

Learning Objective: Document learning objective if possible.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b) (5) 55.43	

Examination Outline Cross-Reference	Level	RO
007 Pressurizer Relief / Quench Tank (SF5)	Tier #	2
	Group #	1
Ability to monitor automatic features of the pressurizer relief / quench tank system, including components that discharge to the PRT / quench tank	K/A #	A3.01
	Rating	3.4

Question 8

Unit 2 is in Mode 1.

PRT level and pressure are rising.

The following annunciators are in alarm:

- 2-ALB-5B 3.3 PRT PRESS HI
- 2-ALB-5B 4.3 PRT LEV HI/LO

Which of the following could cause these indications?

- A. PRT vent isolation valve leakage
- B. CVCS letdown relief valve leakage
- C. Reactor vessel flange inner seal leakage
- D. RHR pump discharge relief valve leakage

Answer: B.

Explanation: The CVCS letdown relief inside containment discharges to the PRT.

A is wrong because leakage through the PRT vent isolation valves to the PRT would not affect PRT level. Plausible because nitrogen system leakage could result in a PRT high pressure alarm.

B is correct. See explanation.

C is wrong because reactor vessel flange leakoff goes to the RCDT. Plausible because like CVCS letdown, it is a contaminated source that is collected inside containment, and a candidate could think it goes to the PRT.

D is wrong because RHR pump discharge relief valve effluents go to the Boron Recycle System. Plausible because the RHR pump suction relief valves discharge to the PRT

Technical References:

LO21.SYS.CS1 (rev. 0) pg. 11 – Chemical and Volume Control System
 LO21.SYS.RC1 (rev. 02.0001) pg. 61, 93 – Reactor Coolant System
 LO21.SYS.RWS (rev. 0) pg. 15 – Radioactive Waste Systems

References to be provided to applicants during exam: None

Learning Objective: DESCRIBE the components of the Reactor Coolant system including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-0250, Reactor Coolant System. (LO21.SYS.RC1.OB03)

Question Source:	Bank #	Wolf Creek – 2017-11 Q7 Changed alarms and distractors A and D for plant applicability. Made editorial modifications.
	Modified Bank # New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.(b) (7)	

Examination Outline Cross-Reference	Level	RO
007 Pressurizer Relief / Quench Tank (SF5)	Tier #	2
	Group #	1
Ability to monitor automatic features of the pressurizer relief / quench tank system in the control room - nitrogen block valve	K/A #	A4.03
	Rating	3.4

Question 9

Concerning the admission of nitrogen to the PRT through 1NG-0028, (N2 SPLY SYS X-01 TO RC PRZR RLF TK 1-01 ISOL VLV):

Procedurally, which of the following situations would require this valve to be closed?

- A. While raising PRT level following a low-level indication
- B. While performing mitigating actions for a leaking PORV
- C. While performing recovery actions after a PORV discharge
- D. While venting PRT fission gasses to the Waste Gas System

Answer: D

Explanation: One of the automatic functions of the PRT system is that the N2 blanket is normally maintained at a prescribed setpoint using a pressure regulator. The 1NG-0028 is the N2 supply isolation valve for this regulator. According to SOP-109, 1NG-0028 is closed to permit venting PRT fission gasses to the Waste Gas System.

A is incorrect but plausible since raising PRT level would likely also raise PRT pressure. Isolating the nitrogen supply is a plausible action to prevent potentially radioactive gases from being discharged back the nitrogen supply system.

B is incorrect because a leaking PORV is mitigated by cycling block valves. Plausible since isolating nitrogen supply would prevent water discharge to the nitrogen system if PRT level continues to rise from the leaking PORV.

C is incorrect because recovery actions to cool the PRT after a PORV discharge will likely require water level to be initially lowered. As level is lowered, PRT pressure would decrease, requiring nitrogen to remain available to the PRT to maintain pressure. Plausible because N2 would not be needed to maintain PRT pressure if the tank pressure was significantly higher than normal operating pressure. It could also be thought that the N2 system would need to be protected from backflow.

D is correct. 1NG-0028 is a normally open valve that must be closed to permit venting the PRT.

Technical References:

LO21.SYS.RC1 (rev. 02.0001) pg. 61, 93 – Reactor Coolant System
SOP-109A (rev. 15) pp. 15, – Pressurizer Relief Tank (see attached)
ALM-0052A (rev. 6) pp. 63-66
SOP-110A (rev. 10) pp. 14-17

References to be provided to applicants during exam: None

Learning Objective: DESCRIBE the components of the Reactor Coolant system including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-0250, Reactor Coolant System. (LO21.SYS.RC1.OB03)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41.(b) (7)	

Examination Outline Cross-Reference	Level	RO
008 - Component Cooling Water System (SF8)	Tier #	2
	Group #	1
Ability to (a) predict the impacts of the following on the Component Cooling Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Effect of loss of instrument and control air on the position of air-operated CCW valves	K/A #	A2.05
	Rating	3.2

Question 10

If the instrument air header pressure is at 40 psig and continuing to lower:

1. What would be the **immediate effect** on the Component Cooling Water regulating valves for the Safety Chilled Water Condensers?
2. Independent of the above, what mitigating actions, per CPNPP procedures, may be taken to manually maintain a Safety Chilled Water condenser's temperature if an abnormal condition were to occur?
 - A. 1. CCW outlet temperature from the Safety Chilled Water condenser would lower.
2. Locally control the CCW AOV regulating valve to the Safety Chilled Water condenser.
 - B. 1. CCW outlet temperature from the Safety Chilled Water condenser would not change.
2. Locally control the CCW AOV regulating valve to the Safety Chilled Water condenser.
 - C. 1. CCW outlet temperature from the Safety Chilled Water condenser would lower.
2. Feed and Bleed the Safety Chilled Water System.
 - D. 1. CCW outlet temperature from the Safety Chilled Water condenser would not change.
2. Feed and Bleed the Safety Chilled Water System.

Answer: B

Explanation: (1) the CCW regulating valves to the Safety Chilled Water Condensers have a 30-minute accumulator supply of air and only fail open after the accumulator air supply is exhausted; therefore, upon a loss of air, the position of the affected regulating valves would remain unchanged initially and the CCW outlet temperature from the Safety Chilled Water condenser would remain unchanged. (2) Per ABN-301, the CCW AOV regulating valves to

the Safety Chilled Water condenser, u-PV-4552 & 4553, may be manually manipulated to maintain outlet temperature after loosening the valve's locknut.

A Incorrect. (1) This portion is wrong. Plausible because the CCW reg. valves are fail open valves and would go to that position if an accumulator were not present. (2) This portion is correct.

B is correct, see Explanation section above.

C Incorrect. (1) This portion is wrong, see A(1). (2) This portion is wrong because feed and bleed is only performed to address chemistry concerns or to flush the piping. Plausible because a feed and bleed of a running Safety Chiller could potentially alter the temperature of the system; however, this would not be in accordance with CPNPP procedures. See SOP-815A.

D Incorrect. (1) This portion is correct, see Explanation section above. (2) This portion is wrong, see C(2)

Technical References:

ABN-301 (rev. 15) pg. 19
LO21SYSCC1 (rev. 00) pg. 14
SOP-815A (rev. 17) pg. 36

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.301.OB04 – Analyze the response to loss of Instrument Air in accordance with ABN-301 Instrument Air System Malfunction.

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41(b) (5)	

Examination Outline Cross-Reference	Level	RO
EPE 007 – Reactor Trip	Tier #	1
	Group #	1
Knowledge of the relationship between reactor trip, stabilization and recovery and the CVCS system	K/A #	EK2.13
	Rating	3.5

Question 11

Per EOS-0.1A, Reactor Trip Response Step 3c. RNO, 1/1-LCV-112A, VCT LVL CTRL VLV is placed in AUTO for which of the following conditions?

- A. If Recycle Holdup Tank high level alarm is received
- B. If a Reactor Coolant Pump is required to be restarted
- C. If Volume Control Tank level is lowering
- D. If Centrifugal Charging Pump suction is aligned to Refueling Water Storage Tank

Answer: D

Explanation: According to EOS-0.1A, Step 3 Basis, LCV-112 is placed in AUTO when the CCP suction is aligned to the RWST, to prevent overfilling the VCT.

A is wrong because in AUTO mode, LCV-112A automatically diverts to RHUT on increasing VCT Level. Therefore, a high RHUT level would not be prevented by placing the valve in AUTO. Plausible because the relationship between these two tanks could be confused.

B is wrong. See Explanation above. Plausible because backpressure from VCT affects seal leakoff flow.

C is wrong. See Explanation above. Plausible because the purpose of the LCV-112 AUTO position may be confused.

D is correct. The normal CCP flow path from the VCT to the CCPs is not in service when the CCPs are aligned to the RWST, so VCT inventory needs to be controlled by an alternate method. In AUTO mode, LCV-112A will divert to the RHUT as level rises.

Technical References:

EOS-0.1A (Rev. 9), Attachment 4, pg. 10, 29

References to be provided to applicants during exam: None.

Learning Objective: [LO21.SYS.CS1.OB04] - EXPLAIN the normal, abnormal and emergency operation of the Chemical and Volume Control system in accordance with referenced procedures.

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41(b)(8,10)	

Examination Outline Cross-Reference
(000008) (APE 8) Pressurizer Vapor Space
Accident

Level
Tier #
Group #
K/A #
Rating

RO
1
1
AA1.02
3.6

Ability to operate and/or monitor the following
as it applies to a Pressurizer vapor space
accident: CVCS system to control PZR
level/pressure

Question 12

A Pressurizer PORV is partially open.

The crew is performing ECCS flow reduction actions per EOS-1.2A, Post-LOCA Cooldown and Depressurization.

All charging and SI pumps are running.

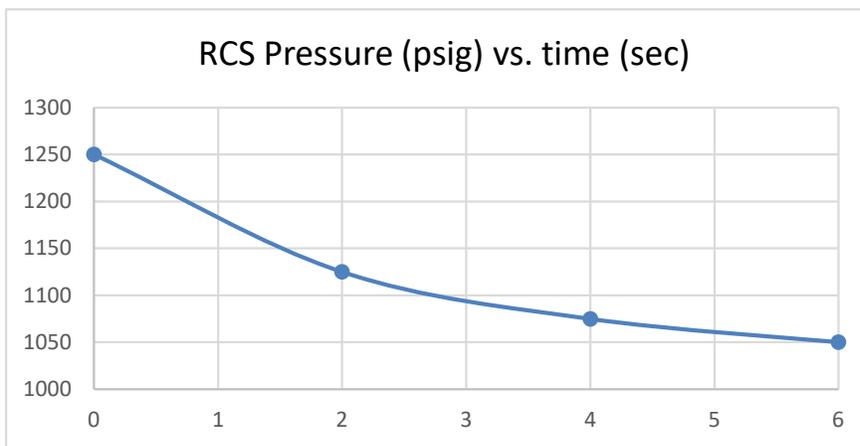
Both RHR pumps are secured.

The following parameters are observed:

- RCS subcooling 100°F
- RCS temperature 360°F
- Pressurizer level 55%

Conditions are met for securing one of the running Charging pumps per Step 16.

When the charging pump is secured, the following trend of RCS pressure is observed:



Per EOS-1.2A, which of the following actions should be taken **NEXT**?

- Restart the RHR pumps.
- Immediately restart the Charging pump that was stopped.

- C. Stop an additional Safety Injection pump.
- D. Verify RCS pressure stabilizes or increases before stopping another pump.

Answer: D

Explanation:

A is wrong. The note prior to step 16 states pressure should be allowed to stabilize. Plausible - RHR pumps are restarted if subcooling is low when checking reduction criteria.

B is wrong. The note states pressure should be allowed to stabilize or increase. Plausible – if pressure is dropping, it could be thought that the pump must be restarted in an attempt to stop the pressure drop.

C is wrong. A period of time should pass to ensure RCS pressure will stabilize. Plausible – this is the action to take once pressure has stabilized.

D. Correct. Per EOS-1.2A NOTE - After stopping any ECCS Pump, RCS Pressure should be allowed to stabilize or increase before stopping another ECCS Pump. From the bases: After an SI pump is stopped, RCS pressure may decrease rapidly to a new equilibrium value where the reduced SI flow again matches leakage from the RCS. The criteria for stopping the next SI pump

have 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

Technical References:

EOS-1.2A, Revision 9, page 13.

References to be provided to applicants during exam: None.

Learning Objective: ANALYZE the Pressurizer Vapor Space LOCA IAW the CPNPP FSAR and Westinghouse ERG background documents. (LO21.MCO.TAA.OB06)

Question Source:	Bank # Modified Bank # New	DC-2007-04 Q40
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (5)	

Examination Outline Cross-Reference (000009) (EPE 9) Small Break LOCA	Level Tier # Group #	RO 1 1
Knowledge of the reasons for the following responses and/or actions as they apply to a small-break LOCA: Maintenance of heat sink	K/A # Rating	EK3.22 3.9

Question 13

Unit 2 has been operating at 100% power for several months.

An RCS leak develops, and operators enter ABN-103, Excessive Reactor Coolant Leakage.

With both centrifugal charging pumps in service and letdown reduced to 45 gpm, Pressurizer level continues to trend down at approximately 1% per minute.

A manual reactor trip and Safety Injection is initiated per ABN-103.

MSIVs are currently open.

Due to equipment failures, AFW flow is unable to be maintained greater than 460 GPM.

As directed by EOP-0.0B, Reactor Trip or Safety Injection, FRH-0.1B, Response to Loss of Secondary Heat Sink is entered.

Based on the above scenario, why would loss of a secondary heat sink be a concern?

- A. Reactor Coolant System pressure may remain too high to provide adequate core cooling from Emergency Core Cooling System.
- B. Refueling Water Storage Tank volume may not be sufficient to provide adequate core cooling until recirculation is established.
- C. Steam Generator tubes may be damaged or rupture upon reintroduction of Auxiliary Feed Water flow.
- D. In service Safety Injection systems are susceptible to loss of power should a Station Blackout event occur.

Answer: A

Explanation: At normal operating pressure, max charging flow with two CCPs would be approximately 220 – 230 gpm due to line restrictions. Pressurizer level drop corresponds to approximately 70 gpm so total leakage would be approximately 230 gpm – 45 gpm (letdown) – 12 gpm (RCP seal Leakoff) + 70 gpm (Pressurizer level drop) = 243 gpm, which would be considered a small break. An SI will start all ECCS pumps, including the CCPs, but the flow of RWST water into the RCS at the maximum rate of the charging pumps near operating pressure would not be adequate to remove all decay heat with break flow and allow for a subsequent cooldown if a secondary heat sink were not available.

A is correct. See Explanation.

B is wrong, but plausible because the finite amount of water in the RWST is a concern in large break LOCAs and recirculation must be established to provide adequate decay heat removal.

C is wrong. SG tube damage is of minimal concern in this scenario, but plausible because SG tubes are susceptible to damage upon reintroduction of AFW flow. However, this is only stated as a caution in the bases of FRH 0.1B and does not prevent the reestablishment of AFW flow.

D is wrong. The likelihood of a loss of both safety busses is of minimal concern in this scenario. Plausible because an actual SBO event would cause a loss of all SI pumps.

Technical References:

LO21MCOTAA (Rev. 0) pp. 7-9

FRH-0.1B (Rev. 9) pg. 55 – Response to Loss of Secondary Heat Sink

References to be provided to applicants during exam: None.

Learning Objective: LO21.MCO.TAA.OB03 – ANALYZE the core cooling mechanisms of a LOCA IAW the CPNPP FSAR and Westinghouse ERG background documents.

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b) (5 & 10)	

Examination Outline Cross-Reference (000011) (EPE 11) Large Break LOCA	Level Tier # Group #	RO 1 1
Ability to determine and/or interpret the following as they apply to large-break LOCA: Significance of ECCS pump operation	K/A # Rating	EA2.05 4.4

Question 14

A large-break LOCA occurred on Unit 2.

EOS-1.3B, Transfer to Cold Leg Recirculation has just been completed through Step 4 and both trains of Emergency Core Cooling System and Containment Spray have been aligned for cold leg recirculation.

Severe oscillations are now noted on 2-02 Residual Heat Removal (RHR) pump suction and discharge pressure, discharge flow, and motor current.

No other abnormal indications are observed.

Per EOS-1.3B, how should operators respond to these indications?

- A. Reduce 2-02 RHR pump flow per ABN-104, Residual Heat Removal System Malfunction.
- B. Stop 2-02 RHR pump. Shift its suction back to the RWST, then restart 2-02 RHR pump.
- C. Stop 2-02 RHR pump. Monitor 2-01 RHR pump, both Centrifugal Charging pumps, and both Safety Injection pumps.
- D. Stop 2-02 Centrifugal Charging pump and 2-02 Safety Injection pump and verify 2-02 RHR pump oscillations improve.

Answer: C

Explanation: The indications of fluctuating amps and flows on an RHR pump indicate possible sump screen blockage and pump cavitation. According to EOS-1.3B, step 5, when the RHR pump starts to cavitate and there are two trains running with only indications of blockage on one sump or train, then only the affected train RHR pump should be stopped to prevent potential pump damage. In a Cold-Leg recirculation lineup with both ECCS trains in service, CCPs and SI pump suctions are normally cross-connected between both RHR pumps. One RHR pump can supply both trains of ECCS, therefore, an immediate trip of an RHR pump is required when its suction is lost.

A is wrong, but plausible core cooling is a priority and because they might believe that this is directed in the EOS-1.2B, but it is not part of the EOP mitigation strategy. With sump blockage indicated, the RHR pump 2 should be stopped per step 5 RNO. ABN-104 is not applicable in Modes 1, 2 or 3 and thus should not be entered.

B is wrong, but plausible because core cooling is a high priority and the RWST suction would not be compromised with strainer debris. However, one train of ECCS is designed to provide full heat removal capability, so preventing imminent damage to a cavitating pump is a much higher concern.

C is correct. See explanation.

D is wrong, but plausible because these are the actions that would be taken if only one RHR pump was in service, to protect the suctions of the CCP and SI pumps. Taking these actions would unnecessarily delay the immediate securing of the cavitating pump.

Technical References:

EOS-1.3B (Rev. 9) page 9, pages 25-26.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.SI1.OB05 – EXPLAIN the normal, abnormal and emergency operation of the Emergency Core Cooling system in accordance with SOP-201, SOP-202, and the ERG procedure network.

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (10)	

Examination Outline Cross-Reference
(000015) (APE 15) Reactor Coolant Pump
Malfunctions

Level	RO
Tier #	1
Group #	1
K/A #	G2.4.2
Rating	4.5

Knowledge of system setpoints, interlocks,
and automatic actions associated with
emergency and abnormal operating
procedures entry conditions for Reactor
Coolant Pump Malfunctions

Question 15

Unit 1 is operating at 100% power in a normal alignment when a rupture develops in RCP 1-04 Thermal Barrier Heat Exchanger.

A CCW radiation monitor has just gone into alarm.

CCW surge tank level is 78% and rising slowly.

CCW flow from RCP 1-04 Thermal Barrier has increased to 60 gpm and is stable.

CCW Thermal Barrier return temperature is 190°F and increasing.

1-HV-4694, RCP 4 THBR CLR CCW RET VLV, (1) and 1-HV-4696, THBR CLR CCW RET ISOL VLV (IRC) (2) .

- A. (1) closes
(2) closes
- B. (1) remains open
(2) closes
- C. (1) closes
(2) remains open
- D. (1) remains open
(2) remains open

Answer: C

Explanation: With a thermal barrier HX leak in progress on RCP 1-04, CCW surge tank level will begin to rise and 1-HV-4694, RCP 4 THBR CLR CCW RET VLV will close on greater than 182.5F CCW Thermal Barrier return temperature. 1-HV-4696, THBR CLR CCW RET ISOL VLV (IRC) will close on greater than 64 gpm CCW flow from RCP 1-04 thermal barrier. Therefore, only 1-HV-4694 will close, while 1-HV-4696 remains open.

A. First part is correct, see above explanation. Second part is incorrect but plausible since 1-HV-4696 does close on high return flow, however, the setpoint has not been exceeded yet.

B first part is incorrect but plausible because 1-HV-4694 closes at 182.5F. It is also plausible if the candidate thinks **only** the main thermal barrier return isolation (1-HV-4709) isolates. The second part is incorrect but plausible, see A.

C is correct, see above explanation.

D is incorrect but plausible. See B for first part, and A for second part.

Technical References:

LO21SYSCC1 (Rev. 0 Chg 1) Pgs 21, 23

ABN-101 (Rev. 15) Pg. 39

References to be provided to applicants during exam: None.

Learning Objective: DESCRIBE the components of the Component Cooling Water system including interrelations with other systems to include interlocks and control loops as described in DBD-ME-229. (LO21.SYS.CC1.OB03)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b) (7)	

Examination Outline Cross-Reference
(000022) (APE 22) Loss of Reactor Coolant
Makeup

Level	RO
Tier #	1
Group #	1
K/A #	AK1.03
Rating	3.6

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to a loss of reactor coolant makeup: Relationship between charging flow and PZR level

Question 16

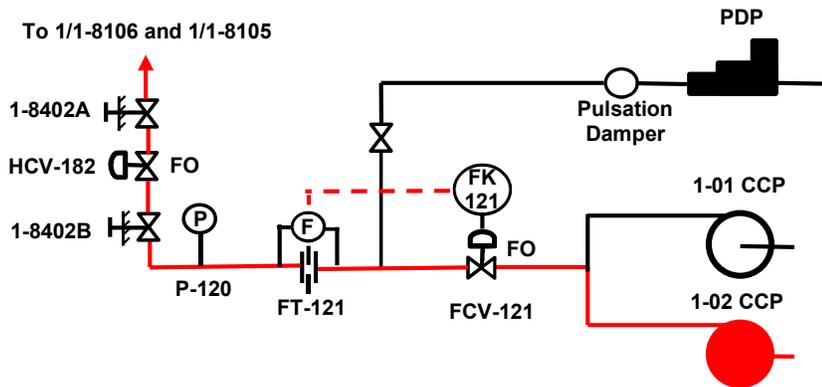
Unit 1 is at 90% power.
1-02 CCP is in service with letdown flow at 45 gpm.
All control systems are in automatic.

A 60 gpm leak occurs at a piping weld near 1-8402A, U1 RC PMP SL WTR PRESS CTRL VLV DNSTRM ISOL VLV.

- The Unit Supervisor enters ABN-103, Excessive Reactor Coolant Leakage.
- Total seal injection flow is maintained at 32 gpm.

ABN-103, Section 2.3, Step 2 states, "VERIFY PRZR level – STABLE OR TRENDING TO NORMAL LEVEL".

With no other operator actions, expected Pressurizer level would ___(1)___.



After a short time, 1-ALB-6A Window 1.5 VCT PRESS HI/LO alarm annunciates.

- 1-PI-115, VCT PRESS indicates 14 psig and begins slowly rising due to automatic makeup.

ALM-0061A, Alarm Procedure 1-ALB-6A Window 1.5, Step 1 states, "Monitor 1-PI-115, VCT PRESS".

The purpose for maintaining VCT pressure above the low pressure alarm is to ___(2)___.

- A. (1) initially lower, then return to original level

- (2) prevent gas binding of charging pumps
- B. (1) lower to a stable level less than prior to the leak
(2) prevent gas binding of charging pumps
- C. (1) initially lower, then return to original level
(2) maintain minimum backpressure on RCP seals
- D. (1) lower to a stable level less than prior to the leak
(2) maintain minimum backpressure on RCP seals

Answer: C

Explanation:

(1) A leak of approximately 60 gpm downstream of the charging pumps would initially cause a slightly lower PZR level, which would cause the PI (Proportional + Integral) type PZR level controller to throttle open FCV-121 more to raise charging flow. The increased flow would compensate for the leakage losses and bring PZR level back to program level. A leak of approximately 60 gpm, plus approximately 45 gpm letdown, plus 12 gpm RCP seal leakoff is within the rated capacity of a CCP, (150 gpm at 2590 psig, higher flow for a primary pressure of 2235 psig)

(2) According to SOP-103A, pressure in the VCT should be maintained high enough to provide a minimum back pressure on the RCP seals. Without adequate backpressure, reverse pressurization and seal damage could occur.

A. (1) This portion is correct. (2) Is wrong because preventing gas binding of charging pumps is accomplished by suction lines equipped with suction line vents aligned to the VCT that allow gases from pump suction lines to be continuously vented. In addition, automatic swapping of the charging pump suction line to the RWST is designed to provide a continued suction source to maintain RCS charging should VCT level become low enough to potentially cause gas binding. These design features should reduce the incidence of vapor binding. Plausible because the ALM and SOP-103 state vapor binding and cavitation as a concern, but this would normally be associated with a high VCT pressure or a low VCT level without automatic charging pump suction swap.

B. (1) Is wrong because the PI controller will always bring PZR level back to setpoint as long as flow demand is within the capacity and above the minimum required flow of the charging pumps. Plausible because a leak greater than the capacity of the running CCP or a different kind of controller could result in a lower PZR level. Also, the location of the leak could lead to confusion as to how the system might respond. PZR level is an indication that is often used to diagnose a larger LOCA and the distinction between a small LOCA within the capacity of the CCPs and a larger LOCA might be lost. (2) This portion is correct.

C is correct, see explanation.

D is wrong. See statements for A and B for plausibility.

References to be provided to applicants during exam: Drawing in stem.

Question Source:	Bank Mod New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Mem/Fund Higher/Cog	3
10 CFR Part 55 Content:	55.41(b)5 & 7	

Examination Outline Cross-Reference
(000025) (APE 25) Loss of Residual Heat
Removal System

Level	RO
Tier #	1
Group #	1
K/A #	AK1.06
Rating	3.8

Knowledge of the operational implications
and/or cause and effect relationships of the
following concepts as they apply to LOSS OF
RESIDUAL Heat Removal System: Time to
boiling determinations

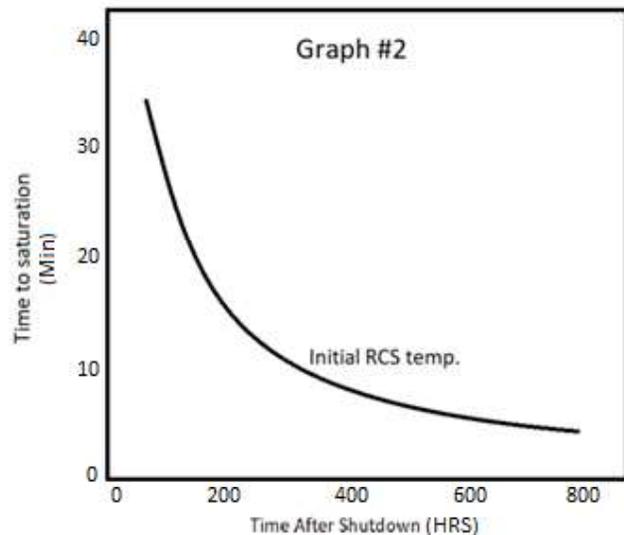
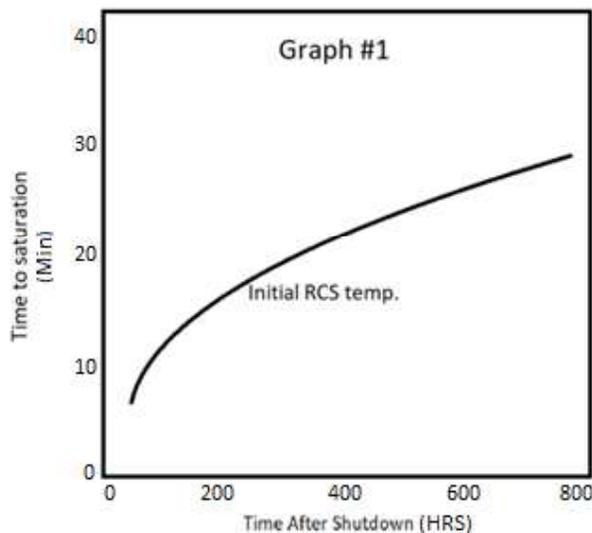
Question 17

Given the following:

- Unit 1 shutdown and started a refueling outage 4 days ago.
- The Unit is currently in Mode 6 with the RCS at reduced inventory.
- RCS temperature was 120°F when a loss of both trains of RHR occurred.
- ABN-104, Residual Heat Removal System Malfunction is entered.

(1) Which of the two graphs below most closely represents the time to saturation in the RCS given the conditions above?

(2) Would a line representing a lower initial temperature than the graphed line be below or above the line shown?



- A. (1) Graph #1
(2) Below
- B. (1) Graph #1
(2) Above
- C. (1) Graph #2

(2) Below

- D. (1) Graph #2
(2) Above

Answer: B

Explanation: The initial temperature line of Graph #1 most closely represents the time to saturation for a given time after shutdown after RHR is lost. As time after shutdown increases, decay heat decreases, therefore, the amount of time required to reach saturation in the RCS increases. A lower initial RCS temperature would be represented by a line above the one shown due to being further away from saturation conditions initially.

A is wrong. For part (2), a lower initial RCS temperature would be represented by a line above the one shown. Plausible because the idea of a lower temperature could be thought to be represented by a lower line. Also, the dashed line in Attachment 5 of ABN-104 is a lower line and might be recalled incorrectly.

B is correct, see explanation.

C is wrong, graph #2 is basically opposite to the correct correlation of the variables as explained in the explanation section above. See question A for reasoning of Part (2). Plausible because, (1) Graph #2 looks similar to the familiar “time after shutdown vs. decay heat” graph and might be confused, (2) see plausibility statement for question A.

D is wrong, see question C for explanation and plausibility for part (1) and question A for explanation and plausibility for part (2).

Technical References:

ABN-104 (Rev. 9) Residual Heat Removal System Malfunction

References to be provided to applicants during exam: Graphs in stem.

Learning Objective:

LO21.SYS.RC1.OB05 – EXPLAIN the normal, abnormal and emergency operation of the Reactor Coolant system in accordance with referenced normal, abnormal, and emergency operating procedures.

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (5 & 7)	

Examination Outline Cross-Reference
(000026) (APE 26) Loss of Component
Cooling Water

Level	RO
Tier #	1
Group #	1
K/A #	AA1.05
Rating	3.5

Ability to operate and/or monitor the following as they apply to a loss of Component Cooling Water: The CCWS surge tank, including level control, level alarms, and a radiation alarm

Question 18

Unit 2 is at full power.

The CCW surge tank makeup valve 2-LV-4500, CCW SRG TK 2-01 RMUW SPLY VLV can be operated _____ (1)

AND

the 'CCW SRG TK TRN A/B EMPTY' annunciator will _____ (2) if the current train A level is 56%.

- A. 1. locally only
2. be illuminated
- B. 1. locally and in the control room
2. be illuminated
- C. 1. locally only
2. be dark
- D. 1. locally and in the control room
2. be dark

Answer: D

Explanation:

- A Incorrect – First part is plausible since 2-LV-4500 may be operated locally using a wrench. However, the valve is normally controlled via a handswitch located on the MCB (CB-03). Second part is plausible since the annunciator would be illuminated if Unit 1 level was 50%.
- B Incorrect – First part is correct since the valve can be controlled locally or from the MCB (see A). Second part is plausible since the annunciator would be illuminated if Unit 1 level was 50%.
- C Incorrect – First part plausible (see A). Second part is correct since the Unit 2 empty alarm setpoint is 33%.

D Correct – First part is correct (see A). Second part is correct since the Unit 2 empty alarm setpoint is 33% so the light will be out or extinguished.

Technical References:

ALM-0032B (Rev. 4) pg. 58-61

ABN-301 (Rev 15) pp. 20

LO21SYSCC1 (Rev. 1)

References to be provided to applicants during exam: None.

Learning Objective: (LO21.ABN.501.OB106), and objective 7 (alarms and indications for CCW)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) (5 & 7)	

Examination Outline Cross-Reference
(000027) (APE 27) Pressurizer Pressure
Control System Malfunction

Level	RO
Tier #	1
Group #	1
K/A #	AA2.09
Rating	4.0

Ability to determine and/or interpret the
following as they apply to Pressurizer
Pressure Control System Malfunction: Reactor
power

Question 19

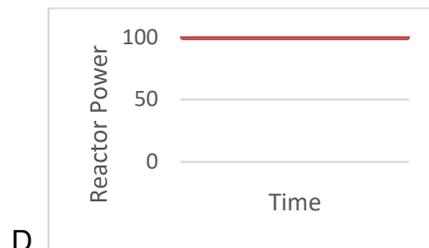
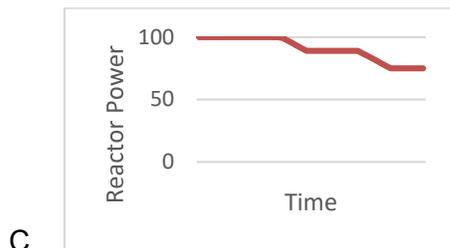
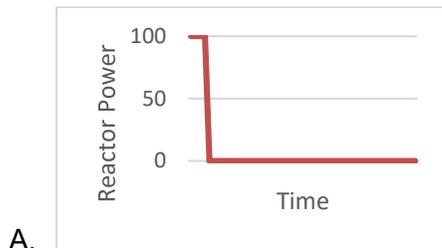
Unit 1 is operating at 100% reactor power.

Switch 1/1-PS-455F, PRZR PRESS CTRL CHAN SELECT is in the 455/456 position.

1-PI-455A, PRZR PRESS CHAN I fails high causing the following:

- 1-ZL-455B, RC LOOP 1 PRZR SPR VLV – Opens
- 1-ZL-455C, RC LOOP 4 PRZR SPR VLV – Opens
- 1/1-PCV-455A, PRZR PORV – Opens

With NO operator actions taken, which of the following graphs represents how Reactor Power will respond to this failure?



Answer: B

Explanation: In the above scenario, at 100% reactor power (2235 psig RCS press.), Pressurizer Pressure Channel 455 failed high (the selected channel). According to ABN-705, the plant response would be for one PORV and both spray valves to open. This would cause actual pressurizer pressure to lower, which would lower the OTN16 trip setpoint. Eventually an OTN16 runback would occur, lowering reactor power. PCV-455A would close

when pressure falls below 2185 psig, however the reactor would eventually trip due to both sprays still being open.

A is wrong, but plausible since the reactor does trip for this event.

B is correct, see explanation.

C is wrong, but plausible since power does lower during this event due to an OTN16 Runback.

D is wrong but plausible since a failure of a pressurizer pressure control channel has very little **direct** impact on reactor power lowering. Power only noticeably lowers due to the OTN16 runback.

Technical References:

ABN-705 – Pressurizer Pressure Malfunction (Rev. 130) pg. 4 (see below)

LO21SYSPP1 – Pressurizer Pressure and Level (Rev. 1) pg. 18 (see below)

References to be provided to applicants during exam: Graphs in stem.

Learning Objective: DESCRIBE the instrumentation and controls of the Pressurizer Pressure Control System including the system response in accordance with the CPNPP FSAR and DBD-ME-250, Reactor Coolant System. (LO21.SYS.PP1.OB04)

EXPLAIN the normal, abnormal and emergency operation of the Pressurizer Pressure and Level Control System in accordance with ABN-705, Pressurizer Pressure Malfunction, and ABN-706, Pressurizer Level Instrumentation Malfunction. (LO21.SYS.PP1.OB07)

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (10)	

Examination Outline Cross-Reference	Level	RO
029 Anticipated Transient Without Scram	Tier #	1
	Group #	1
Knowledge of the relationship between Anticipated Transient Without Scram and the following systems or components: MFW System	K/A #	EK2.15
	Rating	3.3

Question 20

Unit 1 is at full power when a reactor trip is required due to the most challenging transient for an Anticipated Transient Without Trip (ATWT).

This transient event is defined in the AMSAC training and the bases steps for FRS-0.1A, Response to Nuclear Power Generation/ATWT.

Which of the following is **one part of the most limiting event initiators** coupled with the ATWT?

- A. Loss of all main feedwater
- B. Loss of offsite power
- C. Rod ejection accident
- D. Main steam line break

Answer: A

Explanation:

Per the training for AMSAC (LO21.SYS.PQ1) and the bases document for FRS-0.1A, the limiting event initiator for an ATWT is a Loss of all MFW & Electrical Load Without a Rx Trip, Turbine Trip & AFW Flow.

A is correct because (see above) this is the AMSAC design purpose and the timing bases for the order of the EOP steps in EOP-0.0 and FRS-0.1A.

B is wrong because this is not the most challenging event initiator for ATWT. Plausible because this is a valid USAR defined transient but not most limiting or challenging for ATWT.

C is wrong because this is not the most challenging event initiator for ATWT. Plausible because this is a valid USAR defined transient but not most limiting or challenging for ATWT.

D is wrong because this is not the most challenging event initiator for ATWT. Plausible because this is a valid USAR defined transient but not most limiting or challenging for ATWT.

Technical References:

FRS-0.1A, Rev 9, page 18 (bases).
 LO21.SYS.PQ1, July 2020, slide 5.
 USAR Section 7.8, Amendment 104

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.PQ1.OB01

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)7	

Examination Outline Cross-Reference	Level	RO
E12 Uncontrolled Depressurization of All Steam Generators	Tier #	1
	Group #	1
	K/A #	EK3.07
Knowledge of the reasons for the following responses and/or actions as they apply to Uncontrolled Depressurization of All Steam Generators: controlling S/G feed flow	Rating	4.1

Question 21

Unit 2 is operating at 100% power when a full load rejection occurs.

The reactor automatically trips and pressure in all four steam generators is decreasing.

Reports from the field indicate that multiple safety valves on all four steam generators appear to have stuck open.

Post-trip actions have been performed and contingency action procedures have just been entered.

- Pressure in all four steam generators is 500 psig and decreasing.
- RCS pressure is 1700 psig and decreasing.
- All steam generator Narrow Range levels are approximately 5%.

The Unit Supervisor enters ECA-2.1B, Uncontrolled Depressurization of All Steam Generators and is at step 2, Control AFW Flow to Minimize RCS Cooldown.

Per ECA-2.1B Caution before step 2 and bases for this Caution, AFW flow should be adjusted to (1) .

The reason for controlling AFW in these given conditions above is to (2) .

- (1) 100 gpm per steam generator
(2) prevent exceeding maximum cooldown rate
- (1) 460 gpm total flow to all steam generators
(2) allow steam generator components to remain in a wet condition
- (1) 100 gpm per steam generator
(2) allow steam generator components to remain in a wet condition
- (1) 460 gpm total flow to all steam generators
(2) prevent exceeding maximum cooldown rate

Answer: C

Explanation:

In accordance with ECA-2.1B, step 2 bases, the Caution before step 2 states that “If AFW flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable AFW flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.”

The basis for step 2 states “The 100 gpm value is representative of a minimum measurable feed flow to a steam generator.” This is a minimum flow rate that is acceptable when entering this step.

A is wrong because part 2 is wrong. The basis for step 2 states that “Depending upon the size of the effective break areas for the steam generators, the cooldown rate experienced after reactor trip could exceed 100°F/hr. With four SG reliefs open and failure to close MSIV’s from control room (and the reason to have to enter this EOP) will most likely cause you to exceed the cooldown rate with these conditions. There is no guarantee that you could prevent exceeding this cooldown rate and the bases step offers this additional insight into why throttling AFW to 100 gpm is beneficial: “A reduction of AFW flow to the steam generators has three primary effects (one of which is): to **minimize any additional cooldown** resulting from the addition of AFW. Part 1 is correct.

B is wrong because part 1 is wrong. Part 2 is correct.

C is correct per the explanation above.

D is wrong because part 1 and part 2 are wrong (see explanation above)

Technical References:

ECA-2.1B, “Uncontrolled Depressurization of All Steam Generators,” Revision 9, Step 2
ECA-2.1B Bases for Step 2 and Caution before step 2

References to be provided to applicants during exam: None.

Learning Objective: LO 21.ERG.C21.OB02

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) 10	

Examination Outline Cross-Reference	Level	RO
054 Loss of Main Feedwater	Tier #	1
	Group #	1
Ability to operate and/or monitor the following as they apply to Loss of Main Feedwater:	K/A #	AA1.07
MFW pumps	Rating	3.4

Question 22

Unit 1 is at 100% power when the following conditions are observed:

- 1-ALB-7B Window 1.12 FWPT A TRIP annunciates
- 1-HS-2109, FWP A DISCH VLV - CLOSED
- 1-ZL-2111A, FWPT A LP STOP VLV - CLOSED
- 1-ZL-2111B, FWPT A HP STOP VLV - CLOSED

Per ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, operators should ensure 1/1-RBSS, CONTROL ROD BANK SELECT is in AUTO and _____.

- A. turbine runback to 700 MW is in progress
- B. turbine runback to 800 MW is in progress
- C. manually runback turbine power to 900 MW
- D. reduce turbine power in 50 MW increments to maintain reactor power < 100%

Answer: A

Explanation:

A is correct because section 2.3, feedwater pump trip, has the immediate action step (annotated by a diamond) as ensuring the turbine runs back to 700MW.

B is wrong because the ABN-302, section 4.3, immediate action for a heater drain pump trip is to ensure the turbine runs back to 70% turbine power which is approximately 812 MW; and is plausible because it is close to 700MW and is also an immediate action limit in ABN-302.

C is wrong because the ABN-302, section 6.3, immediate action for a heater drain tank level control system malfunction/loss of forward flow direction is to manually runback turbine power to 900MW; is plausible because it is an immediate action limit in ABN-302.

D is wrong because the ABN-302, section 5.3, immediate action for a feedwater heater or drain cooler tube leak is to reduce power in increments of 50MW to maintain reactor power less than 100 percent; is plausible because it is an immediate action limit in ABN-302.

Technical References:

ABN-302, "Feedwater, Condensate, Heater Drain System Malfunction," rev 14, section 2.3

References to be provided to applicants during exam: None.

Learning Objective: LO21SYSMF1 LC29

Enabling objective 5: The following objectives are evaluated from memory:
EXPLAIN the normal, abnormal and emergency operation of the Main Feedwater system in accordance with SOP-302, ABN-302 and FRH series procedures. (LO21.SYS.MF1.OB05)
Given system parameter indications and plant conditions, **ASSESS** from memory any required TS/TR entries, including any actions which must be completed within one hour in accordance with Technical Specifications or TRM. (LO21.SYS.CS1.OB06)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)4	

Examination Outline Cross-Reference	Level	RO
058 Loss of DC Power	Tier #	1
	Group #	1
Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to Loss of DC Power: Loss of breaker protection	K/A #	AK1.04
	Rating	3.7

Question 23

Unit 1 is operating at 100% power with all systems in a normal alignment.

While alternating battery chargers to 125 VDC Switchboard 1ED1, the equipment operator inadvertently opens 1ED1/1-1/DSW, 125 V DC STATION BATTERY BT1ED1 FUSED DISCONNECT SWITCH with no battery charger aligned to the bus.

Which of the following is an expected response to a loss of voltage on 1ED1?

- A. CS-1EA1-1, INCOMING BREAKER 1EA1-1 will not trip on bus fault.
- B. 1/1-PCV-456 PRZR PORV will not open on high Pressurizer pressure.
- C. 1/1-PCPX1, RCP 1 will not trip on overcurrent.
- D. 1-HS-2111C, FWPT A TRIP will not trip MFP A if depressed.

Answer: A

Explanation:

A Correct – Loss of DC power from 1ED1 results in loss of control power to Train A 6.9 kV bus breakers. Without control power, the breakers are unable to close or open remotely or due to automatic trips. Since 1EA1-1 is normally closed in a normal alignment, it would not open due to bus faults.

B Incorrect – Plausible since 1/1-PCV-456 response would be as described if 1ED2 was the affected bus. However, since 1/1-PCV-456 is powered from Train B, it will continue to function normally following a loss of Train A 125 VDC and would still open on high Pressurizer pressure.

C Incorrect – Plausible since loss of control power voltage to the RCP breaker would result in the stated response. However, RCP breaker control power is supplied by 1D3 vice 1ED1.

D Incorrect – Plausible since loss of DC voltage will prevent the MFP trip pushbutton from tripping the MFP. However, MFP A trip circuit is supplied by 1D2 thus the pushbutton will continue to function with a loss of 125 VDC from 1ED1.

Technical References: DC Electrical Load List spreadsheet with LO21.SYS.DC1

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.DC1

Enabling Objective 2: **DESCRIBE** the components of the DC Electrical Distribution system including interrelations with other systems to include interlocks and control loops in accordance with DBD-EE-044, DC Power Systems. (LO21.SYS.DC1.OB02)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b).7	

Examination Outline Cross-Reference	Level	RO
062 Loss of Nuclear Service Water	Tier #	1
	Group #	1
Knowledge of the relationship between	K/A #	AK2.02
Knowledge of the relationship between Loss of	Rating	3.0
Service Water and the following systems or		
components: IAS		

Question 24

Unit 2 is operating at 100% power during the Summer.

A loss of which of the following will result in a degradation of cooling to 2-02 Instrument Air Compressor?

1. Unit 2 Component Cooling Water
2. Ventilation Chilled Water
3. Turbine Plant Cooling Water

- A. 1 only.
- B. Either 1 or 2.
- C. Either 1 or 3.
- D. Either 2 or 3.

Answer: B

Explanation:

The unit-specific Instrument Air Compressors are cooled by their associated CCW non-safeguards loop, but because of high ambient temperatures in the turbine building in the Summer, the IACs have an additional trim cooler on the CCW inlet side to provide pre-cooling from Ventilation Chilled Water. Therefore a loss of either CCW or Ventilation Chilled Water will cause a degradation in cooling to a unit-specific IAC.

A is wrong. While CCW does provide the direct cooling to IAC 2-02, a loss of Ventilation Chilled Water subsystem 2 would cause a loss of cooling to the trim cooler, also degrading cooling.

B is correct. See above.

C is wrong because TPCW does not provide cooling to the unit specific IACs. Plausible because it does provide cooling to the Common IACs X-01 and X-02.

D is wrong because TPCW does not provide cooling to the unit specific IACs. Plausible because it does provide cooling to the Common IACs X-01 and X-02.

Technical References:

LO21.SYS.IA1, Instrument Air, Rev 1, pg 18.

References to be provided to applicants during exam: None.

Learning Objective: DESCRIBE the components of the Instrument Air system including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-218. (LO21.SYS.IA1.OB03)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b).4	

Examination Outline Cross-Reference	Level	RO
065 Loss of Instrument Air	Tier #	1
	Group #	1
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	K/A #	G2.1.7
	Rating	4.4

Question 25

Given the following conditions in Unit 2:

- 2-01 Instrument Air Compressor (IAC) is aligned as LEAD compressor.
- 2-02 Instrument Air Compressor is aligned as BACKUP compressor.
- X-02 Instrument Air Compressor is aligned as STANDBY compressor.

How would the Instrument Air (IA) system be expected to respond to a slow leak in Unit 2?

- 2-01 IAC increases motor frequency to try to maintain pressure at 110 +/- 5 psig. When IA header pressure drops to 100 psig, 2-02 IAC will start and load. When IA header pressure drops to 95 psig, X-02 IAC will start and load to maintain pressure.
- 2-01 IAC loads at a constant speed at 105 psig. When IA header pressure drops to 100 psig, 2-02 IAC will start and load. When IA header pressure drops to 95 psig, X-02 IAC will start and load to maintain pressure.
- 2-01 IAC increases motor frequency to try to maintain pressure at 110 +/- 5 psig. When IA header pressure drops to 95 psig, 2-02 IAC will start and load. When IA header pressure drops to 85 psig, X-02 IAC will start and load to maintain pressure.
- 2-01 IAC loads at a constant speed at 105 psig. When IA header pressure drops to 95 psig, 2-02 IAC will start and load. When IA header pressure drops to 85 psig, X-02 IAC will start and load to maintain pressure.

Answer: B

Explanation:

Unit 1 has Kobelco IACs, which can operate either in a variable frequency drive, or constant speed load/unload mode of operation. Unit 2 and the Common IACs are Atlas Copco models, which can only operate in a constant speed load/unload mode of operation. The lead compressor loads at 105 psig, a Unit 2 backup loads at 100 psig, and a Common standby loads at 95 psig. All 3 would unload at 115 psig if header pressure were restored.

A is wrong. Variable frequency drive is only a mode 1 of operation for the Unit 1 IACs, which are a different model (Kobelco) than the Unit 2 IACs (Atlas Copco). Unit 2 IACs can only load and unload at a constant speed.

B is correct. See above.

C is wrong. Variable frequency drive is only a mode 1 of operation for the Unit 1 IAC. Unit 2 IACs can only load and unload at a constant speed. Also, IAC 2-02 in BACKUP mode loads at 100 psig, and X-02 loads at 95 psig. 85 psig is the Instrument Air Header Pressure Low alarm setpoint, not a loading setpoint.

D is wrong. IAC 2-02 in BACKUP mode loads at 100 psig, and X-02 loads at 95 psig. 85 psig is the Instrument Air Header Pressure Low alarm setpoint, not a loading setpoint.

Technical References:

LO21.SYS.1A1, Instrument Air, Rev 1, pp 7-8.

References to be provided to applicants during exam: None.

Learning Objective:

EXPLAIN the normal, abnormal and emergency operation of the Instrument Air system in accordance with SOP-509, ALM-0011A/B and ABN-301. (LO21.SYS.IA1.OB05)

DIFFERENTIATE between the Unit 1 and 2 Instrument Air systems in accordance with DBD-ME-218. (LO21.SYS.IA1.OB07)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b).4	

Examination Outline Cross-Reference	Level	RO
W E04 LOCA Outside Containment	Tier #	1
	Group #	1
Knowledge of nonlicensed operator responsibilities during an emergency	K/A #	2.4.35
	Rating	3.8

Question 26

ECA-1.2A, LOCA Outside Containment Step 1. b directs 1/1-8840, RHR TO HL 2& 3 INJ ISOL VLV be verified closed.

If 1/1-8840 is open and cannot be manually closed from the Control Room, ECA-1.2A directs locally closing the valve.

Which of the following locations must be accessible to permit closing the valve locally?

- A. Remote Shutdown panel
- B. Safeguards Building 790' valve isolation tank room
- C. Safeguards Building 810' south penetration valve room
- D. Safeguards Building 832' penetration valve room

Answer: C

Explanation:

A is wrong because the step 1 valves can be controlled from the RSP but not this valve.
 B is wrong because the valve cannot be controlled from this room since it is located in the south penetration room just outside Containment. Plausible because other ESF isolation valves are located and can be locally controlled in this room that is adjacent to Containment.
 C is correct because this is the location.
 D is wrong because you cannot control this valve from this penetration room. Plausible because other RCS and ESF valves are located and can be controlled in this room that is adjacent to Containment.

Technical References:

ECA-1.2A, Revision 9.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.C12.Obj 5

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.41(b)3

Examination Outline Cross-Reference	Level	RO
WE11 Loss of Emergency Coolant Recirculation	Tier #	1
	Group #	1
	K/A #	K3.15
Knowledge of the reasons for the following responses and/or actions as they apply to Loss of Emergency Coolant Recirculation: Isolating or venting SI accumulators	Rating	3.3

Question 27

Unit 1 experienced a small break LOCA.

While performing step 29 of ECA-1.1, Loss of Emergency Coolant Recirculation, Safety Injection Accumulators will be isolated when two RCS hot leg loops are less than a MAXIMUM of _____ (1) _____ and the reason for this is to _____ (2) _____.

- A. (1) 200 °F
(2) ensure RCS saturation pressure is higher than accumulator pressure **after** accumulator water has been discharged
- B. (1) 405 °F
(2) ensure RCS saturation pressure is higher than accumulator pressure **after** accumulator water has been discharged
- C. (1) 200 °F
(2) ensure RCS saturation pressure is higher than accumulator pressure **before** accumulator water has been discharged
- D. (1) 405 °F
(2) ensure RCS saturation pressure is higher than accumulator pressure **before** accumulator water has been discharged

Answer: B

Explanation:

According to ECA-1.1, step 29, two hot leg loops should be less than 405 °F before isolating the accumulators.

A Incorrect – First part is plausible because 200°F is checked to determine whether continued cooldown via steam generators is required. Second part is correct since the goal of ECA-1.1 is to empty liquid contents of the accumulators, because it extends time that core cooling can be maintained, but still prevent nitrogen from injecting into RCS. Consequently, a minimum RCS hot leg temperature is specified so that even with the RCS saturated at that temperature, enough RCS pressure would exist to prevent accumulator nitrogen from injecting into the RCS after the liquid volume has been displaced to the RCS.

B Correct – Per ECA-1.1, step 29, two hot leg loops should be less than 405°F before isolating accumulators. Step 29 basis states hot leg temperature of 405°F should be verified so that the RCS saturation pressure exceeds the accumulator pressure after the accumulator water has been discharged to preclude nitrogen injection into the RCS.

C Incorrect - First part is plausible because 200°F is checked to determine whether continued cooldown via steam generators is required. Second part is plausible since preventing accumulator water from injecting would also prevent nitrogen injection. However, this is incorrect since with loss of emergency coolant recirculation capability, the goal is to maximum time available with core cooling maintained to permit recovery of emergency coolant recirculation capability. Consequently, RCS hot legs are allowed to lower to 405°F, which permits the accumulator liquid volume to discharge to the RCS but remain high enough such that even at saturated conditions, RCS pressure would be high enough to prevent nitrogen injection into the RCS.

D Incorrect – First part is correct (see B). Second part is plausible but incorrect (see C).

Technical References:

ECA-1.1, Loss of Emergency Coolant Recirculation, Step 29, Page 18.

References to be provided to applicants during exam: None.

Learning Objective: LO24.ERG.C11. Though not specifically an entry condition, it is a continuous action step.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)10	

Examination Outline Cross-Reference (W E05) Inadequate Heat Transfer	Level Tier # Group #	RO 1 1
Ability to determine and/or interpret the following as they apply to (W E05) Loss of Secondary Heat Sink (CFR: 41.10): CCW flow to RHR heat exchangers	K/A # Rating	A2.15 2.9

Question 28

Unit 1 experienced an automatic Safety Injection following a small break LOCA. Due to equipment failures, the crew entered FRH-0.1A, Response to Loss of Secondary Heat Sink.

Step 1 of FRH-0.1A states, "Check if Secondary Heat Sink is Required." Plant conditions require placing Train A RHR in service while continuing with this procedure.

Prior to placing RHR in service, how should proper CCW flow to 1-01 RHR HX be confirmed or attained for these plant conditions?

- A. Manually throttle 1-HS-4572, RHR HX 1 CCW RET VLV to 7600 gpm.
- B. Manually throttle 1-HS-4572, RHR HX 1 CCW RET VLV to 2500 gpm.
- C. Verify 40% FLO light above 1-HS-4572, RHR HX 1 CCW RET VLV is lit.
- D. Verify 1-HS-4572, RHR HX 1 CCW RET VLV automatically positions to 2500 gpm.

Answer: C

Explanation:

A is wrong because this is the design flow limit for one CCW pump flow to the heat exchanger and with an SI signal (given in stem) the HV-4572 throttles to 40% design flow to prevent overheating CCW. This amount of flow is approximately 3140 gpm...

B wrong because 2500 gpm is the low flow alarm and no manual throttling is necessary.

C is correct because the blue 40% FLO light above the hand switch lights up indicating design flow of 40% is met and valve is opened to the proper position for the SI signal.

D is wrong because 2500 gpm is the low flow alarm and it repositions to 40% design rated flow, or 3140 gpm.

Technical References:

FRH-0.1A, page 3, Rev 9.

LO21.SYS.CC1, rev 5-11-2016, slide 34 (with notes).

SOP-102A, page 16, Rev 20.

CCW study guide, page 18, rev 5-1-2011.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CC1.OB05.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)10	

Examination Outline Cross-Reference (010) (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM	Level Tier # Group # K/A # Rating	RO 2 1 K6.01 3.9
(010K6.01) Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the PRESSURIZER PRESSURE CONTROL SYSTEM: PZR Pressure channels		

Question 29

Unit 1 is proceeding to MODE 5 per IPO-005A, Plant Cooldown from Hot Standby to Cold Shutdown.

Pressurizer pressure is 350 psig and stable.

RCS temperature is 340°F and stable.

1-PT-403, HL 4 PRESS (WR) fails high.

How will Pressurizer Pressure Control System respond, specifically Power Operated Relief Valve positions, following the instrument failure?

- A. 1-PCV-455A OPEN, 1-PCV-456 OPEN
- B. 1-PCV-455A OPEN, 1-PCV-456 CLOSED
- C. 1-PCV-455A CLOSED, 1-PCV-456 OPEN
- D. 1-PCV-455A CLOSED, 1-PCV-456 CLOSED

Answer: C

Explanation:

A is wrong because the high failure of WR pressure channel PT-403 with Low Temperature Over-Pressure Protection System (LTOP) armed will cause PCV-456 to open. LTOP auto arms when RCS temperature goes below 350°F and in the stem it is given as 340°F. This failure does not affect the other PORV (PCV-455A will remain closed). It is controlled with the other WR instrument PT-405. Plausible if you don't remember the logic or think that there is only one WR RCS pressure input to LTOP.

B is wrong the PORV valves are switched. PORV PCV-456 will be open not closed and PORV PCV-455 will be closed not open. Plausible because this is what happens when PT-405 fails HIGH.

C is correct. The high failure of WR pressure channel PT-403 with Low Temperature Over-Pressure Protection System (LTOP) armed causes PCV-456 to open. LTOP auto arms when

below 350°F and in the stem it is less than 350°F. This failure does not affect the other PORV (PCV-455A will remain closed).

D is wrong because only one PORV opens. Plausible because both spray valves open with the failure and with no operator invention, the reactor trips on low pressure but PORV-455 remains closed the entire time. (see C for detailed explanation).

Technical References:

ABN-715, Revision 6, page 3.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.PP2.OB03.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	NRC 2014-6 Q34
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)7	

Examination Outline Cross-Reference
012 REACTOR PROTECTION SYSTEM

Level
Tier #
Group #
K/A #
Rating

RO
2
1
G2.1.31
4.6

(G2.1.31) Ability to locate control room switches, controls, and indications and to determine whether they correctly reflect the desired plant lineup

Question 30

Unit 2 is performing a plant shutdown.

Reactor power is 45% lowering slowly.

According to IPO-003B, Power Operations, what is the expected condition of the four (4) Reactor Protection System Permissive RX \geq 50% PWR P-9 status lights AND where are these indications located?

- A. ON;
CB-07 PCIP
- B. ON;
CB-04 2-TSLB-3
- C. OFF;
CB-07 PCIP
- D. OFF;
CB-04 2-TSLB-3

Answer: D

Explanation:

- A. INCORRECT – Plausible since the permissive RX \leq 50% PWR TURB TRIP PERM P-9 light on CB-07 PCIP will be on; however, the individual permissive P-9 status lights on CB-04 2-TSLB-3 will be off.
- B. INCORRECT – Plausible since the permissive RX \leq 50% PWR TURB TRIP PERM P-9 light on CB-07 PCIP will be on. Second part is correct since individual permissive P-9 status lights are located on CB-04 2-TSLB-3.
- C. INCORRECT – Plausible since first part is correct. Second part is plausible since RX \leq 50% PWR TURB TRIP PERM P-9 status light is located on CB-07 PCIP and indicates permissive status, but not individual channel input status to the permissive.
- D. CORRECT – With reactor power less than 50% power, the individual RX \geq 50% PWR P-9 status are verified OFF and the light indication is located on CB-04 2-TSLB-3.

Technical References:

IPO-003B, Revision 27, pages 118-119

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.ES1.OB03.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)10	

Examination Outline Cross-Reference (013) (ESFAS) ENGINEERED SAFETY FEATURES ACTUATION SYSTEM	Level Tier # Group # K/A #	RO 2 1 K4.08
Knowledge of Engineered Safety Features Actuation System design features and/or interlocks that provide for the following: Redundancy	Rating	3.7

Question 31

Having two separate Solid State Protection Systems (SSPS) on an Engineered Safety Feature (ESF) Actuation System performing the same functions would be an example of ___(1)___.

The effect of removing one train of SSPS from service would be that the number of inputs required for an ESF actuation ___(2)___.

- A. (1) redundancy
(2) would be reduced
- B. (1) coincidence
(2) would not be reduced
- C. (1) redundancy
(2) would not be reduced
- D. (1) coincidence
(2) would be reduced

Answer: C

Explanation: According to the explanation provided by the TS Bases, redundancy requirements are met in ESF systems containing SSPSs by having two separate trains. One purpose of this redundancy is to ensure protection is provided when one train is not in service. To accomplish this, both trains receive the same system inputs to provide signals for their independent logic circuits.

A is wrong. (1) This portion is correct. (2) All inputs are provided to the operational side when a train of SSPS is out of service. Plausible because a misconception of how redundancy is implemented (vice logic) could lead one to believe that each SSPS train receives separate inputs. For example, when an instrument is removed from service, the total number of inputs to a logic that results in an ESF actuation is reduced.

B is wrong. (1) According to LO21SYSES1, the definition of coincidence is: "The minimum number of channel[s] that must sense [a] process signal to cause component actuation". Only one side of SSPS is required to provide protection, so it is not a coincident system. Plausible because this term is often used in related ESF systems to provide protection. (2) This portion is correct.

C is correct, see explanation.

D is wrong. (1) See explanation for B. (2) See explanation for A.

Technical References:

CPNPP Technical Specifications (Rev. 77) pg. B 3.3-61 (see below)

LO21SYSES1 – RPS/ESFAS (Rev. 1) pg. 8 (see below)

References to be provided to applicants during exam: None.

Learning Objective: STATE the function of the Reactor Protection and Engineered Safeguard Actuation Systems in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems. (LO21.SYS.ES1.OB01)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) (7)	

Examination Outline Cross-Reference
(013) (ESFAS) ENGINEERED SAFETY
FEATURES ACTUATION SYSTEM

Level	RO
Tier #	2
Group #	1
K/A #	K5.02
Rating	3.7

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Engineered Safety Features Actuation System: Safety system logic and reliability

Question 32

Unit 1 is operating at 100% power.

1. Regarding the “fail safe” concept in Engineered Safety Features Systems, the Containment HI-3 Pressure signal bistable must....

1-PI-935, CNTMT PRESS (IR) CHAN III has now failed low and its associated bistables have NOT yet been placed in a “tripped” condition.

2. **Based on the above conditions**, the logic for the Engineered Safety Features Containment HI-3 pressure signal would require how many remaining Containment Pressure Channels to actuate?
 - A. (1) energize to actuate
(2) 1 of 2
 - B. (1) energize to actuate
(2) 2 of 3
 - C. (1) deenergize to actuate
(2) 1 of 2
 - D. (1) deenergize to actuate
(2) 2 of 3

Answer: B

Explanation: IAW Reactor Protection and Engineered Safety Features Systems concepts, Containment HI-3 Pressure signal’s bistable is an exception to the “fail safe” principle, thus making it energize to actuate. Also, with 1-PI-935 failed low, the coincidence for HI-3 containment pressure will change to 2/3, regardless of whether the bistables have been placed in the “tripped” position. The normal coincidence and setpoint for HI-3 containment pressure is 2/4 >18.2 psig.

A first part is correct, see above explanation. Second part is wrong but plausible if the candidate assumed the incorrect knowledge of the signal's logic. This logic is correct for other signals in the Reactor Protection and Engineered Safety Features Systems.

B is correct, see above explanation.

C first part is incorrect but plausible because most bistables are deenergized to actuate. Second part is wrong but plausible for reasons listed in A.

D first part is incorrect, see 'C'. Second part is correct, see explanation above.

Technical References:

LO21SYSES1 – Reactor Protection and ESFAS (Rev. 1) pg. 8, 19 (See below)

LO21SYSES2 – Solid State Protection System (SSPS) (Rev. 0) pg. 9 (see below)

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the normal, abnormal and emergency operation of the Reactor Protection and Engineered Safeguard Actuation Systems in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems.
(LO21.SYS.ES1.OB05)

EXPLAIN the instrumentation and controls of the Reactor Protection and Engineered Safeguard Actuation Systems and predict the system response in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems and Westinghouse Drawings 7247D05. (LO21.SYS.ES1.OB04)

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41(b) (5)	

Examination Outline Cross-Reference (022) (SF5) CONTAINMENT COOLING SYSTEM	Level Tier # Group # K/A # Rating	RO 2 1 A3.01 4.2
Ability to monitor automatic features of the Containment Cooling System including initiation of ESFAS mode of operation		

Question 33

Both units are at 100% power.

All systems are normally aligned when a Loss of Offsite Power occurs.

Five minutes later, which of the following describes the status of Containment Air Cooling and Recirculation System Cooling Units and Fans?

- A. Cooling Units and Fans are load shed and must be manually reset.
- B. Cooling Units and Fans are load shed and CANNOT be restarted.
- C. Cooling Units and Fans are tripped and then sequenced onto the safety-related electrical buses.
- D. Two of the four Cooling Units and Fans are sequenced onto the safety-related electrical buses after all four have been tripped.

Answer: C

Explanation:

A is wrong. Plausible because CPNPP has ventilation systems which respond to a Loss of Offsite Power as stated.

B is wrong. Plausible because CPNPP has ventilation systems which cannot be manually restarted until electrical realignments are performed in the field.

C is Correct. The Containment Air Cooling and Recirculation System responds to a Loss of Offsite Power in this manner.

D is wrong. Plausible because this would be representative of restoring adequate Containment Cooling but not all cooling is restored in order to prevent overload of the Emergency Diesel Generators.

Technical References:

ABN 602 – Response to a 6900/480V System Malfunction (Rev. 10) pg. 78, 80 (See below)
SOP 801A – Containment Ventilation System (Rev. 14) pg. 11 (See below)

References to be provided to applicants during exam: None.

Learning Objective: COMPREHEND the normal, abnormal and emergency operation of the Containment Ventilation System.

Question Source:	Bank #	CPNPP 2012-06, Q13
Updated references revisions	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b) (7)	

Examination Outline Cross-Reference (026) (SF5 CSS) CONTAINMENT SPRAY SYSTEM	Level Tier # Group # K/A # Rating	RO 2 1 K4.07 3.8
(026K4.07) Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Adequate level in containment sump for suction		

Question 34

To ensure adequate inventory for containment spray pump suction, EOS-1.3B, Transfer to Cold Leg Recirculation directs monitoring _____, then initiate realignment of suction from the RWST to the sump.

- A. all four RWST level instruments and when the first one goes below 9%
- B. all four RWST level instruments and when the first one goes below 6%
- C. one of the four RWST instruments and when it starts to go below 6%
- D. one of the four RWST instruments and when it starts to go below 9%

Answer: C

Explanation:

Per EOS-1.3A/B Bases on page 28, you pick one RWST instrument to monitor during the drain down of the RWST and when level goes below 6% on that instrument, you swap the suction path for the containment spray pumps from the RWST to the sump.

A is wrong because both aspects are wrong. You pick one RWST instrument to monitor during the drain down of the RWST per EOS-1.3A/B Bases on page 28. The second part is wrong because it states that when RWST level goes below 6%, then you swap the suction path. Plausible because 9% is the RWST empty alarm.

B is wrong the first part is wrong. The second part is correct.

C is correct (see above explanation).

D is wrong because the 9% is wrong. Plausible because 9% is the RWST empty alarm. The first part is correct (see above explanation).

Technical References:

EOS-1.3B, revision 9, page 28.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CT1.OB05.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)7	

Examination Outline Cross-Reference (039) (SF4S MSS) MAIN AND REHEAT STEAM SYSTEM	Level Tier # Group # K/A # Rating	RO 2 1 K6.12 3.8
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the main and reheat steam system: MSIVs		

Question 35

Unit 1 has reached 100% power following a refueling outage.

Assuming nitrogen pressure is maintained to the valve, which of the following MSIV component malfunctions, if not corrected, would eventually cause steam flow to isolate from the affected Steam Generator?

- 1) Loss of air to the MSIV hydraulic oil pump.
 - 2) Loss of power to the MSIV hydraulic oil pump air supply solenoid.
 - 3) Loss of power to Train A and Train B MSIV hydraulic oil dump solenoids.
- A. 1) Only
- B. 2) Only
- C. 1) and 2) Only
- D. 1), 2) and 3)

Answer: A.

Explanation:

The MSIV is a hydraulically operated valve that opens and closes using a piston. A nitrogen overpressure applied to the top of the piston must be overcome by hydraulic oil pressure under the piston to cause the valve to open. An air-operated positive-displacement hydraulic oil pump provides the hydraulic pressure. Oil is vented back to a hydraulic oil sump to close the valve. The hydraulic oil pump air supply solenoid is de-energized to admit air to the pump. The pump then operates to raise oil pressure under the piston, overcoming nitrogen pressure, and opens the MSIV. With the MSIV open, the oil pump air supply remains de-energized allowing the oil pump to continue to operate until the pump reaches a stalled condition. If hydraulic oil pressure lowers, the pump operates to restore pressure until the pump reaches a stalled condition again. Two parallel hydraulic oil dump solenoids powered from different electrical buses are provided to drain oil back to the hydraulic oil sump to permit closing the valve. The hydraulic oil dump solenoids are normally de-energized and must be energized to close the MSIV.

A. Correct – Loss of air to the hydraulic oil pump would prevent the pump from making up for oil leakage and the MSIV would eventually close.

B. Incorrect – Plausible since fail safe components are normally designed to fail to their safe position on loss of power. In this case, power must be applied to the hydraulic oil dump solenoids to cause the MSIV to close.

C Incorrect – Plausible since 1) is correct but 2) is incorrect (see B).

D. Incorrect – Plausible since 1) is correct, but 2) is incorrect (see B). 3) is incorrect but plausible since most pumps are started and stopped based on power supplied to the pump driver. In this case, the pump air supply solenoid is normally de-energized to maintain air pressure to the pump when the MSIV is open.

Technical References:

LO21SYSMR1 – Main Steam (Rev. 2) pg. 85 (see below)

IPO-001B – Plant Heatup From Cold Shutdown to Hot Standby (Rev. 14)

LO21SYSES1 – RPS/ESFAS (Rev. 0) pp. 20, 21 (see below)

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the normal, abnormal and emergency operation of the Main Steam system in accordance with SOP-301, IPO-001, IPO-002, IPO-003, ABN -707, ABN-709, ABN-710, OPT-504 and OPT-509. (LO21.SYS.MR1.OB05)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41(b) (7)	

Examination Outline Cross-Reference	Level	RO
059 Main Feedwater System	Tier #	2
	Group #	1
Knowledge of the physical connections and/or cause and effect relationships between the Main Feedwater System and the following systems: S/GB system	K/A #	K1.13
	Rating	2.9

Question 36

Unit 1 is operating at 100% power.

What is the NORMAL discharge path of Steam Generator Blowdown at full power operations?

- A. Main Condenser
- B. Condensate Storage Tank (CST)
- C. Turbine Building Sump
- D. Heater Drain Tank (HDT)

Answer: D

Explanation:

A is wrong. Main condenser is an alternate discharge path used to utilize additional purification of condensate polishers.

B is wrong. CST is an alternate discharge path used to maintain hotwell level during plant startup while maintaining CST chemistry by using demins.

C is wrong. TB Sump is an alternate discharge path that can be used to drain one or more SGs during plant shutdown (allows for solids reduction without sacrificing filters). Also used to flush beds. It is not the normal discharge path though.

D is correct. Heater Drain Tanks are the normal discharge path, also used during plant startup.

Technical References:

LO21.SYS.SB1, Steam Generator Blowdown System Lesson Plan, Rev 0, Page 14

References to be provided to applicants during exam: None.

Learning Objective: DESCRIBE the components of the Steam Generator Blowdown system including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-239 and FSAR Section 10.4.8. (LO21.SYS.SB1.OB03)

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.41(b).4

Examination Outline Cross-Reference	Level	RO
061 Auxiliary / Emergency Feedwater System	Tier #	2
	Group #	1
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AUXILIARY/EMERGENCY FEEDWATER SYSTEM: Back leakage through discharge check valves (CFR: 41.5 / 45.3)	K/A #	K 5.07
	Rating	3.1

Question 37

Unit 1 is 80% power.

1-TI-2471A, SG 1 MDAFW TEMP and 1-TI-2472A, SG 2 MDAFW TEMP are reading significantly higher than ambient and slowly trending up.

The Unit Supervisor has just entered ABN-305, Auxiliary Feedwater System Malfunction, Section 2.0, Indicated Main Feedwater Backleakage.

The AFW system and test header have not been realigned or tested within the last 48 hours.

At which of the following temperatures will ABN-305 **first** require AFW supply lines to be cooled?

- A. 175°F
- B. 250°F
- C. 270°F
- D. 295°F

Answer: B

Explanation:

IAW ABN-305 guidance, AFW supply lines are **required** to be cooled if temperature exceeds 250°F since power is >30%. In this case, the cooling will first be attempted by running cooler water through the lines.

A is wrong but plausible because 175°F represents the maximum temperature allowed for the AFW flow to SG temperature when plant heatup or shutdown are in progress per Step 12c.

B is correct, see answer explanation above.

C is wrong but plausible since this would be permissible if power was < 30%

D is wrong but plausible because this is the temperature that triggers the SG AFW NZL TEMP HI annunciator.

Technical References:

ABN-305, "Auxiliary Feedwater System Malfunction"

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.305.OB02, "From memory analyze the response to indicated main feedwater back leakage IAW ABN-305."

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference (061) (SF4S AFW) AUXILIARY / EMERGENCY FEEDWATER SYSTEM	Level Tier # Group # K/A # Rating	RO 2 1 K5.08 3.9
(061K5.08) Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AFW: Expected AFW flow rates based on plant conditions		

Question 38

Unit 2 experiences a loss of all main feedwater from 30% power.

Steam Generator levels are as follows:

2-01 SG	37%
2-02 SG	33%
2-03 SG	38%
2-04 SG	34%

With no operator actions, what is the total (approximate) AFW flow for these conditions?

- A. 570 gpm
- B. 1140 gpm
- C. 1715 gpm
- D. 2285 gpm

Answer: D

Explanation:

ALL three pumps start (both MD AFW pumps and TD AFW pump). Usually loss of MFW (both pumps tripped) will cause just the MD AFW pumps to start however in this case 2 of 4 SGs are below 35.4% (unit 2 setpoint) which will cause the TDAFWP to start also. One MD AFW pump provides approximately 570 gpm and the TD AFW pump provides approx. 1145 gpm. Adding $2 \times 570 \text{ gpm} + 1145 \text{ gpm} = 2285 \text{ gpm}$, answer D.

A is wrong because this is the flow for one MD AFW pump and all three start.

B is wrong because this is the flow for both MD AFW pumps and all three start

C is wrong because this is the flow if only the TD AFW pump and one MD AFW pump starts and all three start

D is correct because all three pumps start and a MD AFW pump provides approx. 570 gpm while the TD AFW pump provides approx. 1145 gpm, totaling 2285 gpm.

Technical References:

LO21.SYS.AF1, slides 21, 23, 33, and 38, revision May 2012.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.AF1.OB04 and OBO7

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)5	

Examination Outline Cross-Reference (062) (SF6 ED AC) AC ELECTRICAL DISTRIBUTION SYSTEM	Level Tier # Group # K/A # Rating	RO 2 1 K2.02 3.8
(062K2.01) Knowledge of electrical power supplies to the following: Major bus or motor control center power supplies		

Question 39

What is the power supply to X-03 Ventilation Chiller?

- A. XA1
- B. XA3
- C. 1EA1
- D. 2EA1

Answer: D

Explanation:

Ventilation Chiller X-03 is powered from 2EA1. Some of the other chillers are powered from other buses listed as distracters (such as X-01 powered from 1EA1 and X-06 powered from XA2)

A is wrong because (see above explanation) but credible because XA1 powers some of the safety related ventilation chillers...

B is wrong because (see above explanation) but credible because it is an odd powered bus and some of these chillers are powered from XA1 and XA2.

C is wrong because (see above explanation) but credible because X-01 is powered from bus 1EA1.

D is correct because ventilation Chiller X-03 is powered from bus 2EA1.

Technical References:

SOP-814, page 174, revision 22

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.AC2.OB03.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
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Question History:	Last NRC Exam
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Question Cognitive Level:	Memory/Fundamental	3
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Comprehensive/Analysis

10CFR Part 55 Content:

55.41(b)7

Examination Outline Cross-Reference	Level	RO
063 DC ELECTRICAL DISTRIBUTION SYSTEM	Tier #	2
	Group #	1
	K/A #	A2.03
(063A2.03) Ability to (a) predict the impacts of the following on the DC ELECTRICAL DISTRIBUTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations Malfunction of battery chargers	Rating	3.9

Question 40

Unit 1 is operating at 100%.

Battery charger BC1ED1-1 is in service.

1-ALB-10B Window 1.14 BATT CHRGR BC1ED1-1/BC1ED1-2 TRBL alarms.

An equipment operator is dispatched to ECB 792 safeguard battery charger room and reports battery charger BC1ED1-1 has NO fans operating.

Per ALM-0102A, Alarm Procedure 1-ALB-10B and SOP-605A, 125 VDC Switchgear and Distribution Systems, Batteries and Battery Chargers, what action is required?

- Transfer battery chargers using the following sequence: (1)
- When placing BC1ED1-2 in service, the (2)
 - A. (1) Remove BC1ED1-1 from service, then place BC1ED1-2 in service.
(2) AC input breaker is closed first, prior to closing the DC output breaker.
 - B. (1) Remove BC1ED1-1 from service, then place BC1ED1-2 in service
(2) DC output breaker is closed first, prior to closing the AC input breaker.
 - C. (1) Place BC1ED1-2 in service, then remove BC1ED1-1 from service
(2) AC input breaker is closed first, prior to closing the DC output breaker.
 - D. (1) Place BC1ED1-2 in service, then remove BC1ED1-1 from service
(2) DC output breaker is closed first, prior to closing the AC input breaker.

Answer: A

Explanation:

- A. CORRECT: (1) ALARM PROCEDURE, 1-ALB-10B, provides direction, IF the affected battery charger has NO fans operating, THEN PLACE standby charger in service per SOP-605A for Placing 125 VDC Switchboard 1ED1 Battery Chargers in Service. SOP-605A Section 5.5.1 Alternating 125 VDC Switchboard 1ED1 Battery Chargers removes the in-service BC, step 5.5.1.G before placing standby BC in service step 5.5.1 J.

(2) The following Note is contained in Section 5.5.1: the AC input breaker is CLOSED first, prior to closing the DC output breaker, to charge the output filter capacitors slowly and prevent the output fuse or circuit breaker from being blown due to heavy surge currents

- B. INCORRECT: (1) Correct. (2) DC output and AC input breakers are reversed. This distractor is plausible because (1) is correct and (2) is the breaker sequence for removing a BC from service.
- C. INCORRECT: (1) SOP-605A removes the operating BC before placing the standby BC in service. (2) Correct. This distractor is plausible because (2) is correct and the applicant may believe maintaining a BC in service throughout the evolution is required, which it is by Tech Specs. As a result, an LCO is entered during the transfer.
- D. INCORRECT: (1) SOP-605A removes the operating BC before placing the standby BC in service. (2) DC output and AC input breakers are reversed. This distractor is plausible because the applicant may believe maintaining a BC in service throughout the evolution is required, which it is by Tech Specs. As a result, an LCO is entered during the transfer. Additionally, (2) is the breaker sequence for removing a BC from service.

Technical References:

SOP-605A, 125 VDC SWITCHGEAR AND DISTRIBUTION SYSTEMS, BATTERIES AND BATTERY CHARGERS, Revision 13, pages 67-72
 ALARM PROCEDURE, 1-ALB-10B, Revision 16, pages 73-78

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.DC1.OB04

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)8	

Examination Outline Cross-Reference	Level	RO
064 Emergency Diesel Generator System	Tier #	2
	Group #	1
Ability to predict and/or monitor changes in parameters associated with operation of the Emergency Diesel Generators, including:	K/A #	A1.09
Diesel engine operating parameters	Rating	3.6

Question 41

Unit 1 is at full power.

Diesel Generator 1-01 Channel I Fast/Slow Start Handswitch is in FAST.

Diesel Generator 1-01 Channel II Fast/Slow Start Handswitch is mispositioned in SLOW.

An operator performs a remote normal start of 1-01 Diesel Generator from the control room per SOP-609A, Diesel Generator System.

1-01 Diesel Generator will ramp up to rated speed in approximately (1).

If the diesel generator is synchronized with offsite power with no other operator actions taken, the diesel generator is considered (2).

- A. (1) 6 seconds
(2) inoperable
- B. (1) 40 seconds
(2) operable
- C. (1) 6 seconds
(2) operable
- D. (1) 40 seconds
(2) inoperable

Answer: A

Explanation:

A is correct because one of the selector switches is in FAST which will make the EDG perform a fast start and bring it to 450 rpm in approximately 6 seconds. Despite the EDG performing a fast start, the EDG will be considered inoperable since one of the selector switches is in SLOW per SOP-609A.

B is wrong because applicant must know that if any one of the selector switches is in FAST then the EDG will perform a fast start. Additionally, the EDG is considered inoperable since one of the selector switches is in SLOW. However, the applicant may mistakenly believe it is still operable since T.S. 3.8.1 has a note allowing for one EDG to be synchronized with offsite power under administrative controls.

C is wrong because the EDG is considered inoperable since one of the selector switches are in SLOW. However, the applicant may mistakenly believe it is still operable since T.S. 3.8.1 has a note allowing for one EDG to be synchronized with offsite power under administrative controls. The EDG reaching 450 RPMs in about 6 seconds is correct since it will perform a fast start.

D is wrong because the EDG will perform a fast start and will bring it up to 450 RPMs in about 6 seconds. However the EDG is considered inoperable because of the selector switch positions.

Technical References:

Comanche Peak Technical Specifications
SOP-609A, "Diesel Generator System" Revision 25

References to be provided to applicants during exam: None.

Learning Objective: LOT21.SST.ED2.PPT: From memory, discuss the EDG systems in accordance with the applicable CPNPP procedures and Technical Specifications

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)(7), (8)	

Examination Outline Cross-Reference	Level	RO
073 PROCESS RADIATION MONITORING SYSTEM	Tier #	2
	Group #	1
	K/A #	A4.02
(073A4.02) Ability to manually operate and/or monitor the PROCESS RADIATION MONITORING SYSTEM in the control room: RMS control panel	Rating	3.6

Question 42

I&C receives permission from the Shift Manager to place X-RE-5570A, South Plant Vent Stack Wide Range Gas Monitor OUT of POLL at the PC-11 console.

X-RE-5570A PC-11 POLL STATUS-MONITOR OFF-LINE alarm actuates as a result of I&C actions.

Per ALM-3200, Alarm Procedures DRMS:

Which of the following correctly describes radiation monitoring system response if an actual high radiation condition is detected at the South Plant Vent Stack?

Radiation level indication _____ (1) _____ available on the RM-23, Safety-Related Display and Control Module.

Automatic closure of X-HCV-014, GWPS DISCH TO PLT EXH PLNM ISOL VLV _____ (2) _____ occur.

- A. (1) is
(2) will
- B. (1) is
(2) will NOT
- C. (1) is NOT
(2) WILL
- D. (1) is NOT
(2) will NOT

Answer: A

Explanation:

A is Correct: (1) With 5570A OUT of POLL, according to ALM-3200, For monitors with an associated RM-23, data may still be obtained from the RM-23. 5570A is a safety related rad monitor which is displayed on RM-23. (2) With 5570A OUT of POLL, the monitor which provides automatic actuation on HIGH radiation OR loss of OPERATE status will still function WITHOUT causing the PC-11 console alarm. The automatic action for closure of X-HCV-014, will occur.

B is wrong: (1) Correct. (2) With 5570A OUT of POLL, the monitor which provides automatic actuation on HIGH radiation OR loss of OPERATE status will still function WITHOUT causing the PC-11 console alarm. The automatic action for closure of X-HCV-014, will occur. This distractor is plausible because (1) is correct and with the monitor OUT of POLL, an applicant may believe the automatic action is disabled.

C is wrong: (1) With 5570A OUT of POLL, according to ALM-3200, For monitors with an associated RM-23, data may still be obtained from the RM-23. 5570A is a safety related rad monitor which is displayed on RM-23. (2) Correct. This distractor is plausible because (2) is correct and with the rad monitor OUT of POLL, the applicant may believe indication is not available on RM-23.

D is wrong: (1) With 5570A OUT of POLL, according to ALM-3200, For monitors with an associated RM-23, data may still be obtained from the RM-23. 5570A are safety related rad monitors which are displayed on RM-23. (2) With 5570A OUT of POLL, the monitors which provide automatic actuation on HIGH radiation OR loss of OPERATE status will still function WITHOUT causing the PC-11 console alarm. The automatic action for closure of X-HCV-014, will occur. This distractor is plausible because the applicant may believe indication and automatic actions are disabled when OUT of POLL.

Technical References:

ALM-3200 Revision 5, Page 7

ABN-902, Revision 7, Page 4

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.RM1.OB05.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)5...	

eous Waste Processing System.

Examination Outline Cross-Reference	Level	RO
(076) (SF4S SW) SERVICE WATER SYSTEM	Tier #	2
	Group #	1
(076) (SF4S SW) SERVICE WATER SYSTEM	K/A #	K5.02
(G2.1.30) CONDUCT OF OPERATIONS: Ability to locate and operate components, including local controls	Rating	4.4

Question 43

OPT-502A, AFW/SSW CROSSTIE VALVES is being performed to stroke test 1-HV-4395, SSW TO AFWP CLOSED SUCT VLV.

Per OPT-502A, this valve is stroke tested by operating the _____ (1) _____.

This valve can be operated locally _____ (2) _____ .

- A. 1. key switch on main control board 1-CB-01
2. in 810' safeguards building hallway
- B. 1. key switch on main control board 1-CB-01
2. in 790' safeguards building hallway
- C. 1. key switch on main control board 1-CB-09
2. in 790' safeguards building hallway
- D. 1. key switch on main control board 1-CB-09
2. in 810' safeguards building hallway

Answer: C

Explanation:

OPT-502A directs stroke testing this valve 1-HV-4395 from the control room at 1CB-09, but most of the SSW valves are operated from 1CB-01 so this is a plausible control board location. This valve (1-HV-4395) is operated locally in the 790' hallway of the safeguards building. The 810' elevation is plausible because other SSW MOVs are located here (such as Diesel Generator SSW outlet valve MOVs).

A is wrong because part 1 and part 2 are wrong.(see above explanation

B is wrong because part 1 is wrong (see explanation above

C Correct (see above explanation)

D is wrong because part 2 is wrong. Part 1 is correct (see above explanation).

Technical References:

OPT-502A, revision 13, page 9.

SOP-501, revision 21

L021.SYS.SW1

M1-2234 sh. 002
DBD-ME-233

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.SW1.OB03

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)7	

Examination Outline Cross-Reference	Level	RO
076 SW Service Water System	Tier #	2
	Group #	1
Knowledge of the physical connections and/or cause and effect relationships between the Service Water System and the following systems: EDG	K/A #	K1.05
	Rating	4.1

Question 44

According to ABN-501, Station Service Water System Malfunction, an EDG can be operated, with load, for a MAXIMUM of approximately (1) minute(s) without service water system flow and not affect diesel performance.

When an 86-1 fault exists on the 6.9 KV safeguards bus, EDG running time should be limited to a MAXIMUM of approximately (2) minutes to prevent damage to the EDG.

- A. (1) 1
(2) 13
- B. (1) 1
(2) 15
- C. (1) 4
(2) 13
- D. (1) 4
(2) 15

Answer: B

Explanation:

A is wrong but (2) may be confused with the maximum running time for a charging pump without SW prior to bearing damage of 13 minutes

B is correct

C is wrong but (1) may be confused with the maximum running time for an RCP without CCW prior to seal damage of 4 minutes

D is wrong per A and C above

Technical References:

ABN-501, "Station Service Water System Malfunction," Revision 10, Section 2.3 NOTE

References to be provided to applicants during exam: None.

Learning Objective: LO 21.ABN.501.OB101.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b) 7	

Examination Outline Cross-Reference	Level	RO
078 Instrument Air System	Tier #	2
	Group #	1
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Instrument Air System: Loss of instrument air	K/A #	K5.03
	Rating	3.9

Question 45

Unit 1 is initially operating at 100% power.

The following conditions are noted:

- 1-ALB-1 Window 3.3 INSTR AIR HDR PRESS LO alarms.
- Instrument air header pressure is 45 psig and slowly lowering.
- The reactor has not been tripped.

Assuming instrument air header pressure continues to slowly lower (and no reactor trip is initiated), how will Unit 1 RCS charging be aligned in accordance with ABN-301, Instrument Air System Malfunction?

Charging pump suction will be aligned to the ___(1)___ using ___(2)___ pump

- (1) VCT
(2) a centrifugal charging
- (1) VCT
(2) the positive displacement
- (1) RWST
(2) a centrifugal charging
- (1) RWST
(2) the positive displacement

Answer: C

Explanation:

A is wrong: (1) Charging pump suction is aligned to the RWST IAW ABN-301. (2) Correct charging pump. This distractor is plausible because (2) is correct and (1) is the normal suction source for the charging pumps.

B is wrong: (1) Charging pump suction is aligned to the RWST IAW ABN-301. (2) a centrifugal pump is used and the PDP is tripped. This distractor is plausible because (1) is the normal suction source for the charging pumps and (2) the PDP is a viable charging pump for RCS charging.

C is Correct: (1) The RWST is selected as the makeup source since loss of instrument air will potentially disable air-operated valves that provide the ability to makeup to the VCT. (2) The centrifugal pump is used and the PDP is tripped since instrument air is used to provide remote PDP speed control capability. Without speed control, the PDP is no longer a reliable charging pump. The CCP does not rely on instrument air for speed control and local control of valves in the charging line permit charging flow to be controlled.

D is wrong: (1) ABN-301 Section 2.3 Operator Actions Steps 8a/b provides direction to OPEN RWST pump suction and to CLOSE VCT pump suction to the charging pumps. (2) Steps 8f/g provide direction to start a centrifugal charging pump and trip the positive displacement pump.

Technical References:

ABN-301 Revision 15, Pages 12-13

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.301.OB04

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)5...	

Examination Outline Cross-Reference	Level	RO
103 Containment System	Tier #	2
	Group #	1
Knowledge of Containment System design features and/or interlocks that provide for the following (CFR 41.7): Personnel access hatch and emergency access hatch	K/A #	K4.04
	Rating	3.3

Question 46

The Unit 2 **Emergency Air Lock** design prevents the inner door from opening while the outer door is open by which of the following means?

- A. Electrical interlock defeats inner door opening button.
- B. Hydraulic interlock on inner door from cam valves actuated by the equalizing valve.
- C. Mechanical cams position themselves above opening gear trains of inner door.
- D. Programmable Logic Controller interlock based on valve position and switch settings.

Answer: C

Explanation:

A is wrong because this is an interlock that exists only with the Personnel Air Lock

B is wrong because this is an interlock that exists only with the Personnel Air Lock

C is correct because the Emergency Air Lock uses only a mechanical interlock system to prevent both doors from being open (unless overridden)

D is wrong because this interlock is only for the Unit 2 Personnel Air Lock

Technical References:

SOP-907B, "Containment Personnel Airlocks"

LO21SYSCY1, "Containment Systems"

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CY1.OB02: Describe the components of the Containment System including interrelations with other systems to include interlocks and control loops as described in DBD-ME-008.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental

2

Comprehensive/Analysis

10CFR Part 55 Content:

55.41(b)(7)

Examination Outline Cross-Reference (000005) (APE 5) Inoperable/Stuck Control Rod	Level Tier # Group # K/A # Rating	RO 1 2 G2.4.3 3.7
Inoperable / Stuck Control Rod emergency procedures / plan: Ability to identify post- accident instrumentation		

Question 47

The following conditions are observed 10 minutes after a major seismic event:

- Unit one experienced a small break LOCA.
- Safety injection actuated and ECCS systems are in service.
- The reactor failed to trip but most rods dropped.
- Containment pressure is 6 psig.

Per FRS-0.1A, Response to Nuclear Power Generation/ATWT, which of the following nuclear instrument readings would indicate the subcriticality safety function is satisfied?

- Source Range SUR zero or negative
- Power Range indication less than or equal to 5%
- Intermediate Range SUR more negative than - 0.2 DPM
- Neutron Flux Source Range not on scale with Neutron Flux Wide Range < 5%

Answer: D

Explanation: The only available post-DBA qualified nuclear instruments are the Gamma-metric Neutron Flux Monitors. A containment pressure above or equal to 5 psig is given by FRS-0.1A as the point when these instruments must be used instead of the normal Westinghouse excore nuclear instruments. According to the Subcriticality Status Tree in the same procedure, if the Neutron Flux Wide Range indication is less than 5%, then the Neutron Flux Source Range indication should be used to monitor subcriticality. The Critical Safety Function is then considered satisfied if less than 15 minutes have elapsed since the time of trip, and the Neutron Flux Source Range instrument is not on scale, since no SR indication is expected for 12-15 minutes after the trip.

A is wrong because the SR Westinghouse instruments are not to be used for the specified conditions in FRS-0.1A. With adverse containment conditions, the logic tree goes to the top section of the flow chart. Plausible because the SR instruments are normally preferred to monitor shutdown conditions and provide an input to the subcriticality CSF when adverse conditions are not met.

B is wrong. The Power Range instruments should not be used in this scenario based on containment pressure > 5 psig. Plausible because at containment pressures below 5 psig, the Power Range instruments do provide an input to the subcriticality CSF. This is also a continuous action step of FRS 0.1A

C is wrong because adverse containment conditions exist, therefore the logic tree goes to the top section of the flow chart. With the top section used, Intermediate range is not used at all. Plausible because the IR instruments are used without adverse conditions in containment however the requirement is less than negative 0.2 DPM sur on IR instruments.

D is correct, see Explanation section.

Technical References:

FRS-0.1A – Response to Nuclear Power Generation / ATWT (Rev. 9) pp. 11, 28-30 (see below)

LO21SYSSEC1 – Excore Nuclear Instrumentation (Rev. 1) pp. 8-9 (see below)

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the normal, abnormal and emergency operation of the Excore Instrumentation system in accordance with SOP-703, ABN-701, 702 and 703. (LO21.SYS.EC1.OB05)

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (6)	

Examination Outline Cross-Reference (000033) (APE 33) Loss of Intermediate Range Nuclear Instrumentation	Level Tier # Group # K/A # Rating	RO 1 2 AK2.03 3.8
Knowledge of the relationship between loss of intermediate range Nuclear Instrumentation and the following systems or components: RPS		

Question 48

Unit 1 shutdown to 5% Turbine Power is in progress per IPO-003A, Power Operations.

The following indications are observed:

- 1-NI-41B, PR CHAN I 9%
- 1-NI-42B, PR CHAN II 8%
- 1-NI-43B, PR CHAN III 9%
- 1-NI-44B, PR CHAN IV 10%

Subsequently, 1-NI-35B, IR CURRENT CHAN I fails upscale.

What is the **highest priority** procedure required to be entered for the above conditions?

- EOP-0.0A, Reactor Trip or Safety Injection
- ALM-0064A, Alarm Procedure 1-ALB-6D Window 1.14 IR HI FLUX ROD STOP C-1
- ALM-0064A, Alarm Procedure 1-ALB-6D Window 3.2 IR CHAN I CMPNSATING VOLT FAIL
- ABN-702, Intermediate Range Instrument Malfunction

Answer: A

Explanation: Below or at the P-10 setpoint (10% Rx power), the IR High Flux reactor trip occurs if 1 out of 2 channels are above a 25% Rx power equivalent threshold. The above scenario would cause the IR HI FLUX ROD STOP C-1 alarm and a reactor trip, therefore EOP-0.0A is the correct procedure for these conditions.

A is correct. See explanation above.

B is wrong. The IR HI FLUX ROD STOP C-1 alarm would annunciate for the above scenario. However, the rod block alarm response would not be required since the reactor would trip due to the instrument failure.

C is wrong because this is not the highest priority procedure. Plausible since loss of compensating voltage could cause detector output to rise. However, the loss of compensating voltage alarm response would not be the highest priority procedure since the

intermediate range channel failure will result in a reactor trip below P-10. and EOP-0.0A is the highest-level procedure to be entered for the reactor trip condition.

D is wrong because this is not the highest priority procedure. Plausible because this procedure would be required to be entered for the intermediate range instrument failure but EOP-0.0A is the highest-level procedure for the reactor trip condition..

Technical References:

ALM-0063A – Alarm Procedure 1-ALB-6C (Rev. 5) pg. 31 (see below)

LO21SYSES1 – RPS / ESFAS (Rev. 1) pg. 14 (see below)

References to be provided to applicants during exam: None.

Learning Objectives: EXPLAIN the instrumentation and controls of the Reactor Protection and Engineered Safeguard Actuation Systems and predict the system response in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems and Westinghouse Drawings 7247D05. (LO21.SYS.ES1.OB04)

EXPLAIN the normal, abnormal and emergency operation of the Reactor Protection and Engineered Safeguard Actuation Systems in accordance with DBD-EE-021, Reactor Protection and NSSS Related Control Systems. (LO21.SYS.ES1.OB05)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (8 & 10)	

Examination Outline Cross-Reference (000036) (APE 36; BW/A08) Fuel-Handling Incidents	Level Tier # Group # K/A # Rating	RO 1 2 AK3.04 3.9
Knowledge of the reasons for the following responses and/or actions as they apply to fuel handling incidents: Establishing containment isolation or closure		

Question 49

Unit 2 is in MODE 6.

A fuel assembly is being removed from the core when a fault in a Programmable Logic Card (PLC) causes the Refueling Machine trolley to abruptly move and then stop.

When the trolley stopped, a camera mounted to the trolley broke loose and splashed water out of the refueling cavity.

Visual observations indicate the Refueling Machine mast is slightly bent.

An RP Technician reports elevated area radiation levels at 200 mRem/hr above the Refueling Cavity.

Containment purge is in progress.

Multiple Containment radiation monitors are alarming locally and on the PC-11.

The Unit Supervisor enters ABN-908, Fuel Handling Accident.

What type of evacuation would be required for the above scenario? ___(1)___

Per ABN-908, which of the following containment systems should be isolated/secured first? ___(2)___.

- A. (1) Evacuate containment and survey all personnel for contamination.
(2) Containment Penetrations.
- B. (1) Evacuate personnel near Refueling Cavity only and survey for contamination.
(2) Containment Penetrations.
- C. (1) Evacuate containment and survey all personnel for contamination.
(2) Containment Ventilation.
- D. (1) Evacuate personnel near Refueling Cavity only and survey for contamination.
(2) Containment Ventilation.

Answer: C

Explanation: (1) According to ABN-908, Section 2.3, Steps 2 and 6, a personnel evacuation of containment must be performed for a fuel handling accident in containment and all personnel must be surveyed for contamination. ABN-908, Section 1 describes actions taken for “An unintentional force applied to a fuel assembly in which sufficient damage may have been sustained that could cause a breach in fuel assembly cladding”. In the scenario above, a bent mast of the Refueling Machine caused by lateral movement of the fuel assembly as it is being withdrawn from a core slot would be indicative of a substantial force applied to the assembly. (2) In the above scenario, containment ventilation would be required to be secured and isolated. (See ABN-908, Section 2.3, Step 3) Containment penetrations are also required to be isolated but the step requiring open Containment penetrations to be closed is later in the procedure at Step 8.

A (1) This portion is correct, see explanation section above. (2) This portion is wrong. Plausible because Containment penetrations are directed to be closed at Step 8 after Containment Ventilation Isolation has been directed at Step 3..

B (1) This portion is wrong, but plausible because one might think that high area radiation levels are originating from the refueling cavity such that only a local evacuation of the affected area would be required. (2) This portion is wrong, see explanation for A.

C This is correct, see explanation section above.

D (1) This portion is wrong, see explanation for B. (2) This portion is correct, see explanation section above.

Technical References:

ABN-908 – Fuel Handling Accident (Rev. 5) pp.2-4

References to be provided to applicants during exam: None.

Learning Objective: Given that a Fuel Handling accident is in progress SUMMARIZE the expected control room response in accordance with ABN-908, “Fuel Handling Accident.” (LO21.RFO.FH2.OB3)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (5 & 10)	

Examination Outline Cross-Reference
(000037) (APE 37) Steam Generator Tube
Leak

Level	RO
Tier #	1
Group #	2
K/A #	AK3.10
Rating	3.5

Knowledge of the reasons for the following responses and/or actions as they apply to STEAM Generator Tube Leak: Automatic actions associated with high radioactivity in S/G sample lines

Question 50

Unit 1 is operating at 100% reactor power.

The following radiation monitors are alarming:

- 1-RE-2959, UNIT 1 CONDENSER OFF GAS RADIATION DETECTOR – Yellow ALERT
- 1-RE-2328, MAIN STEAM LINE 1-04 RADIATION DETECTOR – Red ALARM
- 1-RE-2328A, MAIN STEAM LINE 1-04 STEAM GENERATOR LEAK RATE MONITOR DETECTOR – Red ALARM

ABN-106, High Secondary Activity has been entered and preparations to reduce power to 50% are underway.

ABN-106 directs Chemistry to sample steam generators to verify which steam generator is the source of the primary-to-secondary leak and to determine the leak rate.

Considering the conditions above, what information will be provided to the Chemistry Technician per ABN-106, pertaining to the steam generator grab sample?

- A. The blowdown sample radiation monitor may alarm and sample isolation valves will have to be held open based on process radiation levels.
- B. An additional cation and mixed-bed demineralizer may need to be placed in service based on sample activity levels.
- C. Blowdown outlet flow may need to be aligned to the main condenser hotwell based on sample activity levels.
- D. The current value of N16 leak rate indication based on power $\geq 40\%$.

Answer: A

Explanation: With secondary plant radiation detectors being elevated and 3 detectors in alarm as stated in the above scenario, there is significant evidence of a Pri-Sec U-Tube leak.

With a shutdown of the plant imminent, ABN-106 directs Chemistry to implement CHM-113 to determine which S/G is the source of the leakage and to aid in post-shutdown planning. It is highly plausible, given the radiological conditions, that the 1-RE-4200 Sample Rad Monitor would alarm, before or during the grab sample. If this were to occur, then the note provided by ABN 106 before step 6 would apply, "Sample isolation valves will have to be held open while sampling if isolated due to high radiation on u-RE-4200".

A is correct, see explanation above.

B is wrong. Per SOP-305A, only one set Demins will be placed in service at a time. Plausible because one set of demins are only designed to handle a 20 gpd pri-sec leak with a 1% fuel defect. Though a fuel defect is not presented in the above scenario, If one was not familiar with the normal lineup of SOP-305A, they might suppose that additional demins would be necessary to handle the increased activity.

C is wrong. Per SOP-305A, Steam Generator Activity levels do not determine the alignment of blowdown outlet flow. It is based on plant mode and Demin status. Plausible because it is required to align blowdown outlet flow to the Hotwell while the Demins are out of service. The basis of this alignment is to provide some cleaning of outlet flow as it passes through the Condensate Polishing system. One might think that the increased activity levels of a pri-sec leak would necessitate additional cleaning from this alignment.

D is wrong. N16 leak rate indication is not required to be relayed to the chemistry technician performing the grab sample per ABN-106. Plausible because the procedure requires comparing N16 leak rate monitor indication to Chemistry leak rate calculations using the COG radiation levels if all main steamline radiation monitors were clear.

Technical References:

ABN-106 – Abnormal Conditions Procedures (Rev. 13) pp. 4, 19 (see below)
SOP-305A – Steam Generator Blowdown and Cleanup System (Rev. 23)

References to be provided to applicants during exam: None.

Learning Objective: ANALYZE the response to Steam Generator Tube Leakage greater than or equal to 75 gpd in accordance with ABN-106, High Secondary Activity. (LO21.ABN.106.OB02)

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (5 & 10)	

Examination Outline Cross-Reference	Level	RO
068 Control Room Evacuation	Tier #	1
	Group #	2
Ability to operate and/or monitor the following as they apply to Control Room Evacuation:	K/A #	AA1.33
RPS	Rating	3.5

Question 51

While operating in Mode 1:

- A security compromise occurs that requires evacuation of the control room.
- There is no indication of fire within the control room or cable spreading room.

Per ABN-905, Loss of Control Room Habitability, the Reactor Operator is directed to perform ___(1)___ and will ___(2)___.

- (1) Attachment 1
(2) trip the reactor from the control room
- (1) Attachment 2
(2) trip the reactor from the control room
- (1) Attachment 1
(2) trip the reactor from the remote shutdown panel
- (1) Attachment 2
(2) trip the reactor from the remote shutdown panel

Answer: A

Explanation:

RPS includes the reactor trip circuit breakers. A is correct because ABN-905, Loss of Control Room Habitability, Attachment 1, directs that steps 1-4 will be performed by the reactor operator prior to evacuating the Control Room;. ABN-905 is applicable because of the plant conditions in the cue and the applicability section of the procedure.

B is wrong because part (1) is incorrect – Attachment 2 is performed by the relief reactor operator at the remote shutdown panel; is plausible because part (2) is correct.

C is wrong because part (2) is incorrect; is plausible because part (1) is correct.

D is wrong because parts (1) and (2) are incorrect; is plausible because Attachment 2 is part of ABN-905 and the reactor trip breakers will be verified open at the remote shutdown panel per step 2.a of the procedure.

Technical References:

ABN-905, Loss of Control Room Habitability, Attachment 1.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.803/905

Terminal Objective: From memory, analyze the sequence of events and operator actions in accordance with ABN-803, Response to a Fire in the Control Room or Cable Spreading Room, and ABN-905, Loss of Control Room Habitability.

Enabling Objective b: Analyze the response to a Loss of Control Room Habitability in accordance with ABN-905, Loss of Control Room Habitability. (LO21.ABN.803.OB02)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b).10	

Examination Outline Cross-Reference	Level	RO
E03 LOCA Cooldown and Depressurization	Tier #	1
	Group #	2
Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to LOCA Cooldown and Depressurization: Blocking low steam line pressure SI when the PZR pressure lowers less than P-11 setpoint during cooldown	K/A #	EK1.06
	Rating	3.9

Question 52

While operating at full power a small break LOCA occurred.

- EOS-1.2A, Post LOCA Cooldown and Depressurization has been completed up to Step 11, Initiate RCS Cooldown to Cold Shutdown.
- Pressurizer pressure is 1950 psig and slowly lowering.

Step 11c states: BLOCK low steam line pressure SI signal.

Per EOS-1.2A step 11 **bases**, what is the **reason** for blocking low steam line pressure SI signal to prevent Main Steam Isolation Valve automatic closure?

- precludes violation of Integrity Status Tree thermal shock limits.
- permits controlled cooldown to prevent exceeding cooldown rate.
- maintains preferred steam path to condenser to conserve inventory.
- minimizes any radiation release.

Answer: C

Explanation:

Per the basis statement, the reason for blocking the low steam pressure SI signal is to use the preferred steam path to the condenser to preserve inventory.

A is wrong but plausible because the rate of cooldown could affect the Integrity safety function but this not the reason for blocking the low steam pressure SI signal.

B is wrong but plausible because this will allow a more controlled cooldown than the ARVs. This is in fact the alternate pathway to cooldown if by chance the crew misses it and the MSIV's go shut.

C is correct (see above explanation).

D is wrong but plausible because dumping steam to condenser will minimize any release that otherwise would go out the ARV but this is not the reason for this step per the bases.

Technical References:

EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION, Revision 9, step 11 on page 8 and bases for step 11 on page 44.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG12.E12.OB05

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)8	

Examination Outline Cross-Reference (W E09) Natural Circulation	Level	RO
	Tier #	1
	Group #	2
(WE09 EA1.08) Ability to operate and/or monitor the following as they apply to NATURAL CIRCULATION OPERATIONS: CRDM Cooling fans	K/A #	EA1.20
	Rating	3.0

Question 53

Unit 1 reactor trip occurred due to loss of offsite power and the crew has reached step 16 of EOS-0.2A, Natural Circulation Cooldown, to continue RCS cooldown and depressurization.

Plant conditions are as follows:

- | | |
|--|------------------|
| • Letdown | In service |
| • CRDM fan | Only one running |
| • Core Exit Thermocouple (CET) temperature | 560°F |
| • RCS pressure | 1625 psig |
| • Auxiliary spray valve | Closed |

Based on these conditions, what is the next step that the crew should take?

- Continue cooldown to establish greater than 75°F subcooling based on CETs.
- Ensure normal spray valves are closed and then open auxiliary spray valve.
- Continue cooldown to establish greater than 125°F subcooling based on CETs.
- Open one pressurizer PORV to continue cooldown and depressurization.

Answer: C

Explanation:

In step 16b of EOS-0.2A states to maintain subcooling as required in step 15. In step 15 it requires greater than 125° F when both CRDM fans are NOT running, which is the case indicated in the stem. In steam table for 1610 psia (converting 1610 psig to psia or absolute) saturation temperature would be roughly 606° F, or approximately 46° F subcooling. This is well below the required Subcooling Margin (SCM) of 125° F, so answer C would be correct.

A is wrong because this is the subcooled margin required when both CRDM fans are running. Plausible if you don't remember this limit for two fans versus one fan running.

B is wrong because this is step 16e where you continue to lower pressure and you can't do that with this subcooling margin per RNO 16b.

C is correct because this is the correct action per RNO step 16b to restore SCM to greater than 125° F prior to continuing down with pressure

D is wrong because letdown is in service and therefore a PORV is not used when let down is in service. The aux spray valve would be used to continue the depressurization as stated in step 16 of EOS-0.2A.

Technical References:

EOS-0.2A, Rev 9, pages 16-17

CE Steam Tables, Revision 15 (15th printing)

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E02.OB02

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)5 and 10	

Examination Outline Cross-Reference	Level	RO
W E08 RCS Overcooling – Pressurized	Tier #	1
Thermal Shock	Group #	2
	K/A #	EA2.10
Ability to determine and/or interpret the following as they apply to Pressurized Thermal Shock: VCT level and/or pressure	Rating	3.0

Question 54

Unit 1 initiated a Reactor Trip and Safety Injection due to a small break LOCA in Containment.

- FRP-0.1A, Response to Imminent Pressurized Thermal Shock is in progress.
- Auxiliary spray is being used to depressurize the RCS.

Step 23 “Establish Letdown” instructs the operator to ensure that 1/1-LCV-112A, VCT LVL CTRL VLV is in ___(1)___ to ___(2)___.

- (1) AUTO
(2) prevent overfilling the VCT
- (1) MAN
(2) minimize thermal stress on auxiliary spray nozzle
- (1) AUTO
(2) minimize thermal stress on auxiliary spray nozzle
- (1) MAN
(2) prevent overfilling the VCT

Answer: A

Explanation:

A is correct because FRP 0.1A, step 3(l) has the operator place LCV-112A in AUTO if the CCP suction is NOT aligned to the VCT. The applicant must know that SI initiation would have realigned CCP suction from the VCT to the RWST. FRP 0.1A, Attachment 4 mentions the reason for placing it in AUTO would be to prevent overfilling the VCT.

B is wrong because the procedure requires this to be in AUTO. The applicant may mistakenly think it should be in manual since the same step requires that letdown pressure and temperature control should be placed in manual. The second part is incorrect since minimizing thermal stress is accomplished by placing letdown in service NOT due to putting VCT level control in AUTO.

C is wrong because minimizing thermal stress is accomplished by placing letdown in service NOT due to putting LCV-112A in AUTO. The first part of the answer is correct.

D is wrong because the procedure requires this to be in AUTO. The applicant may mistakenly think it should be in manual since the same step requires that letdown pressure and temperature control should be in manual. The second part of the answer is correct.

Technical References:

FRP-0.1A, "Response to Imminent Pressurized Thermal Shock Condition"
LO21.SST.SI1 "Safety Injection Systems and Accumulators"

References to be provided to applicants during exam: None.

Learning Objective: Document learning objective if possible.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)... 55.43	

Examination Outline Cross-Reference (011) (SF2 PZR LCS) PRESSURIZER LEVEL CONTROL SYSTEM	Level Tier # Group # K/A # Rating	RO 2 2 K6.16 3.6
Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Pressurizer Level Control System: PZR PCS		

Question 55

Consider the following:

- Unit 2 was operating at 50% steady-state reactor power.
- Switch 1/2-PS-455F PRZR PRESS CTRL CHAN SELECT is in the 455/456 position.
- Rod control is in AUTO.

A grid perturbation occurred causing a sudden change to Main Turbine load.

Main Turbine load is now stable at the new load.

Concurrent with the load change, 2-PI-455A, PRZR PRESS CHAN I failed low.

Reactor power and Tave are now stable at new values, but Pressurizer parameters are still recovering.

- Current Tave is 566.4°F.
 - Current Pressurizer level is 43%.
 - Current Pressurizer pressure is 2248 psig.
1. Based on the above indications, what type of Main Turbine load change occurred?
 2. If switch 1/2-PS-455F, PRZR PRESS CTRL CHAN SELECT is taken to the 457/456 position, what will be the status of the Pressurizer Backup Heaters?
 - A. (1) Load increase
(2) One group of Backup Heaters on
 - B. (1) Load reduction
(2) All Backup Heaters on
 - C. (1) Load increase
(2) All Backup Heaters on
 - D. (1) Load reduction
(2) One group of Backup Heaters on

Answer: B

Explanation:

(1) The given scenario describes a rapid load change from 50% to 29%. The initial power of 50% is given in the stem, but the final power of 29% must be derived by calculating programmed Tave. Since the Tave of 566.4F is given as the final steady state value, the following calculation would be used to determine the final reactor power.

Tave range (557-589.2) °F for Rx power ranges 0-100% (note: Unit 2 max temp is greater than max Unit 1 temp. of 585.4F)

$$\begin{aligned} \text{Tave prog} &= (589.2 - 557)(\text{Rx pwr}) + 557 \\ \text{Rx pwr} &= (\text{Tave prog} - 557) / (589.2 - 557) && \text{Rearranged for Rx pwr} \\ \text{Rx pwr} &= (566.4 - 557) / (589.2 - 557) \\ \text{Rx pwr} &= 29.2\% && \text{Solved} \end{aligned}$$

Based on the above calculation, it is shown that the plant has experienced a turbine load reduction.

(2) When the PS-455F switch is taken to the P457/P456 position to mitigate the failure of PT-455, Pressurizer pressure control functions return to normal. However, the backup heaters, which were formally on due to the instrument failure, remain energized because of a greater than 5% level above the post transient program level. The new program level would be quantified based on the previously calculated Rx pwr of 29.2% as follows:

$$\begin{aligned} \text{Pzr prog lvl} &= (60 - 25)(\text{Rx pwr}) + 25 \\ \text{Pzr prog lvl} &= 35(.292) + 25 \\ \text{Pzr prog lvl} &= 35.22\% \end{aligned}$$

The given Pressurizer level of 43% is greater than 5% above the calculated program level, therefore, the backup heaters will energize to mitigate the insurge.

Given the fact that 1-PI-455A has failed low, one of the first implications to consider is that all Pzr backup heaters energize and variable heater fully energize. The energized heaters do not produce an immediate effect, however. Based on the implied timing of the instrument failure concurrent with the initiation of the transient to the stabilization of the plant, no more than a minute or two would have elapsed and Pressurizer pressure would not have increased substantially. When the PS-455F switch is taken to the P457/P456 position the PT-455 signal is no longer used and heater functions return to normal.

A (1) This portion is wrong, but plausible because the final given Pressurizer level is higher than the initial (calculated) program level. (2) This portion is wrong. Plausible because once the PS-455F switch is in the correct position, the backup heaters would return to their normal configuration if it were not for the >5% difference between program level and actual.

B this answer is correct. See explanation section above.

C (1) This portion is wrong. See explanation for A. (2) This portion is correct. See explanation section.

D (1) This portion is correct. See explanation section. (2) This portion is wrong. See explanation for A.

Technical References:

LO21SYSPP1 – Pressurizer Pressure and Level Control (Rev. 0) pp. 7, 9, 22 (see below)

References to be provided to applicants during exam: None.

Learning Objective:

DESCRIBE the components of the Pressurizer Level Control System including interrelations with other systems to include interlocks and control loops in accordance with the CPNPP FSAR and DBD-ME-250, Reactor Coolant System. (LO21.SYS.PP1.OB05)

EXPLAIN the normal, abnormal and emergency operation of the Pressurizer Pressure and Level Control System in accordance with ABN-705, Pressurizer Pressure Malfunction, and ABN-706, Pressurizer Level Instrumentation Malfunction. (LO21.SYS.PP1.OB07)

DIFFERENTIATE between the Unit 1 and Unit 2 Pressurizer Pressure and Level Control Systems in accordance with CPNPP Emergency Operating Procedures. (LO21.SYS.PP1.OB09)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	4
10CFR Part 55 Content:	55.41(b) (7)	

Examination Outline Cross-Reference (015) (SF7 NI) NUCLEAR INSTRUMENTATION SYSTEM	Level Tier # Group # K/A # Rating	RO 2 2 K5.06 3.7
Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Nuclear Instrumentation System: Subcritical multiplications and NIS indications		

Question 56

After a three-week outage, Unit 1 is in the process of performing a reactor startup.

At time 0915, the following instrument readings are observed:

- 1-NI-31B – SR COUNT RATE CHAN I: 400 cps and stable
- 1-NI-32B – SR COUNT RATE CHAN II: 450 cps and stable

At time 0916, the Reactor Operator performs a rod withdrawal per the reactivity plan.

As the control rods are withdrawn, a maximum count rate of approximately 800 cps is observed on both SR instruments.

10 seconds after rod withdrawal is complete; Source Range Instruments settle at the following values:

- 1-NI-31B – SR COUNT RATE CHAN I: 500 cps and stable
- 1-NI-32B – SR COUNT RATE CHAN II: 550 cps and stable

At time 0917, the Reactor Operator performs another rod withdrawal, adding a similar amount of reactivity as the prior withdrawal.

1. How long after the second rod withdrawal is complete will a steady count rate be achieved?
2. In the scenario above, what caused the Nuclear Instruments to return to stable values?
 - A. (1) Greater than 10 seconds
(2) Subcritical multiplication
 - B. (1) Less than 10 seconds
(2) Subcritical multiplication
 - C. (1) Greater than 10 seconds
(2) Delayed neutron precursors
 - D. (1) Less than 10 seconds

(2) Delayed neutron precursors

Answer: A

Explanation: (1) With the reactor being in the source range, the neutron population is due to source neutrons, which are assumed constant, and neutrons produced by fission, etc. An equilibrium count rate is produced when the total neutrons produced by fissions and source neutrons make up for neutron losses, such as leakage out of the core. When reactivity is “added” by withdrawing control rods, more neutrons from fission are produced and the ratio of fission neutrons to the original source neutrons becomes greater and produces an increased equilibrium count rate.

Neutrons produced by fission produce both prompt and delayed neutrons. The prompt neutrons cause neutron production to “spike” quickly when additional reactivity is added, then to lower as delayed neutrons begin to have an effect. As criticality is approached, the population due to fission neutrons compared to the source neutrons is greater and the effect of delayed neutrons becomes more pronounced, causing the amount of time to reach an equilibrium state to be greater every time reactivity is added.

Since the initial rod withdrawal caused an equilibrium count rate 10 seconds after the rods stopped; the second rod withdrawal, which added a similar amount reactivity, would have to take longer than 10 seconds as criticality is approached and the neutron population ratio of source to fission neutrons increases.

(2) In a subcritical reactor, equilibrium is achieved when the neutron population produced by fissions and source neutrons is equal to neutron losses. Subcritical Multiplication is the process by which this phenomenon occurs. Subcritical Multiplication can be defined as: How reactivity changes affect the degree in which source neutrons make up the total neutron population in a subcritical reactor.

—

A is correct, see explanation section above.

B (1) This portion is wrong, but plausible as the effect of K_{eff} and reactivity might be confused. It could be thought that as criticality is approached, K_{eff} would be greater and would cause a faster return to equilibrium. (2) This portion is correct, see explanation section above.

C (1) This portion is correct, see explanation section above. (2) This portion is wrong. In a subcritical reactor, delayed neutrons allow reactivity changes to be slower and controlled, and they eventually add to the total neutron equilibrium population, but they only provide a small contribution to the total neutrons. Without them, equilibrium would still be achieved, though much more rapidly, especially as criticality is approached. Plausible because delayed neutrons are very critical for reactor control and play an important role in the subcritical multiplication process as a whole.

D (1) This portion is wrong. See explanation for B. (2) This portion is wrong. See explanation for C.

Technical References:

LO21GFRKIN (Reactor Theory – Reactor Kinetics & Neutron Sources) (Rev. 1) pp. 24-37

References to be provided to applicants during exam: None.

Learning Objective: LO21.GFR.KIN.TLO2

Question Source:	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b) (5)	

Examination Outline Cross-Reference (016) (SF7 NNI) NONNUCLEAR INSTRUMENTATION SYSTEM	Level Tier # Group # K/A # Rating	RO 2 2 K3.07 3.7
(016K3.07) Knowledge of the effect that a loss or malfunction of the NONNUCLEAR INSTRUMENTATION SYSTEM will have on the following systems or system parameters: ECCS		

Question 57

Unit 1 just completed heat up to MODE 3 after an extended refueling shutdown.

- RCS temperature stable at 557°F
- Steam line pressures approximately 1092 psig

0803: 1-PI-515A, MSL 1 PRESS CHAN 1 lowered to 590 psig and stabilized.

0810: Another MSL pressure channel failure occurs.

0813: Automatic Safety Injection occurs.

- NO operator actions have been taken for either failure.

Based on the conditions above, which of the following instrument failures **at 0810** would have caused the Safety Injection?

- A. 1-PI-516A, MSL 1 PRESS CHAN IV failing high
- B. 1-PI-524A, MSL 2 PRESS CHAN I failing low
- C. 1-PI-2325, MSL 1 PRESS failing low
- D. 1-PI-2326, MSL 2 PRESS failing high

Answer: D

Explanation: 1-PI-515A provides 1 of 3 input for low steam line pressure safety injection. A second protection channel failure low on the same steam line would result in safety injection actuation. 1-PI-2325 and 1-PI-2326 are non-safety pressure instruments that control their respective Steam Generator Atmospheric Relief Valves (ARV). If the instrument fails high, the associated ARV will open, and with little decay heat, a safety injection from the resulting cooldown will occur due to low Steam Line pressure or low Pressurizer pressure.

- A. Incorrect – plausible since two channel failures on the same steam line could result in a low steam line pressure safety injection. However, the channel would need to fail low to cause actuation.
- B. Incorrect – plausible since two channels failing low on the same steam line would result in low steam line pressure safety injection. However, 1-PI-524A is on main steam line 2.
- C. Incorrect – plausible since two protection channels failing low on the same steam line would result in low steam line pressure safety injection. However, 1-PI-2325 is a control channel; not a protection channel.
- D. Correct – 1-PI-2326 failing high would result in 1-PCV-2326, SG 2 ATMOS RLF VLV fully opening. Without operator intervention and low decay heat, the excessive cooldown will result in a low Steam Line pressure or low Pressurizer pressure safety injection.

Technical References:

LO98AUDTPC – Trips, Permissives and Controls (Rev 0) pg. 56 (see attached)
LO21SYSES1 – RPS/ESFAS (Rev 0) pg. 91 (see attached)
LO21SYSMR1 – Main Steam (Rev 0) pp 66-67 (see attached)
LO21MCOTA8 – Increased Heat Removal Accident pp 10-11 (see attached)
ABN-709 – STM LINE INSTR MALFUNCTION (Rev. 10)

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the instrumentation and controls of the Main Steam system and the system response in accordance with DBD ME 202. (LO21.SYS.MR1.OB04)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b) (7)	

Examination Outline Cross-Reference (029) (SF8 CPS) CONTAINMENT PURGE SYSTEM	Level Tier # Group # K/A # Rating	RO 2 2 K1.06 3.3
(029K1.06) Knowledge of the physical connections and/or cause and effect relationships between the (SF8 CPS) CONTAINMENT PURGE SYSTEM and the following systems: CNT		

Question 58

Unit 1 refueling cavity is being prepared for drain following core offload.

- The equipment hatch is off.
- The personnel airlock is open.

Radiation Protection requests that 1-HS-5572, AIR PRG SPLY DMPR be CLOSED per SOP-801A, Containment Ventilation System.

1. What is the reason for this request from Radiation Protection?
2. How are proper building pressures maintained if additional fans are needed?
 - A. 1. prevent spread of contamination into Safeguards building
2. by using SOP-801A, Containment Ventilation System
 - B. 1. prevent spread of contamination into Safeguards building
2. by using SOP-816, Primary Plant Ventilation System
 - C. 1. prevent spread of contamination into Containment
2. by using SOP-816, Primary Plant Ventilation System
 - D. 1. prevent spread of contamination into Containment
2. by using SOP-801A, Containment Ventilation System

Answer: B

Explanation: According to SOP-801A, Note 1 on page 38, "IF requested by RP OR Decon, THEN the air purge supply damper (1-HS-5572) may be left closed during periods that the equipment hatch is off AND personnel airlock is open to ensure air flow through the personnel airlock is into Containment. This will prevent the spread of contamination into the Safeguards Building."

Per SOP-801A, page 39, the building pressures are maintained (when additional fans are needed) by using SOP-816.

A is wrong because part 2 is wrong. Part 1 is correct (see above). Plausible because this is the procedure used to start the purge.

B is correct (see above explanation).

C is wrong because part 1 is wrong but plausible if you think you close the damper to containment to prevent the spread in the Containment building. Part 2 is correct (see above).

D is wrong because part 1 and part 2 are wrong (see explanation above).

Technical References:

SOP-801A, revision 14.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CL1.OB03.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)9	

Examination Outline Cross-Reference (045) (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM	Level Tier # Group # K/A # Rating	RO 2 2 A3.04 3.8
(045A3.04) Ability to monitor automatic features of the (SF4S MTG) MAIN TURBINE GENERATOR SYSTEM, including: Main turbine trip		

Question 59

Unit 1 is at 100% power when an automatic reactor and turbine trip occur.

What is the status of the main generator output breakers for this condition?

- A. Main generator breaker trip is delayed 30 seconds to sustain power to RCPs.
- B. Main generator breaker trip is delayed 30 seconds to allow proper bus transfers first.
- C. Main generator breaker trip is delayed 11.5 seconds to sustain power to RCPs.
- D. Main generator breaker trip is delayed 11.5 seconds to allow proper bus transfers first.

Answer: A

Explanation: According to training material slide LO21.SYS.MT1, slide on Turbine trip design:

“After an automatic Main Turbine trip, the Generator breaker trip is delayed 30 seconds to furnish uninterrupted power to the RCP motors without relying on the success of a bus transfer provided the Main Generator conditions permit this.

Likewise, following a manual Main Turbine trip, or a turbine trip due to certain turbine faults or certain Generator protection signals, the Main Generator trip is delayed approximately 11.5 seconds.”

A is correct (see above explanation).

B is wrong because the reason is to sustain RCP power without relying on bus transfers.

C is wrong because the time is wrong. Plausible because this is the time for the delay if a manual trip of the turbine is initiated. The reason is correct.

D is wrong because the time is wrong and the reason is wrong. See above explanations. Plausible because time is the time for delay of a manual turbine trip is initiated.

Technical References:

LO21.SYS.MT1, revision 4/25/2022, slide 152.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.MT1.OBO3

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)7	

Examination Outline Cross-Reference	Level	RO
071 Waste Gas Disposal System	Tier #	2
	Group #	2
Ability to manually operate and/or monitor in the control room: Meteorological data	K/A #	A4.06
	Rating	2.7

Question 60

Given the following conditions:

- Unit 2 is at full power
- Gas Decay Tank X-01 is discharging to the Ventilation System per RWS-201, Gaseous Waste Processing System.

An operator reports that X-RE-5701, AUX BLDG VENT DUCT radiation monitor has increased to the HIGH alarm limit and the release was terminated.

Per ABN-902, Release of Radioactive Gas, the Unit Supervisor requests an operator to verify weather conditions have not changed significantly.

The following data is observed on the Plant Computer EP Screen:

WIND SPD	10.3 MPH, GREEN
WIND DIR	191.7 DEG FROM, DARK BLUE
DELTA T-A	-1.0 DEGF, GREEN
DELTA T-B	-1.0 DEGF, GREEN
TEMP	105 DEGF, YELLOW

Which of the following is correct regarding the current weather data from the plant computer?

- All data points are valid and can be used to determine weather conditions.
- ONLY** wind direction is bad quality and cannot be used to determine weather conditions.
- ONLY** temperature is bad quality and cannot be used to determine weather conditions.
- Both wind direction and temperature are bad quality and cannot be used to determine weather conditions.

Answer: B

Explanation:

Dark blue letters on a plant computer point indicate that the point is not functional, or of bad quality. All other points are either green or just above an alarm threshold.

A is incorrect but plausible, since if wind direction was cyan, dependent on plant conditions it could be utilized.

B is correct, see explanation.

C is wrong but plausible because yellow color indicates caution. This does not mean it is bad quality.

D is wrong but plausible if the candidate believes both yellow and dark blue are indicative of bad quality.

Technical References:

ABN-902, Release of Radioactive/Toxic Gas, page 5, 6, 8 of 23
LO21.SYS.IC5, "Meteorological Monitoring, page 9.

References to be provided to applicants during exam: None.

Learning Objective: Describe the components of the Meteorological Monitoring system including interrelations with other systems to include interlocks and control loops as described in DBD-EE-034.

DESCRIBE the components of the Plant Computer system including interrelations with other systems to include interlocks and control loops in accordance with DBD-EE-056.
(LO21.SYS.PT1.OB02)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)11	

Training Material "Meteorological Monitoring", page 9

Examination Outline Cross-Reference	Level	RO
072 Area Radiation Monitoring System	Tier #	2
	Group #	2
Ability to predict and/or monitor changes in parameters associated with operation of the Area Radiation Monitoring System, including:	K/A #	A1.01
Radiation levels	Rating	3.4

Question 61

Recovery from a Loss of Coolant Accident is in progress on Unit 1.

When monitoring Grid 4 radiation levels on the PC-11, the Reactor Operator notes that Containment High Range Radiation Monitor CTE116 (1-RE-6290A)'s value has a white background.

Which of the following identifies the reason for this indication on CTE116?

- A. PC-11 POLL STATUS
- B. OPERATE FAILURE
- C. Channel ALERT Alarm
- D. Channel HIGH Alarm

Answer: A

Explanation:

When an Area Radiation monitor displays a value with a white background, it is indicative of PC-11 POLL STATUS. This is an indication that the monitor has been removed from the communication loop of the PC-11. NOTE - This is an instrumentation and control function of the DRMS system. The Loss of Coolant Accident recovery is only mentioned for plausibility.

A is correct, see answer explanation.

B is incorrect but plausible because there is an alarm associated with OPERATE FAILURE, but it is dark blue.

C is incorrect but plausible because there is an alarm associated with Channel ALERT, however, it is a yellow indication.

D is incorrect but plausible because there is an alarm associated with Channel HIGH, however, it is a red indication.

Technical References: LO21SYSRM1 Digital Radiation Monitoring Rev.01, Pg. 20

References to be provided to applicants during exam: None.

Learning Objective: LO21SYSRM1 Digital Radiation Monitoring

EXPLAIN the normal, abnormal and emergency operation of the Digital Radiation Monitoring System in accordance with SOP-706, ALM-3200, ABN-902, ABN-903 and EOP-1.0.
(LO21.SYS.RM1.OB05)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
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Question History:	Last NRC Exam	No
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Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
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10CFR Part 55 Content:	55.41(b)11	
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Examination Outline Cross-Reference	Level	RO
086 Fire Protection System	Tier #	2
	Group #	2
Ability to (a) predict the impacts of the following on the Fire Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations:	K/A #	A2.02
Low FPS header pressure	Rating	3.4

Question 62

Unit 1 is operating at 100% power.

A fire was reported in the Auxiliary Building.

The Fire Brigade used local hose stations to extinguish the fire.

Fire header pressure initially dropped to 138 psig for 11 seconds and began slowly recovering. Once the fire was extinguished, fire header pressure returned to 160 psig after water flow was manually stopped.

Assuming all pumps were in AUTOMATIC prior to the fire, which of the following correctly describes condition of the Fire Pumps AFTER fire header pressure recovered?

- A. X-01 emergency fill fire pump running, X-04 electric fire pump running, X-05 diesel fire pump running, X-06 diesel fire pump off
- B. X-01 emergency fill fire pump running, X-04 electric fire pump running, X-05 diesel fire pump off, X-06 diesel fire pump running
- C. X-01 emergency fill fire pump off, X-04 electric fire pump running, X-05 diesel fire pump running, X-06 diesel fire pump off
- D. X-01 emergency fill fire pump off, X-04 electric fire pump running, X-05 diesel fire pump off, X-06 diesel fire pump running

SOP-904, FIRE PROTECTION MAIN WATER SUPPLY AND FIRE PUMPS SYSTEM, Rev 18, Page 7

Answer: C

Explanation:

A. is wrong but plausible since X-04 and X-05 pumps would start in response to low pressure but would have to be manually stopped. The Emergency Fill pump is an electric pump but it must be started manually.

B is wrong but plausible since X-04 fire pump would start in response to low pressure and X-06 diesel fire pump would start at 140 psig if pressure remained low for 20 seconds.

However, X-01 emergency fill pump only starts manually and X-05 diesel fire pump should start for these conditions.

C. is correct. X-04 electric fire pump starts at 142 psig after 3 seconds and X-05 diesel fire pump starts at 140 psig if pressure remains low for 10 seconds. Both pumps must be manually stopped. X-01 emergency fill pump only starts manually, and X-06 diesel fire pump would only start at 140 psig if pressure remained low for 20 seconds.

D. is wrong but plausible because X-01 emergency fill pump would remain off since it must be started manually and X-04 electric fire pump will start at 142 psig for 3 seconds. However, X-05 diesel fire pump would start at 140 psig for 10 seconds.

Technical References:

References to be provided to applicants during exam: None

Learning Objective: EXPLAIN the normal operations of the Fire Protection system in accordance with DBD-ME-104 and DBD-ME-225. (LO21.SYS.FP1.OB04)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b).5	

Examination Outline Cross-Reference	Level	RO
002 Reactor Coolant System	Tier #	2
	Group #	2
Knowledge of Reactor Coolant System design features and/or interlocks that provide for the following: Overpressure protection	K/A #	K4.10
	Rating	4.1

Question 63

Regarding Low Temperature Overpressure Protection (LTOP):

Train A LTOP is ARMED by ___(1)___ auctioneered RCS temperature, at a setpoint of ___(2)___.

- A. Train A (1) (2)
 ≤ 350 degrees F
- B. Train A ≤ 320 degrees F
- C. Train B ≤ 350 degrees F
- D. Train B ≤ 320 degrees F

Answer: C

Explanation:

Train A LTOP is armed by Train B, and vice versa. This occurs automatically at ≤ 350F. 320F is the Tech Spec 3.4.12 applicability setpoint.

A is wrong. Opposite trains arm LTOP. Part 2 correct.

B is wrong. Opposite trains arm LTOP. 320F is the TS 3.4.12 applicability.

C is correct.

D is wrong. Part 1 correct. 320F is the TS 3.4.12 applicability.

Technical References:

LO21SYSP2, LTOP, Rev 1 ch 1, pp 9-10.

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the instrumentation and controls of the Low Temperature Overpressure Protection System including the system response in accordance with ABN-714 and ABN-715. (LO21.SYS.PP2.OB03)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3

Comprehensive/Analysis

10CFR Part 55 Content:

55.41(b).7

Examination Outline Cross-Reference System is NA	Level Tier # Group #	RO 3
(G2.1.4) Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, and maintenance of active license status	K/A # Rating	2.1.4 3.3

Question 64

Per STA-121, License Operator Physicals and License Application Process, all NRC licensed operators are required to report any change in prescribed medicines from the family doctor to _____ if it is to treat a new condition.

- A. the Shift Operations Manager immediately
- B. the Shift Operations Manager prior to taking shift
- C. the Shift Operations License Coordinator immediately
- D. the Shift Operations License Coordinator prior to taking shift

Answer: C

Explanation:

IAW STA-121, all NRC Licensed Operators are responsible to notify their supervisor and the Shift Operations License Coordinator immediately of any changes to their physical and/or mental condition or treatment of existing conditions.

A is wrong but plausible since the Shift Operations Manager is responsible for Submitting Certifications of Operators' Medical Examinations to the NRC, and notifying the NRC of the occurrence of any disability after the submission of the licensee's medical examination form. However, the two people required to be notified immediately are the immediate supervisor and the Shift Operations License Coordinator.

B is wrong but plausible since the Shift Operations Manager is responsible for Submitting Certifications of Operators' Medical Examinations to the NRC, and notifying the NRC of the occurrence of any disability after the submission of the licensee's medical examination form. The reporting time is plausible because the operator should not take the shift with a disqualifying medical condition.

C is correct. (see above explanation).

D is wrong because it must be done immediately, not prior to taking the shift.

Technical References:

STA-121, Revision 7, page 6.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ADM.OLR.OB02, OB04

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference	Level	RO
	Tier #	3
Knowledge of how to conduct system lineups, such as valves, breakers, or switches	Group #	
	K/A #	2.1.29
	Rating	4.1

Question 65

Regarding system status control, which of the following conditions requires recording as a discrepancy when performing an initial system lineup?

- A. A hose is installed that is not needed for system fill and vent.
- B. A position indicator is broken, and cannot be repaired.
- C. A required valve cap was missing, and the cap was replaced.
- D. A component was not in the correct position but has been realigned.

Answer: B

Explanation:

A is wrong but plausible, since it would be recorded if the hose could not be removed.

B is correct

C is wrong but plausible, since it would be recorded if the valve cap could not be replaced.

D is wrong but plausible, since the component was not in the correct position. However, it should not be recorded as a discrepancy since the component may be expected to be in the wrong position on an initial lineup and is in the correct position.

Technical References:

ODA-410, System Status Control, Revision 16, pages 19-20

References to be provided to applicants during exam: None.

Learning Objective: LO22.ADM.XA1.RO5005

Question Source: (note changes; attach parent)	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(10)	

Examination Outline Cross-Reference System is NA	Level Tier # Group #	RO 3
(G2.2.39) EQUIPMENT CONTROL: Knowledge of less than or equal to 1 hour TS action statements (does not include action statements of 1 hour or less that follow the expiration of a completion time for a TS condition for which an action statement has already been entered)	K/A # Rating	2.2.39 3.9

Question 66

Unit 1 is in MODE 6.

Core offload is in progress.

Chemistry reports boron concentration from the last refueling canal sample is less than required by Technical Specification LCO 3.9.1, Boron Concentration.

Which of the following describes the REQUIRED ACTION per Technical Specification LCO 3.9.1, Boron Concentration?

- A. Within 1 hour, verify all dilution paths isolated.
- B. Within 1 hour, initiate boration until boron concentration is within limits specified in COLR.
- C. Immediately suspend all operations involving CORE ALTERATIONS and positive reactivity changes and initiate boration of the reactor coolant system.
- D. Immediately suspend all movement of fuel assemblies in the refueling canal and within 4 hours restore boron concentration within limits.

Answer: C

Explanation:

A is wrong but plausible because this is an immediate action requirement (not 1 hour) and isolating dilution paths would assist in performing the action to stop all positive reactivity additions.

B is wrong but plausible because boration must be performed to restore boron concentration to greater than the limit specified in the COLR, however it must be immediately restored to within limits.

C is correct.

D is wrong but plausible because the movement of fuel assemblies must be suspended however boron concentration must be immediately restored to within limits.

Technical References:

TS LCO 3.9.1, Amendment 156, page 3.9-1.

References to be provided to applicants during exam: None.

Learning Objective: Apply administrative requirements of the SFPC and cleanup system, including TS, TRM, and ODCM.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	2014 NRC Q69
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Question History:	Last NRC Exam	No
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Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
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10CFR Part 55 Content:	55.41(b)10
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Examination Outline Cross-Reference	Level	RO
	Tier #	3
Knowledge of tagging and clearance procedures	Group #	
	K/A #	2.2.13
	Rating	4.1

Question 67

As part of a clearance, several normally sealed open valves were closed. During restoration, a _____ seal should be reapplied to these valves.

- A. Red
- B. Green
- C. Blue
- D. Yellow

Answer: A

Explanation:

A is correct because red is the color applied to sealed open positions.

B is wrong because red is used for open positions; is plausible because this is the color used to identify normally sealed closed positions.

C is wrong because red is used for open positions; is plausible because this is the color used for throttled positions.

D is wrong because this is the color used for personal safety.

Technical References: ODA-403, Operations Department Locked Component Control, Rev 8, section 4.5, page 3

References to be provided to applicants during exam: None.

Learning Objective: Lesson LO21ADMXA1

Enabling objective H: EXPLAIN the administrative controls of valves, breakers and other equipment required to be secured-in-position in accordance with ODA-403 and OWI-103.

Question Source:
(note changes; attach parent)

Bank #
Modified Bank #

2016 NRC exam Q69
and 2019 NRC exam
Q69

New

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.41(b)10

Examination Outline Cross-Reference:	Level	RO	SRO
Rev:0	Tier	3	
	Group		
Level of Difficulty: 2	K/A		2.2.13
	Importance Rating	4.1	
K/A Statement: Knowledge of tagging and clearance procedures.			

Question 69

As part of a clearance, several normally sealed throttled valves were closed. During restoration, a _____ seal should be reapplied to these valves.

- A. Red
- B. Green
- C. Blue
- D. Yellow

Answer: C

K/A Match: This question matches the KA by requiring knowledge of tagging procedures.

Explanation:

- A. Incorrect. Plausible as this is the color used to identify normally sealed open or off positions.
- B. Incorrect. Plausible as this is the color used to identify normally sealed closed positions.
- C. Correct. This is the color used to identify normally sealed throttled positions.
- D. Incorrect. Plausible as this is the color used for personal safety.

Technical Reference(s): ODA-403, Rev 8 PCN-2 Attached w/ Revision # See
Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Remove plant equipment from service as required for maintenance (OPD1ADMXA1OB09).

Question Source: **Bank #** 2016 NRC Exam Q69 (note changes or
Modified Bank #

attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge Comprehensive or Analysis X

10CFR Part 55 Content: 55.41 10
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 4	Tier	3		
	Group			
	K/A	G.2.2.13		
	Importance Rating	4.1		

Knowledge of tagging and clearance procedures	
Question # 69	
<p>As part of a clearance several normally sealed throttled valves were closed. During restoration, a _____ seal should be reapplied to these valves.>></p> <p>A. Red</p> <p>B. Green</p> <p>C. Blue</p> <p>D. Yellow</p>	
Answer: C	

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate knowledge the color of seal that would be applied when restoring a clearance.

Explanation:

A. Incorrect. Plausible as this is the color used to identify normally sealed open or off positions.

B. Incorrect. Plausible as this is the color used to identify normally sealed closed positions.

C. Correct. This is the color used to identify normally sealed throttled positions.

D. Incorrect. Plausible as this is the color used for personal safety

Technical Reference(s)	ODA-403	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **REMOVE** plant equipment from service as required for maintenance; **BRIEFING** affected personnel, **REVIEWING** and **AUTHORIZING** the appropriate work documents and permits. (OPD1.ADM.XA1.OB09)>>

Question Source: Bank # _____
 Modified Bank # _____ (Note changes or attach parent)
 New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
 Comprehension or Analysis _____
 Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
 55.43 _____

Examination Outline Cross-Reference	Level	RO
No System	Tier #	3
	Group #	NA
(G2.3.5) RADIATION CONTROL: Ability to use RMSs, such as fixed radiation monitors and alarms or personnel monitoring equipment (CFR: 41.11 / 41.12 / 43.4 / 45.9)	K/A #	G2.3.5
	Rating	2.9

Question 68

A portable frisker is being used to perform a whole-body frisk.

Background radiation is 100 counts per minute.

Which of the following is the MINIMUM continuous count rate at which an individual must notify Radiation Protection for assistance per STA-653, Contamination Control Program?

- A. 155 counts per minute
- B. 185 counts per minute
- C. 225 counts per minute
- D. 255 counts per minute

Answer: C

Explanation: According to STA-653, 100 counts above background is considered contaminated and must notify RP for assistance, which is 100 cpm + 1—cpm, or 200 cpm. The MINIMUM value that is at or above 200 cpm is 225 cpm, or C.

A is wrong because the math is wrong. Plausible if 50 cpm above background is used.

B is wrong because the math is wrong. Plausible if used 75 cpm above background.

C is correct (see above explanation).

D is wrong because the math is wrong. Plausible if used 150 cpm above background.

Technical References:

STA-653, revision 21, page 18.

References to be provided to applicants during exam: None.

Learning Objective: Explain how to monitor personnel for contamination, incl using friskers.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

2014 NRC Q71

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)11	

Examination Outline Cross-Reference	Level	RO
No System	Tier #	3
(G2.4.34) EMERGENCY	Group #	NA
PROCEDURES/PLAN: Knowledge of RO responsibilities outside the main control room during an emergency (CFR: 41.10 / 43.5 / 45.13)	K/A #	G2.4.34
	Rating	4.2

Question 69

For a fire in the cable spreading room, ABN-803A, Response to a Fire in the Control Room or Cable Spreading Room has been entered.

The Relief Reactor Operator performs _____.

- A. Attachment 1 to trip the reactor
- B. Attachment 2 to reposition transfer switches
- C. Attachment 3 to trip reactor coolant pump breakers
- D. Attachment 4 to trip diesel generator feeder breakers

Answer: B

Explanation: According to ABN-803A, the Relief Reactor Operator performs Attachment 2, the Reactor Operator performs Attachment 1 prior to leaving the control room, Nuclear Equipment Operator # 1 performs Attachment 3 and NEO #2 performs Attachment 4.

A is wrong because RO performs Attachment 1 (see above). Plausible if don't remember who performs which attachment.

B is correct (see above explanation)

C is wrong because NEO # 1 performs Attachment 3. Plausible if don't remember who does which attachment.

D is wrong because NEO # 2 performs Attachment 4. Plausible if don't remember who does which attachment.

Technical References:

STA-803A, Revision 16, page 32.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.803.OB01.

Question Source:

Bank #

(note changes; attach parent)

Modified Bank #
New

X

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.41(b)10

Examination Outline Cross-Reference	Level	RO
192006 Reactor Theory	Tier #	4
(192006K1.03) FISSION PRODUCT	Group #	NA
POISONS (CFR: 41.1): Describe the	K/A #	K1.03
production of xenon 135	Rating	2.8

Question 70

Unit 1 just tripped from full power where it had sustained operation for the past six months.

The immediate Xenon-135 concentration will:

- A. decrease due to decay of Cesium-136.
- B. increase due to decay of Iodine-135.
- C. decrease due to Xenon-135 production from fission has stopped.
- D. increase due to differences in the half lives of Cesium-136 and Xenon-135.

Answer: B

Explanation: Xenon-135 is produced in a reactor by two primary methods. One is directly from fission; the other is from the decay of iodine-135. Immediately after a reactor trip from full power, xenon-135 will rise and this is because the iodine-135 production term is much bigger than the loss term from fission being significantly reduced when the reactor tripped.

A is wrong because xenon-135 does not decrease immediately after a reactor trip, it rises or increases.

B is correct (see above explanation).

C is wrong because xenon-135 does not decrease immediately after a reactor trip, it rises or increases.

D is wrong because it increases because of the decay from iodine-135. Second aspect is incorrect as well.

Technical References:

LO21.GFR.FPP, revision 3.2, slides 15, 18

References to be provided to applicants during exam: None.

Learning Objective: Document learning objective if possible.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

GFE P59

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)1	

TOPIC: 192006
KNOWLEDGE: K1.03 [2.7/2.8]
QID: P59

Immediately after a reactor trip from sustained high power operation, xenon-135 concentration in the reactor will...

- A. increase, due to the decay of iodine-135.
- B. decrease, because xenon-135 production from fission has stopped.
- C. remain the same, because the decay of iodine-135 and xenon-135 balance each other out.
- D. decrease initially, and then slowly increase due to the differences in the half-lives of iodine-135 and xenon-135.

ANSWER: A.

Xenon and Samarium

- On a reactor trip
 - Xenon will peak with a negative reactivity of almost 5,000 pcm and decay in 3 days back to 0 pcm

Production and Removal of Xenon

ELO 1.3 – Explain how Xe-135 is produced and removed in the core of a nuclear reactor.

- Xe-135 production terms:
 - Direct term
 - Xe-135 - 0.3% fission yield
 - Indirect term
 - I-135/Te-135/Sb-135
 - 5.6% fission yield
- Xe-135 removal terms:
 - Absorption
 - Neutron capture reaction which creates Xe-136 (burnout)
 - Decay
 - Beta-Minus decay to Cesium-135

Examination Outline Cross-Reference	Level	RO
192007 Fuel Depletion and Burnable Poisons	Tier #	4
	Group #	N/A
Define burnable poisons and state its use in the reactor	K/A #	K1.01
	Rating	2.5

Question 71

Integrated Fuel Burnable Absorbers are used in the Comanche Peak reactor core to provide an even ___(1)___ flux distribution.

This allows for ___(2)___.

- A. (1) axial
(2) higher fuel enrichment usage
- B. (1) radial
(2) minimizing control rod usage
- C. (1) axial
(2) minimizing control rod usage
- D. (1) radial
(2) higher fuel enrichment usage

Answer: A

Explanation:

A is correct because Integrated Fuel Burnable Absorbers (IFBAs) are pellets placed in the middle of fuel rods to provide an even axial flux pattern. The advantage of using burnable poisons allows for use of higher fuel enrichment.

B is wrong because Burnable Poison Rod Assemblies (BPRAs) are used to provide a flatter radial power distribution. Also, minimizing control rod usage is an advantage of using chemical shims (using boric acid) and is not associated with the advantages of using burnable poisons.

C is wrong because minimizing control rod usage is an advantage of using chemical shims (using boric acid) and is not associated with the advantages of using burnable poisons.

D is wrong because Burnable Poison Rod Assemblies (BPRAs) are used to provide a flatter radial power distribution NOT the use of IFBAs.

Technical References:

LO21.GFR.DEP – 192007 – Fuel Depletion and Burnable Poison

References to be provided to applicants during exam: None.

Learning Objective: ELO 1.3 – Describe the design and use of burnable poisons in a reactor core.

Question Source:
(note changes; attach parent)

Bank #
Modified Bank #
New

X

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.41(b)(1)

Examination Outline Cross-Reference	Level	RO
192008 Reactor Theory	Tier #	4
(192008K1.21) REACTOR OPERATIONAL PHYSICS (POWER OPERATION): Explain the relationship between steam flow and reactor power given specific conditions	Group #	NA
	K/A #	K1.21
	Rating	3.8

Question 72

Unit 1 has been operating at 90% power for the last week at EOL.

Rod control is in manual.

If a digital turbine control system malfunction opens the main turbine steam inlet valves an additional 5 percent, reactor power will...

- A. increase, because the rate of neutron absorption in the moderator initially decreases.
- B. increase, because the rate of neutron thermalization in the moderator initially increases.
- C. decrease, because the rate of neutron absorption in the moderator initially increases.
- D. decrease, because the rate of neutron thermalization in the moderator initially decreases.

Answer: B

Explanation When the main turbine steam inlet valves open an additional 5 percent, the RCS cold leg temperatures will decrease. As the coolant temperature decreases, the coolant density will increase and the coolant will become a more efficient moderator, requiring fewer collisions for a fission neutron to become a thermal neutron. Fewer collisions means that fission neutrons will exist at fewer intermediate kinetic energy levels while slowing down, which decreases the probability that a fission neutron will be absorbed by U-238 at a resonance energy.

A is wrong because the reason for the increase is wrong (see above explanation)

B is correct (see above explanation)

C is wrong because it increases not decreases and the reason is also wrong.

D is wrong because it increases and the reason it increases is correct for this distracter.

Technical References:

LO21.GFR.LPP-1902008, revision 3.2, slide 57 and notes.

References to be provided to applicants during exam: None.

Learning Objective: LO21.GFR.PHY.TL03.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	P2372
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)1	

*NRC Generic Fundamentals Examination Question Bank—PWR
May 2021*

TOPIC: 192008
KNOWLEDGE: K1.21 [3.6/3.8]
QID: P2372

A nuclear power plant is operating at steady-state 90 percent power near the end of a fuel cycle with manual rod control when a turbine control system malfunction opens the main turbine steam inlet valves an additional 5 percent. Reactor power will initially...

- A. increase, because the rate of neutron absorption in the moderator initially decreases.
- B. increase, because the rate of neutron absorption at U-238 resonance energies initially decreases.
- C. decrease, because the rate of neutron absorption in the moderator initially increases.
- D. decrease, because the rate of neutron absorption at U-238 resonance energies initially increases.

ANSWER: B.

DISCUSSION: When the main turbine steam inlet valves open an additional 5 percent, the RCS cold leg temperatures will decrease. As the coolant temperature decreases, the coolant density will increase and the coolant will become a more efficient moderator, requiring fewer collisions for a fission neutron to become a thermal neutron. Fewer collisions means that fission neutrons will exist at fewer intermediate kinetic energy levels while slowing down, which decreases the probability that a fission neutron will be absorbed by U-238 at a resonance energy.

Steam Plant Startup

Knowledge Check – NRC Bank

A nuclear power plant has been operating at 80 percent power for several weeks when a partial steam line break occurs that releases 2 percent of rated steam flow. Main turbine load and control rod position remain the same.

Assuming no operator or protective actions occur, when the plant stabilizes reactor power will be _____; and average reactor coolant temperature will be _____.

- A. higher; higher
- B. unchanged; higher
- C. higher; lower
- D. unchanged; lower

With the reactor stable (critical) at 80%, a partial steamline break results in more steam being withdrawn from the S/G's, thus cooling off the RCS. A reduction in moderator temperature inserts positive reactivity, initially resulting in a positive SUR. This causes reactor power to rise (approximately 2% power) and stabilize. Therefore, power will increase and Tavg will decrease.

Examination Outline Cross-Reference
(193003K1.16) STEAM (CFR: 41.14): Define
the following term: -- subcooled and
compressed liquids

Level	RO
Tier #	4
Group #	N/A
K/A #	K1.16
Rating	2.7

Question 73

Unit 1 had a reactor trip and safety injection due to a loss of coolant accident.

- RCS pressure is currently stable at 1915 psig.
- RCS temperature is currently stable at 620°F.

Based on these conditions, the RCS is

- A. Superheated
- B. Subcooled
- C. Saturated
- D. Supersaturated

Answer: B

Explanation:

In the steam table, saturated conditions exist at 1900 psia and approximately 630° F. At 620° F and 1900 psia, the RCS would be subcooled or compressed for this pressure. Knowledge of the definition is rolled into applying the definition to the actual plant conditions and determining the answer from that point.

A is wrong because it is not superheated at this pressure and temperature. Applicant could pick this answer due to incorrect application of the steam tables.

B is correct because (see explanation above)

C is wrong because (see explanation above)

D is wrong because (see explanation above)

Technical References:

CE steam tables, revision 15.
ACAD "Thermodynamic Properties of Steam," revision 4.2, page 19.

References to be provided to applicants during exam: None.

Learning Objective: Document learning objective if possible.

Question Source: Bank #

(note changes; attach parent)	Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)14	

Examination Outline Cross-Reference 193004 Thermodynamic Process -Throttling and the Throttling Process	Level Tier # Group # K/A # Rating	RO 4 K1.15 2.8
Determine the exit conditions for a throttling process based on the use of steam and/or water		

Question 74

Unit 2 RCS heatup and pressurization is in progress.

RCS pressure is 785 psig with a steam bubble (96.0 percent quality saturated steam) in the Pressurizer.

1/2-PCV-0455A, PRZR PORV tailpipe temperature has been steadily rising.

PRT pressure is 5 psig.

If 1/2-PCV-0455A is leaking by, which one of the following will be the approximate PORV tailpipe temperature?

- A. 228°F
- B. 258°F
- C. 284°F
- D. 320°F

Answer: B

Explanation:

$$\begin{aligned}
 h_v @ 800 \text{ psia (100\%)} &= 1199.3 \text{ Btu/lbm} \\
 h_l @ 800 \text{ psia (0\%)} &= 509.8 \text{ Btu/lbm} \\
 h @ 800 \text{ psia (96\%)} &= 1199.3 - (.04)(1199.3 - 509.8) \\
 &= 1199.3 - 27.6 \\
 &= 1172 \text{ Btu/lbm}
 \end{aligned}$$

Using Mollier Diagram, 1172 Btu/lbm at 20 psia results in a superheated vapor at 258°F

A is incorrect but plausible because that is the saturation temperature at 20 psia.

B is correct

C is incorrect but plausible as that is the saturation temperature for enthalpy of 1172 BTU/lbm

D is incorrect but plausible as that is the saturation temperature for enthalpy of 1199.3 BTU/lbm

Technical References: Steam tables

References to be provided to applicants during exam: Steam tables provided

Learning Objective: NANT Thermodynamic process, TLO 1.f

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b).14	

*NRC Generic Fundamentals Examination Question Bank—PWR
May 2021*

TOPIC: 193004
KNOWLEDGE: K1.15 [2.8/2.8]
QID: P1577

A heatup and pressurization of the reactor coolant system (RCS) is in progress following a maintenance shutdown. RCS pressure is 800 psia with a steam bubble (96.0 percent quality saturated steam) in the pressurizer. Pressurizer power-operated relief valve (PORV) tailpipe temperature has been steadily rising. PORV downstream pressure is 20 psia.

Which one of the following will be the approximate PORV tailpipe temperature if a PORV is leaking by?

- A. 226°F
- B. 258°F
- C. 284°F
- D. 320°F

ANSWER: B.

SOLUTION: $h_v @ 800 \text{ psia (100\%)} = 1199.3 \text{ Btu/lbm}$
 $h_l @ 800 \text{ psia (0\%)} = 509.8 \text{ Btu/lbm}$
 $h @ 800 \text{ psia (96\%)} = 1199.3 - (.04)(1199.3 - 509.8)$
 $= 1199.3 - 27.6$
 $= 1172 \text{ Btu/lbm}$

Using Mollier Diagram, 1172 Btu/lbm at 20 psia results in a superheated vapor at $\approx 258^\circ\text{F}$

Examination Outline Cross-Reference	Level	RO
193009 Thermodynamics	Tier #	4
	Group #	NA
(193009K1.07) CORE THERMAL LIMITS	K/A #	K1.07
(CFR: 41.14): Describe factors that affect peaking and hot channel factors	Rating	3.3

Question 75

For Comanche Peak, which one of the following describes fuel thermal conductivity with the RCS through core life?

(Assume the reactor operates at steady-state 100 percent power throughout the fuel cycle.)

- A. Smaller at EOL, due to more fuel pellet densification.
- B. Smaller at EOL, due to more contamination of fuel rod fill gas with fission product gases.
- C. Greater at EOL, due to a smaller gap between fuel pellets and fuel rod cladding.
- D. Greater at EOL, due to greater temperature difference between fuel pellets and reactor coolant.

Answer: C

Explanation:

There are many factors that affect the fuel to pellet thermal conductivity (and therefore the peaking factors) during a fuel cycle, but the reduction in pellet to cladding gap size is the dominant factor, leading to a greater fuel-to-coolant thermal conductivity at the End of Life (EOL).

A is wrong because the thermal conductivity is greater at EOL.

B is wrong because the thermal conductivity is greater at EOL

C is correct (see above explanation).

D is wrong because the reason is incorrect (the fuel to coolant delta T is smaller at EOL).

Technical References:

LO21.GFT.LIM Revision 3.3, slide 49

References to be provided to applicants during exam: None.

Learning Objective: LO21.GFT.LIM.ELO3.

Question Source:	Bank #	P1984
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.41(b)14

*NRC Generic Fundamentals Examination Question Bank—PWR
May 2021*

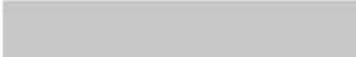
TOPIC: 193009
KNOWLEDGE: K1.07 [2.9/3.3]
QID: P1894 (B1395)

Which one of the following describes the fuel-to-coolant thermal conductivity for a fuel rod near the end of a fuel cycle (EOC) when compared to the beginning of the same fuel cycle (BOC)? (Assume the reactor operates at steady-state 100 percent power throughout the fuel cycle.)

- A. Smaller at EOC, due to more fuel pellet densification.
- B. Smaller at EOC, due to more contamination of the fuel rod fill gas with fission product gases.
- C. Greater at EOC, due to a smaller gap between the fuel pellets and the fuel rod cladding.
- D. Greater at EOC, due to a greater temperature difference between the fuel pellets and the reactor coolant.

ANSWER: C.

DISCUSSION: There are many factors that affect the fuel-to-coolant thermal conductivity during a fuel cycle, but the reduction in pellet-to-cladding gap size (option C) is the dominant factor, leading to a greater fuel-to-coolant thermal conductivity at the EOC. Options A and B are incorrect because the thermal conductivity is greater at EOC. Option D is partially correct, but the reason is incorrect because the fuel-to-coolant ΔT is smaller at EOC.



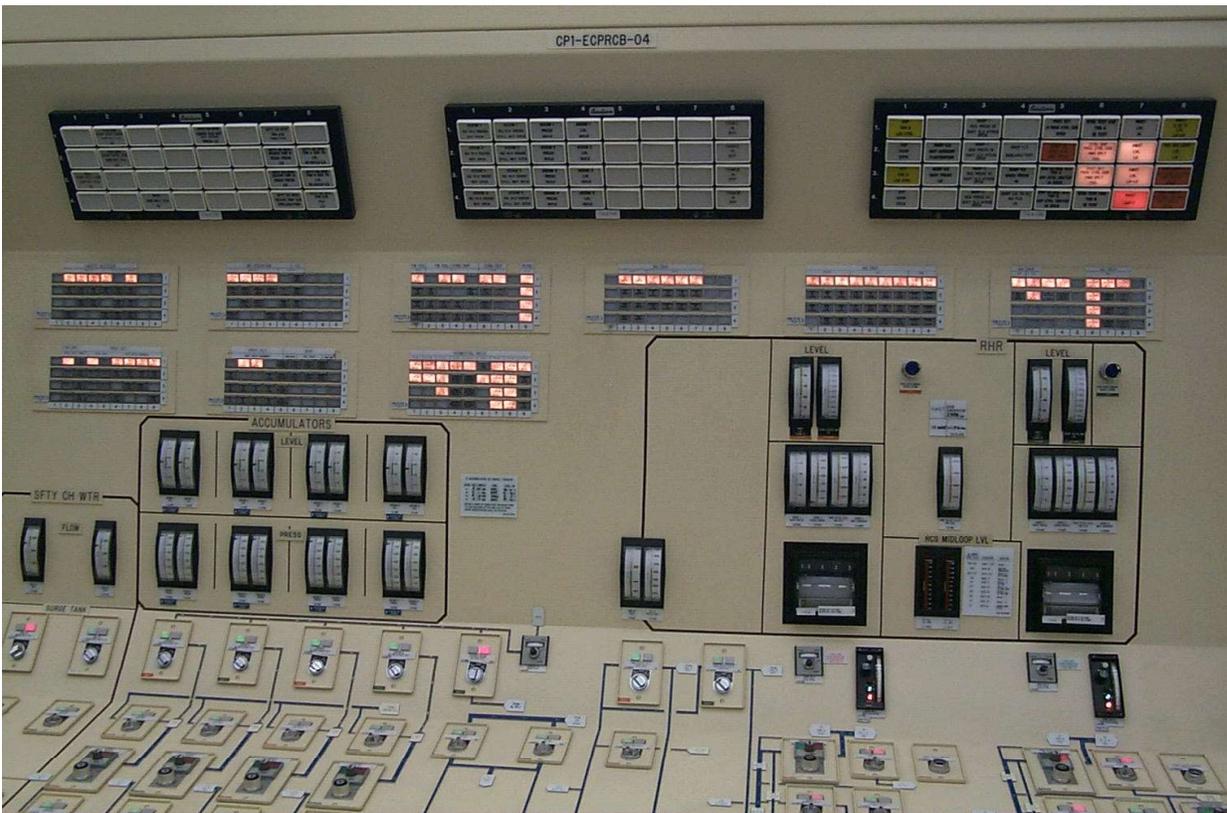
Examination Outline Cross-Reference
(010) (SF3 PZR PCS) PRESSURIZER
PRESSURE CONTROL SYSTEM

Level	SRO
Tier #	2
Group #	1
K/A #	A2.09
Rating	3.8

(010A2.09) Ability to (a) predict the impacts of the following on the (SF3 PZR PCS) PRESSURIZER PRESSURE CONTROL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations (CFR: 43.5): RPS failure

Question 76

- Unit 1 is at full power.
- Pressurizer pressure channel selector switch is selected to 455/458.
- Trip Status Light Box lights change status as indicated in the picture below and the plant remains at power:



1. What is the impact on pressurizer pressure control for this event?
 2. From the choices provided below, which of the following is the most restrictive (shortest completion time) TS LCO for this event?
- A. 1. PCV-456 will not open in AUTO

- 2. Enter 3.8.7 Condition A.1 to Restore Inverter to OPERABLE Status
- B. 1. PCV-456 will not open in AUTO
 - 2. Enter 3.8.9 Condition B.1 to Restore AC vital bus subsystem to OPERABLE status
- C. 1. PCV-455 will not open in AUTO
 - 2. Enter 3.8.9 Condition B.1 to Restore AC vital bus subsystem to OPERABLE status
- D. 1. PCV-455 will not open in AUTO
 - 2. Enter 3.8.7 Condition A.1 to Restore Inverter to OPERABLE Status

Answer: C

Explanation:

Picture is from training material, LO21.SYS.ES4 for a loss of PC1. Loss of PC1 causes PT-455 to fail low, which prevents PCV-455 from operating in AUTO, therefore C1 and D1 are true. Loss of PC3 would cause PT-456 to fail low and consequently PCV_456 would not open in AUTO (not true in this question so A1 and B1 are incorrect). For the part 2 information, Technical Specification 3.8.7 Condition A.1 has an LCO completion time of 24 hours to restore the inverter and 3.8.9 Condition B.1 has a completion time of 2 hours to get the PC1 bus back, So selection C is the correct answer for both parts.

A is wrong because it is not a failure of the inverter nor is it a failure of PC3 (see above explanation).

B is wrong because it is not a failure of PT-456 but the part 2 information is correct, but for loss of PC1, not PC3.

C is correct because part 1 and part 2 are both correct (see above explanation)

D is wrong because part 2 is incorrect. (see above explanation).

Technical References:

LO21.SYS.ES4, Revision 7/23/2020, slide 8, 18, and 44.

TS 3.8.7 and 3.8.9, Amendment 156

References to be provided to applicants during exam: Provided picture in stem

Learning Objective: LO21.SYS.ES4.OB02

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference (064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM	Level Tier # Group # K/A # Rating	SRO 2 1 G2.2.12 4.1
(064) (SF6 EDG) EMERGENCY DIESEL GENERATOR SYSTEM (G2.2.12) Knowledge of surveillance procedures		

Question 77

Unit 2 is heating up to recover from a 5-day shutdown.

Tave is currently being maintained between 440°F and 460°F.

2-02 MD AFWP has been removed from service due to a bearing failure.

2-01 DG Fuel Oil Storage Tank (FOST) Level is 89,000 gallons (56”).

2EA1 is currently powered from XST1.

Unit 1 is at 100% power, but 1-02 DG is inoperable.

It is desired to perform a slow start operability test of 2-01 DG per OPT-214B, Diesel Generator Operability Test, section 8.1.

Per OPT-214B, 2-01 DG **CANNOT** be removed from service for which of the following reasons?

- A. 2-02 MD AFWP is inoperable.
- B. 1-02 DG is inoperable.
- C. 2-01 DG FOST level is below minimum level.
- D. 2EA1 is powered by its non-preferred source.

Answer: A

Explanation: Per OPT-214B, Section 5.1.3, DO NOT remove a DG from service if the plant is in Mode 1-4 AND any ESF equipment on the opposite train that relies on the OPERABLE DG as a source of emergency power is INOPERABLE. MD AFWP 2-02 (powered by DG 2-02) is INOPERABLE due to a bearing failure.

A is correct, see explanation above.

B is wrong but plausible since 1-02 DG is inoperable. OPT-214B Section 5.1.3 discusses not removing a DG from service if it is the ONLY operable DG. However, 2-02 DG remains operable.

C is wrong. The tech spec minimum for associated DG FOST Level is 86,000 gallons (63"). It is plausible because at <90,000 gallons, OPT-214B directs you to order fuel for the tank.

D is wrong because XST1 is currently powered by its preferred source. It is plausible because per OPT-214B, it is required that all busses for both units are required to be aligned to their preferred sources prior to DG start per the surveillance.

Technical References:

OPT-214B – Diesel Generator Operability Test (Rev. 20) pg. 19 (see below)

References to be provided to applicants during exam: None

Learning Objective: DELINEATE the role of surveillances into Operability in accordance with the Technical Specifications. (LO21.ADM.XA5.OB04)

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference
(073) (SF7 PRM) PROCESS RADIATION
MONITORING SYSTEM

Level	SRO
Tier #	2
Group #	1
K/A #	A2.01
Rating	3.1

(073A2.01) Ability to (a) predict the impacts of the following on the (SF7 PRM) PROCESS RADIATION MONITORING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: PRM component failures

Question 78

Unit 1 experienced a large break LOCA followed by fuel damage due to loss of core cooling.

Several radiation monitors have failed due to excessive radiation.

1. Per ODA-407, Operations Department Procedure Use and Adherence Attachment 8A, "ERG Rules of Usage", exceeding what Containment radiation dose rate value can affect instrumentation?
2. At this dose rate value, per EPP-201, Assessment of Emergency Action Levels, Emergency Classification and Plan Activation, the Containment barrier _____ considered to be potentially lost.
 - A. 1. 20 R/HR
2. is
 - B. 1. 20 R/HR
2. is NOT
 - C. 1. 10^5 R/HR
2. is
 - D. 1. 10^5 R/HR
2. is NOT

Answer: C

Explanation:

When containment radiation exceeds $10E^5$ R/hr or 100,000 R/HR, this environment begins to affect instrumentation located inside containment (as specified in the ERG's and ODA-407). As specified in EPP-201, when dose rate in containment has exceeded 1,110 R/HR, containment barrier is potentially lost.

A is wrong. It is plausible because part 1 is the radiation field required to enter the FRZ procedures for high containment radiation and part 2 is correct for any dose rate value over 1,110 R/HR.

B is wrong because both parts are wrong. Part 1 is plausible (see A), and Part 2 would be correct for 20 R/HR.

C is correct, see answer explanation.

D both parts are incorrect, refer to A and B for plausibility.

Technical References:

ODA-407, revision 19, page 40.

ERG rules of usage

LO21.SYS.PA1, revision 4/14/2022, slide 9.

EPP-201, revision 13, page 7.

References to be provided to applicants during exam: None

Learning Objective: LO21.SYS.PA1.OB04

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference
(103) (SF5 CNT) CONTAINMENT SYSTEM

Level
Tier #
Group #
K/A #
Rating

SRO
2
1
G2.3.14
3.8

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures or to analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only)
(CFR: 43.4 / 45.10)

Question 79

The RCS specific activity LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture OR a (1).

The SGTR safety analysis assumes an existing reactor coolant steam generator tube leakage rate of (2).

- A. (1) main steam line break
(2) 1 gpm
- B. (1) main steam line break
(2) 10 gpm
- C. (1) loss of coolant accident
(2) 1 gpm
- D. (1) loss of coolant accident
(2) 10 gpm

Answer: A

Explanation:

The LCO limits on the specific activity of the reactor coolant ensures that the resulting offsite and Control Room doses meet the appropriate Standard Review Plan acceptance criteria following a SGTR or a MSLB accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at, or more conservative than, the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm.

A is correct (see above explanation).

B is wrong. Part 1 is correct. Part 2 is incorrect but plausible because it is 10 times the correct leak rate, as well as a leakage value per LCO 3.4.16 for identified leakage.

C is wrong because Part 1 is wrong but plausible since RCS activity may be elevated during a loss of coolant accident. Part 2 correct (see explanation).

D is wrong because Part 1 is wrong but plausible since RCS activity may be elevated during a loss of coolant accident. Part 2 is incorrect but plausible because it is 10 times the correct leak rate, as well as a leakage value per LCO 3.4.16 for identified leakage.

Technical References:

TSB B 3.4-82, revision 83.

References to be provided to applicants during exam: None.

Learning Objective: LO21.MCO.MIB.OB02

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.4	

Examination Outline Cross-Reference	Level	SRO
(028) (SF5 HRPS) HYDROGEN	Tier #	2
RECOMBINER AND PURGE CONTROL	Group #	2
SYSTEM	K/A #	A2.02
	Rating	3.4

(028A2.02) Ability to (a) predict the impacts of the following on the (SF5 HRPS) HYDROGEN RECOMBINER AND PURGE CONTROL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: LOCA with significant hydrogen production

Question 80

Unit 1 experienced a loss of coolant accident 18 hours ago.

Containment Pressure is 3 psig and slowly rising.

Containment Temperature is 114F and slowly rising

1-ALB-3A, 3.7 CNTMT H₂ TRN A/B CONC HI is in alarm and ALM-0031A, Alarm Procedure 1-ALB-3A has been entered.

Hydrogen monitor readings have been assessed per ALM-0031A:

- 1-AI-5506A/B CNTMT H₂ CONC: 3.5%
- 1-AI-5506C/D CNTMT H₂ CONC: 3.7%

Based on the current hydrogen levels in containment, the containment atmosphere _____ (1) _____ considered combustible.

To lower hydrogen concentration below levels at which equipment damage could occur due to a flame propagation or detonation event, the Unit Supervisor should start the Hydrogen Purge Supply and Exhaust system per _____ (2) _____.

- (1) is
(2) EOP-0.0A, Reactor Trip or Safety Injection
- (1) is NOT
(2) EOP-0.0A, Reactor Trip or Safety Injection
- (1) is
(2) SOP-205, Hydrogen Purge Supply and Exhaust System
- (1) is NOT
(2) SOP-205, Hydrogen Purge Supply and Exhaust System

Answer: D

Explanation:

Based on the current hydrogen levels in containment, per the bases of EOP-1.0A, Loss of Reactor or Secondary Coolant, the containment atmosphere is NOT considered combustible. This occurs at 4%. Also, since containment pressure is < 5 psig and containment temperature is between 50-160F, SOP-205 can be used to reduce hydrogen concentration.

A first part is incorrect but plausible because H2 is elevated. Second part is incorrect but plausible because EOP-0.0A would be used to reduce containment pressure using the Containment Spray system if certain conditions were met per the ALM.

B first part is correct, second part is incorrect. See A for plausibility.
C first part is incorrect, see A for plausibility. Second part is correct.
D is correct. See above explanation.

Technical References:

- ALM-0031A – Alarm Procedure 1-ALB-3A (Rev. 8), pg. 77 (see below)
- SOP-205 – Hydrogen Purge Supply and Exhaust System (Rev. 10)
- EOP-0.0A – Reactor Trip or Safety Injection (Rev. 9)
- EOP-1.0A – Loss of Reactor or Secondary Coolant (Rev. 9)

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E1A.OB04

Question Source:	Bank #	
Original bank question attached	Modified Bank #	X
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43 (b) (5)	

ORIGINAL BANK QUESTION

Examination Outline Cross-Reference	Level	SRO
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control	Tier #	2
	Group #	2
	K/A #	A2.03
Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: A2.03 The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment	Rating	4.0
	QREV	6

Question 92

Given the following conditions:

- Unit 1 experienced a Large Break Loss of Coolant Accident 18 hours ago
- Containment Pressure is now 6 psig and rising
- Containment Hydrogen alarms “CNTMT H2 TRN A/B CONC HI” are both in Alarm in the control room, and alarm procedure ALM-0032 was entered
- The confirmed Hydrogen Microprocessor readings for both channels are:
 - Train A Hydrogen is 3.5%
 - Train B Hydrogen is 3.7%
- TSC has been notified

To lower hydrogen concentration below levels at which equipment damage could occur due to a flame propagation or detonation event, the US, with TSC concurrence, should start the (1) _____ system per (2) _____.

- E. (1) Hydrogen Purge Supply and Exhaust
(2) EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT
- F. (1) Hydrogen Purge Supply and Exhaust
(2) EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION
- G. (1) Containment Spray
(2) EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT
- H. (1) Containment Spray
(2) EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION

Answer: D

Explanation:

A is wrong because containment pressure is too high to use the HPSE system
B is wrong because containment pressure is too high to use the HPSE system.

C is wrong because it does not contain the steps to reinitiate CS
D is correct because it (EOP-0.0A) is specifically listed in the ALM-0031A for Hi H2 CONC as the procedure to use with TSC direction, which was given in the stem.

Technical References:

ALM-0031A, rev 8, page 77, SOP-205, rev 10, EOP-0.0A, rev 9, EOP-1.0A, rev 9.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E1A.OB04.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)5	

Examination Outline Cross-Reference (014) (SF1 RPI) ROD POSITION INDICATION SYSTEM	Level Tier # Group # K/A # Rating	SRO 2 2 G2.2.7 3.6
(G2.2.7) Knowledge of the process for conducting infrequently performed tests or evolutions (CFR: 41.10 / 43.3 / 45.13)		

Question 81

Per STA-122, Infrequently Performed Tests or Evolutions, which of the following activities is considered an infrequently performed test or evolution (IPTE)?

- A. Removal of the Reactor Vessel upper internals
- B. Control rod drive surveillance testing
- C. Reactor protection system surveillance testing
- D. Backfeeding and restoration of Main Transformers

Answer: A

Explanation:

A is correct because Removal of the Reactor Vessel upper internals is considered an IPTE per STA-122, page 20 of the list of included IPTE (Attachment 8.A). This ties to the KA because CPNPP has specific OE on inadvertently leaving a DRPI assembly attached when removing the upper internals.

B is wrong because this is an example of what is NOT an infrequently performed test in Attachment 8.A of procedure STA-122.

C is wrong because this is an example of what is NOT an infrequently performed test in Attachment 8.A of procedure STA-122.

D is wrong because this is an example of what is NOT an infrequently performed test in Attachment 8.A of procedure STA-122.

Technical References:

STA-122, Attachment 8.A, Revision 3, pages 1-4.

References to be provided to applicants during exam: None.

Learning Objective: Document learning objective if possible.

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental
Comprehensive/Analysis

3

10CFR Part 55 Content:

55.43.3

Examination Outline Cross-Reference	Level	SRO
027 Containment Iodine Removal System	Tier #	2
	Group #	2
Knowledge of conditions and limitations in the facility license	K/A #	2.2.38
	Rating	4.5

Question 82

In accordance with Technical Specification Bases and the FSAR, Technical Specification 3.6.7, Spray Additive System surveillance requirement, SR 3.6.7.1 ensures:

An equilibrium containment recirculation sump ___(1)___ pH using NaOH.

This pH band minimizes the evolution of iodine as well as ___(2)___.

- A. 1. Alkaline
2. the occurrence of general corrosion and hydrogen production
- B. 1. Alkaline
2. the occurrence of chloride and caustic stress corrosion
- C. 1. Acidic
2. the occurrence of general corrosion and hydrogen production
- D. 1. Acidic
2. the occurrence of chloride and caustic stress corrosion

Answer: **B**

Explanation:

Per the bases of LCO 3.6.7, SR 3.6.7.1 ensures an equilibrium containment recirculation sump alkaline pH (>7.1) using NaOH and this minimum band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion.

- A. First part is correct. Second part is incorrect but plausible since this is the basis for the maximum containment sump pH for LCO 3.5.4.
- B. Is correct, see explanation above.
- C. First part is incorrect but plausible if the candidate does not understand the pH band and whether 7.1 is considered alkaline or acidic. Second part is incorrect, see A for plausibility.
- D. First part is incorrect, see C for plausibility. Second part is correct, see explanation above.

Technical References:

Tech Spec Bases page 412, Rev. 84

CPNPP/FSAR Section 6.5.2.1 Design Bases page 2675, Amendment 104.

References to be provided to applicants during exam: None.**Learning Objective:** LO21.MCO.MIF.OB01

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.43(b).1	

Additional Material:

T.S. Bases:

SR 3.6.7.1 This surveillance verifies that the available buffering agent is sufficient to ensure that the equilibrium containment sump pH is greater than or equal to 7.1. Details of the bases for performance test requirements that ensure sump pH requirements are met are included in the bases of the Technical Requirements Manual.

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA). Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. When NaOH is added to the spray, a pH value of greater than or equal to 7.1 of the solution recirculated from the containment sump is ensured. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

Examination Outline Cross-Reference
(008) (SF8 CCW) COMPONENT COOLING
WATER SYSTEM

Level	SRO
Tier #	2
Group #	1
K/A #	G2.2.3
Rating	3.8

(008) (SF8 CCW) COMPONENT COOLING
WATER SYSTEM (G2.2.3) (Multi-unit license)
Knowledge of the design, procedural, and/or
operational differences between units
(CFR:43.5)

Question 83

IPO-005B, Plant Shutdown from Hot Standby to Cold Shutdown is in progress.

Unit 2 Tave is 330°F.

All RCPs have been stopped.

All SGs have been drained to 5% narrow range level.

2-01 and 2-02 CCW Pumps are in service.

CCW Surge Tank Level begins slowly lowering and the following annunciators alarm:

- 2-ALB-3B, Window 2.4 - CCW SRG TK TRN A LVL HI-HI/LO
- 2-ALB-3B, Window 1.3 - CCW SRG TK TRN A/B LVL LO-LO

Per ABN-502, Component Cooling Water System Malfunctions, when CCW Surge Tank falls below a MAXIMUM level of _____ (1) _____ the affected safeguards loop will isolate.

After the leak is isolated and the affected train CCW pump is placed in PULL OUT, _____ (2) _____ Limiting Condition(s) for Operation must be entered.

- A. 1. 33%
- 2. ONLY 3.7.7, Component Cooling Water System

- B. 1. 57%
- 2. ONLY 3.7.7, Component Cooling Water System

- C. 1. 33%
- 2. 3.7.7, Component Cooling Water System, and
3.4.6, RCS Loops – MODE 4

- D. 1. 57%
- 2. 3.7.7, Component Cooling Water System, and
3.4.6, RCS Loops – MODE 4

Answer: C

Explanation:

- A. Is wrong. Part 1 is correct as described in 'C' below. Part 2 is wrong but plausible as the applicant must determine that both RHR Loops are the required OPERABLE Loops and therefore in accordance with the NOTE in TS 3.7.7 Condition A, TS 3.4.6 must also be entered.
- B. Is wrong. Part 1 is incorrect but plausible as 57% is the Unit 1 number in accordance with ABN-502. Part 2 is incorrect but plausible as described in 'A' above.
- C. Is Correct. The tank is common above 37% on Unit 2 and the leak cannot be identified using this methodology until level reaches 33% as specified in ABN-502.. Part 2 is correct in that the NOTE in Condition A of TS 3.7.7 requires that the applicable Conditions of LCO 3.4.6 be entered. As all RCPs have been secured and the Steam Generator Levels are below the 10% required by SR 3.4.6.2, the LCO must be entered as only a single RCS Loop is OPERABLE, one RHR loop. With the loss of CCW surge tank level one CCW train is inoperable. As CCW is a support system for RHR, the RHR train becomes inoperable. Per normal TS rules of usage this would not require entry into the RCS loops Tech Spec. However, in this case it is directed from the Condition A of TS 3.7.7 which is entered. This will force the loss of one of the operable loops per TS 3.4.6 since all RCPs are off and thus a TS 3.4.6 entry is also required.
- D. Is wrong. Part 1 is incorrect but plausible as 57% is the Unit 1 number in accordance with ABN-502. Part 2 is correct as described in 'C' above.

Technical References:

ABN-502, Revision 12, page 14
 TS 3.7.7 and 3.4.6, Amendment 156

References to be provided to applicants during exam: None

Learning Objective: LO21ABN501OB106)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	2016 NRC Q87
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference (000008) (APE 8) Pressurizer Vapor Space Accident	Level	SRO
	Tier #	1
	Group #	1
	K/A #	A2.27
(000008A2.27) (APE 8) Pressurizer Vapor Space Accident	Rating	3.6
Ability to determine and/or interpret the following as they apply to PZR vapor space accident: PZR pressure and/or level due to sensing line leakage		

Question 84

Unit 1 power reduction was in progress to allow operators to isolate a large break on the LOWER sensing line of Pressurizer level transmitter 1-LT-0459.

- The affected channel has been de-selected from control.

During power reduction, a secondary transient caused a Pressurizer PORV to open and it stuck partially open.

- The affected PORV Block Valve could not be closed so a manual Reactor Trip and Safety Injection actuation were performed.

The crew is currently performing EOP-1.0A, Loss of Reactor or Secondary Coolant, Step 6, Check If ECCS Flow Should Be Reduced.

Current plant conditions:

- Containment Pressure 0.6 psig
- Pressurizer Level (1-LI-460 & 461) 85%
- RCS Subcooling 35°F
- All SG Narrow Range (NR) Levels 60% and stable
- RCS Pressure 1605 psig and rising

Based on these conditions:

1-LT-0459 _____(1)_____ and the Unit Supervisor should _____(2)_____.

1. is indicating **higher** than 1-LI-460/461
2. transition to EOS-1.1A, Safety Injection Termination
1. is indicating **lower** than 1-LI-460/461
2. stay in EOP-1.0A, Loss of Reactor or Secondary Coolant
1. is indicating **higher** than 1-LI-460/461
2. stay in EOP-1.0A, Loss of Reactor or Secondary Coolant
1. is indicating **lower** than 1-LI-460/461
2. transition to EOS-1.1A, Safety Injection Termination

Answer: D

Explanation:

With the lower sensing line leak, LT-459 indicates lower than actual level. Note that this is not the same as a reference line leak because that is considered the UPPER tap. Actual level will be off-scale high because of the stuck open PORV (vapor space event). With actual PZR level being off scale high all criteria in step 6 support a transition to EOS-1.1A, SI Termination.

The only combination that is correct is D.

A is wrong because part 1 is incorrect. However, part 2 is correct for procedure selection.

B is wrong because part 2 is incorrect. Correct procedure is to transition to EOS-1.1A

C is wrong because part 1 and part 2 are both incorrect (see above explanation)

D is correct because a transition to EOS-1.1A is required and the indicated level is lower than actual level.

Technical References:

EOP-1A, Revision 9, page 8.

EOS-1.1A, Revision 9.

References to be provided to applicants during exam: None

Learning Objective: LO21.ERG.E11.OB103

Question Source:

(note changes; attach parent)

Bank #

Modified Bank #

New

X

Question History:

Last NRC Exam

No

Question Cognitive Level:

Memory/Fundamental

Comprehensive/Analysis

3

10CFR Part 55 Content:

55.43.5

Examination Outline Cross-Reference (000038) (EPE 38) Steam Generator Tube Rupture	Level Tier # Group # K/A # Rating	SRO 1 1 G2.4.18 4.0
(000038) (EPE 38) Steam Generator Tube Rupture (G2.4.18) Knowledge of the specific bases for emergency and abnormal operating procedures		

Question 85

Unit 2 is performing EOP-3.0B, Steam Generator Tube Rupture due to a rupture on Steam Generator 2-04.

The crew is at the step to verify Ruptured Steam Generator Pressure prior to commencing the max-rate cooldown.

Steam Generator 2-04 pressure is 850 psig and stable.

1. Per EOP-3.0B, what is the basis for this step?
2. Based on the plant conditions above, the Unit Supervisor will...
 - A. 1. Ensures that the SI accumulators will not inhibit the subsequent cooldown.
2. Stay in EOP-3.0B and commence the max-rate cooldown.
 - B. 1. Ensures that the SI accumulators will not inhibit the subsequent cooldown.
2. Transition to ECA-3.1B, SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired.
 - C. 1. Ensures that the subsequent cooldown will not result in an Orange condition on the Integrity status tree.
2. Stay in EOP-3.0B and commence the max-rate cooldown.
 - D. 1. Ensures that the subsequent cooldown will not result in an Orange condition on the Integrity status tree.
2. Transition to ECA-3.1B, SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired.

Answer: C

Explanation:

If the pressure is less than 550 psig, this cooldown could result in an ORANGE priority on the Integrity Status Tree. To avoid this condition the operator is transferred to ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED, which limits the cooldown rate to less than 100°F/hr.

A first part is wrong but plausible since RCS pressure is related to the SI accumulators for preventing nitrogen injection which might inhibit natural circulation, but the pressure is higher (800 psig). Second part is correct, see above explanation.

B first part is wrong, see A. Second part is incorrect but plausible since this transition would be required if pressure was less than 550 psig.

C is correct, see explanation above.

D first part is correct, see explanation above. Second part is incorrect but plausible, see B.

Technical References:

EOP-3B, Revision 9, page 7, 63.

LO21.ERG.E3A.LP, Revision 00.0001, page 4

References to be provided to applicants during exam: None

Learning Objective: LO21.ERG.E3A.OB06

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference (000055) (EPE 55) Station Blackout	Level	SRO
	Tier #	1
	Group #	1
(000055EA2.04) Ability to determine and/or interpret the following as they apply to (EPE 55) Station Blackout: Instruments and controls operable with only DC battery power available	K/A #	A2.04
	Rating	3.9

Question 86

All offsite power was lost five hours ago.

Unit 1 was unable to restore power to either safeguard bus.

ECA-0.0A, Loss of All AC Power is in progress.

DC bus voltage has reached 110 vdc and Plant Staff has determined additional DC load shedding should be performed.

1. When **1EC1** is de-energized, which of the following loads is lost?
2. What procedure has guidance for restoring or compensating for a loss of **1EC1**?
 - A. 1. Train A EDG emergency start relay
2. ECA-0.0, Attachment 2.C, DC Load Shed When DC Voltage Less Than 110 V
 - B. 1. 1-TR-433A, HL 3 WR TEMP
2. ECA-0.0, Attachment 2.C, DC Load Shed When DC Voltage Less Than 110 V
 - C. 1. Train A EDG emergency start relay
2. ABN-603, Loss of Protection or Instrument Bus
 - D. 1. 1-TR-433A, HL 3 WR TEMP
2. ABN-603, Loss of Protection or Instrument Bus

Answer: C

Explanation:

As described in ABN-603, page 123, The emergency start relay for train A EDG is lost, which affects its start circuit. It is lost on loss of 1EC1. WR hot leg temperature instrument would be a desired instrument during cooldown but this instrument is powered from 1EC2 and would be unaffected. If you did not know that this, then it is plausible since a hot leg instrument is needed for the cooldown.

A is wrong because part 2 is incorrect. Part 1 is correct, the emergency start relay is lost as described in ABN-603. EDG train A can still be started manually in emergency mode and this information is provided in the ABN-603, not in ECA-0.0.

B is wrong because parts 1 and 2 are wrong. WR hot leg TR-433A is not lost on 1EC1 loss, but on 1EC2 loss. Attachment 2.C of ECA-0.0 is plausible because it will need to be

performed for additional load shedding but does not have the required guidance for restoring the Train A diesel and the associated 1EC1 bus as required in part 1 of the question.

C is correct because part 1 and part 2 are both correct (see above explanation)

D is wrong because part 1 is incorrect. (see above explanation).

Technical References:

ABN-603, revision 8, pages 16 and 26.

ECA-0.0, revision 9, page 123.

LO21.ERG.C00, revision 4/12/2021

References to be provided to applicants during exam: None

Learning Objective: LO21.ERG.C00.OB06

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	RO
0056 Loss of Offsite Power	Tier #	1
	Group #	1
Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines, such as operating procedures, abnormal operating procedures, or severe accident management guidelines	K/A #	G2.4.16
	Rating	4.4

Question 87

Unit 1 has been operating at 100% power for three months.

During the previous shift, a tornado caused extensive damage to the switchyard resulting in a loss of all offsite power.

1-02 DG was tagged out of service prior to the event and remains unavailable.

1-01 DG tripped on overspeed during startup and has not been recovered.

Unit 1 Core Exit Thermocouples now read 1100°F and stable.

Unit 2 is stable in Mode 3 with both diesel generators aligned to the buses.

Following the trip of 1-01 DG, which procedure(s) is (are) required to be performed?

- (1) ECA-0.0A, Loss of All AC Power
- (2) SAG-1.0, Control Room Severe Accident Initial Response
- (3) ABN-601, Response to a 138/345 KV System Malfunction
- (4) ECA-0.2A, Loss of All AC Power Recovery with SI Required

- A. (1) Only
- B. (2) Only
- C. (1) and (3) Concurrently
- D. (3) and (4) Concurrently

Answer: C

Explanation:

A is wrong because the Applicability Note in ABN-601 states that actions of ECA-0.0A are required to be run in parallel with ABN-601 when BOTH Safeguards buses are de-energized or know the transition flow chart in ECA 0.0A; is plausible because the unit is in a loss of all AC power.

B is wrong because SACRG applicability/entry condition is CETs greater than 1200°F; It is plausible because 1100°F is close to the applicability temperature.

C is correct because the Applicability Note in ABN-601 states that actions of ECA-0.0A are required to be run in parallel with ABN-601 when BOTH Safeguards buses are de-energized, and because the transition flow chart of ECA-.0A directs continuing that procedure while enacting ABN-601.

D is wrong because you have to enter ECA-0.0A and the ABN-601. ECA-0.2A is plausible since ECA-0.0A actions to depressurize SGs will result in safety injection. However, a transition to ECA-0.2A, if required, is not directed until AC power is recovered since ECA-0.2A recovery actions require AC power to be available.

Technical References:

ABN-601, "Response to a 138/345 KV System Malfunction," rev 18, page 2

References to be provided to applicants during exam: None.

Learning Objective: LOT Lesson ABN-601/602 (Code LO21ABN601) rev 2

1. Terminal: a. From memory, ANALYZE the response to events described in accordance with ABN-601, Response to 138/345 kV System Malfunction and ABN-602, Response to 6900/480 V System Malfunction.

2. Enabling: The following objectives are evaluated from memory:
 a. ANALYZE the response to Loss of All Offsite AC Power in accordance with ABN-601, Response to 138/345 KV System Malfunction. (LO21.ABN.601.OB01)

	Bank #	
Question Source: (note changes; attach parent)	Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41(b)... 55.43	

SRO level because the applicant must assess plant conditions and choose which procedures to enact based on a note in the procedures and the transition flow chart of ECA-0.0A.

Parent Question from 2014 CPNPP NRC Exam:

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 03/03/2014	Tier	_____	<u>3</u>
Change: 2	Category	_____	<u>4</u>
	K/A	<u>G 2.4.16</u>	
Level of Difficulty: 2	Importance Rating	_____	<u>4.4</u>

Emergency Procedures/Plan: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as operating procedures, abnormal operating procedures, and severe accident management guidelines
Proposed Question: 100

Given the following conditions:

- A Station Blackout has been in progress for several hours.
- Unit 1 is responding to the Station Blackout in accordance with ECA-0.0A, Loss of All AC Power.
- While checking Core Exit Thermocouple (CET) temperatures they are found to be 1220°F and rising.

Which of the following is the required action?

The Unit Supervisor should...

- ...remain in ECA-0.0A, Loss of All AC Power and ensure actions required to restore power to any AC Safeguards bus are in progress.
- ...transition to SACRG-1, Severe Accident Control Room Guideline Initial Response and verify a GENERAL EMERGENCY has been declared.
- ...enter ABN-601, Response to a 138/345 KV System Malfunction concurrent with ECA-0.0A and restore power to any AC Safeguards bus.
- ...transition to ECA-0.2A, Loss of All AC Power Recovery With SI Required and manually align SI valves in preparation for power restoration.

Proposed Answer: B

Explanation:

- A. Incorrect. Plausible because this action would continuously be performed, however, once CET temperatures exceed 1200°F entry into SACRG-1 is required.
- B. Correct. IAW Step 24 of ECA-0.0A when CET temperatures exceed 1200°F entry into SACRG-1 is required.
- C. Incorrect. Plausible because this action would be in progress, however, once CET temperatures exceed 1200°F entry into SACRG-1 is required.
- D. Incorrect. Plausible because entry into ECA-0.2A would be required because a SI would be needed with CET temperatures greater than 1200°F, however, at least one safeguards bus is needed to enter ECA-0.2A.

Technical Reference(s) ECA-0.0A, Flowchart Attached w/ Revision: See
ECA-0.0A, Step 24 Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: IDENTIFY the proper transitions out of ECA-0.0, Loss of All AC Power.

Question Source: Bank ILOT8323
 Modified Bank _____ (Note changes or attach parent)
 New _____

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge X
 Comprehension or Analysis _____

10 CFR Part 55 Content: 55.41 _____
 55.43 5

Examination Outline Cross-Reference (000057) (APE 57) Loss of Vital AC Instrument Bus (G2.2.25) Knowledge of the bases in TS for limiting conditions for operation and safety limits (SRO Only)	Level Tier # Group # K/A # Rating	SRO 1 1 G2.2.25 4.2
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Question 88

Unit 1 is at 100% power.

Instrument distribution panel 1EC1 de-energized due to inverter failure.

After 1EC1 is re-energized from its alternate power supply, the Unit Supervisor directs placing the swing inverter in service to supply 1EC1.

1. What procedure contains steps to place the swing inverter in service and transfer 1EC1 to the swing inverter output?
2. Per Tech Spec 3.8.7 bases, what additional actions must be completed when 1EC1 is energized from its alternate power supply?

Note: SFDP is Safety Function Determination Program

- A. 1. ABN-603, Loss of Protection or Instrument Bus
2. Perform a risk assessment for 1-02 Emergency Diesel Generator.
- B. 1. SOP-607A, 118 VAC Distribution System and Inverters
2. Perform a risk assessment for 1-02 Emergency Diesel Generator.
- C. 1. ABN-603, Loss of Protection or Instrument Bus
2. Evaluate impacted equipment under SFDP of Technical Specification 5.5.15.
- D. 1. SOP-607A, 118 VAC Distribution System and Inverters
2. Evaluate impacted equipment under SFDP of Technical Specification 5.5.15.

Answer: D

Explanation:

Per TS bases for 3.8.7, when this inverter is swapped to its alternate power supply, the associated EDG and Blackout Sequencer must be declared inoperable and a SFDP assessment per 5.5.15 is required. In TS section 3 it states that a risk assessment for the affected inoperable components would be required but the inoperable component in this case is 1-01 EDG, not 1-02 EDG. SOP-607A has the detailed steps to place a swing inverter in service and restore realign the panel to the inverter.

A is wrong because both part 1 and part 2 are wrong but plausible because you do have to enter ABN-603 for loss of the safety bus 1EC1 and it contains instructions for aligning the panel to an alternate power supply. However, it does not contain the detailed steps to complete the swap to a swing inverter.

B is wrong because part 2 is incorrect (see above). Part 1 is correct
C is wrong because part 1 is incorrect. Part 2 is correct.
D is correct because both parts 1 and 2 are correct (see explanation above).

Technical References:

TS Bases, B3.8.7, page B3.8.69, Revision 82
ABN-603, Revision 8
SOP-607A, page 55, Revision 25

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.603.OB02.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.2	

Examination Outline Cross-Reference	Level	SRO
077 Generator Voltage and Electric Grid Disturbances	Tier #	1
	Group #	1
	K/A #	AA2.05
Ability to determine and/or interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Status of grid	Rating	3.6

Question 89

Unit 1 is in Mode 1 at 100% power.

Excessive ambient temperatures over the past several weeks have resulted in grid instability.

When the safeguard bus frequency indicators drop below _____ (1) _____ for greater than 2 minutes, the Unit Supervisor must _____ (2) _____.

- A. (1) 58.5 Hz
(2) declare both offsite transmission networks inoperable
- B. (1) 59.4 Hz
(2) declare both offsite transmission networks inoperable
- C. (1) 58.5 Hz
(2) declare Train A and Train B AC vital bus electrical power distribution subsystems inoperable
- D. (1) 59.4 Hz
(2) declare Train A and Train B AC vital bus electrical power distribution subsystems inoperable

Answer: B

Explanation:

ABN-601 section 9 has 4 continuous action steps (steps 1 thru 4) and step 4 is to “Check Frequency – GREATER THAN 59.4 Hz (1782 rpm).” When frequency goes below 59.4 Hz for greater than two minutes, the procedure directs placing the DGs on the safeguards buses and divorcing the safeguards buses from offsite power. Declaring both offsite transmission networks inoperable is correct for part 2 because grid frequency disturbance affects both offsite transmission networks.

- A. is wrong because Part 1 is wrong. It is plausible since automatic grid frequency relays provide load automatic load shedding between 59.3 Hz and 58.5 Hz. Part 2 is correct.
- B. is correct (see explanation above).
- C. is wrong because parts 1 and 2 are wrong. It is plausible since automatic grid frequency relays provide load automatic load shedding between 59.3 Hz and 58.5 Hz. Second part is plausible since the buses are potentially affected by loss of power; however, Technical Specifications define the AC electrical power subsystem for each train as consisting of a

primary Engineered Safety Feature 6.9 kV bus and secondary load centers and 480 and 120 V buses. Power sources to the buses are controlled per other Technical Specifications.

D. is wrong because part 2 is wrong. It is plausible as discussed in the explanation above.

Technical References:

ABN-601, Revision 18, Page 130-134

TS, Amendment 150, Pages 245, 262

References to be provided to applicants during exam: None

Learning Objective: LO21.ABN.601.OB01

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)5	

Examination Outline Cross-Reference (000024) (APE 24) Emergency Boration	Level	SRO
	Tier #	1
	Group #	2
(000024AA2.04) Ability to determine and/or interpret the following as they apply to (APE 24) EMERGENCY Boration: Availability of the RWST	K/A #	AA2.04
	Rating	3.7

Question 90

While Unit 2 is operating at 100% Reactor Power, the following sequence of events occurs:

- 1021: Reactor Operator trips the reactor due to feedwater flow control valve failure, but two control rods remain stuck fully withdrawn.
- 1022: A 86-1 lockout occurs on 2EA2, Train B Safeguard Bus.
- 1023: Reactor Operator dispatches an operator to manually open 2CS-8439, U2 CVCS CHRG PMP EMER BORATE MAN VLV per ABN-107, Emergency Boration due to loss of power to 1/2-8104, EMER BORATE VLV.
- 1024: Pressurizer pressure begins lowering uncontrollably and the Reactor Operator manually initiates Safety Injection.
- 1024: Reactor Operator notes initial RWST level is 96%.
- 1034: Reactor Operator verifies that based on initial RWST volume, ECCS injected at least 19,823 gallons, which is ECCS equivalent of 3600 gallons of Boric Acid Storage Tank boron per TDM-201B, CVCS Calculations/Blended Flow.
- 1037: Equipment operator manually opens 2CS-8439.

Which of the following LCO action statements is applicable and NOT met at Time 1034?

- A. Two boration injection subsystems inoperable OR required actions and associated completion times not met (Modes 1, 2, 3, and 4):

Initiate action to restore at least one boration injection subsystem to operable status AND initiate engineering evaluation to identify compensatory actions to be completed in a timely manner, immediately.

- B. Required ECCS residual heat removal (RHR) subsystem inoperable (Mode 4):

Initiate action to restore required ECCS RHR subsystem to OPERABLE status, immediately.

- C. RWST inoperable for reasons other than boron concentration or temperature (Modes 1, 2, 3, and 4):

Restore RWST to operable status within 1 hour.

- D. SDM shall be within the limits provided in the COLR (Mode 2 with $k_{\text{eff}} < 1.0$, Modes 3, 4, and 5):

Initiate boration to restore SDM to within limit within 15 minutes.

Answer: C

Explanation: In the scenario above, the RWST would be considered inoperable due to a low water level and Train B CCP, 2-02 Boric Acid Transfer Pump, and 1/2-8104 for the required boration injection system would be considered inoperable due to loss of power. The TS LCO for RWST level (3.5.4-B.1) is based on availability for both ECCS and boration functions and is entered anytime indicated level is below 95%. The ECCS injection volume given in the stem of 19823 gals represent almost 4.0% of an RWST level change based on a 95% level equivalency of 500772 gals. This amount would bring the RWST level below the required LCO level with initial level at 96%.

A is wrong because only one boration injection system is inoperable due to the inoperable CCP. Two suction flowpaths (112D and Train A) remain operable even with 1/2-8104 de-energized since the Train A flowpath can be manually aligned. Plausible because the inability to operate the emergency boration system using the preferred emergency borate valve 1/2-8104 or 112E, due to loss of the B Safeguards 6.9 KV bus could be thought to also make that subsystem inoperable. However, according to the TRM 13.1.31 basis, manual valve operation is allowable for the system to remain operable, as long as the operation has assurance of being completed within 15 minutes.

B is wrong because the plant would not be in Mode 4 by time 1034, given the plant conditions present. Plausible because RHR train B components would be inoperable because of the lockout of 2EA2.

C is correct, see Explanation section above.

D is wrong because boration has already been initiated with the ECCS injection. Plausible because if the ECCS injection is not considered to count as a source of emergency boration then boration would not commence until the Manual Emergency Boration valve is opened at time 1037.

Technical References:

ABN-107 – Emergency Boration (Rev. 9)

TDM-201B – CVCS Calculations / Blended Flow (Rev. 4)

Technical Specification for Comanche Peak Nuclear Power Plant Units 1 and 2

TS 3.1.4, TS 3.1.1, TS 3.5.4

Technical Requirements Manual for Comanche Peak Steam Electric Station Units 1 and 2

TR 13.1.31

References to be provided to applicants during exam: none

Learning Objective: (SRO Only) Given Reactor Makeup system operability status or parameter indications, various plant conditions, and a copy of regulatory requirements (Technical Specifications, TRM, etc.), ASSESS any LCO entries, applicable conditions, and required actions including completion time in accordance with the associated regulatory requirements and their bases. (LO21.SYS.CS2.OB07)

Question Source:

Bank #

	Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference (W E15) Containment Flooding / 5	Level	SRO
	Tier #	1
	Group #	2
(W E15) Containment Flooding (G2.4.47) EMERGENCY PROCEDURES/PLAN: Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential)	K/A #	G2.4.47
	Rating	4.2

Question 91

A large break LOCA has occurred on Unit 1.

Containment spray automatically actuated and was verified in EOP-0.0A, Reactor Trip or Safety Injection.

The crew is currently performing steps in EOP-1.0A, Loss of Reactor or Secondary Coolant.

The following conditions exist when the STA completes a review of Status Trees:

- Pressurizer level 0%
- Containment recirc sump water level 818 feet
- Containment pressure 12 psig and decreasing
- Containment area radiation monitors 25 R/hr
- RVLIS All lit except 49 in above flange light DARK
- CETs 500°F

What is the **highest** priority procedure to be entered for the above conditions?

- A. FRI-0.2A, Response to Low Pressurizer Level
- B. FRZ-0.2A, Response to Containment Flooding
- C. FRZ-0.1A, Response to High Containment Pressure
- D. FRZ-0.3A, Response to High Containment Radiation Level

Answer: B

Explanation:

With 818 feet of water in containment, this is an orange condition for containment flooding. With containment pressure less than 18 psig, this does not require entry to FRZ-0.1A. Containment radiation greater than 20R/Hr would require entry into FRZ-0.3A but it would be yellow and therefore a lower priority than the orange for flooding. With pressurizer level below 17 inches this is a yellow path for FRI-0.2 Low Pressurizer Level but it is not above an orange path priority for flooding. This makes B correct.

A is wrong (see above explanation)

B is correct (see above explanation)
C is wrong (see above explanation).
D is wrong (see above explanation).

Technical References:

LO21.ERG.FZ2, Revision 2/24/2021
FRZ-0.2A, Revision 9, page 5.

References to be provided to applicants during exam: None

Learning Objective: LO21.ERG.FZ2.OBO3

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
069 W E14 Loss of Containment Integrity	Tier #	1
	Group #	2
Loss of Containment Integrity RADIATION	K/A #	G2.3.11
CONTROL: Ability to control radiation releases	Rating	4.3

Question 92

Unit 1 is in Mode 1.

1-HS-5157, CNTMT SMP DRN ISOL VLV (ORC) is mechanically stuck open and 1-HS-5158, CNTMT SMP DRN ISOL VLV (IRC) has been closed from the Control Room.

1-HS-5540, H2 PRG EXH ISOL DMPR (ORC) exhibited leakage in excess of limits and a blind flange has been installed.

RWST boron concentration is 2385 ppm.

1-01 Safety Injection accumulator pressure is 600 psig.

Which one of the following Technical Specification Action Statements is applicable AND has the most limiting COMPLETION TIME?

- A. 3.5.1, Accumulators Condition B
- B. 3.5.4, Refueling Water Storage Tank Condition A
- C. 3.6.3 Containment Isolation Valves Condition A (two containment isolation valves)
- D. 3.6.3 Containment Isolation Valves Condition D (containment ventilation leakage limit)

Answer: C

Explanation:

A is wrong because the accumulator pressure is low out of spec, but its completion time is 24 hours (not the most time limiting from the choices)

B is wrong because the boron concentration is low out of spec, but its completion time is 8 hours.

C is correct because both valves are considered containment isolation valves. With HV-5157 stuck open, the flow path in that line needed to be isolated by shutting AND de-energizing HV-5158. In this case they merely shut HV-5158 and did not secure power to it. This action statement has a completion time of 4 hours.

D is wrong because this action statement has satisfied 3.6.3 D.1. As a result they are committed to D.2 which is to be performed every 31 days.

Technical References:

Comanche Peak Technical Specifications

References to be provided to applicants during exam: None.

Learning Objective: LO21.RLS.SL1, "Technical Specifications": From memory, DEMONSTRATE the ability to apply the T.S. as required in the determination of operability and required actions.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
069 W E14 Loss of Containment Integrity	Tier #	1
	Group #	2
Loss of Containment Integrity RADIATION	K/A #	G2.3.11
CONTROL: Ability to control radiation releases	Rating	4.3

Question 92

Unit 1 is in Mode 1.

1-HS-5157, CNTMT SMP DRN ISOL VLV (ORC) is mechanically stuck open and 1-HS-5158, CNTMT SMP DRN ISOL VLV (IRC) has been closed from the Control Room.

1-HS-5540, H2 PRG EXH ISOL DMPR (ORC) exhibited leakage in excess of limits and a blind flange has been installed.

RWST boron concentration is 2385 ppm.

A loss of Inverter 1PC1 just occurred. Bus 1PC1 is being supplied from bypass power.

Which one of the following Technical Specification Action Statements is applicable AND has the most limiting COMPLETION TIME?

- A. 3.5.4, Refueling Water Storage Tank Condition A
- B. 3.6.3, Containment Isolation Valves Condition A (two containment isolation valves)
- C. 3.6.3, Containment Isolation Valves Condition D (containment ventilation leakage limit)
- D. 3.8.7, Inverters – Operating Condition A.

Answer: B

Explanation:

A is wrong because the boron concentration is low out of spec, but its completion time is 8 hours.

B is correct because both valves are considered containment isolation valves. With HV-5157 stuck open, the flow path in that line needed to be isolated by shutting AND de-energizing HV-5158. In this case they merely shut HV-5158 and did not secure power to it. This action statement has a completion time of 4 hours.

C is wrong because this action statement has satisfied 3.6.3 D.1. As a result they are committed to D.2 which is to be performed every 31 days.

D is wrong but plausible because the condition is applicable, but its completion time is 24 hours (not the most time limiting from the choices)

Technical References:

Comanche Peak Technical Specifications

References to be provided to applicants during exam: None.

Learning Objective: LO21.RLS.SL1, "Technical Specifications": From memory, DEMONSTRATE the ability to apply the T.S. as required in the determination of operability and required actions.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)(2)	

Examination Outline Cross-Reference	Level	SRO
E07 Saturated Core Cooling	Tier #	1
	Group #	2
Ability to determine and/or interpret the following as they apply to Saturated Core Cooling: ECCS Flow	K/A #	EA2.05
	Rating	3.7

Question 93

Unit 1 has initiated a Reactor Trip due to a large break LOCA in Containmentment.

- Safety Injection has been initiated.
- Core exit thermocouples indicate a temperature of 1100 degrees F
- Containmentment pressure is 14.0 psig.
- The crew is responding per FRC-0.2A, Response to Degraded Core Cooling.

Per FRC-0.2A bases:

If core exit subcooling is less than (1) degrees F, then ECCS flow should be maintained to the RCS to provide inventory make-up and the CORE COOLING Critical Safety Function is not satisfied.

When RVLIS indication has been restored with 11 IN ABOVE PLATE LIGHT LIT and core exit TC temperatures are less than (2) degrees F, then ECCS has been successful in restoring RCS inventory and core cooling and the procedure may be exited.

- A. (1) 25
(2) 750
- B. (1) 55
(2) 750
- C. (1) 25
(2) 1200
- D. (1) 55
(2) 1200

Answer: B

Explanation:

A Incorrect – Plausible since the subcooling limit of 25F would be correct below adverse containmentment pressure. Second part is correct since temperature must be reduced below 750F to exit FRC-0.2A.

B Correct – With Containmentment pressure at 14 psig, the adverse Containmentment subcooling limit is 55F. Second part is correct since temperature must be reduced below 750F to exit FRC-0.2A.

C Incorrect – Plausible since the subcooling limit of 25F would be correct below adverse containment pressure. Second part is plausible since RVLIS must be restored with 11 inch above core plate light lit and CET must be reduced below 1200F to exit FRC-0.1A. However, once FRC-0.2A is entered, the temperature limit for exiting FRC-0.2A is reduced to 750F.

D Incorrect – Plausible since the first part is correct. Second part is plausible since RVLIS must be restored with 11 inch above core plate light lit and CET must be reduced below 1200F to exit FRC-0.1A. However, once FRC-0.2A is entered, the temperature limit for exiting FRC-0.2A is reduced to 750F.

Technical References:

FRC-02.A-R9-P0, "Response to Degraded Core Cooling," Revision 9, Page 25

FRC-03.A-R9-P0, "Response to Saturated Core Cooling," Revision 9, Page 9

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.FC1.OB08.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference
System is NA

Level
Tier #
Group #
K/A #
Rating

SRO
3
NA
G2.1.5
3.9

(G2.1.5) Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential)

Question 94

Unit 1 is in Mode 1.

Unit 2 is in Mode 4.

1. The Operations shift crew is required to have a MINIMUM of (1) NEOs to satisfy On-Shift staffing requirements of ODA-102, Conduct of Operations.
2. Per Technical Specification 5.2.2 Unit Staff, the shift crew composition may be ONE less than the minimum requirement for a period of time NOT TO EXCEED (2) hours.
 - A. (1) 7
 (2) four
 - B. (1) 4
 (2) two
 - C. (1) 4
 (2) four
 - D. (1) 7
 (2) two

Answer: D

Explanation:

A is wrong because the 2nd part is incorrect but plausible as an operator can work as much as 4 hours in excess of the normal shift without exceeding work hour rules. Part 1 is correct for the stated conditions, a total of 7 NEOs are required per ODA-102.

B is wrong because 1st part is incorrect because for the stated conditions, 7 NEOs are required. It is plausible because ODA-102 states that 4 NEOs are required for Safe shutdown, two for each unit when in Modes 1, 2, 3, or 4. 2nd part is correct per ODA-102 the operations shift crew composition may be one less than the minimum required for period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

C is wrong because 1st part is incorrect but plausible (see B). 2nd part is incorrect but plausible (see A).

D is correct 1st part is correct for the stated conditions, a total of 7 NEOs are required per ODA-102. 2nd part is correct (see B).

Technical References:

ODA-102, Revision 38, page 39.

TS 5.2.2, Amendment 150, page 5.2-2

References to be provided to applicants during exam: None

Learning Objective: (LO21.RLS.SL1.OB16)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	2017 NRC Q95
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference	Level	SRO
	Tier #	3
Knowledge of RCS or balance-of-plant chemistry controls, including parameters measured and reasons for the control	Group #	
	K/A #	2.1.34
	Rating	3.5

Question 95

While Unit 1 heatup is in progress coming out of a forced outage.

RCS temperature is stable at 225°F and Pressurizer pressure is stable at 320 psig.

Per the Technical Requirements manual, which of the following chemistry parameters have applicable actions if they exceed their specified limits under stated conditions?

What is the basis for establishing limits on these parameters?

- A. Oxygen only.
Reduces potential for fuel cladding oxidation and subsequent failure under accident conditions.
- B. Chloride and fluoride only.
Reduces potential for Reactor Coolant System leakage or failure due to stress corrosion.
- C. Chloride, fluoride, and oxygen.
Reduces potential for fuel cladding oxidation and subsequent failure under accident conditions.
- D. No chemistry parameter limits are applicable for these conditions.
Reduces potential for Reactor Coolant System leakage or failure due to stress corrosion.

Answer: B

Explanation:

RCS Chemistry limits are always applicable. If RCS chloride and fluoride limits are exceeded in any mode, TRM 13.4 has prescribed actions to reduce concentrations within limits. A Chemistry limit is also established for dissolved oxygen; however, oxygen monitoring is not required and the associated limit on oxygen is not applicable with $T_{avg} < 250^{\circ}\text{F}$. Per the TRM basis, the limits on RCS chemistry parameters reduce potential for Reactor Coolant System leakage or failure due to stress corrosion.

A is incorrect. The first part is incorrect but plausible since a chemistry limit is established for oxygen; however, the limit is not applicable for the specified plant conditions. The second part is incorrect but plausible since clad oxidation is a potential issue under accident conditions but the process is not impacted by dissolved oxygen levels in the RCS at the onset of an accident.

B is correct. See explanation.

C is incorrect. The first part is plausible since all three parameters have limits established per TRM requirements; however, the oxygen limit is not applicable for the stated plant conditions. Second part is incorrect but plausible (see A).

D is incorrect. The first part is incorrect but plausible since at least one of the parameters is not applicable for the stated plant conditions and, with the RCS at low pressure, stress corrosion would not be of great concern. Second part is correct (see B).

Technical References:

Technical Requirements Manual, TR 13.4.33 Reactor Coolant System (RCS) Chemistry

References to be provided to applicants during exam: None.

Learning Objective: APPLY the administrative requirements of Primary Chemistry Control including Technical Specifications, TRM and ODCM.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)(2)	

Examination Outline Cross-Reference System is NA	Level Tier # Group #	SRO 3 NA
(G2.2.6) EQUIPMENT CONTROL: Knowledge of the process for making changes to procedures	K/A # Rating	G2.2.6 3.6

Question 96

You must stop a task in progress due to a procedure error that requires a procedure change (PCN).

The PCN does not change intent of the procedure and must be implemented immediately.

Per STA-202, Nuclear Generation Procedure Change Process, interim changes should be approved by _____ (1) and final review and approval of the original change package must be completed _____ (2) .

- A. 1. the procedure supervisor
2. within 14 days of interim approval
- B. 1. the procedure supervisor
2. within 14 days from procedure usage
- C. 1. the on-duty SRO
2. within 14 days of interim approval
- D. 1. the on-duty SRO
2. within 14 days from procedure usage

Answer: C

Explanation:

Per training slide 36, "If the PCN does not change the intent of the procedure and must be implemented immediately, interim approval may be obtained and documented in Section II of STA-202-7. "An Interim Approved Change requires a 10CFR50.59 review and is approved by the Shift Manager or an on-duty SRO. The original change package must be forwarded for final review and approval within 14 days of interim approval."

A is wrong because part 1 is wrong. Procedure supervisor is not the on-duty SM or on-duty SRO. Part 2 is correct.

B is wrong because part 1 and part 1 are wrong. (see above explanation)

C is correct because part 1 and part 2 are both correct (see above explanation)

D is wrong because both part 1 and part 2 are wrong (see above explanation).

Technical References:

LO21.ADM.XA2, Revision 7/23/2020, slides 36 and 37.
STA-202, Revision 42.

References to be provided to applicants during exam: None

Learning Objective: LO21.ADM.XA2.OB5

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.3	

Examination Outline Cross-Reference	Level	SRO
	Tier #	3
Knowledge of maintenance work order requirements	Group #	
	K/A #	2.2.19
	Rating	3.4

Question 97

A clearance is being prepared for work inside MCC 1B1-1.

Per STI-605.01, Work Control and Clearance and Safety Tagging, an independent review of the clearance shall be documented by (1) , and clearance placement shall be authorized by (2) .

- A. (1) A currently-licensed SRO only
 (2) A currently-licensed SRO only
- B. (1) A currently-licensed or previously-licensed SRO
 (2) A currently-licensed SRO only
- C. (1) A currently-licensed SRO only
 (2) A currently-licensed or previously-licensed SRO
- D. (1) A currently-licensed or previously-licensed SRO
 (2) A currently-licensed or previously-licensed SRO

Answer: B

Explanation:

Per STA-605.1, "For clearances that provide personnel protection, a SRO/previous SRO shall document an independent review of the clearance (e.g., an RO creates the Clearance and an SRO/previous SRO reviews the clearance and a current SRO authorizes the placement.) This is not required when requested by Operations for a clearance providing configuration control for a LCOAR, Caution Tag or due to plant conditions."

MCC 1B1-1 is a non-safeguards 480V bus. Work inside this cabinet requires a clearance for personnel protection.

A is wrong, because the independent review may also be performed by a previous SRO. Part 2 correct.

B is correct. See above.

C is wrong because the independent review may also be performed by a previous SRO, and the authorization must come from a current SRO.

D is wrong because the authorization must come from a current SRO. Part 1 correct.

[Note to chief: Part 1 of this question could also be modified if desired to alternate between "A current RO, or a previously-licensed SRO, or a currently-licensed SRO", and the current correct answer.]

Technical References:

STA-605, CLEARANCE AND SAFETY TAGGING, Rev 21
STA-605.1, WORK CONTROL AND CLEARANCE AND SAFETY TAGGING, Rev 12,
Section 6.2.1

References to be provided to applicants during exam: None.

Learning Objective: DESCRIBE the process of creating a clearance IAW STA-605 and STI-605.01. (LO21.WCC.XA1.OB01)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	<u>55.43(b).5</u>	

Examination Outline Cross-Reference
System NA

Level
Tier #
Group #
K/A #
Rating

SRO
3
NA
G2.3.14
3.8

(G2.3.14) RADIATION CONTROL: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures or to analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only)

Question 98

Unit 1 plant shutdown was in progress due to an RCS specific activity level exceeding Technical Specification LCO 3.4.16 limit for greater than 48 hours:

- Dose equivalent I-131 75 $\mu\text{Ci/gm}$
- XE-133 400 $\mu\text{Ci/gm}$

During the shutdown, a small break LOCA occurred initiating an automatic reactor trip and safety injection.

The crew has now transitioned to EOS-1.2A, Post LOCA Cooldown and Depressurization:

- CCW to RCPs was maintained throughout the event.
- Normal charging flowpath and RCP seal injection flow have been established.
- The crew is preparing to establish RCP seal return flow by opening 1/1-8100 and 1/1-8112, RCP SEAL WTR RET ISOL VLVs.

Based on these conditions, complete the following:

1. Opening RCP seal water return isolation valves will increase _____ (1) _____.
 2. The isotope that exceeds LCO 3.4.16 limits requiring plant shutdown is _____ (2) _____.
- A. 1. safeguards building radiation levels
2. XE-133
- B. 1. reactor coolant system inventory losses
2. XE-133
- C. 1. safeguards building radiation levels
2. I-131
- D. 1. reactor coolant system inventory losses
2. I-131

Answer: C

Explanation:

A. is wrong. 1st part is correct (See C). 2nd part is incorrect (See C).

B. is wrong. 1st part is incorrect but plausible as it could be thought that restoring seal water return would be a loss of RCS that was previously isolated. However, with the seal water return valves closed, seal water return is lost from the RCS to the PRT via a relief valve. Once normal charging and seal return are established, seal return water is routed to the charging pump suction. Consequently, restoration of normal seal water return flow does not increase RCS inventory loss.

C. is Correct. 1st part is correct as the bases for step 29 of EOS-1.2A states that restoring normal RCP seal water return flow to the VCT may raise radiation levels in the safeguards building, especially with high RCS activity. 2nd part is correct as Technical Specification LCO 3.4.16 requires I-131 or XE-133 above limit to be restored to with limit with 48 hours or initiate a unit shutdown. Also if I-131 is above 60 $\mu\text{Ci/gm}$ a unit shutdown is required. XE-133 great than 500 $\mu\text{Ci/gm}$ is needed to initiate a unit shutdown due to XE-133.

D. is wrong. 1st part is incorrect (See B). 2nd part is (See C).

Technical References:

EOS-1.2, Revision 9, pages 25 and 61.

TS 3.4.16, Amendment 156, pages 159-160.

References to be provided to applicants during exam: None.

Learning Objective: OPD1.ADM.XA1.OB17

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	2015-06 Q98
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43.4	

Examination Outline Cross-Reference	Level	SRO
System is NA	Tier #	3
(G2.4.14) EMERGENCY	Group #	NA
PROCEDURES/PLAN: Knowledge of general guidelines for emergency and abnormal operating procedures usage	K/A #	G2.4.14
	Rating	4.5

Question 99

Unit 2 was at 100% power when 1/2-PCV-456, PRZR PORV failed open.

Unit 2 reactor reactor trip and safety injection occurred.

The associated block valve has since been closed for 1/2-PCV-456.

EOP-1.0B, LOSS OF REACTOR OR SECONDARY COOLANT, Step 3 is in progress to Check Intact SG Levels.

In accordance with ODA-407, Attachment 8A, ERG Rules of Usage regarding continuous action steps...

In the ERG network, continuous action steps are designated by a (an) ___1__.

Upon subsequently transitioning to EOS-1.1B, SAFETY INJECTION TERMINATION, the continuous action step to Check Intact SG Levels ___2___ applicable.

- A. 1. asterisk (*)
2. is
- B. 1. asterisk (*)
2. is NOT
- C. 1. [C]
2. is
- D. 1. [C]
2. Is NOT

Answer: A

Explanation:

Per ODA-407, Attachment 8.A, ERG Rules of Usage A Continuous Action Step is applicable from the point at which it is first encountered until superseded by alternate instruction OR stated to be no longer applicable. The course of direction stated by the Continuous Action Step should be followed until subsequent procedure instruction OR a change in the response and recovery strategy is encountered that supercedes the previous instruction. In this specific case, SG Levels will be maintained per the direction given by the continuous action step in EOP-1.1B until otherwise directed. These steps are annotated by an asterisk and shading on the procedure flow chart.

A is correct, see explanation above.

B first part is correct, second part is incorrect but plausible because a new procedure is being entered. Some procedures in the ERG network will annotate when a continuous action step is no longer applicable.

C first part is incorrect but plausible because [C] stands for commitments, and they are also applicable to other attachments in the ERG network. Lastly, the candidate may link the corresponding letter to the continuous action annotation.

D both parts are incorrect, see B and C.

Technical References:

ODA-407, Revision 19, page 21 and 22.

References to be provided to applicants during exam: None

Learning Objective: LO21.ERG.XD2.OB06

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference Emergency Procedures/Plan	Level Tier # Group #	SRO 3 N/A
EMERGENCY PROCEDURES/PLAN: Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions (CFR 41.7 / 43.5 / 45.12)	K/A # Rating	G2.4.21 4.6

Question 100

Using ERG rules of usage guidance in ODA-407, Operations Department Procedure Use and Adherence, consider the following conditions:

A unit tripped with no safety injection actuation, and safety injection was not required at the time. The Unit Supervisor should transition to the appropriate procedure and _____ (1) _____.

A second event occurs that now requires safety injection and results in a Critical Safety Function Status Tree (CSFST) yellow path. Based on these new conditions, the next procedure the Unit Supervisor should implement is _____ (2) _____.

- A. (1) implement FRG procedures as required
(2) the yellow path FRG
- B. (1) monitor CSFSTs only
(2) EOP-0.0A/B, Reactor Trip or Safety Injection
- C. (1) implement FRG procedures as required
(2) EOP-0.0A/B, Reactor Trip or Safety Injection
- D. (1) monitor CSFSTs only
(2) the yellow path FRG

Answer: C

Explanation:

C is correct for this question.

A is wrong because part 2 is wrong. Part 1 is correct, when you enter EOS-1.0 you also must implement the FRGs (not monitor them). Part 2 is wrong because if it is red or orange path you would be required to implement the FRG first but with a yellow path you are required to go back to EOP-0.0 and monitor CSFST's. The applicant must know that they are in EOS-1.0 with the first part of the stem (this is the appropriate procedure).

B is wrong because part 1 is wrong. When you enter EOS-1.0 you also must implement the FRGs (not monitor them). Part 2 is correct You must enter EOP-0.0 again for the Safety Injection and Monitor CSFST's only for a yellow condition.

C is correct. When you enter EOS-1.0 you also must implement the FRGs (not monitor them). This makes part 1 correct. For part 2, with a valid SI actuation and no red or orange path FRG you are required to go to EOP-0.0 and Monitor CSFST's only.

D is wrong because part 1 and part 2 are incorrect (see above explanations).

Technical References:

ODA-407, Rev 19, page 30.
ERG User Guide Study Guide

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.XD2.OB016: Given a set of plant conditions and a set of Critical Safety Function Status Trees, correctly DETERMINE the status of the Critical Safety functions and IDENTIFY the applicable FRGs.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)(5)	