CPNPP NRC 2018 Written Examination
Senior Reactor Operator

No.	A	B	C	D
1.	A	B	C	●
2.	●	B	C	D
3.	A	B	●	D
4.	●	B	C	D
5.	A	B	●	D
6.	A	B	C	●
7.	A	●	C	D
8.	A	B	C	●
9.	A	B	●	D
10.	A	B	C	●
11.	●	B	C	D
12.	A	●	C	D
13.	●	B	C	D
14.	A	B	C	●
15.	●	●	C	D
16.	A	B	●	D
17.	A	B	●	D
18.	A	B	C	●
19.	A	●	C	D
20.	A	●	C	D
21.	A	●	C	D
22.	●	B	C	D
23.	A	B	C	●
24.	A	B	C	●
25.	A	B	●	D
26.	A	B	●	D
27.	A	B	●	D
28.	A	B	●	D
29.	A	●	C	D
30.	A	●	C	D
31.	A	B	●	D
32.	A	B	●	D
33.	●	B	C	D
34.	●	B	C	D
35.	A	B	C	●
36.	●	B	C	D
37.	A	B	C	●
38.	A	B	C	●
39.	●	B	C	D
40.	A	B	●	D
41.	A	B	●	D
42.	A	●	C	D
43.	●	B	C	D
44.	A	B	●	D
45.	●	B	C	D
46.	A	B	●	D
47.	●	B	C	D
48.	A	●	C	D
49.	●	B	C	D
50.	A	●	C	D
51.	●	B	C	D

Accept Answers A&B
JR
6/21/18

Name ______________________ Date ______________

CPNPP NRC 2018 Written Examination
Senior Reactor Operator

52. A B ● D
53. A ● C D
54. A B C ●
55. ● B C D
56. A B C ●
57. ● B C D
58. A ● C D
59. A ● C D
60. A B ● D
61. A B C ●
62. A B C ●
63. A B ● D
64. A ● C D
65. A B ● D
66. A ● C D
67. ● B C D
68. ● B C D
69. ● B C D
70. ● B C D
71. A B ● D
72. A ● C D
73. A B C ●
74. A B ● D
75. A B C ●
76. ● B C D
77. A B ● D
78. A ● C D
79. A B C ●
80. ● B C D
81. ● B C D
82. A B ● D
83. A B ● D
84. A ● C D
85. A B C ●
86. ● B C D
87. A ● C D
88. ● B C D
89. A B ● D
90. A ● C D
91. ● B C D
92. A B C ●
93. A B ● D
94. A ● C D
95. A ● C D
96. A B C ●
97. A B ● D
98. ● B C D
99. A B C ●
100. A B C ●

Name ____________________ Date ______________

Examination Outline Cross-Reference	Level	RO
076 (SF4S SW) Service Water	Tier #	2
	Group #	1
2.1.32 Ability to explain and apply system limits and precautions.	K/A #	G2.1.32
	Rating	3.8
	QREV	6

Question 1

Unit 2 is in an extended outage and SSW Pump 2-02 needs to be secured for maintenance. The BOP operator places 2-HS-4251A, SSWP 2 to the STOP position on the MCB.

1) Per the Precautions and Limitations of SOP-501B, Station Service Water System, when SSW Pump 2-02 handswitch is placed in the STOP position; the Discharge Valve, 2-HV-4287, will travel ____(1)____ CLOSED.

Subsequently:

SSW Pump 2-02 is to be restarted following several days of maintenance.
One minute after SSW Pump 2-02 is started; the operator inadvertently stops the pump due to a human performance error.

2) With the cause of the human performance error corrected, SSW Pump 2-02 may be re-started ____(2)______ , per the Precautions and Limitations of SOP-501B.

A. (1) 100%
(2) 30 minutes after the motor stops

B. (1) 90%
(2) 30 minutes after the motor stops

C. (1) 100%
(2) immediately after the motor stops

D. (1) 90%
(2) immediately after the motor stops

Answer: D

Explanation:

Per the P&L of SOP-501B, when the SSW Pump(s) discharge valve(s) is/are CLOSED via motor operator, THEN the limit switches stop the valve at approximately 10% open. Should an activity, such as the placement of a clearance, require this valve to be fully CLOSED, it must be manually CLOSED. When the valve is CLOSED manually, it should be reopened to 10% when being returned to service.

An SSW pump is allowed 3 consecutive starts after the motor stops. Subsequent starts require either 30 minutes of run time or 60 minutes idle time. Since this is only the second start, the SSW pump may be started immediately on motor stop.

A is wrong. Part 1 is incorrect but plausible as this is what happens on the Condensate pumps when they are stooped. Part 2 is incorrect but plausible because a 30 minute wait would be required after subsequent starts.
B is wrong. Part 1 is correct, Part 2 is incorrect but plausible, see A
C is wrong. Part 1 is incorrect but plausible see A. Part 2 is correct.
D is correct.

Technical References:
SOP-501B, Station Service Water System, Rev 12, pg. 7.

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the normal, abnormal and emergency operation of the Station Service Water system IAW SOP-501 and ABN-501. (LO21.SYS.SW1.OB05)

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
	Tier #	2
059 (SF4S MFW) Main Feedwater	Group #	1
	K/A #	A3.02
Ability to monitor automatic operation of the MFW, including: A3.02 Programmed levels of the S/G	Rating	2.9
	QREV	6

Question 2

Unit 1 and Unit 2 steam generators are both at 65.5% narrow range level.

Both units are at full power.

What is the expected response of each unit's steam generator water level control system with the control system in automatic?

Raise/Lower Unit 1 steam generator level to (1) narrow range level.
Raise/Lower Unit 2 steam generator level to (2) narrow range level.

A. (1) 67%
(2) 64%

B. (1) 64%
(2) 67%

C. (1) 75%
(2) 60%

D. (1) 60%
(2) 75%

Answer: A

Explanation:
A is correct because these are the stated program levels in the steam generator water level control study guide and main feedwater study guide.
B is wrong because they are the incorrect levels for the particular steam generators.
C is wrong because they are the incorrect levels for the particular steam generators. They are plausible since they correspond to wide range levels that ensure coverage of the tube bundles. Both values are correct for their corresponding steam generators.
D is wrong because they are incorrect levels. See plausibility in C.

Technical References:
Steam Generator Water Level Control Study Guide page 5
Main Feedwater Study Guide page 26

References to be provided to applicants during exam: None.

Learning Objective: SYS.SN1 2.d EXPLAIN the instrumentation and controls of the Steam Generator Water Level Control system and PREDICT the system response in accordance with DBD-ME-203. (LO21.SYS.SN1.OB04)

SYS.MF1 2.g DIFFERENTIATE between the Unit 1 and 2 Main Feedwater systems in accordance with DBD-ME-203. (LO21.SYS.MF1.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	 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 IAS design feature(s) and/or interlock(s) which provide for the following: K4.01 Manual/automatic transfers of control	K/A #	K4.01
	Rating	2.7
	QREV	6

Question 3

When will Instrument Air Compressor 1-02 load/start, if it is selected as the backup instrument air compressor?

A. 85 psig

B. 95 psig

C. 100 psig

D. 105 psig

Answer: C

Explanation:
A is incorrect but plausible because 85 psig is the pressure that ABN-301 attempts to establish.
B is incorrect but plausible because 95 psig is the pressure that air compressors X-01 and X-02 would load/start if they were the backup compressor.
C is correct because 100 psig is the pressure that air compressor 1-02 loads/starts when it is in backup.
D is incorrect but plausible because 105 psig is the pressure that air compressor 1-02 would load/start if it were the lead compressor.

Technical References:
LO.21.SYS.IA1 powerpoint, slide 32, rev 7/20/2017

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.IA1.OB05

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: 55.41.7

Examination Outline Cross-Reference	Level	RO
012 (SF7 RPS) Reactor Protection	Tier #	2
	Group #	1
Ability to predict and/or monitor Changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: A1.01 Trip setpoint adjustment	K/A #	A1.01
	Rating	2.9
	QREV	6

Question 4

The Over-Temperature N-16 trip nominal setpoint of ____1_____ varies with several primary parameters. The core design limits are additionally protected with this trip setpoint by reducing the OT N-16 trip when ______2____.

A. 115% of rated thermal power
   the margin to DNB is reduced

B. 115% of rated thermal power
   the margin to RIL is reduced

C. 112% of rated thermal power
   the margin to DNB is reduced

D. 112% of rated thermal power
   the margin to RIL is reduced

Answer: A

Explanation:

A is correct because the setpoint for the OTN-16 overtemperature limit is 115% of rated thermal power and the setpoint for OTN-16 temp varies (reduced) when the margin to DNB is reduced.

B is wrong (see A) but plausible because the second part is wrong.

C is wrong (see A) but plausible because 112% is the OPN-16 power trip, second part is correct.

D is wrong (see A) but plausible because 112% is the overpower limit for OPN-16 and the second part is wrong.

Technical References:

LO21.SYS.NT1 LOPC.ppt, revision 5-31-1997, slide 20

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.NT1.OB05

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

Question History:	Last NRC Exam	
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)5	

Examination Outline Cross-Reference	Level	RO
063 (SF6 ED DC) DC Electrical Distribution	Tier #	2
	Group #	1
K2.01 Knowledge of bus power supplies to the following: Major DC loads (CFR: 41.7)	K/A #	K2.01
	Rating	3.7
	QREV	6

Question 5

Which of the following components are powered by DC Bus 1D2?

(1) Main Generator Emergency Seal Oil Pump
(2) Main Turbine Seal Steam Control Unit
(3) Feedwater Pump 1A Emergency Lube Oil Pump
(4) Feedwater Pump 1B Emergency Lube Oil Pump

A. (1) and (3) only.

B. (1), (2), and (3) only.

C. (1), (3), and (4) only.

D. (3) and (4) only.

Answer: C

Explanation:

DC Bus 1D2 is unique in that it supplies both 125 VDC and 250 VDC loads, and is the only source of 250 VDC power in the unit.

A is wrong because while both components are supplied by 1D2, (4) Feedwater Pump 1B Emx Lube Oil Pump is also supplied by 1D2. This is unique to have both trains of the same component supplied by the same bus.
B is wrong because the Main Turbine Seal Steam Control Unit is powered by 125VDC Bus 1D1, not 1D2.
C is correct.
D is wrong because 1D2 additionally supplies (1) Main Generator Emergency Seal oil pump. Plausible because the normal Main Generator Seal Oil Pumps are supplied by AC buses 1B3 and 1B4, not DC.

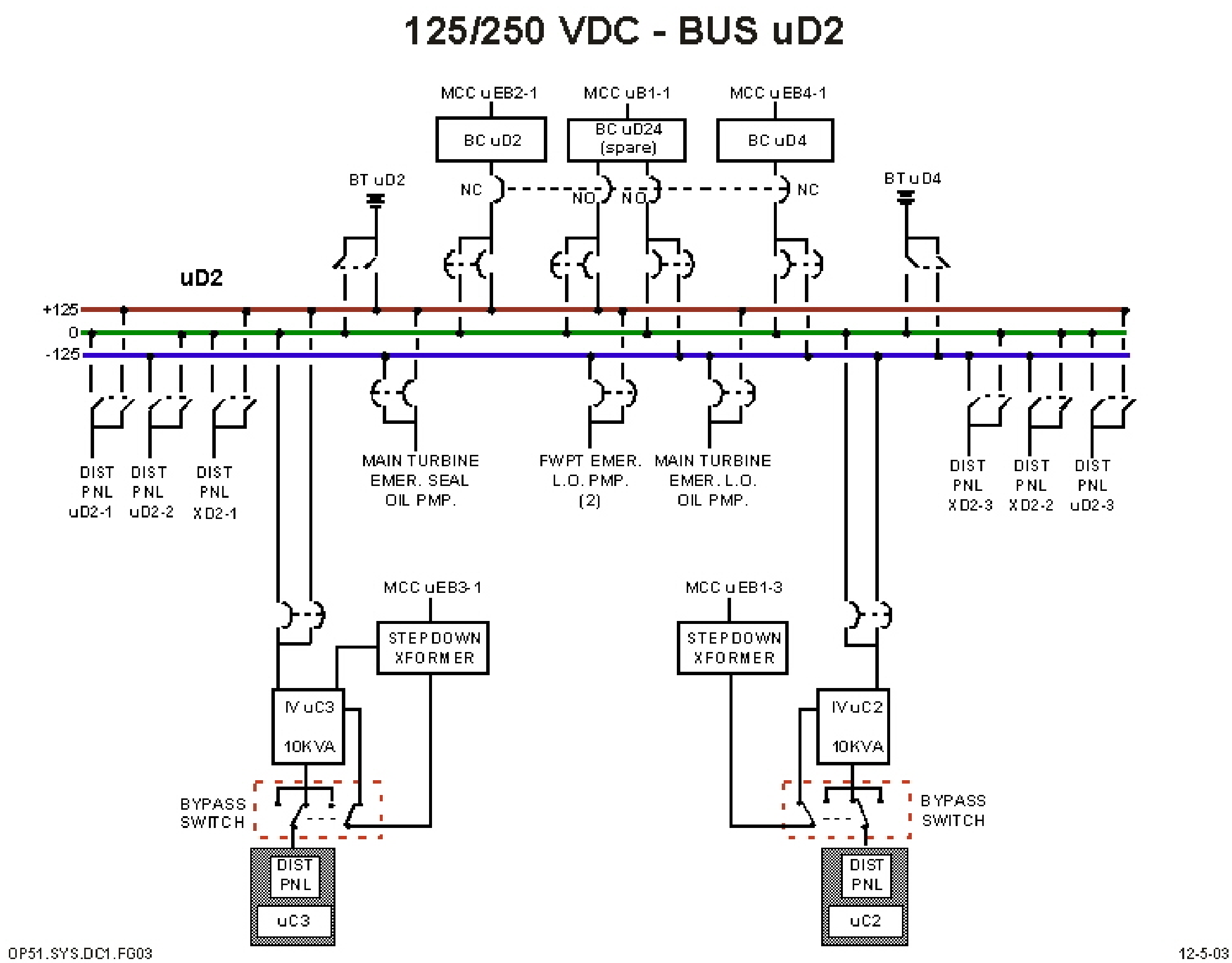
Technical References:
1D2 Load List
DC Electrical Distribution Study Guide, 5-5-2011 page 8

References to be provided to applicants during exam: None.

Learning Objective: DEMONSTRATE an understanding of the components of the DC Electrical Distribution system including interrelations with other systems to include interlocks and control loops.

(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) 55.43	



Examination Outline Cross-Reference	Level	RO
076 (SF4S SW) Service Water	Tier #	2
Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.01 Loss of SWS	Group #	1
	K/A #	A2.01
	Rating	3.5
	QREV	6

Question 6

- Unit 1 at 70%
- An 86-2 Lockout Relay (LOR) fault occurs on Safeguards Bus 1EA2
- Both 1EA2-1, INCOMING BKR, and 1EA2-2, INCOMING BKR trip open
- EDG 1-02 automatically starts and DG 2 BKR 1EG2 automatically closes
- 1-ALB-01, Window 1.8 – SSWP 1/2 OVERLOAD/TRIP, alarms
- SSW Pump 1-02 handswitch indications are as follows:
  - Amber MISMATCH light LIT
  - White TRIP light LIT

Per ABN-501, Station Service Water System Malfunction, which of the following actions should be taken?

A. Wait for the Blackout Sequencer to complete its timing and verify proper Service Water Flow to EDG 1-02.
Continue plant operations at approximately 70% power.

B. Wait for the Blackout Sequencer to complete its timing and verify proper Service Water Flow to EDG 1-02.
Enter and perform the actions of EOP-0.0A, Reactor Trip or Safety Injection.

C. Place EDG 1-02 EMER STOP/START handswitch in PULL-OUT.
Enter and perform the actions of EOP-0.0A, Reactor Trip or Safety Injection.

D. Place EDG 1-02 EMER STOP/START handswitch in PULL-OUT.
Continue plant operations at approximately 70% power.

Answer: D

Explanation:
A is wrong. Plausible because the BOS will fire to load the EDG and the SSW Pump will have an amber MISMATCH light until the sequencer sends a signal to start the pump, however, the white TRIP light is not expected and this is an indication that the pump has tripped and will not start on the sequencer timing. The immediate action per ABN-501 is to place the EDG in PULLOUT.

B. is wrong. Plausible because the BOS will fire to load the EDG and the SSW Pump will have an amber MISMATCH light until the sequencer sends a signal to start the pump, however, the white TRIP light is not expected and this is an indication that the pump has

tripped and will not start on the sequencer timing. The immediate action per ABN-501 is to place the EDG in PULLOUT.

C. is wrong. Plausible because with the SSW Pump tripped, the EDG must be immediately tripped whether or not it is carrying the bus, however, a loss of a single Safeguards Bus does not require the Unit be tripped.

D, Is correct. With the SSW Pump tripped, the EDG must be immediately tripped whether or not it is carrying the bus. A loss of a single Safeguards Bus does not require the Unit be tripped....

Technical References:
ABN-501, rev 9, page 4, ALB-0011.,

References to be provided to applicants during exam: None.

Learning Objective: ABN.501.OB01.006.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X (23103)
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
	Tier #	2
007 Pressurizer Relief/Quench Tank	Group #	1
	K/A #	K5.02
Knowledge of the operational implications of the following concepts as the apply to PRTS: K5.02 Method of forming a steam bubble in the PZR	Rating	3.1
	QREV	7

Question 7

Given the following conditions:

- Unit 1 is in MODE 5
- RCS Pressure is 5 psia

Which of the following describes the PREFERRED method for establishing a steam bubble in the Pressurizer in accordance with SOP-101A, Reactor Coolant System?

A. Maintain pressurizer full, maintain vacuum in the PRT, pressurizer heaters on with a MAXIMUM heatup of 100°F in one hour, saturation temperature approximately 429°F.

B. Maintain pressurizer level at 50%, maintain PRT nitrogen blanket at 3 psig, pressurizer heaters on with a MAXIMUM heatup of 100°F in one hour, saturation temperature approximately 162°F.

C. Maintain pressurizer full, maintain PRT nitrogen blanket at 3 psig, pressurizer heaters on with a MAXIMUM heatup of 50°F in one hour, saturation temperature approximately 162°F.

D. Maintain pressurizer level at 50%, maintain vacuum in the PRT, pressurizer heaters on with a MAXIMUM heatup of 50°F in one hour, saturation temperature approximately 429°F.

Answer: B
Explanation:
A is wrong because this is a method for establishing a steam bubble but not the preferred method. These steps were found in Section 5.6 and require taking the pressurizer solid.
B is correct because these are the steps found in Section 5.5 of the procedure and are done during a vacuum fill of the RCS.
C is wrong because this contains a mix and match of the requirements in A and B. The 50°F is a number that was picked but seemed plausible. Could change to 80°F for better plausibility.
D is wrong because similar reasoning to C.

Technical References:
SOP-101A, Reactor Coolant System, Rev 18, Sections 5.5 and 5.6

References to be provided to applicants during exam: None.

Learning Objective: SYS.RC1.OB05.024.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X (ILOT6363)
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41(b)(5)	

Question #7 K/A 007 PRT (SF5) K5.02

1	ID: ILOT6363	Points: 1.00

Given the following conditions:

- After being used to draw a vacuum in the Reactor Coolant System (RCS), the Pressurizer Relief Tank has been isolated from the Pressurizer (PRZR).
- RCS pressure is 5 psia.
- RCS and PRZR temperatures are equalized at 130ºF.
- Actual PRZR level is 50%.

Which of the following describes the PREFERRED method of establishing a steam bubble in the Pressurizer in accordance with SOP-101A, Reactor Coolant System?

Adjust Charging and Letdown to…

A. raise PRZR level to 100% and raise PRZR pressure to approximately 50 psig. Energize PRZR heaters to heat up the PRZR, establishing a steam bubble at approximately 298ºF in the PRZR.

B. maintain PRZR level constant and raise PRZR pressure to approximately 50 psig.
Energize PRZR heaters to heat up the PRZR, establishing a steam bubble at approximately 298ºF in the PRZR.

C. raise PRZR level to 100% and maintain PRZR pressure constant. Energize PRZR heaters to heat up the PRZR, establishing a steam bubble at approximately 162ºF in the PRZR.

D. maintain PRZR level constant and maintain PRZR pressure constant. Energize PRZR heaters to heat up the PRZR, establishing a steam bubble at approximately 162ºF in the PRZR.

Answer: D

Answer Explanation
A. Incorrect. Plausible since this is similar to an alternate method of establishing a bubble, however, the preferred method is by establishing a bubble while a vacuum still exists in the RCS. B. Incorrect. Plausible since this is similar to an alternate method of establishing a bubble, however, the preferred method is by establishing a bubble while a vacuum still exists in the RCS. C. Incorrect. Plausible since the bubble is established under vacuum conditions, but the level in the PRZR is maintained constant at approximately 50% instead of filling solid. D. Correct. During vacuum fill of the RCS a bubble in the PRZR is established while the RCS is at a vacuum. Charging and Letdown are adjusted as needed to maintain level and pressure constant in preparation for establishing a bubble.

Question 1 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	3.00
System ID:	23551
User-Defined ID:	ILOT6363
Cross Reference Number:	SYS.RC1.OB05.024
Topic:	Given the following conditions: After being used to draw a vacuum in the Reactor Coolant System (
K/A:	SF5.007.K5.02
Question Reference:	
SRO (YES):	
Comments:	LC20 Audit; R/S21E18 (Comp); R22E41 (NRC Remedial 1) REF: SOP-101; OP51.SYS.RC1

Examination Outline Cross-Reference	Level	RO
062 (SF6 ED AC) AC Electrical Distribution	Tier #	2
	Group #	1
A3.04 Ability to monitor automatic operation of the ac distribution system, including: Operation of inverter (e.g., precharging synchronizing light, static transfer) (CFR: 41.7)	K/A #	A3.04
	Rating	2.7
	QREV	6

Question 8

Which one of the following conditions will directly cause an automatic transfer of Inverter IV1PC1 Static Switch from "INVERTER SUPPLYING LOAD" to "BYPASS SOURCE SUPPLYING LOAD" if the IN SYNC light is lit?

A. AC OUTPUT Frequency meter reads 57 Hz.

B. AC OUTPUT Ammeter reads 150 Amps.

C. FAN FAILURE red lamp lit. Main Control Board INVERTER TROUBLE alarm lit.

D. DC INPUT Voltmeter reads 104 VDC.

Answer: D

Explanation:

A is wrong because the static switch doesn't automatically transfer on low AC output frequency.

B is wrong because the static switch automatically transfers on high AC OUTPUT current of 175 Amps, 120% of rated load.

C is wrong because a FAN FAILURE is caused by high temperatures inside the inverter cabinet, and also causes a MCB INVERTER TROUBLE alarm (which also comes in when the static switch is aligned to bypass), but this does not directly cause a static switch transfer.

D is correct because the static switch auto transfers on DC INPUT VOLTAGE ≤105 VDC

Technical References:
208/120 VAC, 118 VAC Distribution, Inverters & Lighting STUDY GUIDE, 4/11/17, page 12.

References to be provided to applicants during exam: None.

Learning Objective: DESCRIBE the components of the 118 VAC Instrument Power system including interrelations with other systems to include interlocks and control loops as described in DBD-EE-043. (LO21.SYS.AC3.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)(7) 55.43	

Background Information:

Each inverter has two momentary pushbuttons to allow manual operation of the static switch:

INVERTER TO LOAD

BYPASS SOURCE TO LOAD

Indication of the state of the static switch is provided by two lights:

INVERTER SUPPLYING LOAD yellow light

BYPASS SOURCE SUPPLYING LOAD red light

There is also a yellow IN SYNC light, which indicates that voltage and frequency of the inverter circuitry output and the bypass power source are matched (within 0.8 Hz).

When the inverter and bypass source are synchronized, the static switch will automatically transfer to bypass supplying load on the following fault conditions:

DC input voltage ≤105 VDC

AC output voltage drops to 50% normal value (~59 VAC)

Output load current rises to 120% of rated value (~175 Amps)

Each inverter has four meters:

AC OUTPUT Voltmeter

AC OUTPUT Ammeter

AC OUTPUT Frequency meter

BYPASS SOURCE AC INPUT Voltmeter

Alarm indication on the inverters is as follows:

INVERTER FAILURE – red light lit indicates low voltage downstream of CB2 AC OUTPUT breaker.

LOW DC VOLTAGE – red light lit indicates ≤105 VDC on the inverter side of CB1 DC INPUT breaker.

FAN FAILURE – red light lit indicates high temperatures inside the inverter cabinet. This condition may be caused by one or both of the inverter cabinet cooling fans (mounted on top of the cabinet) not running, high ambient temperature or component failure causing high cabinet temperatures.

The following conditions will cause a Main Control Board inverter trouble alarm:

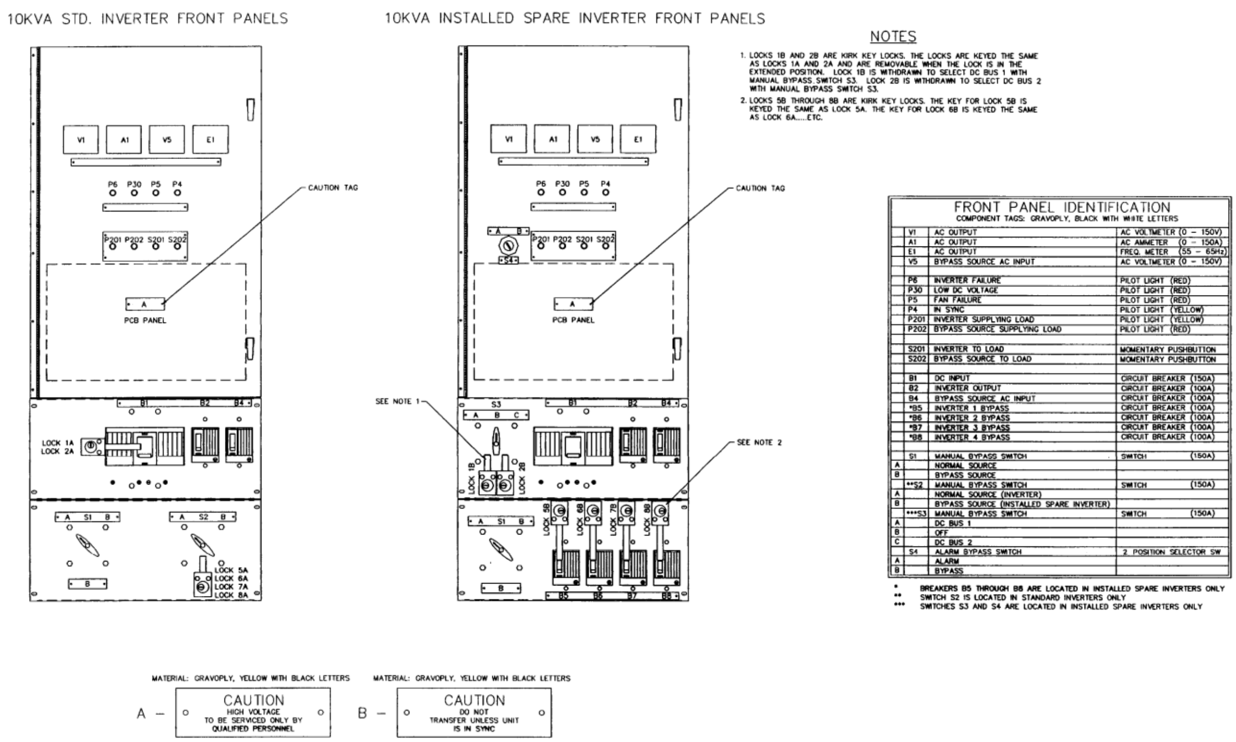
Local FAN FAILURE alarm.

CB1 DC INPUT breaker open.

Local INVERTER FAILURE alarm.

Static switch in BYPASS SOURCE SUPPLYING LOAD.

BYP SW S1 switch in BYPASS SOURCE position.



This drawing reproduced
from vendor drawing
10-102723/001

Figure 11

Examination Outline Cross-Reference	Level	RO
039 Main and Reheat Steam System (MRSS)	Tier #	1
	Group #	1
Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: K3.06 SDS	K/A #	K3.06
	Rating	2.8
	QREV	7

Question 9

Which of the following malfunctions would cause the steam dump controller to generate a demand signal of 1% for every 1.8 pounds of differential pressure detected, in the steam pressure mode in auto?

- PT-505, U1 Turbine Impulse Chamber Pressure Transmitter
- PT-507, U1 Steam Dump Header Pressure Transmitter

A. PT-505 fails high

B. PT-505 fails low

C. PT-507 fails high

D. PT-507 fails low

Answer: C
Explanation:
A is incorrect because PT-505 failing high results in steam dump valves remaining closed until RCS Average Tavg reached five degrees above 100% power programmed Tavg. Plausible because PT-505 is an input to the steam dump control.
B is incorrect because PT-505 failing low results in reference value dropping to approximately 530°F. Plausible because PT-505 is an input to the steam dump control.
C is correct.
D is incorrect because PT-507 failing low results in the steam dump valves closing if in the steam pressure mode. Plausible because PT-507 is the correct instrument.

Technical References:
LO21SYSSD1, Steam Dumps, Revision 30, pages 17 and 18

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.SD1.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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
006 Emergency Core Cooling	Tier #	2
	Group #	1
Knowledge of the operational implications of the following concepts as they apply to ECCS: K5.05 Effects of pressure on a solid system	K/A #	K5.05
	Rating	3.4
	QREV	6

Question 10

A spurious safety injection occurs at 100% power. With NO operator action, the bubble in the pressurizer _____1_____, and an interlock prevents the safety injection signal from being reset for a MINIMUM of _______2_______seconds following the spurious actuation.

A. 1) is maintained by the pressurizer PORV's
 2) 30

B. 1) collapses
 2) 30

C. 1) is maintained by the pressurizer PORV's
 2) 60

D. 1) collapses
 2) 60

Answer: D

Explanation:
A is wrong because the RCS becomes a solid system; the interlock is 60 seconds
B is wrong because the interlock is 60 seconds
C is wrong because the RCS becomes a solid system
D is correct because the ECCS will increase RCS pressure until it becomes a solid system, collapsing the pressurizer bubble; the interlock is 60 seconds

Technical References:
LO21.SYS.SI1, Revision 5/25/17, Page 51

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the instrumentation and controls of the Pressurizer Pressure Control System and **PREDICT** the system response. (LO21.SYS.PP1.OB04).

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)5

Examination Outline Cross-Reference	Level	RO
010 (SF3 PZR PCS) Pressurizer Pressure Control	Tier #	2
	Group #	1
Knowledge of bus power supplies to the following: K2.03 Indicator for PORV position	K/A #	K2.03
	Rating	2.8
	QREV	6

Question 11

Power for the position indicating lights of 1/1-PCV-455A, PRZR PORV on CB-05 are supplied from __________.

A. 1ED1

B. 1ED2

C. 1EC1

D. 1EC2

Answer: A

Explanation:
A is correct because a tripped control power breaker on 1ED1 will cause the valve to lose power and fail shut with no position indication
B is wrong but plausible because this would be the power supply to PORV-456 position indication.
C is wrong but plausible as 1EC1 is similar to 1ED1.
D is wrong but plausible 1EC2 is similar to 1ED2 which would be the power supply to PORV-456 position indication.

Technical References:
E1-0020, Sheet B, Rev CP-20.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.603.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
013 (SF2 ESFAS) Engineered Safety Features Actuation	Tier #	2
	Group #	1
	K/A #	A3.01
Ability to monitor automatic operation of the ESFAS including: A3.01 Input channels and logic	Rating	3.7
	QREV	6

Question 12

The ESF protection scheme for Steam Line Isolation on Negative Steam Line Pressure Rate requires _____1_____ on 1 out of 4 steam generators and has a rate sensitivity of 100 psig decreasing with _____2______.

A. 2 out of 3 pressure signals
a 100 second lead/lag time constant

B. 2 out of 3 pressure signals
a 50 second lead/lag time constant

C. 2 out of 4 pressure signals
a 100 second lead/lag time constant

D. 2 out of 4 pressure signals
a 50 second lead/lag time constant

Answer: B

Explanation:
A is wrong because the second part is wrong (see B below)
B is correct because the logic is 2 of 3 pressure signals at 100 psig and decreasing on 1 out of 4 SGs with a rate constant of 50 seconds lead/lag.
C is wrong because the logic is 2 of 3 not 2 of 4 but plausible because there are other logic schemes that have 4 signals.
C is wrong because the first part is wrong as described in B above and the second part of 100 seconds is wrong but is plausible because the 100 psig and decreasing is the pressure setpoint.

Technical References:
LO21SYSES1.pdf, rev 6/14/17, page 16.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.ES1.OB04

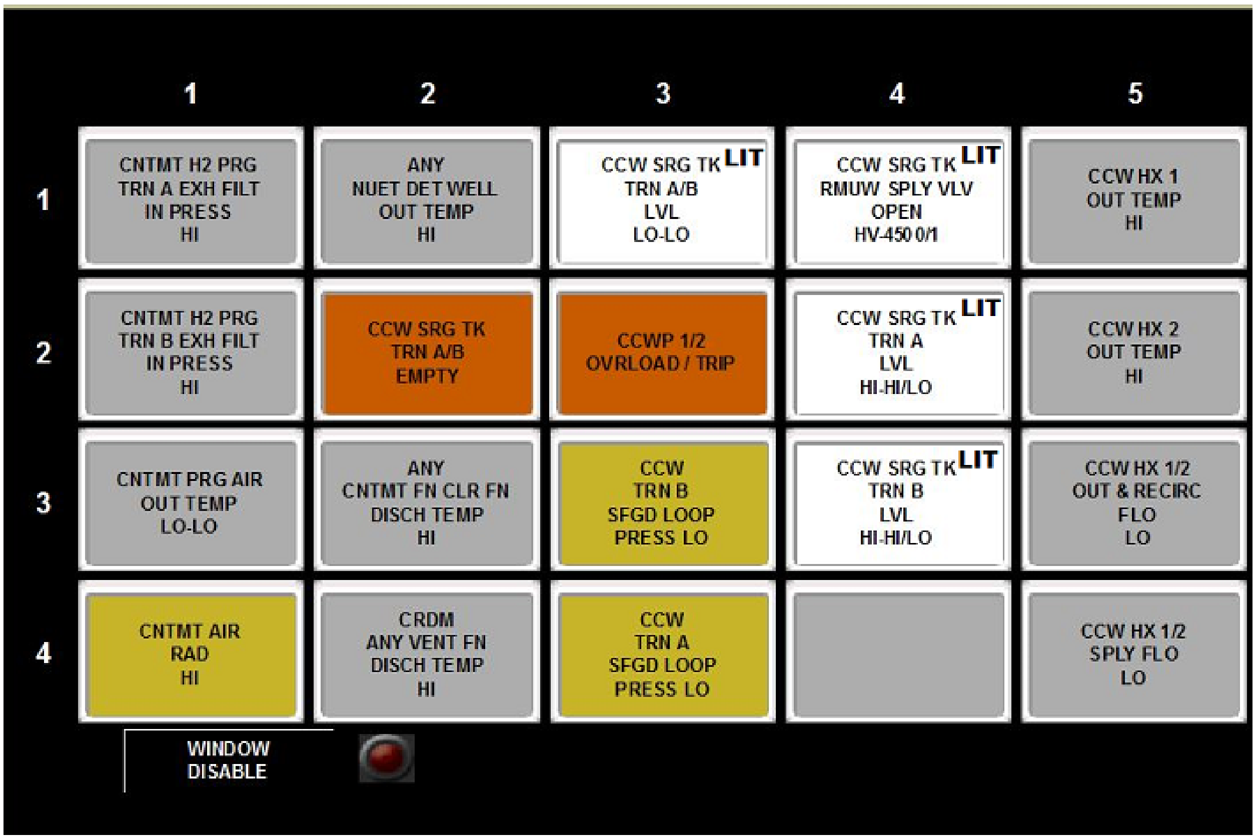
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.41(b)7	

Examination Outline Cross-Reference	Level	RO
008 Component Cooling Water	Tier #	2
	Group #	1
Ability to manually operate and/or monitor in the control room:	K/A #	A4.07
	Rating	2.9
A4.07 Control of minimum level in the CCWS surge tank	QREV	7

Question 13

For **UNIT 1**, the following alarms are LIT solid on 1-ALB-3B.



Based on the conditions provided what CCW Surge Tank level did Make-up START?

(Assume BOTH Trains of the CCW Surge Tank are at the same level)

A. 63%

B. 57%

C. 39%

D. 33%

Answer: A
Explanation:
A is correct for Unit 1
B is wrong because at this level the Unit 1 tank is considered empty.
C is wrong because this is the LO-LO Level setpoint for Unit 2
D is wrong because at this level the Unit 2 tank is considered empty

Technical References:
LO21.SYS.CC1., Revision 5/30/2017, Pages 9-10

References to be provided to applicants during exam: None.

Learning Objective: DIFFERENTIATE between the Unit 1 and 2 Component Cooling Water systems as described in DBD-ME-229 (LO21.SYS.CC1.OB07).

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)7	

Examination Outline Cross-Reference	Level	RO
003 (SF4P RCP) Reactor Coolant Pump	Tier #	2
	Group #	1
Knowledge of the effect of a loss or	K/A #	K6.14
malfunction on the following will have on the	Rating	2.6
RCPS: K6.14 Starting requirements	QREV	6

Question 14
The Unit 1 Reactor Operator is making preparations to start Reactor Coolant Pump (RCP) 1-03. Note the switch and panel below.

The Reactor Coolant Pump

A. Will NOT start because failure to accelerate protection is not enabled

B. Will start because failure to accelerate protection is enabled

C. Will start because the oil pressure interlock is met

D. Will NOT start because the oil pressure interlock is not met

Answer: D

Explanation:
A is wrong because this is not an interlock for starting and the panel on the right is just a distracter panel
B is wrong because this is not an interlock for starting and the panel on the right is just a distracter panel
C is wrong because the blue light is out (see D below).
D is correct because the blue light for the oil pressure interlock is out which is the 600 psig interlock requirement for oil pressure to start the pump.

Technical References:

SOP-108A, rev 12, page 18. and LO21.SYS.RC1.pptx, Rev 6/15/1999, slide 29

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.RC1.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
004 Chemical and Volume Control	Tier #	2
	Group #	1
Knowledge of the effect of a loss or malfunction on the following CVCS components: K6.17 Flow paths for emergency boration	K/A #	004 / K6.17
	Rating	4.4
	QREV	7

Question 15

Unit 1 has a reactor trip from 100% power with the following conditions:

- four control rods failed to insert
- The US has entered EOP-0.0A, "Reactor Trip or Safety Injection," and ABN-107, "Emergency Boration."

Step 1 of ABN-107 states:
"Check RWST TO CHRG PMP SUCT VLVs, 1/1-LCV-112D AND 1/1-LCV-112E – CLOSED."

For some unknown reason the RWST TO CHRG PMP SUCT VLV, 1/1-LCV-112D is OPEN and WILL NOT CLOSE.

The Response Not Obtained (RNO) column of ABN-107 directs the operator to emergency borate _________ because it is the preferred method for these plant conditions.

A. from the RWST via 1/1-LCV-112D OR 1/1-LCV-112E using Attachment 4

B. through emergency borate valve 1-8104 using Attachment 1

C. through manual emergency borate valve 1CS-8439 using Attachment 3

D. through normal boration valves 1-FCV-0110A and 1/u-FCV-0110B using Attachment 2

Answer: A
Explanation:
A is correct because according to ABN-107, step 1 if 1/1-LCV-112D and 1/1-LCV-112E are not closed, the RNO directs you to step 7 which states that Attachment 4 is the preferred method to emergency borate from the RWST. If it is open (LCV-112D) and won't close, the interlock to allow the other valves to open and borate through other normally preferred methods is not available so this is the preferred method.
B is incorrect because according to ABN-107, Attachment 1 is the preferred method if 1/1-LCV-112D and 1/1-LCV-112E are closed, and at least one Boric Acid pump is available.
C is incorrect because according to ABN-107, Attachment 3 is another method if 1/1-LCV-112D and 1/1-LCV-112E are closed, and at least one Boric Acid pump is available.
D is incorrect because according to ABN-107, Attachment 3 is another method if 1/1-LCV-112D and 1/1-LCV-112E are closed, and at least one Boric Acid pump is available.

Technical References:
ABN-107, Emergency Boration, Rev. 9, pages 4-5

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.105.OB06; LO21.SYS.CS1.OB04

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:	55.41.7	

Examination Outline Cross-Reference	Level	RO
026 (SF5 CSS) Containment Spray	Tier #	2
	Group #	1
A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment Sump Level (CFR: 41.5)	K/A #	A1.03
	Rating	3.5
	QREV	6

Question 16

A LOCA inside containment has occurred.

Per EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION, when RWST level reaches a MINIMUM of ______(1)________ level, the crew must initiate the switchover of Containment Spray suction from the RWST to the Containment Sump.

Containment Recirculation Sump Level indication uses _____(2)______ to sense level.

A. (1) 6%
(2) Differential Pressure Transmitters

B. (1) 9%
(2) RTDs

C. (1) 6%
(2) RTDs

D. (1) 9%
(2) Differential Pressure Transmitters

Answer: C

Explanation:

A is wrong the first part is correct. The second part is wrong because recirc sump level is sensed using submergible RTDs, not D/P transmitters.

B is wrong but plausible because the switchover of CT suctions to the sump does not begin until 6% RWST level, however, the 9% is significant as that is when the RWST Empty alarm will be LIT. The second part is correct

C is correct.

D is wrong but plausible because the switchover of CT suctions to the sump does not begin until 6% RWST level, however, the 9% is significant as that is when the RWST Empty alarm will be LIT. Also recirc sump level is sensed using submergible RTDs, not DP transmitters.

Adequate containment recirc sump level to support recirculation mode of operation is determined not explicitly by containment recirc level directly, but rather implicitly by RWST level. **ECCS pump** suction from the containment recirc sump is **automatically** aligned at the RWST 2 OF 4 LVL LO-LO setpoint (33%), and **Containment Spray Pump** suction from the containment recirc pump is **manually** aligned shortly following actuation of the RWST EMPTY alarm (alarm setpoint 9%, procedural requirement to swap suction is 6%).

Technical References:
EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION, REV 9, Attachment 3, Bases

References to be provided to applicants during exam: None.

Learning Objective: STATE the performance and design attributes of the following Containment Spray System components, flowpaths and features: (OP51.SYS.CT1.OB02)
1. Containment Spray Pumps
3. Containment Spray Recirculation Sumps

STATE the performance and overall design criteria of the Containment Spray System. (OP51.SYS.CT1.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	4
10CFR Part 55 Content:	55.41(b)(7) 55.43	

Supporting Information (EOS-1.3A) (Containment Spray Study Guide)

Two trains of Containment Sump level indication are provided on CB-02 & 04. Each train is powered from its respective 1E 120vac power supply. The indicators (2/train) are scaled from 808' elevation to 817' foot elevation. Each level detector consists of 11 separate sensing points mounted in a vertical column, two resistance temperature detectors (RTD) are provided for each sensing point. One of the RTDs is thermally connected to a low power heating element. When the heater is energized, a temperature difference (and therefore a resistance difference) exists between the two RTDs. The magnitude of the temperature difference depends on whether the sensors are wet or dry. Once a sensor is wetted, the heated RTD cools off and the resistance difference between the two RTDs is reduced. The change in the resistance is used as a level change and is reflected on the MCB. The MCB indications make step changes as each RTD pair is covered or uncovered. Upon loss of power to the detectors, the control board indication fails low.

Examination Outline Cross-Reference	Level	RO
	Tier #	2
064 Emergency Diesel Generator	Group #	1
	K/A #	A2.18
Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.18 Consequences of premature opening of breaker under load	Rating	2.6
	QREV	7

Question 17

Given the following conditions:

- Unit 1 is performing OPT-214A, Diesel Generator Operability Test on EDG 1-01
- Shortly after closing CS-1EG1, Diesel Generator Output Breaker, an 86-2 relay, phase-to-ground fault has picked up
- Immediately following, a station blackout occurs
- While investigating the 86-2 Lockout Relay, the US asks you to verify EDG 1-01 is running

Per ABN-602, Response to a 6900/480V System Malfunction you would expect to…

A. observe the emergency diesel generator shutting down and place DG EMER STOP/START handswitch to PULL-OUT.

B. observe CS-1EG1 in the open position and subsequently close CS-1EG1.

C. observe CS-1EG1 in the closed position with the diesel still running.

D. observe the emergency diesel generator running and place DG EMER STOP/START handswitch to PULL-OUT.

Answer: C
Explanation:
A is wrong because this would be performed if the diesel had not started. Since the diesel was started and already running you wouldn't need to shutdown the EDG
B is wrong because after the time delay the output breaker would be expected to be closed since the 86-2 relay actuated and is overridden by the emergency start sign that is in due to the station blackout.
C is correct because with the 86-2 relay actuated the emergency start signal would override the relay and the breaker would close back in.
D is wrong because of part 2 (you would not shut it down). Plausible if someone thinks they need to shutdown the diesel due to the 86 relay.

Technical References:
Opt-214A, Diesel Generator Operability Test, Rev 22, page 17
ECA 0.0, Loss of all AC Power, Rev 9, page 6

Emergency Diesel Generator Study Guide, page 100

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.ED1.OB22, EXPLAIN the instrumentation and controls of the Emergency Diesel Generator system and PREDICT the system response in accordance with DBD-ME-011.

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)(8)	

Examination Outline Cross-Reference	Level	RO
006 (SF2; SF3 ECCS) Emergency Core Cooling	Tier #	2
	Group #	1
Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: K4.13 Reset of containment isolation	K/A #	K4.13
	Rating	3.8
	QREV	6

Question 18

Given the following conditions:

- A LOCA has resulted in both a Phase A and Phase B Containment Isolation.
- RCS pressure is currently 65 psig.
- Containment pressure is currently 30 psig.

Which of the following describes what will happen if both the Phase A and Phase B Containment Isolation signals are attempted to be reset?

A. NO Phase A and Phase B valves would be capable of being repositioned

B. ONLY Phase A valves would be capable of being repositioned

C. ONLY Phase B valves would be capable of being repositioned

D. BOTH Phase A and Phase B valves would be capable of being repositioned

Answer: D

Explanation:
A is wrong. Plausible since containment pressure is still above the setpoints where Phase A and Phase B isolations occur, but controlled action by the operators is permitted to restore systems to a configuration to allow plant recovery.
B is wrong. Plausible since Phase A valves would be capable of being repositioned, but Phase B valves would also be capable of being repositioned.
C is wrong. Plausible since Phase B valves would be capable of being repositioned, but Phase A valves would also be capable of being repositioned.
D is correct. Valves would be capable of being reset to allow the operators to restore systems to a configuration required to allow plant recovery even with pressure above the Phase A and Phase B setpoints....

Technical References:
7247D05, Sheet 1 (Rev CP-1) and Sheet 8, (Rev CP-2).

References to be provided to applicants during exam: None.

Learning Objective: SYS.ES1.OB04.002.

Question Source:	Bank #	X (23462)
(note changes; attach parent)	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
022 Containment Cooling (SF5)	Tier #	1
	Group #	1
2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions	K/A #	G2.2.44
	Rating	4.2
	QREV	7

Question 19

Given the following conditions:

- Unit 1 at 100% power
- A containment recirculation fan is taken to start from the main control room in accordance with SOP-801A, CONTAINMENT VENTILATION SYSTEM

Per SOP-801A, what does the operator expect to see and do in order to verify proper operation of the system with respect to the chill water supply and return valves for the selected unit/fan?

A. The supply valve is verified to be open; the return valve must be manually opened once the fan starts

B. The supply valve is verified to be open ; the return valve automatically opens once the fan is started

C. The supply valve automatically opens once the fan is started; the return valve is verified to be open

D. The supply valve must be manually opened once the fan starts; the return valve is verified to be open

Answer: B
Explanation:
A is wrong. First part is correct. Second part is incorrect. Plausible if you don't remember that the return valve auto opens once the fan starts.

B is correct; In accordance with SOP-801-A, section 5, the supply valve is expected to be open but has an "ensure" with the step and the return valve auto opens once the fan starts.

C is wrong. Both parts are incorrect but plausible if you get the two valves backwards (see B above).

D is wrong; Both parts are incorrect but plausible if you get the two valves backwards (see B above).

Technical References:
SOP-801-A, Revision 14, Page 11-12

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CL1.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)5	

Examination Outline Cross-Reference	Level	RO
064 (SF6 EDG) Emergency Diesel Generator	Tier #	2
	Group #	1
2.2.12 Knowledge of surveillance procedures.	K/A #	2.2.12
(CFR 41.10 / 45.13)	Rating	3.7
	QREV	6

Question 20

Per the definition of “FAST START” contained in OPT-214B, “Diesel Generator Operability Test,” this type of surveillance test is normally performed:

A. Every month or 31 days

B. Every six months or 184 days

C. Every eighteen months or before/after the outage

D. Every thirty-six months or every other outage per machine

Answer: B

Explanation:

A is wrong because the definition in OPT-214B states that the FAST start is normally done every six months.
Also, the monthly EDG start / load / run surveillance testing is allowed to be performed in “SLOW START” Mode (SR 3.8.1.2 / 3.8.1.3). The semi-annual surveillance run (SR 3.8.1.7) is the most limiting FAST START requirement.

B is correct. The definition in OPT-214B states that the FAST start is normally done every six months. No allowance for SLOW START is made for the semi-annual surveillance run (SR 3.8.1.7)

C is wrong but credible because there are other tests that get encompassed with this test. The various 18 month FAST START surveillance tests are less limiting than the SR 3.8.1.7 semi-annual test.

D is wrong but credible because the various 18 month / STAGGERED FAST START surveillance tests are less limiting than the SR 3.8.1.7 semi-annual test. 36 months is plausible because the majority of 18 month surveillances are performed on a STAGGERED TEST BASIS, meaning that an individual EDG is tested every 36 months (hence the inclusion of “EACH” at the start of the stem).

Technical References:
OPT-214B, “Diesel Generator Operability Test,” Rev 16, page 3.
Emergency Diesel Generator Study Guide, 5-2-2011, page 76
Technical Specifications 3.8.1.2 and 3.8.1.7, Amendment 156
Technical Requirements Manual Table 15.5.21-1, Revision 87

References to be provided to applicants during exam: None.

Learning Objective: APPLY the administrative requirements of the Emergency Diesel Generator system including Technical Specifications, TRM and ODCM. (LO21.SYS.ED1.OB24)

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) 55.43	

Supporting Information:

The "traditional" full cold start used to be performed every month. With this modification in place, the test is now only performed on each diesel once every 6 months, reducing the required number of periodic high stress test starts from 12 per year down to just 2 per year. In place of the traditional monthly test, we now perform a "slow" start of each engine for 5 out of every 6 months.

When the two position fast/slow start keyswitches are placed in SLOW, solenoids operate to isolate boost air to the governor boost cylinder. The keyswitches also change the 701 digital governor to operate in a slow start mode. With the keyswitches in SLOW, the engine can be started with either a normal or emergency start signal. The start signals will act in the same manner as before but the engine will not immediately ramp to 450 rpm. Instead, engine speed will stabilize at approximately 215 rpm and then to ramp slowly on to 450 rpm over approximately 30 seconds. More detail is provided in the section on governor operation.

Keyswitch contacts in-line with the 9B (and 9A, Channel 1) field flash solenoids open in SLOW to delay field flash. In this arrangement the field will not flash until engine speed exceeds 425 rpm. At that time, relay 1-HX-3419-9 (and 1-HX-3413-9, Channel 1) will actuate, energizing the field flash relays.

Examination Outline Cross-Reference		Level	RO
059 (SF4S MFW) Main Feedwater		Tier #	2
		Group #	1
Knowledge of the effect that a loss or malfunction of the MFW will have on the following: K3.02 AFW system		K/A #	K3.02
		Rating	3.6
		QREV	6

Question 21

While operating at 6% power, the only running Main Feed Water Pump trips on Unit 2. Steam Generator levels are as follows:

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

Which of the following describes 1) The response of the Auxiliary Feed Water System to the stated conditions and 2) If sufficient Auxiliary Feedwater make-up capability exists to restore Steam Generator Water levels?

A. 1) ONLY the Motor Driven Auxiliary Feed Water Pumps automatically start
2) Sufficient AFW make-up capability DOES exist to restore SG water levels

B. 1) BOTH the Motor Driven AND Turbine Driven Auxiliary Feed Water Pumps automatically start
2) Sufficient AFW make-up capability DOES exist to restore SG water levels

C. 1) BOTH the Motor Driven AND Turbine Driven Auxiliary Feed Water Pumps automatically start
2) Sufficient AFW make-up capability DOES NOT exist to restore SG water levels

D. 1) ONLY the Motor Driven Auxiliary Feed Water Pumps automatically start
2) Sufficient AFW make-up capability DOES NOT exist to restore SG water levels

Answer: B

Explanation:
A is wrong. Part 1 is incorrect because ALL three start. Plausible because both MFPs will be tripped as the other MFP must have its trip oil pressure switches isolated per TSs. Usually both MFPs tripping will cause just the MD AFW pumps to start however in this case 2 of 4 SGs are below 35.4% which will cause the TDAFWP to start also. Part 2 is correct because per ABN-403 AFW pumps can supply approximately 6% reactor power.
B is correct. Part 1 is correct because on a trip of both MFW pumps and with SG level (2/4) in at at least two SGs below 35.4%. ALL AFW pumps receive an autostart signal. The design of the AFW system is to ensure the system is capable of supplying the minimum required flow to at least two SGs. Part 2 is correct because per ABN-403 AFW pumps can supply approximately 6% reactor power.

C is wrong. Part 1 is correct, see B above. Part 2 is incorrect but plausible because if power were 1% higher sufficient AFW make-up capability would not exist to restore SG water levels.
D is wrong. Part 1 is incorrect but plausible, see A above. Pat 2 is incorrect but plausible, see C above

Technical References:
L021.SYS.AF1.ppt, rev 7/11/2017, slide 25 and 38.
ABN-403, rev 10, p 4

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.AF1.OB04.

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)7	

Examination Outline Cross-Reference	Level: RO
026 Containment Spray	Tier #: 2
	Group #: 1
Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following:	K/A #: K4.05
	Rating: 2.8
K4.05 Prevention of material from clogging nozzles during recirculation	QREV: 7

Question 22

Which of the following accurately describes a design feature of the Unit 1 containment spray recirculation system (sump) installed to reduce the possibility of clogging the spray nozzles when in the recirculation mode?

A. a trash rack on three sides of the sump

B. sump strainer orifice openings at 1.15 inches

C. a large sump surface area (8000 square feet)

D. both fine and course strainers in the sump

Answer: A
Explanation:

A is correct. Installed a trach rack on three sides of the sump..
B is wrong because the sump strainer hole size is 0.115 inches. A 1.15 inch hole would be way too big to prevent clogging.
C is wrong because 8000 is double the square footage for a single unit.
D is wrong because .the fine and course screens were removed and just a single strainer was used with roughly 0.115 inch holes.

Technical References:
LO21.SYS.CT1.PPT, Revision 7/30/15, slides 18 and 52.

References to be provided to applicants during exam: None.

Learning Objective: DESCRIBE the components of the Containment Spray system including interrelations with other systems to include interlocks and control loops in accordance with the FSAR and DBD-ME-0232. (LO21.SYS.CT1.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	Level	RO
	Tier #	2
005 Residual Heat Removal	Group #	1
	K/A #	A2.04
Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.04 RHR valve malfunction	Rating	2.9
	QREV	7

Question 23

Given the following conditions for Unit 2:

- RHR pump 2-01 is out of service unexpectedly due to maintenance issues.
- Unit 2 is in Mode 5 with reduced inventory.
- Level is 55 inches above the core plate as read on LI-3615B, RX VSL LVL (WR)
- RHR pump 2-02 is in service WHEN the flow control valve 2-HC-607, RHR HX 2 FLO CTRL, loses control power.

(1) What would be the expected FINAL position of 2-HC-607, RHR HX 2 FLO CTRL valve?

(2) Per Attachment 12 of IPO-010B, what can be done to prevent cavitation of the running RHR pump 2-02 for these plant conditions?

A. (1) Fails AS IS.
(2) Provide alternate power to HC-607 using a shared junction box with 2-FK-619 and use controls at the Remote Shutdown Panel.

B. (1) Fails AS IS.
(2) Throttle the 1/2-8809B valve using the power switch and its associated open/close switch in the Control Room.

C. (1) Fails OPEN.
(2) Provide alternate power to HC-607 using a shared junction box with 2-FK-619 and use controls at the Remote Shutdown Panel.

D. (1) Fails OPEN.
(2) Throttle the 1/2-8809B valve using the power switch and its associated open/close switch in the Control Room.

Answer: D
Explanation:
A is wrong because part 1 and part 2 are incorrect.
B is wrong because part 1 is incorrect.
C is wrong because part 2 is incorrect.
D is correct because according to the system training slides, HC-606 fails open on loss of control power, Attachment 12 of IPO-010B states that to prevent cavitation while at reduced inventory, the 8809 valve can be throttled using its power button and the open/closed switch for the valve.

Technical References:
AIPO-010B, Rev 14, page 152
LO21.SYS.SRH1, revision 8/28/2017, slide 38.

References to be provided to applicants during exam: None.

Learning Objective:
EXPLAIN the instrumentation and controls of the Residual Heat Removal system and PREDICT the system response in accordance with DBD-ME-260. (LO21.SYS.RH1.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
061 Auxiliary / Emergency Feedwater (AFW) System	Tier #	1
	Group #	1
	K/A #	K6.02
Knowledge of the operational implications of the following concepts as the apply to the AFW: K6.02 Pumps	Rating	2.6
	QREV	7

Question 24

Given the following Unit 1 plant conditions:

- Reactor tripped
- A complete Feedwater Isolation has occurred
- TDAFWP is tagged out for maintenance
- MDAFWP 1-02 trips shortly after starting

Per ABN-305, AUXILIARY FEEDWATER SYSTEM MALFUNCTION,…

1) a "CAUTION" states, placing 1-HS-2451A, MD AFWP 2 in the STOP or PULL-OUT position will reset the ____(1)_____ relay and may result in an automatic restart if the handswitch is returned to AUTO.

2) MDAFWP 1-01 flow shall NOT exceed ____(2)_____.

A. (1) 50/51 overcurrent
(2) 600 gpm

B. (1) 86M lockout
(2) 600 gpm

C. (1) 50/51 overcurrent
(2) 800 gpm

D. (1) 86M lockout
(2) 800 gpm

Answer: D
Explanation:
A is wrong. Part 1 is incorrect but plausible because this pump does have a 50/51 overcurrent relay, however, in order to reset this relay it must be performed locally at the breaker. Part 2 is incorrect but plausible because 600 gpm is a reasonable number greater than the minimum for heat removal but less than the 800 gpm for runout.
B is wrong. Part 1 is correct, per ABN-305 if the handswitch is taken to STOP or PULL-OUT and then returned to the AUTO position the 86M relay will reset and could result in an automatic pump re-start if there are no other dropped relays on the breaker. Part 2 is incorrect but plausible, see A above.
C is wrong. Part 1 is incorrect but plausible, see A above. Part 2 is correct, per ABN-305 flow is not to exceed 800 gpm when a single MDAFWP is running to prevent a runout condition..

D is correct. Part 1 is correct, see B above. Part 2 is correct see C above.

Technical References:
LO21SYSAF1, Auxiliary Feedwater, Revision date 6/8/17, page 13; ABN-305, Auxiliary Feedwater System Malfunction, Revision 8, page 47-48

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.AF1.OB04; LO21.ABN.305.OB03

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:	55.41.7	

Examination Outline Cross-Reference		Level	RO
003 (SF4P RCP) Reactor Coolant Pump		Tier #	2
		Group #	1
K3.03 Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: Feedwater and emergency feedwater (CFR: 41.7)		K/A #	K3.03
		Rating	2.8
		QREV	7

Question 25

[REFERENCE PROVIDED]

The crew is performing actions of EOS-0.2A, NATURAL CIRCULATION COOLDOWN, following a loss of offsite power to Unit 1.

At Time = 0, a loss of offsite power occurs and the following RCS Loop Delta T readings are recorded below:

- Loop 1: 17 °F
- Loop 2: 17 °F
- Loop 3: 25 °F
- Loop 4: 25 °F

At Time = 10 minutes, all feedwater capability is lost to S/G #1.

(1) What is the APPROXIMATE maximum cooldown rate allowed by EOS-0.2A at T = 0?
(2) What is the APPROXIMATE maximum cooldown rate allowed by EOS-0.2A at T= 10 minutes?

A. (1) 50 °F/ hr
(2) 20 °F/ hr

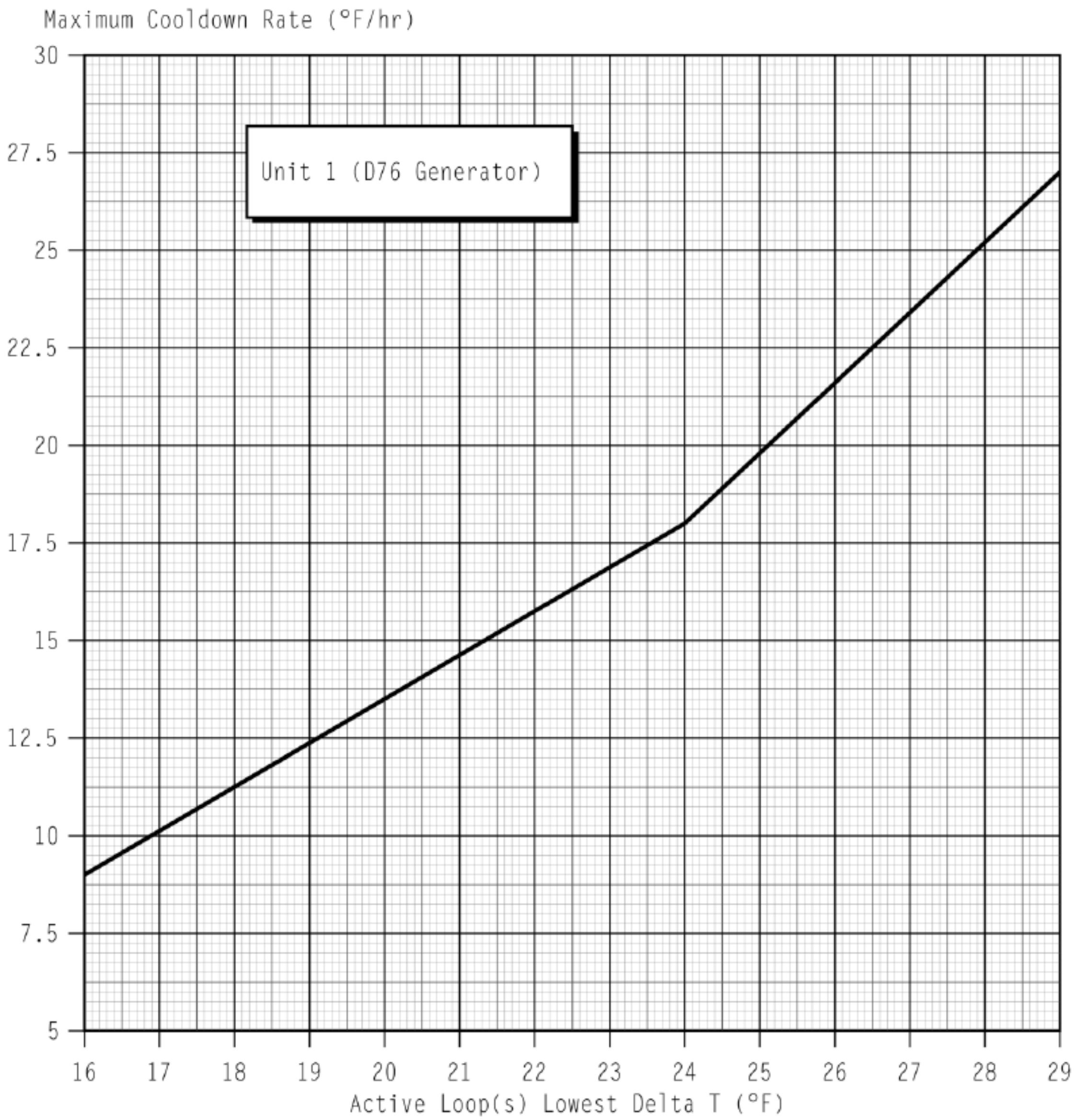
B. (1) 10 °F/ hr
(2) 5 °F/ hr

C. (1) 50 °F/ hr
(2) 10 °F/ hr

D. (1) 20 °F/ hr
(2) 10 °F/ hr

ATTACHMENT 5
PAGE 1 OF 1

COOLDOWN RATE AS A FUNCTION OF DECAY HEAT/ACTIVE LOOP ΔT



Answer: C
Explanation:

This question tests the applicant's understanding of the relationship between feedwater availability and procedurally-allowed cooldown rates following a loss of all RCPs (natural circulation cooldown).

An "inactive RCS loop" exists if either the capability to feed the respective SG or the capability to release steam from the respective SG is not available. [EOS-0.2A Step 8 Note]

If all RCS loops are "active", the maximum permissible cooldown rate is approximately 50°F/HR [EOS-0.2A Step 8.b]

If any RCS loop is inactive, maintain cooldown rate in RCS cold legs less than maximum allowable limit of EOS-0.2A attachment 5. [EOS-0.2A Step 8.RNO.a).2)]

An inactive RCS loop is considered to be experiencing "stagnation" if all RCS hot leg temperatures are NOT all decreasing at same rate. In this case, it is required to decrease cooldown rate by a factor of two [EOS-0.2A Step 9.RNO.c)1)]. No stagnation conditions are given in this question.

In part (1) of this question, all RCS loops are initially considered to be active, therefore the maximum allowed cooldown rate is approximately 50 degrees F.

In part (2) of this question, RCS loop 1 becomes inactive due to a loss of all feedwater capability, therefore the cooldown rate limits of attachment 5 apply. The cooldown rate is determined by the lowest loop Delta T, which is 17 degrees F. Using attachment 5, this correlates to a maximum allowed cooldown rate of approximately 10 degrees F / hr.

A is wrong because although part (1) gives the correct initial cooldown rate based on having all loops active, it incorrectly uses the cooldown rate for the highest Delta T (25F) instead of lowest Delta T (17F) to calculate part (2)

B is wrong because it incorrectly uses Att 5 to determine the initial cooldown rate in part (1), when all loops are active. Part (2) incorrectly applies the ½ requirement reserved for a stagnant loop.

C is correct. With all 4 loops initially active, the initial cooldown rate is limited to 50F. When 1 loop becomes inactive, Att 5 is used to calculate the new cooldown rate based on the lowest loop Delta T, 17F.

D is wrong because it incorrectly uses Att 5 to calculate initial cooldown rate, and incorrectly uses the highest loop Delta T, 25F, to do so. It then incorrectly applies the ½ requirement to calculate the cooldown rate in part 2.

Technical References:
EOS-0.2A, NATURAL CIRCULATION COOLDOWN , Rev 9, steps 8, 9, and Att. 5.

References to be provided to applicants during exam: EOS-0.2A, NATURAL CIRCULATION COOLDOWN , Rev 9, Attachment 5.

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	4
10CFR Part 55 Content:	55.41(b)(10)	
	55.43	

Examination Outline Cross-Reference	Level	RO
103 (SF5 CNT) Containment	Tier #	2
	Group #	1
Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: K1.02 Containment isolation/containment integrity	K/A #	K1.02
	Rating	3.9
	QREV	7

Question 26

Containment penetrations associated with Phase A isolation valves are one of several types of arrangements.

The Phase A CVCS isolation penetration is which type?

A. Two valve isolation arrangement where ONE valve isolates the non-essential portion of the system

B. One valve isolation arrangement with a closed system

C. Two valve isolation arrangement with BOTH an inside and outside containment isolation valve

D. One valve isolation arrangement with an interfacing system isolation

Answer: C
Explanation:
A is wrong but plausible if you don't know the CVCS configuration.
B is wrong (see C below) but plausible if you don't know the CVCS configuration.
C is correct because there is an inside and outside valve (HS-8152 and HS-8160, respectively).
D is wrong but plausible if thought the isolation valve inside containment was an RCS valve as the interfacing system and the valve outside containment was a CVCS valve (i.e. the system switches from RCS to CVCS at the containment boundary)

Technical References:
EOP-0.0A, rev 9, page 111.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E0A.OB04

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)9	

Examination Outline Cross-Reference	Level	RO
013 Engineered Safety Features Actuation	Tier #	2
	Group #	1
Knowledge of the physical connections and/or cause-effect relationships between the ESFAS and the following systems:	K/A #	K1.15
	Rating	3.4
	QREV	6

K1.15 MFW System

Question 27

The following conditions exist on Unit 1 due to main feedwater malfunction:

- SG 1-01 level 44% and increasing at 0.40% per minute
- SG 1-02 level 48% and increasing at 0.38% per minute
- SG 1-03 level 42% and increasing at 0.45% per minute
- SG 1-04 level 39% and increasing at 0.47% per minute

With NO operator action, which of the following is the MAXIMUM amount of time that can elapse BEFORE a feedwater isolation should occur?

A. 85.0 minutes

B. 88.0 minutes

C. 91.0 minutes

D. 94.0 minutes

Answer: C
Explanation:
A is wrong because this would be correct for Unit 2 (MFI at 81.5 % in one SG)
B is wrong but would be correct for Unit 2 IF hi/hi level was required in 2/4 steam generators
C is correct (MFI at 84% in one SG)
D is wrong but would be correct for Unit 1 IF hi/hi level was required in 2/4 steam generators
Calculations
84% level is the target where a MFI will occur.
For SG 1-01 84% - 44% = 40% and 40% divided by 0.40% per minute = 100 minutes
For SG 1-02 84% - 48% = 36% and 36% divided by 0.38% per minute = 94.7 minutes
For SG 1-03 84% - 42% = 42% and 42% divided by 0.45% per minute = 93.3 minutes
For SG 1-04 84% - 39% = 45% and 45% divided by 0.47% per minute = 95.7 minutes
Therefore 91 minutes is the maximum time that can elapse before the feedwater isolation occurs at 93.3 minutes with SG 1-03, therefore C is correct.

Technical References:
LO21.SYS.MF1, Main Feedwater Lesson Plan, Revision 5/25/2017, Page 24

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.MF1.OB03.

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)7	

Examination Outline Cross-Reference	Level	RO
073 (SF7 PRM) Process Radiation Monitoring	Tier #	2
	Group #	1
Knowledge of the physical connections and/or cause-effect relationships between the PRM system and the following systems: K1.01 Those systems served by PRMs	K/A #	K1.01
	Rating	3.6
	QREV	7

Question 28

Unit 1 is at full power when the following occurs:

- 1-RE-0406, Gross Failed Fuel monitor alarms.
- The US enters the abnormal procedure for this activity
- Chemistry reports a sample that is high in Co-60 and Cr-51 and it has been increasing steadily over the past 24 hours to its current state.

Which of the following actions should be taken?

A. Restart the sample pump for Radiation Monitor 1-RE-0406 in order to establish continuous monitoring

B. Start the PDP and secure any CCPs that are running in order to flush the letdown line

C. Do not move any control rods in order to reduce the potential for CRDM mis-stepping from a crud burst

D. Establish more letdown flow to increase the removal of fuel failure isotopes

Answer: C

Explanation:

Cause-effect part of KA met by knowing the isotope is causing the alarm and specifically a crud burst isotope. Physical connection is part of KA by knowing there is no sample pump and that flow is off of letdown line with a stop valve.

A is wrong because the sample flow is off of the letdown line and only stops if you close the inlet valve or flow is too low in solid plant operations.

B is wrong because you do the opposite, secure the PDP and start CCPs. Plausible if you get these two pumps backwards.

C is correct because these are isotopes from a crud burst type of event, and CP has OPEx on CRDM mis-stepping. ABN-102 directs the minimization of tripping rods or moving them due to this risk.…

D is wrong because you do increase letdown flow but not remove failed fuel isotopes inthis case (these are corrosion products). Plausible if you don’t know isotopes for failed fuel versus crud burst.

Technical References:

ABN-102, rev 8, page 4-5.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.103.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)9	

Examination Outline Cross-Reference	Level	RO
	Tier #	2
011 Pressurizer Level Control	Group #	2
	K/A #	K5.06
Knowledge of the operational implications of the following concepts as they apply to the PZR LCS: K5.06 Indicated charging flow: seal flow plus actual charging flow	Rating	2.9
	QREV	6

Question 29

Unit 2 is at full power. Letdown flow is 110 gpm.

RCP	Seal Injection flow (GPM)	Seal Leakoff Flow (GPM)
2-01	9	3
2-02	7	2.5
2-03	8	2
2-04	7	3.5

2-FI-121A, CHRG FLO should be set at _____ gpm to maintain pressurizer level constant.

A. 99

B. 121

C. 140

D. 151

Answer: B

Explanation:
A is wrong because see B. Plausible if someone thinks they need to set charging to letdown minus seal leakoff.
B is correct because charging needs to be set at letdown plus seal leakoff to maintain constant pressurizer level.
C is wrong because see B. Plausible if someone thinks they need to add injection and subtract leakoff.
D is wrong because see C. Plausible if someone thinks they need to set charging to injection plus letdown.

Technical References:
CVCS Study Guide, p 59, rev 4/28/2011

References to be provided to applicants during exam: None.

Learning Objective:
DISCUSS the Pressurizer Level Control system to include:
1) Inputs to the level control system
2) Effects of Pressurizer level on CCP flow and PDP speed

(LO21.SST.CS1.OB04)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X (ILOT6346)
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		Level	RO
014 Rod Position Indication		Tier #	2
		Group #	2
Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:		K/A #	A2.07
		Rating	2.6
		QREV	6

A2.06 Loss of LVDT

Question 30

Given the following conditions:

- Unit 1 is at 100% power.
- The 120 VAC Distribution System is in the Normal alignment.
- A failure of 120/208 VAC DISTRIBUTION PANEL 1C1 (CP1-ECDPNC-02) occurs.

What is the effect on the Digital Rod Position Indication (DRPI) system and the procedural action necessary?

A. DRPI is in Half-Accuracy. Energize Train C 120 VAC Distribution Panel 1C14 from CP1-ECDPNC-03, 120/208 VAC DISTRIBUTION PANEL 1C4.

B. Loss of DRPI. Energize Train C 120 VAC Distribution Panel 1C14 from CP1-ECDPNC-03, 120/208 VAC DISTRIBUTION PANEL 1C4.

C. DRPI is in Half-Accuracy. Energize Train C 120 VAC Distribution Panel 1C14 from CPX-EPDPNB-03 120/208 VAC MISCELLANEOUS POWER PANEL XC4-4.

D. Loss of DRPI. Energize Train C 120 VAC Distribution Panel 1C14 from CPX-EPDPNB-03 120/208 VAC MISCELLANEOUS POWER PANEL XC4-4.

Answer: B

Explanation:
A is plausible if believed that Data A was powered from 1C1 and Data B was powered from 1C14 during normal alignment, however all of DRPI is powered from 1C14 which is normally powered from 1C1.
B is correct. As DRPI is powered from 1C14, which is normally aligned from 1C1, a loss of 1C1 would result in a loss of DRPI. In accordance with ABN-712, the power supply should be shifted to 1C4.
C is plausible if believed that Data A was powered from 1C1 and Data B was powered from 1C14 during normal alignment, however all of DRPI is powered from 1C14 which is normally powered from 1C1. Further XC4-4 could be thought to be the alternate power supply to 1C14.
D is plausible as this is the correct response of DRPI, however, this is the incorrect alternate power supply and XC4-4 could be thought to be the alternate power supply to 1C14.

Technical References:
SOP-608A, 120VAC Distribution System, Revision 12, Sections 5.13 and 5.14; ABN-712 Rod Control System Malfunction, Revision 11, pages 30-31

References to be provided to applicants during exam: None.

Learning Objective: LO21.SST.ROD.OB04.

Question Source:	Bank #	From 2013 Exam
(note changes; attach parent)	Modified Bank #	
	New	
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	Level	RO
086 Fire Protection	Tier #	2
	Group #	2
Ability to monitor automatic operation of the	K/A #	A3.02
Fire Protection System including: A3.02	Rating	2.9
Actuation of the FPS	QREV	6

Question 31

Unit 1 Main Fire Detection Board Safeguards Building Panel, Window 4.2 – 810 SWGR RM TRN A actuated at CV-06.

The Fire Pre-Plan Instruction (FPI) number to correctly address this alarm can be found utilizing ____1____.

The Abnormal Operating Procedure (ABN) to correctly address this alarm can be found utilizing ____2____.

A. 1. the job aid at panel CV-06
   2. Attachment 4, Unit 1 Main Fire Detection Board Safeguards Building Panel

B. 1. Attachment 4, Unit 1 Main Fire Detection Board Safeguards Building Panel
   2. the job aid at panel CV-06

C. 1. Attachment 4, Unit 1 Main Fire Detection Board Safeguards Building Panel
   2. Attachment 4, Unit 1 Main Fire Detection Board Safeguards Building Panel

D. 1. the job aid at panel CV-06
   2. the job aid at panel CV-06

Answer: C
Explanation:
A is wrong because Part 1 is wrong. The specific FPI is located in the specific Attachment of ABN-901 that covers the alarm window. Plausible if confused over what the job aid has or the location of all the items on each alarm window.
B is wrong because Part 2 is wrong. The specific ABN is located in the specific Attachment of ABN-901 that covers the alarm window. Plausible if confused over what the job aid has or the location of all the items on each alarm window.
C is correct because the FPI and ABN cannot be found on the job aid. The specific FPI and ABN are located in the specific Attachment of ABN-901 that covers the alarm window.
D is wrong both parts are wrong. Plausible if confused over what the job aid contains or location of items. The specific FPI and ABN are located in the specific Attachment of ABN-901 that covers the alarm window.

Technical References:
ABN-901, rev 9, page 12.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.FP1.OB04

Question Source:	Bank #	
(note changes; attach parent)	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	Level	RO
016 (SF7 NNI) Nonnuclear Instrumentation	Tier #	2
	Group #	2
K4.01 Knowledge of NNIS design feature(s) and/or interlock(s) which provide for the following: Reading of NNIS channel values outside control room (CFR: 41.7)	K/A #	K4.01
	Rating	2.8
	QREV	6

Question 32

Which of the following plant parameters used to verify Natural Circulation is/are DIRECTLY indicated on BOTH the Main Control Boards AND Remote Shutdown Panel?

A. Core Exit Thermocouple Temperatures, RCS Hot and Cold Leg Temperatures, SG Pressures ONLY.

B. RCS Subcooling, RCS Hot and Cold Leg Temperatures, SG Pressures ONLY.

C. RCS Hot and Cold Leg Temperatures, SG Pressures ONLY.

D. RCS Subcooling, SG Pressures ONLY.

Answer: C

Explanation:

- CET Temp is available only on the MCB, not on the RSP.
- RCS Subcooling is not directly indicated on the RSP, it must be manually calculated using indicated P-PZR, RCS loop temps, and steam tables.
- RCS Hot and Cold Leg temperatures ARE indicated on the RSP, although they use strap-on RTDs instead of thermowell-embedded RTDs (MCB NR Cold Leg and WR Cold and Hot Leg) or N-16 monitoring (MCB NR Hot Leg).
- SG Pressure is indicated on both MCB and RSP; it is also the only parameter that uses the same method of detection for both MCB and RSP indication.

A is wrong because CETs are not available DIRECTLY at RSP

B is wrong because RCS Subcooling is not available DIRECTLY at RSP.

C is correct because RCS Hot and Cold Leg Temps (via strap-on RTDs) and SG pressures are available at both MCB and RSP.

D is wrong because RCS Hot and Cold Leg Temperatures are also directly indicated on the RSP via strap-on RTDs (instead of thermowell-embedded RTDs or N-16 monitors). Also, RCS Subcooling is not available on the RSP.

Technical References:
ABN-905A, Rev 9, Att 1, Instrumentation and Controls available at the RSP

ABN-905A, Rev 9, Att 3, Natural Circulation Verification
EOS-0.1A Rev 9, Attachment 3, Natural Circulation Verification
Reactor Coolant System Study Guide, 4-28-2011, Pages 20-23
Main Steam Study Guide, 6-9-2011, page 43

References to be provided to applicants during exam: None.

Learning Objective: 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:	Bank #	
(note changes; attach parent)	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	

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 9	PAGE 25 OF 40

ATTACHMENT 3
PAGE 1 OF 1

NATURAL CIRCULATION VERIFICATION

The following conditions support or indicate natural circulation flow:

- ☐ RCS subcooling - GREATER THAN 25°F.
- ☐ SG pressures - STABLE OR DECREASING.
- ☐ RCS hot leg temperatures - STABLE OR DECREASING.
- ☐ Core exit TCs - STABLE OR DECREASING.
- ☐ RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR SG PRESSURE

FROM MAIN CONTROL BOARDS

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 2	PROCEDURE NO. ABN-905B
LOSS OF CONTROL ROOM HABITABILITY	REVISION NO. 4	PAGE 40 OF 74

ATTACHMENT 3
PAGE 1 OF 1

NATURAL CIRCULATION VERIFICATION

The following conditions support or indicate natural circulation flow:

- RCS Subcooling - GREATER THAN 25°F
- SG pressures - STABLE OR DECREASING
- RCS hot leg temperatures - STABLE OR DECREASING
- RCS cold leg temperature - AT SATURATION TEMPERATURE FOR SG PRESSURE

FROM REMOTE SHUTDOWN PANEL

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-905A
LOSS OF CONTROL ROOM HABITABILITY	REVISION NO. 9	PAGE 46 OF 74

ATTACHMENT 7
PAGE 1 OF 2

RCS PRESSURE/TEMPERATURE VERIFICATION

Time									
PRZR PRESS	1-PI-455B								
Tsat from Steam Table (2)									
PRZR LVL	1-LI-459B								
NEUT FLUX SR	1-NI-50A-3								
RCS LOOP (4)	CL1								
1 & 2 TEMP	CL2								
1-TR-410F	HL1								
	HL2								
Calculated Subcooling °F									
SG 1 PRESS (2)	1-PI-514B								

CALCULATED SUBCOOLING MARGIN

Cold and Hot Leg Wide Range Temperatures

RTDs, located in thermowells in the cold and hot leg piping of each loop, provide wide range temperature indication over a range of 0-700°F. Operators use this information, during plant heatup and cooldown, to control RCS temperature. These instruments provide input signals to the RCS Low Temperature Overpressure Protection (LTOP) System. Operators monitor wide range temperatures for each loop via indicating recorders at CB05 and plant computer scan points. These instruments are a part of the Post Accident Monitoring System (PAMS). PAMS instrumentation is identified by labeling having a black background and white lettering.

Strap-on RTDs

Strap-on RTDs are located in proximity to inline RTDs, on each cold and hot leg, for cross calibration. These instruments input to temperature recorders located on the remote shutdown panel, providing the only indication for each loops' hot and cold leg temperature at that location.

Steam Pressure Transmitters

Each Main Steam line has at least five steam pressure transmitters. Loop 1 and 2 have one additional steam pressure transmitter each for RSP indication. One transmitter on each steam line is used solely to control the ARV and indication. Another transmitter on each steam line is used solely to provide calorimetric data input to the plant computer. Three channels are used for control, protection and indication. Of these three channels, two channels provide pressure compensation for the steam flow signal. Failure modes for the steam pressure transmitters are the same as the above description for steam flow transmitters.

Indications for four pressure channels per steam line are displayed on the MCB. One chart recorder displays main steam line pressure for all four SGs. Steam pressure is also indicated on the RSP. When the RSP steam pressure indication is used with the steam tables, this indication becomes a faster indication of cooldown rate than using the strap-on wide range temperature indication normally used.

Examination Outline Cross-Reference		Level	RO
033 (SF8 SFPCS) Spent Fuel Pool Cooling		Tier #	2
		Group #	2
2.1.27 Knowledge of system purpose and/or function.		K/A #	G2.1.27
		Rating	3.9
		QREV	6

Question 33

Per the Technical Specifications, one purpose of the Spent Fuel Pool Cooling System is to be able to store a maximum of __________ in BOTH pools with $\geq$ 23 ft of water to ______.

A. 3373 fuel assemblies
Absorb most of the 10% assumed iodine released from a ruptured assembly

B. 1684 fuel assemblies
Moderate enough neutrons to maintain a Keff < 0.95 to prevent a criticality event

C. 3373 fuel assemblies
Moderate enough neutrons to maintain a Keff < 0.95 to prevent a criticality event

D. 1684 fuel assemblies
Absorb most of the 10% assumed iodine released from a ruptured assembly

Answer: A

Explanation:
A is correct. This is a purpose, to store a max of 3373 fuel assemblies in both pools with at least 23 ft of water to absorb 99% of the assumed 10% iodine released from a ruptured fuel assembly.
B is wrong because the fuel assembly count is for only one pool (SFP1) and the boron actually ensures Keff < 0.95, not just the water.
C is wrong because the bundle count is correct but the reason is incorrect.
D is wrong because the first part is wrong, the second part is correct.
Technical References:
L021_SYS_SF1.pptx, rev 07/17/2017, slide 59 and 64, TS amendment 162, page 4.0-3, and TS 3.7.15 Amendment 156, page 3.7-35.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.SF1.OB01.

Question Source:	Bank #	
(note changes; attach parent)	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)13

Examination Outline Cross-Reference	Level	RO
035 (SF 4P SG) Steam Generator	Tier #	2
	Group #	2
Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: K6.03 S/G level detector	K/A #	K6.03
	Rating	2.6
	QREV	6

Question 34

Unit 2 is operating at 50% power.

2-FS-510C, SG 1 FW FLO CHAN SELECT, is in the 2-FY-510C position [NORMAL].
2-FS-512C, SG 1 STM FLO CHAN SELECT, is in the 2-FY-512B position [NORMAL].
2-LS-519C, SG 1 LVL CHAN SELECT, is in the 2-LQY-551 position [NORMAL].

Steam Generator (SG) 2-01 actual level begins to rise with SGWLC program in AUTO.

All other SG levels are stable at approximately 64%.

Which of the following is the cause of the rising level in Steam Generator 2-01?

A. 2-LT-551, SG 1 LVL (NR) CHAN I, has failed low

B. 2-PT-514A, MSL 1 PRESS CHAN I, has failed low

C. 2-FT-512A, SG 1 STM FLO CHAN I, has failed low

D. 2-FT-510A, SG 1 FW FLO CHAN I, has failed high

Answer: A

Explanation:
A is correct because the controlling level channel failing low results in an increase in feed flow to restore level to program. The increase in feed flow causes actual level in the SG to increase.
B is incorrect but plausible because this is an input to level control, but a failed low steam pressure channel causes a corresponding decrease in steam flow and a decrease in feed flow, accompanied by a decreasing actual level.
C is incorrect but plausible because this is an input to level control, but a failed low steam flow channel causes a decrease in feed flow, accompanied by a decreasing actual level.
D is incorrect but plausible because this is an input to level control, but a failed high feed flow channel causes a decrease in feed flow, accompanied by a decreasing actual level.
Technical References:
LO21.SYS.SN1, Steam Generator Water Level Control, Revision date 6/6/17, page 12

References to be provided to applicants during exam: None.

Learning Objective: SYS.SN1.OB04.002

Question Source: Bank # 23412

(note changes; attach parent)	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
	Tier #	2
041 Steam Dump/Turbine Bypass Control	Group #	2
	K/A #	A1.01
Ability to predict or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: A1.01 Tave, verification above low/low set point	Rating	2.9
	QREV	6

Question 35

Unit 1 is tripped and cooling down for a planned refueling outage in accordance with IPO-005A, PLANT COOLDOWN FROM HOTSTANDBY TO COLD SHUTDOWN.

- The balance of plant operator has PK-507 in MANUAL
- Cooldown rate is < 70 ^{0}F per hour with steam dumps in steam pressure mode
- The steam dump interlock select switches are in ON
- Temperature continues to drop until suddenly the steam dumps close
- Tave temporarily stabilizes at 552 ^{0}F

What procedural actions does the operator need to do to continue the cooldown?

Place both steam dump interlock switches to _____1______ and then reopen ___2___ dump valves.

A. OFF/RESET and back to ON
ALL

B. BYPASS INTERLOCK
ALL

C. OFF/RESET and back to ON
BANK 1

D. BYPASS INTERLOCK
BANK 1

Answer: D

Explanation:

A is wrong because taking the switches to this position would close the steam dumps. Also, because Tave is below P-12 (553F) only the group 1 dump valves will reopen at this point. Plausible if someone thinks they need to reset Lo-Lo Tave (P-12) similarly to resetting C-7 and they think that after the reset that they can reopen all dump valves.
B is wrong because the second part is wrong. Plausible if someone thinks that in bypass interlock they can reopen all dump valves.
C is wrong because taking the switches to this position would close the steam dumps. Because below P-12 the second part is correct in that only the Bank 1 valves will reopen.

Plausible if someone thinks they can open all of them after a reset of P-12 (which can't be done at this temp anyway).
D is correct because this will allow ONLY group one valves to open by design and allow the cooldown to continue.

Technical References:
IPO-0005A, rev 26, page 26; Steam Dumps Study Guide, p. 13, rev 5/4/2011.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.SD1.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)(7)	

Examination Outline Cross-Reference		Level	RO
075 (SF8 CW) Circulating Water		Tier #	2
		Group #	2
Knowledge of bus power supplies to the following: K2.03 Emergency/essential SWS pumps		K/A #	K2.03
		Rating	2.6
		QREV	6

Question 36

What are the power supplies to the Service water pumps for Unit 2 (SSWP 01 and SSWP 02, respectively)?

A. 2EA1 and 2EA2

B. 2EB1 and 2EB2

C. 2EB1 and 2EB3

D. 2A1 and 2A2

Answer: A

Explanation:
A is correct because pump 01 is off of vital bus 2EA1 and pump 02 is off of vital bus 2EA2.
B is wrong because these are the wrong buses. Plausible if don't know power supplies.
C is wrong because these are the wrong buses. Plausible if don't know power supplies.
D is wrong because these are non-safety buses of the same voltage service.

Technical References:
L021.SYS.SW1.pptx, rev 8/23/2017, slide 31.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.SW1.OB04.

Question Source:	Bank #	
(note changes; attach parent)	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	Level	RO
027 Containment Iodine Removal	Tier #	2
	Group #	2
Knowledge of the physical connections and/or cause-effect relationships between the CIRS and the following systems: K1.01 CSS	K/A #	K1.01
	Rating	3.4
	QREV	6

Question 37

In the containment spray system at Comanche Peak, _____1_____ is added to containment spray from the spray additive tank to _______2_______ pH in the containment sump to promote iodine hydrolysis during a large break LOCA.

A. 1) Tri Sodium Phosphate 2) lower

B. 1) Sodium Hydroxide 2) lower

C. 1) Tri-Sodium Phosphate 2) raise

D. 1) Sodium Hydroxide 2) raise

Answer: D

Explanation:
A is wrong but plausible, as Tri-Sodium Phosphate is used at some plants for Iodine removal, but Sodium Hydroxide is used at Comanche Peak. Applicant may confuse upper with lower on pH.
B is wrong but plausible as first part is correct, and applicant may confuse upper with lower on pH.
C is wrong but plausible, as Tri-Sodium Phosphate is used at some plants for Iodine removal, but Sodium Hydroxide is used at Comanche Peak. Second part is correct.
D is correct

Technical References:
LO21SYSCT1, Revision May 16, 2017, Page 5

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CT1.OB03.

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)8	

Examination Outline Cross-Reference	Level	RO
071 (SF9 WGS) Waste Gas Disposal	Tier #	2
	Group #	2
Ability to manually operate and/or monitor in the control room: A4.09 Waste gas release rad monitors	K/A #	A4.09
	Rating	3.3
	QREV	6

Question 38

Which of the two following Radiation Monitors will cause HCV-014, Waste Gas Discharge Control Valve to automatically close?

1. X-RE-5700 (FBV088), Fuel Building Vent Exhaust Monitor
2. X-RE-5701 (ABV089), Aux Building Vent Exhaust Monitor
3. X-RE-5567A/B (PVG384/385), Plant Vent Stack Noble Gas Monitor
4. X-RE-5570A/B (PVG684/685), Plant Vent Stack Wide Range Gas Monitor

A. 1 and 3

B. 1 and 4

C. 2 and 3

D. 2 and 4

Answer: D

Explanation:
A is incorrect but plausible because the Fuel Building Vent Exhaust Monitor (FBV088) monitors downstream of HCV-0014, but neither of these monitors will perform this action.
B is incorrect but plausible because the Fuel Building Vent Exhaust Monitor (FBV088) will not perform this action but the Plant Vent Stack Wide Range Noble Gas Monitor (PVG684/685) will automatically close HCV-0014.
C is incorrect but plausible because the Aux Building Vent Duct Radiation Monitor (ABV089) will automatically close HCV-0014, but the Plant Vent Stack Noble Gas Monitor will not perform this action.
D is correct because high radiation alarm on either the Aux Building Vent Duct Radiation Monitor (ABV089) or the Plant Vent Stack Wide Range Noble Gas Radiation Monitor (PVG684/685) will automatically close HCV-0014.

Technical References:
LO21SYSRWS, Radioactive Waste Systems, Revision date 3/6/17, page 10

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.RWS.OB04

Question Source:	Bank #	2012 NRC
(note changes; attach parent)	Modified Bank #	
	New	

Question History:	Last NRC Exam	2012
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b).7	

Examination Outline Cross-Reference	Level	RO
WE12 Uncontrolled Depressurization of all Steam Generators / 4	Tier #	1
	Group #	1
	K/A #	EK2.1
Knowledge of the interrelations between the (Uncontrolled Depressurization of all Steam Generators) and the following: EK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	Rating	3.4
	QREV	7

Question 39

Unit 2 was at full power when a Main Steam Line Break occurred OUTSIDE Containment.

- MSIVs will NOT close from the control room
- An Operator was dispatched to locally close the MSIVs per EOP-0.0B, Reactor Trip Or Safety Injection

Subsequently

- ECA-2.1B, Uncontrolled Depressurization Of All Steam Generators, is entered

A CAUTION statement after step one of this procedure directs the operator to control AFW flow to maintain a minimum flow of 100 gpm to ______ SG with less than ______.

A. Each
10 % Narrow Range level

B. At Least One
10 % Narrow Range level

C. Each
18 % Narrow Range level

D. At Least One
18 % Narrow Range level

Answer: A
Explanation:
Note: because this is manual control of a normally automatic feature (AFW flow for level control at minimum levels) it meets the intent of EK2.1.
A is correct because it states “A minimum AFW flow of 100 gpm must be maintained to each SG with a narrow range level less than 10% (18% FOR ADVERSE CONTAINMENT).” There are no adverse containment conditions because break is outside containment.
B is wrong because (see A above). Plausible because could think of it in terms of heat sink but this caution is to prevent tube damage so it is all four SG’s (ie EACH).
C is wrong because there are no adverse containment conditions given so 18% is wrong but plausible if they don’t recognize this from stem information.
D is correct because both parts are incorrect (see discussion on other distracters and answer above).

Technical References:
ECA-2.1B, rev 9, page 4.

References to be provided to applicants during exam: None.

Learning Objective: ERG.E2B.OB05.003

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)7	

Examination Outline Cross-Reference	Level	RO
000054 (APE 54; ~~CE E06~~) Loss of Main Feedwater /4	Tier #	1
	Group #	1
	K/A #	AK1.02
AK1.02 Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): Effects of feedwater introduction on dry S/G (CFR 41.8 / 41.10)	Rating	3.6
	QREV	6

Question 40

Given:

- A loss of all feedwater has occurred
- FRH-0.1A, Response to Loss of Secondary Heat Sink, is in progress
- RCS Bleed and Feed is in progress
- All Steam Generators are Hot and Dry
- The crew is preparing to re-establish secondary feed flow per FRH-0.1A Step 27.

Answer the following:

(1) A Dry Steam Generator is defined as: ____________________ .
(2) The limiting component(s) of concern when re-introducing feed to a Hot, Dry Steam Generator is thermal shock to the Steam Generator: ________________ .

A. (1) SG Wide Range Level < 43%
(2) Shell

B. (1) SG Wide Range Level < 43%
(2) Tubes

C. (1) SG Wide Range Level < 14%
(2) Shell

D. (1) SG Wide Range Level < 14%
(2) Tubes

Answer: C

Explanation:
A is wrong because part (1) uses the number for SG level (although range was changed to WR from NR for credibility) for a Red Path Secondary Heat Sink CSFST, entry conditions for FRH-0.1A.

B is wrong because part (1) uses the number for SG level (although range was changed to WR from NR for credibility) for a Red Path Secondary Heat Sink CSFST, entry conditions for FRH-0.1A., and part (2) limiting component is the SG shell.

C is correct.

D is wrong because the part (2) limiting component is the SG shell.

BASES

The limiting area with respect to Steam Generator integrity at 550°F was determined to be the SG shell, and the integrity of the tubes and tube sheet should not be significantly affected by thermal shock at this temperature (Reference SAMGS). The tubes are actually more ductile and better able to withstand the thermal shock of cold water since they are thinner.

steam generator. A hot, dry steam generator is defined as a steam generator in which the primary side of the steam generator is above 550°F (550°F is a temperature evaluated to be low enough that thermal stress would not lead to a failure when feedwater is established to any remaining dry steam generator.) and the secondary side has no liquid inventory. (Indicated SG level less than SG wide range level setpoints identified in this step.) Reestablishment of feedwater is the more

TABLE 1		
SG WIDE RANGE LEVEL	RCS Temperature	SG FEED FLOW CONSIDERATIONS
ALL SGs LESS THAN 14% (19% FOR ADVERSE CONTAINMENT)	INCREASING	• Establish maximum available feed flow to ONE SG. • WHEN selected SG wide range level greater than 14% (19% FOR ADVERSE CONTAINMENT), THEN adjust feed flow as necessary to establish narrow range level.
	STABLE OR DECREASING	• Establish feed flow to ONE SG at a rate not to exceed 100 gpm. • WHEN selected SG wide range level greater than 14% (19% FOR ADVERSE CONTAINMENT), THEN adjust feed flow as necessary to establish narrow range level.
ANY SG(s) GREATER THAN 14% (19% FOR ADVERSE CONTAINMENT)	NOT APPLICABLE	• Establish feed flow to affected SG(s) as necessary to establish narrow range level.

Technical References:
FRH-0.1A Rev 9 Step 27 and Step 27 Bases, pages 25, 72 an 74.

References to be provided to applicants during exam: None.

Learning Objective: Given a procedural Step, NOTE, or CAUTION, DISCUSS

the reason or basis for the Step, NOTE, or CAUTION in FRH-0.1 in accordance with FRH-0.1, Loss of Heat Sink. (LO21.ERG.FH1.OB04)

Question Source:	Bank #	
(note changes; attach parent)	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)(4)	
	55.43	

Examination Outline Cross-Reference	Level	RO
000011 (EPE 11) Large Break LOCA / 3	Tier #	1
Ability to determine or interpret the following as they apply to a Large Break LOCA: EA2.09 Existence of adequate natural circulation	Group #	1
	K/A #	EA2.09
	Rating	4.2
	QREV	6

Question 41

Unit 1 has experienced a large break loss of coolant accident and has transitioned to Emergency Procedure EOS-1.2A, POST LOCA COOLDOWN and DEPRESSURIZATION.

- Reactor coolant pumps are still unavailable due to offsite power losses
- Adverse containment conditions DO NOT exist

The US asks you to verify natural circulation exists per Attachment 3 of this procedure. What is one requirement for natural circulation and what action would you take if you don't have it?

A. RCS subcooling greater than 25 ^{0}F
Increase steam dump rate to condenser

B. RCS subcooling greater than 55 ^{0}F
Increase steam dump rate to condenser

C. RCS subcooling greater than 25 ^{0}F
Increase steam dump rate to atmosphere

D. RCS subcooling greater than 55 ^{0}F
Increase steam dump rate to atmosphere

Answer: C

Explanation:
A is wrong because can't dump to condenser with no power on non-safety buses, subcooling is correct.
B is wrong because adverse conditions don't exist as given in stem
C is correct because greater than 25F is the correct subcooling limit for natural circulation and increase dump to atmosphere is correct with no condenser.
D is wrong because subcooling is for adverse conditions.

Technical References:
EOS-1.2A, rev 9, page 37 and page 55.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E12.OB04.

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
WE11 Loss of Emergency Coolant Recirculation / 4	Tier #	1
	Group #	1
	K/A #	EK1.1
Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation) EK1.1 Components, capacity, and function of emergency systems	Rating	3.7
	QREV	6

Question 42

During the performance of ECA-1.1A, Loss of Emergency Coolant Recirculation Capability, the procedure estimates a depletion time for the RWST with **maximum safeguards flow** and a 60 percent depleted RWST from 100 percent full at the beginning of the event.

Given:

- Design basis LBLOCA that depressurizes RCS at the onset of the event
- RWST level is at 40% at the onset of the calculation, when the estimate in ECA-1.1A is done for RWST depletion
- Use DESIGN BASIS pump flows for all pumps, even though flow would be slightly less during the highest RCS pressure and slightly more with a fully discharged RCS.

Based on the conditions given above, what is the depletion time to **completely empty** the RWST?

NOTE: round your answer to the nearest whole number such that if you calculated 18.6 minutes, that rounds up to 19 minutes and if you calculated 18.4 minutes you would round down to 18 minutes)?

A. 5 to 7 minutes

B. 8 to 10 minutes

C. 11 to 13 minutes

D. 14 to 16 minutes

Answer: B
Explanation:
Note: This is higher order because you have to know the capacity of the RWST (500,000 gallons when full, so by this time 60 percent depleted is 200,000 gallons remaining). Max Safeguards flow is all four CS pumps (12 kgpm) and both trains of ECCS (2 CCP pumps at 150gpm each, 2 RHR pumps at 3800 gpm each, and 2 SI pumps at 425 gpm each is approx. 20,750 gpm). 200,000 gal / 20,750 gpm is approximately 9.63 minutes, which rounds up to 10 minutes, answer B. This is also an approximate calc in the back of the EOP (generic to all Westinghouse designs) to illustrate why you need to go to one train of ECCS equipment if you have to enter this procedure (to maximize the time for RWST water while you try to solve the blockage or alignment issue)

A is wrong because the answer is 10 minutes. You could come up with this if you use the RWST empty alarm value for final value in the tank, or the wrong capacity or numbers for the ECCS or CS pumps.
B is correct (see above)
C is wrong but plausible if you use the wrong capacity or numbers for the ECCS or CS pumps.
D is wrong but plausible if you use the wrong capacity or numbers for the ECCS or CS pumps.
.

Technical References:
ECA-1.1A, rev 9, page 63.

References to be provided to applicants during exam: None.

Learning Objective: ERG.C11.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	Level	RO
	Tier #	1
055 Station Blackout	Group #	1
	K/A #	EK1.01
Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: EK1.01 Effect of battery discharge rates on capacity	Rating	3.3
	QREV	7

Question 43

Given the following conditions:

- Unit 1 is in a Station Blackout
- The crew has performed actions in ECA-0.0A, LOSS OF ALL AC POWER, to reduce battery load
- BT1ED1 battery is discharging at 200 amps.

With NO further operator action, what is the MAXIMUM time that the vital DC bus supplied by this battery will be able to supply power to its loads if the battery was fully charged initially?

A. 9.75 hours

B. 8.00 hours

C. 6.00 hours

D. 4.91 hours

Answer: A

Explanation:

A is correct. BT1ED1 is a 1950 amp hour battery. 1950 / 200 = 9.75 hrs

B is wrong (see A above for calc). Plausible if you use the assumed 8 hr coping time, which sets the discharge rate at 244 amps for all batteries, which is not correct for this battery for conditions given in the stem.

C is wrong (see A calc above). Plausible if you use the other two batteries 1200 amp hour rating and divide by 200 amps, you get 6 hrs.

D is wrong (see calc above) if you use the wrong battery size (the other two batteries are 1200 amp hours) and the 244 amp discharge rate you get 4.91 hours.

Technical References:

LO21.SYS.DC1.pptx, slide 24, rev

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.DC1.OB03

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
056 Loss of Offsite Power		Tier #	1
		Group #	1
2.4.18 Knowledge of abnormal condition procedures.		K/A #	G2.4.11
		Rating	4.0
		QREV	6

Question 44

CPNPP has lost all Offsite and Onsite Power.

What is the preferred Switchyard and incoming transmission line for the Black Start Corridor?

A. 1) 138 KV
   2) Venus

B. 1) 345 KV
   2) Venus

C. 1) 138 KV
   2) DeCordova

D. 1) 345 KV
   2) DeCordova

Answer: C
Explanation:
A is wrong; Part 1 is correct 138 Kv is the preferred source of power for the black start corridor per ABN-601. Part 2 is incorrect but plausible because there are several transmission lines in to CPNPP and any one of them could be believed as correct.
B is wrong; Part 1 is incorrect but plausible because there are both 345 Kv and 138 Kv transmission lines in to CPNPP and either voltage could be believed as correct. Part 2 is incorrect but plausible, see A above.
C is correct. Part 1 is correct, see A above. Part 2 is correct, DeCordova is the preferred transmission line for the Black Start Corridor per ABN-601.
D is wrong; Part 1 is incorrect but plausible, see B above. Part 2 is correct, see C above

Technical References:
ABN-601, Revision 13, Attachment 20, Note, page 233

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.601.OB01.

Question Source:	Bank #	
(note changes; attach parent)	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	Level	RO
000065 (APE 65) Loss of Instrument Air / 8	Tier #	1
	Group #	1
Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: AK3.08 Actions contained in EOP for loss of instrument air	K/A #	AK3.08
	Rating	3.7
	QREV	6

Question 45

In ABN-301, INSTRUMENT AIR MALFUNCTION, What is the reason for the concern when throttling MDAFW flow control valves to control steam generator level (step 13) during this event?

The sizing of the AFW flow control valve accumulators will only allow ______.

A. for 5 full cycles over a 30 minute period

B. for 15 re-positions over a 4 hour period

C. for them to be maintained open for 7.5 hours

D. for them to be able to maintain inventory during a 50°F/hr cool down

Answer: A

Explanation:
A is correct because they are designed for 5 full cycles over a 30 minute period
B is wrong because this sizing reason is for the ARV's and is why it is plausible but not correct.
C is wrong because this sizing reason is for the TDAFW steam supply valves and is to keep them closed. Plausible because of the confusion over size/time and because changed it to open.
D is wrong because the ARV's are sized for this cool down but it is not the reason for the sizing of the AFW flow control valves.

Technical References:
LO21.SYS.IA1.pptx, revision 7/20/2017, page 60.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.IA1.OB04

Question Source:	Bank #	
(note changes; attach parent)	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		Level	RO
000007 (EPE 7) Reactor Trip, Stabilization, Recovery / 1		Tier #	1
		Group #	1
EA2.06 Ability to determine or interpret the following as they apply to a reactor trip: Occurrence of a reactor trip (CFR 41.7)		K/A #	EA2.06
		Rating	4.3
		QREV	6

Question 46

Given:
- A manual Reactor Trip is initiated from CB-07 and CB-10.
- Reactor Trip Breaker “A” indicates green.
- Reactor Trip Breaker “B” indicates red.
- All DRPI indication is lost.
- Group Step Counters are unchanged.
- Reactor power indicates 3% and lowering.

The crew is performing EOP-0.0A, Reactor Trip of Safety Injection, Step 1,“Verify Reactor Trip.” Which ONE of the following describes the condition of the reactor and the required action?

A. The Reactor is verified tripped. Adequate Shutdown Margin is verified. Continue to EOP-0.0A, Reactor Trip or Safety Injection, step 2.

B. The Reactor is NOT verified tripped. Adequate Shutdown Margin is verified. Momentarily de-energize 480V normal switchgear 1B3 AND 1B4.

C. The Reactor is verified tripped. Adequate Shutdown Margin is NOT verified. Emergency Borate per ABN-107, Emergency Boration, upon completion of immediate action steps.

D. The Reactor is NOT verified tripped. Adequate Shutdown Margin is NOT verified. Transition to FRS-0.1A, Response to Nuclear Power Generation/ATWT.

Answer: C
Explanation:
A is wrong because without indication of rod position, it cannot be verified that no more than 1 rod failed to fully insert, therefore adequate Shutdown Margin cannot be verified. Emergency Boration of 3600 gal is required by foldout page step 2.

B is wrong because 1B3 and 1B4 are only de-energized if step 1.a is NOT satisfied, meaning no Reactor Trip Breakers are open OR neutron flux is not decreasing.

C is correct. Step 1.a is satisfied with at least 1 reactor trip breaker open and flux decreasing, but Step 1.b is NOT satisfied because rods cannot be verified on the bottom, therefore emergency boration of 3600 gal is required per foldout page step 2 upon completion on immediate actions.

D is wrong because a transition to FRS-0.1A is only performed if EOP-0.0A step 1.a is NOT satisfied, meaning no Reactor Trip Breakers are open OR neutron flux is not decreasing, and de-energizing 480V normal switchgear 1B3 and 1B4 does not trip the reactor.

[Modified from CPNPP 2007 RO Exam Question 39. Step Counter status added to stem, all 4 answers modified and correct answer changed.]

Technical References:
EOP-0.0A, Reactor Trip of Safety Injection Attachment 1.A, FOLDOUT PAGE STEP 2, and Attachment 10, Bases

References to be provided to applicants during exam: None.

Learning Objective: EO0.XG2.OB12

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	NRC 2007 Q39
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41(b)(6) & (10)	
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	007 EA2.06	
	Importance Rating	4.3	

Ability to determine or interpret the following as they apply to a reactor trip: Occurrence of a reactor trip.

Proposed Question: Common 39

Given the following conditions:

- A manual Reactor Trip is attempted from CB-07 and CB-10.
- Reactor Trip Breaker "A" indicates green.
- Reactor Trip Breaker "B" indicates red.
- Reactor power indicates 3% and decreasing.
- All DRPI indication is lost.

Which ONE (1) of the following describes the condition of the reactor and the appropriate action?

A. The Reactor is tripped. Continue in EOP-0.0A, Reactor Trip or Safety Injection.

B. The Reactor is tripped. Actuate Safety Injection to meet Shutdown Margin criteria.

C. The Reactor is **not** tripped. Transition to FRS-0.1A, Response to Nuclear Power Generation/ATWT.

D. The Reactor is **not** tripped. Reattempt to manually trip the reactor and manually initiate Turbine trip.

Proposed Answer: A

Explanation (Optional):

A. Correct. One Reactor Trip Breaker opened and flux is lowering, therefore, the Reactor is tripped.

B. Incorrect. Reactor is tripped, however, action not consistent with the EOP.

C. Incorrect. Action not consistent with the EOP-0.0A RNO.

D. Incorrect. Action not consistent with the EOP-0.0A RNO.

Technical Reference(s) EOP-0.0A, Step 1 & Bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: EO0.XG2.OB12 (As available)

Question Source:
- Bank # X
- Modified Bank # ______ (Note changes or attach parent)
- New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 1, 10

Comments:
Robinson Bank

Examination Outline Cross-Reference	Level	RO
	Tier #	1
025 Loss of Residual Heat Removal System	Group #	1
	K/A #	AA1.12
Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: AA1.12 RCS temperature indicators	Rating	3.6
	QREV	6

Question 47

During mid-loop operations on Unit 1 with the head installed, a COMPLETE loss of shutdown cooling occurs.

Section 8 of ABN-104, “Residual Heat Removal System Malfunction,” is entered for the event and requires the use of which temperature indicator to verify that temperature is < 200 ^{0}F?

A. 1-TI-3611-2, CORE EXIT TEMP

B. 1-TI-411A, CL 1 TEMP (NR CHAN I)

C. 1-TI-413B, CL 1 TEMP (WR)

D. 1-TI-604, RHR HX 1 OUT TEMP

Answer: A

Explanation:
A is correct because this is the indicator called out in the procedure.
B is wrong (see A above) Any RCS temp indicators should be good distractors except for hot leg temps. This would make 2 correct answers.
C is wrong (see A above).
D is wrong (see A above).

Technical References:
ABN-104, Residual Heat Removal System Malfunction, p. 60

References to be provided to applicants during exam: None.

Learning Objective: DISCUSS the response to reduced inventory events utilizing ABN-104, "RHR System Malfunctions".
(LO21.IPO.010.OB06)

Question Source:	Bank #	
(note changes; attach parent)	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	Level	RO
000057 (APE 57) Loss of Vital AC Instrument Bus / 6	Tier #	1
	Group #	1
	K/A #	G2.4.20
2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	Rating	3.8
	QREV	6

Question 48

Unit 2 is at 8% power when the following occurs:

- 118V INV IV2EC1 TRBL alarm is in
- Multiple instruments are alarming or failing

What is a Unit 2 ONLY concern related to this event?

A. EDG 2-02 86-2 lockout relay prevents auto start on loss of power

B. Feedwater containment penetrations could overheat

C. Thermal barrier return isolation valve (HV-4696) closes

D. Black Out Sequencer is inoperable per TS 3.8.1

Answer: B

Explanation:
A is wrong because both Unit 1 and Unit 2 experience this condition for loss of EC1.
B is correct because the FWIV that is unique to Unit 2 causes the containment penetrations to overheat at low power (given in stem).
C is wrong because both Unit 1 and Unit 2 experience this condition for loss of EC1.
D is wrong because both Unit 1 and Unit 2 experience this condition for loss of EC1.
Technical References:
ABN-603, rev 8, page 16.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.603.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
W E04 LOCA Outside Containment	Tier #	1
	Group #	1
Knowledge of the interrelations between the (LOCA Outside Containment) and the following: EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	K/A #	EK2.2
	Rating	3.8
	QREV	6

Question 49

Unit 1 is responding to a Loss of Coolant Accident (LOCA) per ECA-1.2A, LOCA Outside Containment. Attempts are being made to determine if the leak has been isolated.

Which ONE of the following is the PRIMARY indication that the completed actions have been successful?

A. RCS Pressure rising

B. Pressurizer level rising

C. Emergency Core Cooling System flows rising

D. Reactor Vessel Level Indicating System indication rising

Answer: A

Explanation:
A is correct because Attachment 2 of ECA-1.2A instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions.
B is incorrect but plausible because Pressurizer Level may be below indicated level.
C is incorrect but plausible because ECCS flows could indicate that the break is isolated if it were lowering. Rising ECCS flow is indicative of the RCS break becoming worse.
D is incorrect but plausible because Attachment 2 of ECA-1.2A specifically lists RVLIS indication as an alternate method.

Technical References:
ECA-1.2A, LOCA Outside Containment, Rev.9, Page 6

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.C12.OB05

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank 2017 exam	X
	New	
Question History:	Last NRC Exam	No

Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	2
10CFR Part 55 Content:	55.41.7	

Parent Question (2017 exam, question 53)

Examination Outline Cross-reference:	Level	RO		SRO
Rev. Date: Rev. 1	Tier	1		
	Group	1		
	K/A	W/E04 EK2.2		
Level of Difficulty: 3	Importance Rating	3.8		

LOCA Outside Containment: Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Question # 53

Unit 1 plant conditions:

- Unit 1 is responding to a Loss of Coolant Accident (LOCA) per ECA-1.2A, LOCA Outside Containment.
- The crew believes that the leak outside of Containment has been isolated but Reactor Coolant System pressure is not rising.

Which of the following alternate indications may be used to determine if the break has been isolated per ECA-1.2A, LOCA Outside Containment?

A. Refueling Water Storage Tank level stable.

B. Reactor Vessel Level Indicating System indication rising.

C. Emergency Core Cooling System flows rising.

D. Emergency Core Cooling System alignment verification.

Answer: B

K/A Match:

This question matches the KA by requiring knowledge of the interrelations between the LOCA outside containment and heat removal system valves that are cycled in an attempt to locate the leak.

Explanation:

A. Incorrect. Plausible because RWST inventory could be lost to the Safeguards Building from an interfacing system break outside Containment and RWST level stable could indicate that break flow has ceased, however, RWST level may still be lowering with the break isolated as ECCS flow refills the RCS. Therefore, RWST level change is not a good indicator of break isolation.

B. Correct. As stated in Attachment 2, Step 3 Bases, RCS pressure may not initially rise once the break is isolated, due to plant cooldown or when the RCS is saturated. RVLIS indication rising shows that ECCS flow is not leaving the RCS via the break, but rather it is refilling the RCS. This indicates that the break is isolated from the RCS.

C. Incorrect. Plausible because ECCS flows could indicate that the break is isolated if it were lowering. Rising ECCS flow is indicative of the RCS break becoming worse.

D. Incorrect. Plausible because an ECCS valve alignment is performed in Step 1 of ECA-1.2A, and this may isolate the break, however, verifying ECCS alignment alone does not ensure that the break is isolated. RCS parameters, such as pressure and RVLIS trend must be evaluated to verify break isolation.

Technical Reference(s)	ECA-1.2A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: **STATE** the bases for operator actions, notes and cautions from ECA-1.2, LOCA Outside Containment.

Question Source: Bank # 2013 NRC Exam Q54
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2013 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Examination Outline Cross-Reference	Level	RO
W E05- Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4	Tier #	1
	Group #	1
Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations	K/A #	EA2.1
	Rating	3.4
	QREV	7

Question 50

The following conditions exist after a Reactor Trip and Safety Injection on Unit 2. The crew has transitioned from EOP-0.0B and attachment 2 has been completed satisfactory. Critical Safety Function monitoring has been initiated. All available ECCS equipment is operating correctly, AFW flow is maximized, with the following plant conditions:

- RCS Pressure 2235 psig
- PRZR level 95%
- RCS Subcooling 50 ºF
- CNTMT Pressure 4 psig
- SG NR levels AFW flow Pressure
- 2-01 - 25% 235 gpm 1245 psig
- 2-02 - 24% 235 gpm 1235 psig
- 2-03 - 22% 0 gpm 1230 psig
- 2-04 - 25% 0 gpm 1240 psig

Which of the below actions should be done next?

A. Enter FRH-0.1B, Response to Loss of Secondary Heat Sink

B. Enter FRH-0.2B, Response to Steam Generator Overpressure

C. Enter FRH-0.3B, Response to Steam Generator High Level

D. Enter FRH-0.4B, Response to Loss of Normal Steam Release Capabilities

Answer: B

Explanation:
A is wrong because AFW flow is greater than 460gpm. Plausible if you don't recall minimum AFW flow needed for an adequate heat sink.
B is correct because although AFW flow is adequate, one SG (ie 3 of them in this case) has pressure not less than 1235 psig so it requires FRH-0.2B to be implemented.
C is wrong because FRH-0.2B is correct for these conditions. Plausible if you think you need only one SG pressure below 1235 psig to meet the decision point, in which case FRH-0.3B would be correct.
D is wrong because FRH-0.2B is correct for these conditions. Plausible if you forget or don't recall the decision point for FRH-0.3B and FRH-0.4B where the split is over "Pressure in ALL SGs less than 1185 psig", which it is not, so you might pick this distracter because of this fact. This is incorrect because the split occurred earlier for higher pressures than 1235 psig.
Technical References:

FRH-0.2B, revision 9, page 6.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.FH2.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)10	

Examination Outline Cross-Reference	Level	RO
038 Steam Generator Tube Rupture	Tier #	1
	Group #	1
Knowledge of the reasons for the following responses as the apply to the SGTR:	K/A #	K3.09
	Rating	4.1
EK3.09 Criteria for securing/throttling ECCS	QREV	6

Question 51

Given the following Unit 2 conditions:

- Steam Generator 2-02 is ruptured
- Steam Generator 2-01 is faulted inside containment
- EOP-2.0B, Faulted Steam Generator Isolation is complete
- Containment pressure is 7 psig and rising
- RCS Depressurization in EOP-3.0B, Steam Generator Tube Rupture, is complete

In accordance with EOP-3.0B, Steam Generator Tube Rupture…

1. What is the PRIMARY reason for terminating Safety Injection flow when the criteria are met?
2. What MINIMUM Pressurizer Level is required to terminate Safety Injection flow?

A. (1) Prevent ruptured Steam Generator overfill and lifting of the Atmospheric Relief Valves.
(2) 15%

B. (1) Prevent ruptured Steam Generator overfill and lifting of the Atmospheric Relief Valves.
(2) 13%

C. (1) Prevent Pressurizer overfill and lifting of the Pressurizer Safety Valves.
(2) 15%

D. (1) Prevent Pressurizer overfill and lifting of the Pressurizer Safety Valves.
(2) 13%

Answer: A

Explanation:

A. Correct. In accordance with EOP-3.0B, Steam Generator Tube Rupture, Safety Injection flow must be terminated following RCS depressurization to prevent overfilling of the ruptured steam generator and lifting the atmospheric relief valve, thus, causing a release to the environment. Also, during adverse containment conditions 15% pressurizer level is required on Unit 2 vice a 13% requirement for non-adverse conditions.
B. Incorrect. First part is correct as described in “A” above. Second part is incorrect but plausible as 13% would be the pressurizer level at which safety injection flow would be terminated if adverse containment conditions DID NOT exist.
C. Incorrect. First part is incorrect but plausible as the pressurizer does fill during RCS

depressurization and it is a concern, however, there is not a concern with the primary Safety Valves lifting during the depressurization as the PORVs will open and relieve pressure if necessary. Second part is correct as described in “A” above.
D. Incorrect. First part as described in “C” above. Second part as described in “B” above.

Technical References:
EOP 3.0B, Revision 9, Step 23 and Attachment 6 (bases), Pages 15 and 28

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E3A.OB05.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	ON 2015 exam
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 4
10CFR Part 55 Content:	55.41(b)10	

Examination Outline Cross-Reference	Level	RO
	Tier #	1
062 Loss of Nuclear Service Water	Group #	1
	K/A #	AA1.06
Ability to operate and/or monitor the following as they apply to the Loss of Nuclear Service Water: AA1.06 Control of flow rates to components cooled by the SWS	Rating	2.9
	QREV	6

Question 52

Unit 1 is operating at full power when the following alarm is received:

- 1-ALB-1, Window 1.7 – SSW TRN A/B HDR PRESS LO

The US enters ABN-501, "Station Service Water System Malfunction." As the RO you would be expected to verify service water flow is less than (1) gpm. Also, multiple individual component low flow alarms come in. These flows are required to be adjusted with (2).

A. (1) 15,000
(2) TDM-901A,Throttled Valves/Flow Rates

B. (1) 15,000
(2) ALM-0011A, SSW TRN A/B HDR PRESS LO

C. (1) 18,000
(2) TDM-901A,Throttled Valves/Flow Rates

D. (1) 18,000
(2) ALM-0011A, SSW TRN A/B HDR PRESS LO

Answer: C

.

Explanation:
A is wrong because the first part is incorrect. This is the value for CCW HX and it is not the value specified in the ABN. Plausible because of its value and association with a SSW cooled component.
B is wrong because both parts are incorrect. The ALM is credible because it has references to TDM-901A in other alarms such as Window 1.11 – CCP 1 L/O CLR SSW RET FLO LO.
C is correct because this is the value listed in the ABN and the throttle position directions are listed in section 4 of the ABN under LIMITATIONS and require use of TDM-901A.
D is wrong for the second part but plausible see B above

Technical References:
ABN-501,"Station Service Water System Malfunction," rev. 9, p. 9 and 11
ALM-0011A, 1-ALB-1, Window 1.7 – SSW TRAN A/B HDR PRESS LO

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the normal, abnormal and emergency operation of the Station Service Water system IAW SOP-501 and ABN-501. (LO21.SYS.SW1.OB05)

Question Source:	Bank #	
(note changes; attach parent)	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)(10)	
	55.43	

Examination Outline Cross-Reference	Level	RO
029 Anticipated Transient Without Scam	Tier #	1
	Group #	1
	K/A #	2.4.6
Knowledge of EOP Mitigation Strategies	Rating	3.7
	QREV	7

Question 53

Given the following conditions:

- Unit 1 was at 100%
- An ATWT occurred
- The crew enters FRS-0.1A, Response to Nuclear Power Generation/ATWT
- During performance of Step 5 – "Checking PRZR Pressure – LESS THAN 2335 psig," the reactor trip breakers are opened locally
- All control rods to begin to drop into the core

Which of the following describes how the operator exits FRS-0.1A and returns to EOP-0.0A, Reactor Trip or Safety Injection, for this situation?

A. Once all rod bottom lights are lit

B. Only when all required procedure steps are completed

C. Once the RED or ORANGE path condition is cleared on the status tree

D. At Step 8, after verifying the reactor subcritical (<5% power and the IR channels indicate a negative startup rate)

Answer: B
Explanation:
A is incorrect but plausible because verifying the reactor is tripped is a continuous action step, but nothing in FRS-0.1A allows returning to procedure in effect until FRS-0.1A is complete.
B is correct because this the only step (ie the end of FRS-0.1A) that allows returning to procedure in effect.
C is incorrect but plausible because it is logical to exit a procedure once a red or orange path is clear, but nothing in FRS-0.1A allows returning to procedure in effect until FRS-0.1A is complete.
D is incorrect but plausible because verifying the reactor subcritical is a logical step after verifying the reactor is tripped, but nothing in FRS-0.1A allows returning to procedure in effect until FRS-0.1A is complete.

Technical References:
FRS-0.1A, Response to Nuclear Power Generation/ATWT, Revision 9

References to be provided to applicants during exam: None.

Learning Objective: LO21. ERG.FS1.OB04; LO21ERGFS1 Terminal Objective

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	18562
Question History:	Last NRC Exam	None
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
000058 (APE 58) Loss of DC Power / 6	Tier #	1
	Group #	1
Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: AK3.01 Use of dc control power by D/Gs	K/A #	AK3.01
	Rating	3.4
	QREV	6

Question 54

During a Station Blackout for Unit 1 and Unit 2, the Unit 1 US has entered ECA-0.0A and the crew has completed the load stripping attachments within 2 hours for Attachment 2.A and Attachment 2.B. As referenced in ECA-0.0A, as DC voltage continues to lower, what is the MINIMUM voltage for equipment operation and the specific piece of equipment that is limited by this minimum DC voltage?

A. When DC bus voltage drops below 110 Volts DC the battery does not have enough power to open the air start solenoids for the Emergency Diesel Generators

B. When DC bus voltage drops below 110 Volts DC the battery does not have enough power to flash the field for the Emergency Diesel Generators

C. When DC bus voltage drops below 105 Volts DC the battery does not have enough power to open the air start solenoids for the Emergency Diesel Generators

D. When DC bus voltage drops below 105 Volts DC the battery does not have enough power to flash the field for the Emergency Diesel Generators

Answer: D

Explanation:
A is wrong because this is not the concern and the voltage is not below 105vdc.
B is wrong because although the second part is correct for field flash, the first part, the voltage, is still above the 105vdc.
C is wrong because although the first part (voltage is correct) the second part is wrong.
D is correct because once voltage drops below 105 VDC, the major concern for the EDG is field flash, not air start.

Technical References:
ECA-0.0A, rev 9, page 123.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.DC1.OB04

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
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3	Tier #	1
	Group #	1
AA1.04 Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure recovery, using emergency-only heaters (CFR 41.7)	K/A #	AA1.04
	Rating	3.9
	QREV	6

Question 55

The plant is operating at 100% power with the Pressurizer Pressure Control System in its normal at-power control alignment. (PS-455F selected to 455/456)

1-ALB-5C, Window 1.3 – PRZR LVL DEV HI is in alarm due to a previous plant transient.

Subsequently, Pressurizer Pressure channel PT-455 fails high.

(1) What is the status of the Pressurizer Backup Heaters before operator action is taken?

(2) What is the status of the Pressurizer Backup Heaters when operators place the Pressurizer Pressure Master Controller PK-455A in MANUAL to stabilize Pressurizer pressure?

A. (1) Energized
(2) Energized

B. (1) Energized
(2) De-Energized

C. (1) De-Energized
(2) Energized

D. (1) De-Energized
(2) De-Energized

Answer: A

Explanation:

This question tests the applicant's understanding of pressurizer backup heater actuation logic in the presence of a Pressurizer Pressure System Control Malfunction.

The normal at-power alignment for Pressurizer Pressure Control is for PS-455F to be selected to the midposition (455/456). When PT-455 fails high, it causes both spray valves to modulate full-open, PORV PCV-455A to open (and re-close at 2185 psig), and variable pressurizer heaters to go to full-off. Backup pressurizer heaters would normally deenergize as well (at sensed pressure above 2218 psig), but because the PRZR LVL DEV HI alarm is in [> 5% above program level], backup pressurizer heaters remain energized independent of

Pressurizer Pressure Master Controller PK-455A output. When PK-455A is taken to MANUAL control and adjusted, backup heaters remain energized until BOTH pressurizer level lowers to within 5% of program level, AND pressurizer pressure rises above 2218 psig (assuming it dropped below). Backup heaters would normally auto-energize at 2210 psig and lowering (and reset at 2218 psig and rising), so they would not be expected to energize when PK-455A is taken to MANUAL at 2215 psig IF pressurizer level were in band.

A is correct. With pressurizer level > 5% above program level, backup heaters remain energized through the transient.

B is wrong because although the backup heaters do remain energized at the start of the transient, when the master pressure controller is taken to MANUAL they remain energized. Applicant may incorrectly believe that taking PK-455A to MANUAL while ABOVE the heater actuation point of 2210 psig will cause heaters to de-energize.

C is wrong because backup heaters remain energized due to the pressurizer level deviation. Applicant may believe that heaters de-energize initially due to failed instrument upscale, but energize when taken to manual due to pressure dropping below 2218 psig (the heater reset setpoint).

D is wrong because backup heaters remain energized for the transient due to the pressurizer level deviation. If pressurizer level was within 5% of program for this transient, this would be the correct answer.

Technical References:
ABN-705 Pressurizer Pressure Malfunction
Pressurizer Pressure and Level Control Study Guide, 5-5-2011

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	3
10CFR Part 55 Content:	55.41(b)(7)	
	55.43	

SYSTEM RUPTURE	FUNCTION	*MASTER CONTROLLER OUTPUT	
2460	PLANT SAFETIES LIFT		
2385	REACTOR TRIP		
2335	456 POWER OPERATED RELIEF OPEN		
2335	455A POWER OPERATED RELIEF OPEN	81.3 %	
	456 POWER OPERATED RELIEF CLOSED		2315
	455A POWER OPERATED RELIEF CLOSED	75 %	2315
2310	SPRAYS FULLY OPEN	73.4 %	
2260	SPRAYS START TO OPEN	57.8 %	
2250	VARIABLE HEATERS OFF	54.7 %	
2235	N.O.P.	50%	2235
	VARIABLE HEATERS FULLY ON	45.3 %	2220
2218	BACKUP HEATERS OFF	44.7 %	
	BACKUP HEATERS ON	42.2 %	2210
	PORV BLOCKED		2185
	SI BLOCK/UNBLOCK		1960
	#LOW PRESS RX TRIP		1880
	SI ACTUATION		1820
			DNB

#COMPENSATED SIGNAL
SETPOINTS IN PSIG

* CONTROLLED BY MASTER PZR CONTROLLER. ACTUATION PRESSURE CAN VARY SIGNIFICANTLY. PRESSURE SUPPLIED TO SHOW RELATIVE OPERATING SEQUENCE.

PRESSURIZER PRESSURE SETPOINTS

3-1-2003

Examination Outline Cross-Reference	Level	RO
	Tier #	1
077 Generator Voltage and Electric Grid Disturbances	Group #	1
	K/A #	AK2.06
	Rating	3.9
Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: AK2.06 Reactor power	QREV	6

Question 56

A grid disturbance has resulted in a small reduction in incoming switchyard voltage and frequency.

What is the effect on reactor power as a result of the transient?

A. Reactor power will lower slightly due to the reduced speed of the unit motors.

B. Reactive load and reactor power will drop slightly due to lower voltage.

C. As voltage lowers the amps drawn by equipment lowers causing reactive load to lower and reactor power to rise.

D. As grid voltage and frequency drop the main turbine control valves open causing reactor power to rise.

Answer: D

Explanation:
A is wrong because there will be a reduction in motor speed, but power will be higher due to load increasing on the main generator.
B is wrong because an applicant could think that lower voltage could result in less reactive load and reduced power.
C is wrong because a decrease in voltage should result in a rise in amperage and rise in reactive load.
D is correct because the main generator will pickup real load and reactive load from frequency and voltage drops. This will cause a rise in reactor power as stated.

Technical References:
Comanche Peak Exam Bank ILOT8297

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the normal, abnormal and emergency operation of Main Turbine and its support systems.
(LO21.SYS.MT1.OB26)

Question Source:	Bank #	X (2010-04, Q47)
(note changes; attach parent)	Modified Bank #	
	New	

Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	 3
10CFR Part 55 Content:	55.41(b)(5) 55.43	

Examination Outline Cross-Reference	Level	RO
W E01 Rediagnosis	Tier #	1
	Group #	2
Knowledge of the interrelations between the (Reactor Trip or Safety Injection/Rediagnosis) and the following: EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	K/A #	EAK2.2
	Rating	3.5
	QREV	7

Question 57

During a plant transient, the crew enters EOS-0.0A, Rediagnosis, to reevaluate plant status.

Regarding heat removal systems and the major strategy of the Rediagnosis procedure, what is the first check (diagnosis) that is made and if that is not the problem then what is the second check (diagnosis) within this procedure?

A. 1. Faulted steam generator
   2. Ruptured steam generator

B. 1. Loss of coolant accident
   2. Faulted steam generator

C. 1. Loss of coolant accident
   2. Ruptured steam generator

D. 1. Ruptured steam generator
   2. Faulted steam generator

Answer: A
Explanation:
A is correct because the order is Faulted SG, then ruptured SG, then LOCA..
B is wrong because the correct order is listed in A above. Plausible for LOCA to be the first since SI is required to be running or should be running at the beginning of this procedure.
C is wrong because the correct order is listed in A above. Plausible for LOCA to be the first since SI is required to be running or should be running at the beginning of this procedure.
D is correct because the correct order is listed in A above Plausible because they could get the order backwards.

Technical References:
EOS-0.0A, rev 9, pages 3-4, and also bases.

References to be provided to applicants during exam: None.

Learning Objective: ERG.E00.OB01

Question Source:	Bank # 18588	X
(note changes; attach parent)	Modified Bank #	

	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
E10 Natural Circulation with Steam Void in Vessel with/without RVLIS

Knowledge of the reasons for the following responses as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS) EK3.4 RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Level	RO
Tier #	1
Group #	2
K/A #	EK3.4
Rating	3.4
QREV	6

Question 58

During the performance of EOS-0.4A, Natural Circulation with Steam Void in Vessel (without RVLIS), when starting the first Reactor Coolant Pump, ensure pressurizer level is (1) and subcooling is a minimum of (2) :

A. (1) above 90%
(2) 25°F

B. (1) above 90%
(2) 60°F

C. (1) between 30% and 40%
(2) 25°F

D. (1) between 30% and 40%
(2) 60°F

Answer: B

Explanation:
A is incorrect but plausible because pressurizer level must be greater than 90%, but 25°F is the subcooling normally associated with tripping a RCP.
B is correct. According to EOS-0.4A, prior to starting a RCP, check that pressurizer level is above 90% and subcooling is greater than 60°F.
C is incorrect but plausible because IF a Reactor Coolant Pump CANNOT be started, EOS-0.4B contains guidance to maintain Pressurizer level between 30% and 40%, and 25°F is the subcooling normally associated with tripping a RCP.
D is incorrect but plausible because IF a Reactor Coolant Pump CANNOT be started, EOS-0.4B contains guidance to maintain Pressurizer level between 30% and 40%, and 60°F is the correct subcooling value.
Technical References:
EOS-0.4A, Natural Circulation Cooldown with Steam Void In Vessel (Without RVLIS), Revision 9, page 3

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E02.OB04

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:	55.41.5	

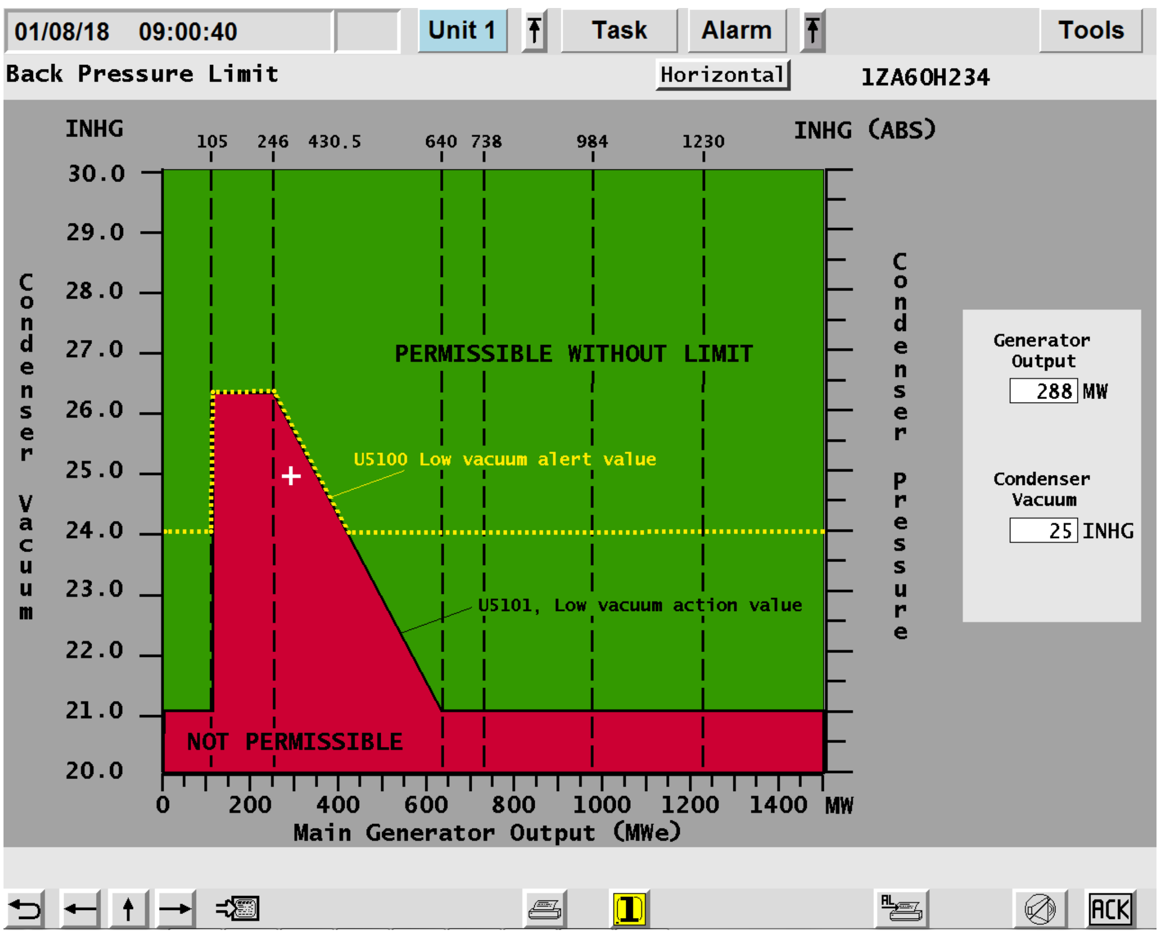
Examination Outline Cross-Reference
000051 (APE 51) Loss of Condenser Vacuum / 4

2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10)

Level	RO
Tier #	1
Group #	2
K/A #	2.4.47
Rating	4.2
QREV	7

Question 59

Given the following conditions have just occurred:



Which actions are required by ABN-304, MAIN CONDENSER AND CIRCULATING WATER SYSTEM MALFUNCTION, and why?

A. Trip the Reactor immediately, to prevent excessive steam erosion on the LOW pressure turbine blades.

B. Decrease Turbine Load or increase Condenser Vacuum within 5 minutes, maximum, to prevent excessive steam erosion on the LOW pressure turbine blades.

C. Trip the Reactor immediately, to prevent impingement due to moisture carryover on HIGH pressure turbine blades.

D. Decrease Turbine Load or increase Condenser Vacuum within 5 minutes, maximum, to prevent impingement due to moisture carryover on HIGH pressure turbine blades.

Answer: B
Explanation:

ABN-304 MAIN CONDENSER AND CIRCULATING WATER SYSTEM MALFUNCTION
Sect. 3 MAIN OR AUXILIARY CONDENSER VACUUM DECREASING

☐ [C]	3	VERIFY Main Condenser Vacuum - GREATER THAN <u>21" HG</u> • Main Cond. Vacuum on TG Control Display • <u>u</u>-PI-2042-1, CNDSR A PRESS • <u>u</u>-PI-2042-2, CNDSR B PRESS	<u>IF</u> Reactor Power is greater than or equal to 10%, <u>THEN</u> TRIP Reactor <u>AND</u> GO TO EOP-0.0A/B while others continue this procedure. <u>IF</u> Reactor Power is less than 10%, <u>THEN</u> TRIP Turbine <u>AND</u> perform ABN-403 while continuing this procedure.
☐	4	VERIFY Main Condenser Vacuum Maintained - GREATER THAN 26.5" HG • Main Cond. Vacuum on TG Control Display • <u>u</u>-PI-2042-1, CNDSR A PRESS • <u>u</u>-PI-2042-2, CNDSR B PRESS	On the Back Pressure Limit Display ENSURE turbine NOT operating for more than 5 minutes in the NOT PERMISSIBLE region (not applicable for ramp rate >60 MW/min) by: a. Decreasing Turbine Load. <u>OR</u> b. Increasing Condenser Vacuum.

Circ Water Study Guide:

Besides condensate depression, the CW system has an effect on the life of the Low Pressure Turbines. By maintaining sufficient backpressure on the Low Pressure Turbines, the Turbine blades will not be damaged from excessive steam erosion. At CPNPP Main Condenser pressure during normal operation is maintained < 3.0 inches Hg absolute. If Main Condenser pressure rises above 3.0 inches Hg absolute then additional CWPs are started. Main Turbine load should be reduced when backpressure pressure reaches 4.5 inches Hg absolute and there are no additional CWPs available. To calculate Condenser pressure inches Hg absolute, the indicated vacuum is subtracted from barometric pressure using the plant computer.

Condenser pressure inches Hg absolute = Barometric pressure - Indicated Vacuum

… If at least 1 CWP is operating, then operator concern will be to ensure Main Condenser vacuum is > 24.5" Hg. A HI prolonged backpressure on the LP Turbines can cause turbine

blade damage. If Main Condenser pressure reaches 4.5 inches Hg absolute, turbine load should be reduced.

A is wrong because condenser vacuum has not degraded to the reactor trip setpoint of 21" Hg.

B is correct. The turbine is operating in the NOT PERMISSIBLE region, but has not exceeded the Reactor / Turbine Trip setpoint of 21" Hg condenser vacuum.

C is wrong because condenser vacuum has not degraded to the reactor trip setpoint of 21" Hg, and the concern with lowering vacuum is damage to the low pressure turbine blades, not impingement of the high pressure turbine blades due to carryover (which is a concern associated with high steam generator level).

D is wrong because the concern with lowering vacuum is damage to the low pressure turbine blades, not impingement of the high pressure turbine blades due to carryover (which is a concern associated with high steam generator level).

Technical References:
ABN-304 MAIN CONDENSER AND CIRCULATING WATER SYSTEM MALFUNCTION
Sect. 3 MAIN OR AUXILIARY CONDENSER VACUUM DECREASING

Circulating Water Study Guide, 5-4-11

References to be provided to applicants during exam: None.

Learning Objective: During normal plant operations, DESCRIBE the Operator response to a Circulating Water Pump Trip, Main or Auxiliary Condenser Vacuum Decreasing, Main or Auxiliary Condenser Tube Leak or Turbine Building Elevation 758' Flooding per ABN-304. (LO21.SST.CO1 OB05)

ANALYZE the response to Main or Auxiliary Condenser Vacuum Decreasing in accordance with ABN-304, Main Condenser and Circulating Water System Malfunction. (LO21.ABN.304.OB02)

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)	
	55.43	

Examination Outline Cross-Reference		Level	RO
01 Continuous Rod Withdrawal		Tier #	1
		Group #	2
Knowledge of the physical connections and/or cause/effect relationships between the CRDS and the following systems:		K/A #	K1.04
		Rating	3.2
		QREV	6

K1.04 RCS

Question 60

Given :

- Unit 2 is operating at 88% power with generator load stable
- The Rod Control System is in Automatic
- Reactor power is RISING
- Tavg is greater than Tref
- Pressurizer level is RISING

Which of the following would cause these symptoms to occur?

A. Power Range Channel N-43 fails high

B. Failed OPEN S/G safety valve

C. Uncontrolled rod withdrawal

D. Turbine load rejection

Answer: C

Explanation:
Continuous rod withdrawal would cause reactor power to rise, and Thot would rise, which would cause Tave to rise above Tref for a constant steam demand. RCS pressure would rise due to pressurizer level rising (RCS less dense).

A is wrong because a failure of the PR channel high would not cause any plant parameters to change initially. As a result of the input of the failed PR channel high to the rate comparator of the Reactor Control Unit, which would anticipate Tave higher than Tref, rods would move in to reduce power to lower Thot and therefore Tavg. Plausible if the candidate does not understand plant response on failure of the PR channel.
B is wrong because this would increase steam demand, and therefore Tc would lower, causing Tave to be less than Tref. Plausible because reactor power would rise.
C is correct (see explanation).
D is incorrect because load rejection would cause steam demand to lower, and less heat would be taken out of the RCS, and Tc would increase, which would make Tave > Tref, and power would lower. Plausible because Tave would initially be greater than Tref.

Technical References:
LO21.MCO.TA7, Revision 12/27/16, Pages 5-6

References to be provided to applicants during exam: None.

Learning Objective: LO21.MCO.TA7.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
(W E15) Containment Flooding / 5	Tier #	1
Ability to operate and / or monitor the following as they apply to the (Containment Flooding) EA1.2 Operating behavior characteristics of the facility.	Group #	2
	K/A #	EA1.2
	Rating	2.7
	QREV	7

Question 61

Containment flooding is occurring in Unit 1 and FRZ-0.2A, "RESPONSE TO CONTAINMENT FLOODING" has been entered. In the Containment Status Tree, Attachment 1.A of this procedure, it has the diagnostic block that states "Containment sump level less than _____," which is approximately the design basis water level for containment and can be read on level instrument ______.

A. 809 feet
LI-5164 on CB-01

B. 809 feet
LI-4779 on CB-02

C. 816 feet
LI-5164 on CB-01

D. 816 feet
LI-4779 on CB-02

Answer: D
Explanation:

Notes for administration:
LI-5164, RX CAV SMP LVL
LI-4779, CNTMT RECIRC SMP LVL

A is wrong because both parts are wrong. The DB level is exactly 816 ft 10 inches and one of the PAM instruments used to measure it is LI-4779 on MCB 02. Plausible because this instrument is used to measure sump level in containment but is not the PAM instrument to be used for the Functional procedures, LI-4779 is the correct one. LI-5164 also goes from 0 to 2 ft indicated (not 808 ft to 817 ft indicated as LI-4779 does).
B is wrong because first part is wrong, second part is correct (see A above for plausibility).
C is wrong because second part is wrong (see A above for plausibility discussion)
D is correct because the containment Design Basis level is exactly 816 ft 10 inches (so approx. 816 ft) and one of the two PAM instruments used to measure it is LI-4779 on MCB 02 (the other is LI-4781 and is on MCB 04).

Technical References:
Post-Accident Instrumentation Study Guide, rev 5/9/2011, page 11. FRZ-0.2A, rev 9, page 5.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CY1.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	Level	RO
WE16 High Containment Radiation	Tier #	1
	Group #	2
Ability to determine and interpret the following as they apply to the (High Containment Radiation) EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations	K/A #	EA2.1
	Rating	2.9
	QREV	6

Question 62

Which of the following parameters by itself results in a Yellow condition for the Critical Safety Function CONTAINMENT Status Tree?

A. Containment Pressure of 5 psig

B. Containment Pressure of 25 psig

C. Containment Radiation Level of 5 R/hr

D. Containment Radiation Level of 25 R/hr

Answer: D

Explanation:
A is incorrect because containment pressure is a parameter for the containment critical safety function, and 5 psig is the value for adverse containment.
B is incorrect because containment pressure is a parameter for the containment critical safety function, but 18 psig results in an orange condition.
C is incorrect because containment radiation must be above 20 R/hr to result in a yellow condition, but plausible if one believes the value to be 5 R/hr.
D is correct because containment radiation above 20 R/hr results in a yellow condition

Technical References:
FRZ-0.3A, Response to High Containment Radiation Level, Revision 9, page 4

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.FZ2.OB07

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43.5	

Examination Outline Cross-Reference		Level	RO
000037 (APE 37) Steam Generator Tube Leak / 3		Tier #	1
		Group #	2
Ability to operate and / or monitor the following as they apply to the Steam Generator Tube Leak: (CFR 41.7 / 45.5 / 45.6)		K/A #	AA1.03
		Rating	3.0
AA1.03 Loop isolation valves		QREV	6

Question 63

A Steam Generator Tube LEAK is in progress on Steam Generator 1-01.

1) Industry operating experience has shown which radiation monitor to be the MOST LIKELY first indication of a Steam Generator Tube Leak?
2) With both Main Steam Line #1 N-16 Radiation Monitor 1-RE-2325A and SG Blowdown Process Sample Radiation Monitor 1-RE-4200 in ALARM, an automatic isolation of which components will occur?

A. (1) Main Steam Line #1 N-16 Radiation Monitor (1-RE-2325A)
(2) SG Blowdown and Sample Isolation Valves for SG 1-01 only.

B. (1) SG Blowdown Process Sample Radiation Monitor (1-RE-4200)
(2) SG Blowdown and Sample Isolation Valves for ALL SGs.

C. (1) Main Steam Line #1 N-16 Radiation Monitor (1-RE-2325A)
(2) SG Blowdown and Sample Isolation Valves for ALL SGs.

D. (1) SG Blowdown Process Sample Radiation Monitor (1-RE-4200)
(2) SG Blowdown and Sample Isolation Valves for SG 1-01 only.

Answer: C
Explanation:

SG Blowdown Process Sample Radiation Monitor 1-RE-4200 is a common process monitor for all 4 SGs, and causes an isolation of blowdown and sample isolation valves for all 4 SGs when its alarm setpoint is exceeded. Industry operating experience has shown that the first radiation monitor to show indication of a SG Tube Leak/Rupture is the Main Steam Line radiation monitor (in this case RE-2325A for SG #1) due to its fast response to increased N-16 levels, following by Condenser Offgas monitor, followed by the SG blowdown rad monitor.

A is wrong. 1ST part correct, 2nd part incorrect. Plausible because of explanation above.
B is wrong. 1st part incorrect, 2nd part correct. Plausible because of explanation above.
C is correct (see above discussion).
D is wrong. 1st and 2nd parts both incorrect. Plausible because of explanation above.

Technical References:
ABN-106, High Secondary Activity, Rev 10, Step 3.2 Automatic Actions
SG Blowdown Study Guide, 5-7-2011, p. 27

References to be provided to applicants during exam: None.

Learning Objective: ANALYZE the response to a 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 #	
(note changes; attach parent)	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)(11)	
	55.43	

Examination Outline Cross-Reference	Level	RO
005 Inoperable/Stuck Control Rod	Tier #	1
	Group #	2
Ability to determine and interpret the following as they apply to the inoperable/stuck control rod:	K/A #	A2.01
	Rating	3.3
	QREV	6

A2.01 Stuck or inoperable rod from in-core and ex-core NIS, incore or loop temperature measurements.

Question 64

Unit 1 is at full power when a DRPI ROD DEV alarm comes in.

In accordance with ABN-712, Rod Control System Malfunction, Section 4.0, Digital Rod Position Indication Malfunction, Step 6 states to:
"CHECK Redundant Indication Demonstrates ALL Rods – ALIGNED"

Which of the following methods is used in this step for a redundant check of rod alignment?

A. Compare initial turbine load to current turbine load

B. Compare initial Plant Computer thermocouple map to current thermocouple map

C. Compare initial Loop Hot Leg temperatures to current Hot Leg temperatures

D. Compare initial Loop Cold Leg temperatures to current Loop Cold Leg temperatures

Answer: B

Explanation:
A. Incorrect. Plausible because Turbine load can affect rod position, however, information is not specified in ABN-712.
B. Correct. Per ABN-712, Step 6 bullet 3 states to "CHECK previous Plant Computer thermocouple map and current thermocouple map approximately equal (Refer to Attach. 3)."
C. Incorrect. Plausible because Rod movement near the Hot Legs could affect Hot Leg temperature but would not be recognizable unless the inserted rod was near the Hot Leg. Information is not specified in ABN-712.
D. Incorrect. Plausible because Rod movement near the Hot Legs could affect Hot Leg temperature which in turn could affect Cold Leg temperatures but is incorrect for same reason as above. Information is not specified in ABN-712.

Technical References:
ABN-712, Revision 11, Section 4.3, Step 6

References to be provided to applicants during exam: None.

Learning Objective: LO21.SST.ROD.OB04.

Question Source: Bank # ON 2015 NRC EXAM

(note changes; attach parent)	Modified Bank #	
	New	
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	Level	RO
	Tier #	1
074 Inadequate Core Cooling	Group #	2
	K/A #	2.4.1
2.4.1 Knowledge of EOP entry conditions and immediate action steps.	Rating	4.6
	QREV	7

Question 65

Given the following conditions:

- Unit 1 has experienced a LOCA
- Several plant challenges have caused the core to start heating up
- The US asks the RO to independently verify that the entry conditions for FRC-0.1A, RESPONSE TO INADEQUATE CORE COOLING, have been met

Per FRC-0.1A bases, which indication provides the MINIMUM required entry conditions?

A. The AVERAGE of the five highest Core Exit Thermocouples displayed on the SPDS CETMAP screen indicate greater than 1200 ^{0}F on the plant computer

B. The AVERAGE of the five highest Core Exit Thermocouples displayed on the SPDS CETMAP screen indicate greater than 1000 ^{0}F on the plant computer

C. BOTH 1-TI-3611-2 and 1-TI-3612-2, CORE EXIT TEMP meters on CB-05 indicate greater than 1200 ^{0}F

D. BOTH 1-TI-3611-2 and 1-TI-3612-2, CORE EXIT TEMP meters on CB-05 indicate greater than 1000 ^{0}F

Answer: C
Explanation:
A is wrong because this isn’t how it is done at CP and these are not the PAM instruments. The CCM analog displays are the PAM instruments and they would be used first. Plausible because this is how the FRC-0.1A bases describes it for a typical Westinghouse plant. This is not how CP does it, however. Temperature is correct.
B is wrong because both first part and second part are incorrect, Tem is 1200F not 1000F.
C is correct because you would use both trains of the CCM analog displays on the MCB to verify this and they would need to be above 1200F.
D is wrong because the temperature is incorrect (first part is correct).

Technical References:
FRC-0.1A, “Response to Inadequate Core Cooling,” rev. 9, p, 42.

References to be provided to applicants during exam: None.

Learning Objective: DEFINE the following situations IAW the FRGs and FSAR:
1) Adequate core cooling.
2) Degraded core cooling.
3) Inadequate core cooling.
(LO21.MCO.MI2.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)(10) 55.43	

Examination Outline Cross-Reference	Level	RO
2.3.15 Radiation Control	Tier #	3
	Group #	
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)	K/A #	2.3.15
	Rating	2.9
	QREV	6

Question 66

An Area Radiation Monitor channel with an "Operate Failure" status will display which color of background on PC-11, Digital Radiation Monitoring System?

A. Red

B. Blue

C. Magenta

D. Light Blue

Answer: B

Explanation:

A is wrong. Red background on PC-11 channel indicates Channel High Alarm.
B is correct.
C is wrong. Magenta background on PC-11 channel indicates PC-11 Communications failure.
D is wrong. Light Blue background on PC-11 channel indications PC-11 Equipment failure.

Technical References:
Digital Rad Monitoring Study Guide, 4-28-2011, p. 39.

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the normal, abnormal and emergency operation of the Digital Radiation Monitoring System. (LO21.SYS.RM1.OB05)

Question Source:	Bank #	
(note changes; attach parent)	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)(12)	
	55.43	

A monitor that is flashing indicates an alarm condition that has not been acknowledged.

- Normal Operation - Green
- PC-11 Poll Status - White (Gray)
- PC-11 Communications - Magenta
- Operate Failure - Blue
- Channel High Alarm - Red
- Channel Alert Alarm - Yellow
- Equipment Failure - Light Blue

The following is a brief description of what each of the monitor status items mean.

Normal Operation - Indicates that the monitor is operating with no abnormal conditions.

PC-11 Poll Status - As discussed earlier, each RM-80 is connected to the PC-11s via a daisy chained communication loop. The PC-11 requests information (polls) from each monitor in the communication loop in turn. A PC-11 Poll Status is indication that the monitor has been removed from the communication loop of the PC-11. A monitor may be removed from the poll status by the RM operator using the PC-11. A monitor removed from poll will not generate any alarms on the PC-11; however, the trend data, sample and process flow will remain accurate. If an RM-23 is connected to the monitor, it will remain operable. Also, any monitor automatic actuations status will remain available unless the monitor has been removed from poll by the PC-11.

PC-11 Communications - Indicates a monitor communication failure or a channel not responding to the poll from either SCADA “A” or SCADA “B”. Possible causes for a monitor communication failure include:

- I&C activities on the monitor,
- Loss of RM-80 power,
- RM-80 Communication ON-LINE switch in the RM-80 in the bypass ON position,
- Malfunction of communication loop.

A channel not responding to poll may be due to maintenance activities, incorrect number of channels defined in the database, or during an RM-80 reset.

Operate Failure - The operate failure alarm was discussed in the RM-80 section. As a review the following items may cause an operate failure:

- Monitor Database Unknown
- Monitor Loss of Sample Flow
- Channel Out of Service
- Channel Filter Not Moving

- Channel No Pulses Received
- Channel Check Source Test Failed
- Channel Loss of Sample Flow
- Channel Operate Failure

Channel High Alarm - Channel has exceeded the database high alarm setpoint.

Channel In Alert Alarm - Channel has exceeded the database alert alarm setpoint.

Equipment Failure - Several items will generate an Equipment Failure. Each of these items is discussed below.

Monitor Loss of Process Flow -Probable causes include the following.

- Process flow rate outside the specified band (high or low),
- Inadequate ventilation alignment,
- Complete loss of process flow input,
- Incorrect value in the RM-80 database,
- Failure of the ADC board in the RM-80

Monitor in Scan Overload - The PC-11 polls the RM-80 at six second intervals to update sample flow, process flow, and radiation levels. Alarm poll status is conducted every two seconds. If the RM-80 fails to perform the activity calculations within the allotted time, the alarm is initiated. This may occur due to a simultaneous operator requested functions or calculation time exceeding the time requirements.

Monitor Loss of Flow Control - This was discussed in the RM-80 section. This alarm may occur due to any of the following conditions.

- Rapid or excessive changes in process flow,
- Faulty flow control valve, relay, or switch,
- Too many primary plant exhaust fans in service, or
- Clogged filters.

Monitor Loss of Isokinetic Control - This was discussed in the RM-80 section. The same conditions as Loss of Flow Control will cause this alarm.

Monitor Loss of RM-23 Communications - This occurs when the RM-80 is no longer able to communicate with the RM-23. Possible causes are as follows.

- Loss of power to the RM-23,
- Failure of the communication line to the RM-23.

Examination Outline Cross-Reference	Level	RO
Tier 3	Tier #	3
	Group #	
2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	K/A #	G2.2.42
	Rating	3.9
	QREV	7

Question 67

The Technical Specification entry-level condition for QPTR is __(1)___ and is only applicable when ___(2)______.

A. Greater than or equal to 1.02
   Thermal Power is greater than 50% RTP

B. Greater than or equal to 0.99
   Thermal Power is greater than 75% RTP

C. Greater than or equal to 0.99
   Thermal Power is greater than 50% RTP

D. Greater than or equal to 1.02
   Thermal Power is greater than 75% RTP

Answer: A
Explanation:
A is correct per TS 3.2.4 (QPTR < 1.02 and is applicable when thermal power is > 50% RTP).
B is wrong (see A above). Credible because 0.99 is another value in the TS and 75% RTP is also a valid TS value for core parameters.
C is wrong (see A above). Credible because 0.99 is another value in the TS and the second part is correct.
D is wrong because although first part is correct the second part is wrong. Credible because 75% RTP is also a valid TS value for core parameters.
Technical References:
TS 3.2.4, Amendment 150, page 3.2-11.

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	2
	Comprehensive/Analysis	

10CFR Part 55 Content: 55.41(b)2…

Examination Outline Cross-Reference	Level	RO
Emergency Procedures / Plan	Tier #	3
	Group #	
Knowledge of fire protection procedures.	K/A #	2.4.25
(CFR: 41.10)	Rating	3.3
	QREV	7

Question 68

In accordance with the Precautions of SOP-904, “Fire Protection Water Supply and Fire Pumps System,” Fire Protection water may be sent to the Vents and Drains Systems associated with which of the following SSCs during normal system alignments?

A. Turbine Building

B. Safeguards Building

C. Fuel Building

D. Auxiliary Building

Answer: A
Explanation:

Per SOP-904, Precaution 3.1: “The fire water supply system contains chemicals that will adversely affect demineralizer resins. Fire Protection water shall not be admitted to the Fuel, Containment, Electrical Control, Auxiliary or Safeguards Buildings Vents and Drains System, as these drains are processed through the Rad. Waste Processing System. Fire Protection water may be sent to the Turbine and Diesel Generator buildings Vents and Drains System, unless these drains are being processed through the Rad. Waste Processing System.”

A is Correct.
B is wrong because Fire Protection water shall not be admitted to the Safeguards Building Vent and Drain system.
C is wrong because Fire Protection water shall not be admitted to the Fuel Building
D is correct because Fire Protection water shall not be admitted to the Auxiliary Building

Technical References:
SOP-904, “Fire Protection Water Supply and Fire Pumps System,” Rev 17, Precaution 3.1

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the administrative requirements for normal operation of the Fire Protection system in accordance with Fire Protection Report, SOP-904 and OPT-220. (LO21.ADM.FP1.OB02)

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
2.3.5 Radiation Control	Tier #	3
	Group #	
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9)	K/A #	2.3.5
	Rating	2.9
	QREV	6

Question 69

You are exiting an RCA and approach the portable frisker which is reading 350 cpm background radiation. Based on this information, answer the following questions:

1) Are you allowed to perform your whole body frisk in an area with background radiation at this level?

2) When you do perform a whole body frisk (in this area or another) you should:

A. (1) NO
(2) Start on the lowest scale and go up one scale at a time until the meter is on scale.

B. (1) NO
(2) Start on the highest scale and go down one scale at a time until the meter is up on scale.

C. (1) YES
(2) Start on the lowest scale and move up one scale at a time until the meter is on scale.

D. (1) YES
(2) Start on the highest scale and go down one scale at a time until the meter is on scale.

Answer: A

Explanation:
This question was used on the 2015 NRC Exam. By changing the value to 350 cpm background, it makes it modified because the answers are now different.

A is correct. Frisking may only be performed in areas with the background radiation < 300 cpm. Frisking should be started on the X1 (lowest) scale.

B is wrong. 1st part is correct. 2nd part is incorrect because frisking should be started on the X1 scale. It is plausible because scales are set up such that they go up by a factor of 10 for every scale. Using that knowledge of how the scales work, it would figure that the applicant could think that X1 (0-10), X10 (0-100), X100 (0-1000) cpm. With this philosophy, it would reason that they would start on the highest scale so as not to “peg” the meter.

C is wrong 1st part is incorrect because background count rates are > 300 cpm, therefore NO (not allowed). It is plausible because it is well over the preferred background level (100 cpm). 2nd part is correct.

D is wrong. 1st part is incorrect but plausible (see C). 2nd part is incorrect but plausible (see B).

Technical References:
STA-653, CONTAMINATION CONTROL PROGRAM, Rev 16, pp. 12, 18.

References to be provided to applicants during exam: None.

Learning Objective: From memory, discuss the radiological principles in place at CPNPP to minimize personnel exposure in accordance with Radiation Protection Procedures and 10CFR20, Standards for Protection Against Radiation. (LO21.ADM.RAD.OB00)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	NRC 2015-03 Q71
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41(b)(12)	
	55.43	

6.6.7 Protective Clothing worn inside a Contaminated Area should be removed at the step off pad.
[CR-2011-005658]

6.6.8 Attachment 2 provides guidance for donning and removing PCs.

[C] 6.7 <u>Personnel Monitoring</u>
[00816]

6.7.1 Contamination monitoring requirements should be posted at the exit of Satellite/Alternate RCA's.
[CR-2011-005658]

6.7.2 Unless otherwise posted or authorized, all personnel shall monitor themselves after handling contaminated materials or exiting a contaminated area, at the nearest available frisker or PCM, and when exiting at the access control point.

6.7.3 The frisker is most commonly used for monitoring after exiting a contaminated area or after handling contaminated material. Frisking should be done with a background count rate of less than 300 counts per minute (cpm).

6.7.4 The Personnel Contamination Monitor (PCM) is most commonly used at the access control point, although it may be used to replace the frisker at locations such as the Reactor Building Personnel Air Lock.

6.7.5 Hand-held friskers should be available at or near normally established step-off pads when PCM's are not readily available.
[CR-2011-005658]

6.7.6 Hand held friskers should be available at or near each normally established RCA egress point for RP Technician use to respond to quantify contamination alarms.
[CR-2011-005658]

6.7.7 Portal Monitors will most commonly be used when exiting the protected area (e.g., PAP, AAP).

6.7.8 See Attachment 3 for guidelines on the use of friskers, PCMs and portal monitors.

6.7.9 When personnel contamination is found, Radiation Protection should evaluate and document the extent of the contamination. Radiation Protection should decontaminate the individual in accordance with RPI-402.

ATTACHMENT 3
PAGE 1 OF 2

GUIDELINES FOR PERSONAL MONITORING

Monitoring With a Frisker

> **NOTE:** Due to background radiation levels some friskers may indicate a background count rate greater than 300 cpm. These friskers may be used to perform a gross contamination check. In-plant low background frisker stations are provided as necessary.

1. Ensure meter is turned on and the scale switch is set at X1. Observe background level momentarily.

2. Without picking up the probe, frisk both sides of one hand. The probe should be about ½ inch away from the surface area being frisked.

3. Pick up probe and frisk remainder of body, scanning at a slow rate. Special attention shall be given to the face, soles of feet, hands, knees, posterior, and any surface left exposed while wearing protective clothing and dosimetry.

4. If an increase in the count rate is noted (visual or audible), return the probe to the spot and verify count rate. A significant and abrupt rise/drop in the count rate may indicate the presence of a DRP. Notify Radiation Protection.

5. If the frisker alarms or a continuous count rate of 100 cpm above background or greater is noted, remain at that point and notify, or have a co-worker notify, Radiation Protection for assistance. If contamination is not detected, proceed as usual.

Examination Outline Cross-Reference	Level	RO
	Tier #	3
	Group #	
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	K/A #	2.1.31
	Rating	4.6
	QREV	6

Question 70

Post accident Monitoring instrumentation can be quickly located in the control room because its labeling is color-coded as ____________.

A. black labels with white letters

B. white labels with black letters

C. red labels with white letters

D. white labels with red letters

Answer: A

Explanation:
A is correct per the reference.
B is wrong because the colors are reversed and plausible because of this also.
C is wrong (see A above) but plausible because other labels have this color scheme in the CR.
D is wrong (see A above)) but plausible because other labels have this color scheme in the CR.

Technical References:
LO21.MCO.MI7.ppt, revision 8/24/2017, slide 8.
References to be provided to applicants during exam: None.

Learning Objective: LO21.MCO.MI7.OB01

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.41.10	

Examination Outline Cross-Reference	Level	RO
Equipment Control	Tier #	3
	Group #	
Knowledge of limiting conditions for operations	K/A #	2.2.22
and safety limits. (CFR: 41.5)	Rating	4.0
	QREV	6

Question 71

In accordance with Safety Limit SL 2.1.2, RCS Pressure, what is the maximum allowable RCS pressure in MODE 2, and the action REQUIRED if this limit is exceeded?

A. 2485 psig; restore compliance and be in MODE 3 within one hour.

B. 2485 psig; restore compliance within 5 minutes.

C. 2735 psig: restore compliance and be in MODE 3 within one hour

D. 2735 psig: restore compliance within 5 minutes.

Answer: C

Explanation:

A is wrong because This is the design pressure. Action is correct for exceeding limit.
B is wrong because This is the design pressure. Also the action listed would be correct for exceeding the limit in MODES 3-5, not MODE 1 or 2.
C is correct. 2735 is the max allowable pressure at MODE 2, calculated from design pressure (2495) x 110%. Action is to restore pressure and be in MODE 3 within 1 hour.
D is wrong because while the pressure is correct, the action is to restore pressure and be in MODE 3 within 1 hour, not 5 minutes. The given action would be correct for exceeding the limit in MODES 3-5.

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The SL on maximum allowable RCS pressure is 2735 psig.

The design pressure of the RCS is 2485 psig. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code. To ensure system integrity, all RCS components are hydrostatically tested at 125% (3107 psig) of design pressure.

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits.

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

Technical References:
Tech Spec Safety Limit 2.1.2 Amendment 150

References to be provided to applicants during exam: None.

Learning Objective: EXPLAIN the term “Safety Limit” as it applies to the Technical Specifications and DESCRIBE the Safety Limits for Comanche Peak. (LO21.RLS.SL1.OB05)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X DCPP 2016-04 Q71
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
G2.2.22 -Knowledge of limiting conditions for operations and safety limits.	**Tier #**	3
	Group #	2
	K/A #	2.2.22
	Rating	4.0

Question 71

What is the maximum allowable RCS pressure, and the action required, if this limit is exceeded in MODE 2, in accordance with Safety Limit SL 2.1.2, RCS Pressure?

A. 2500 psig; restore pressure and be in MODE 3 within one hour

B. 2500 psig; restore pressure within 5 minutes

C. 2735 psig: restore pressure and be in MODE 3 within one hour

D. 2735 psig: restore pressure within 5 minutes

Proposed Answer: C. 2735 psig: restore pressure and be in MODE 3 within one hour

Explanation:

A. Incorrect. This is the design pressure. Action is correct for exceeding limit.
B. Incorrect. This is the design pressure. Action listed is correct for exceeding the limit in Modes 3-5.
C. Correct. 2735 is design pressure (2500) x 110%. Action is to restore pressure and be in MODE 3 within 1 hour.
D. Incorrect. Action is to restore pressure and be in MODE 3 within 1 hour, however the pressure is correct. Action listed is correct for exceeding the limit in MODE 3-5.

Technical References: SL 2.1
References to be provided to applicants during exam: None
Learning Objective: 9701 -Explain Technical Specification 2.1 & 2.2 (Safety limits)

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

Examination Outline Cross-Reference	Level	RO
	Tier #	3
	Group #	
Knowledge of the station's requirements for verbal communications when implementing procedures.	K/A #	G2.1.38
	Rating	3.7
	QREV	6

Question 72

The required communication protocol for steps in a procedure are the following:

1. Steps with a circle around them are _____(1)_____.
2. Steps with a diamond around them are ____(2)_____.

A. 1. performed using three way communication
   2. performed without two or three way communication

B. 1. performed without communication
   2. verbalized without two or three way communication

C. 1. verbalized without two or three way communication
   2. performed without communication

D. 1. performed using three way communication
   2. performed with two or three way communication

Answer: B

Explanation:
A is wrong because diamond steps are ABN initial operator actions and they are verbalized without two or three way communication (it is neither expected nor desired) and circle steps are immediate action steps such as the first four steps in EOP-0.0 and are performed without communications. Plausible to pick this if confused over when three way communications are required, desired, and/or if the step annotations are not known (diamond or circle).
B is correct because diamond steps are ABN initial operator actions and they are verbalized without two or three way communication (it is neither expected nor desired) and circle steps are immediate action steps such as the first four steps in EOP-0.0 and are performed without communications.
C is wrong because each part is backwards (see C below). Plausible for same reasons as plausibility in A.
D is wrong because neither of these are used for either situation. Plausible for same reasons as plausibility in A.

Technical References:
OPD1.ADM.XA1.LN.doc, rev May 8, 2007, page 34

References to be provided to applicants during exam: None.

Learning Objective: OPD1.ADM.XA1.OB00

Question Source: Bank #

(note changes; attach parent)	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	Level	RO
Tier 3	Tier #	**3**
	Group #	
2.1.34 Knowledge of primary and secondary plant chemistry limits.	K/A #	G2.1.34
	Rating	2.7
	QREV	6

Question 73

With a Unit in Mode 5, ________is added to the RCS to scavenge Oxygen and prior to RCS temperature rising above 250ºF, Oxygen levels must be lowered to below the limit of ________.

A. Lithium
2000 ppb

B. Lithium
100 ppb

C. Hydrazine
2000 ppb

D. Hydrazine
100 ppb

Answer: D

Explanation:

A. Incorrect: Part 1 is incorrect but plausible because Lithium is added to the RCS, but it is added for pH control vice Scavenging Oxygen. Part 2 is incorrect but plausible because 2000 ppb is the limit that must be enforced prior to exceeding 180 ºF.

B. Incorrect: 1st part is incorrect but plausible, see A above. Part 2 is correct, prior to RCS temperature exceeding 250 ºF, Oxygen levels must be less than 100 ppb.

C. Incorrect: 1st part is correct Hydrazine is added to the RCS when at reduced temperatures.. 2nd part is incorrect but plausible (see A).

D. CORRECT: 1st part is correct, see C above. Part 2 is correct, see B above

Technical References:

STA-609, REACTOR COOLANT WATER CHEMISTRY CONTROL PROGRAM (Rev 11), Attachment 8.D, REACTOR COOLANT SYSTEM COLD SHUTDOWN (RCS Temperature ≤ 250ºF), page 24

References to be provided to applicants during exam: None.

Learning Objective: LO21.GFE.PC1.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	4
10CFR Part 55 Content:	55.41(b)5	

Examination Outline Cross-Reference	Level	RO
	Tier #	3
Knowledge of abnormal condition procedures.	Group #	
	K/A #	2.4.11
	Rating	4.0
	QREV	6

Question 74

A fire is reported in the plant. After identifying the location you expect the US to report an ABN entry in the ___ series.

A. ABN-600

B. ABN-700

C. ABN-800

D. ABN-900

Answer: C

Explanation:
A is wrong because the 600 series is for electrical malfunctions.
B is wrong because the 700 series is for instrument malfunctions.
C is correct because the 800 series is for fire responses
D is wrong because the 900 series is for various other incidents.

RO only because not asking for specific entry simply asking overall procedure structure.
Technical References:
ABN table of contents

References to be provided to applicants during exam: None.

Learning Objective: LO21.ADM.FP1.OB04

Question Source:	Bank #	
(note changes; attach parent)	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)(10)	
	55.43	

Examination Outline Cross-Reference	Level	RO
Tier 3	Tier #	**3**
	Group #	
2.2.43 Knowledge of the process used to track inoperable alarms.	K/A #	G2.2.43
	Rating	3.0
	QREV	6

Question 75

Out-of-Service Annunciators are identified in the Control Room by…

A. removing the bulbs.

B. blanking the alarm window.

C. flipping down the alarm cover.

D. placing a dot on the alarm window.

Answer: D

Explanation:

A. is wrong but plausible because removing the bulbs would ensure that no field input could cause that alarm to illuminate in the Control Room.

B. is wrong but plausible because there are multiple blank alarms in the Control Room and the alarm window would provide no description of a faulty plant condition.

C. is wrong but plausible because this would be a recognizable way of determining troubleshooting of the alarm was on-going.

D. is correct because per ODA-401 any out-of-service alarm will have dot placed on the alarm window.

Technical References:
ODA-401, Control Of Annunciators, Instruments and Protective Relays, Rev 11.

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)10	

Examination Outline Cross-Reference	Level	SRO
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4	Tier #	1
	Group #	1
	K/A #	G2.2.38
2.2.38 Knowledge of conditions and limitations in the facility license	Rating	4.5
	QREV	6

Question 76

Unit 2 is operating at 100% power.
An inadvertent Train ‘A’ Containment Spray actuation has just occurred.

2-HS-4701, RCP CLR CCW RET ISOL VLV, is closed.
2-HS-4701, RCP CLR CCW RET ISOL VLV, must be overridden and opened within a MAXIMUM of _____(1)_____ or the reactor must be tripped.

When 2-HS-4701 is overridden and opened, the penetration must be isolated within a MAXIMUM of _____(2)_____ to comply with LCO 3.6.3, Containment Isolation Valves.

A. 1) Ten minutes
   2) Four hours

B. 1) Ten minutes
   2) Six hours

C. 1) Six minutes
   2) Four hours

D. 1) Six minutes
   2) Six hours

Answer: A

Explanation:
A is correct because the TCOA table requires CCW be restored within 10 minutes and LCO 3.6.3 condition A requires restoration within 4hours.
B is wrong because part 2 is wrong but is plausible because there are other LCOs that are six hours.
C is wrong because part 1 is wrong (see A above) plausible because other TCOA are six minutes.
D is wrong because both parts are wrong (see A above) and plausible because these choices are available in the TS table and in the TCOA tables.

Technical References:
ABN-101, rev 12, page 42, STI-214.01, rev 1, page 18, TS 3.6.3, Amend 150, page 3.6-8, TRM, rev 88, page 16.6-4.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.101.OB05

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:	55.43(b)2	

Examination Outline Cross-Reference	Level	SRO
000009 (EPE 9) Small Break LOCA / 3	Tier #	1
	Group #	1
EA2.29 Ability to determine or interpret the following as they apply to a small break LOCA: CVCS pump indicating lights for determining pump status. (CFR 43.5 / 45.13)	K/A #	EA2.29
	Rating	3.4
	QREV	6

Question 77

A manual reactor trip and safety injection have occurred due to a Small Break LOCA concurrent with a Loss of Offsite Power.

(1) The monitor light box on CB-02 will show pump running indication for which charging pumps?
(2) The Positive Displacement Charging Pump may be credited as a required boration injection subsystem in which Modes?

A. (1) Centrifugal Charging Pumps and Positive Displacement Charging Pump.
(2) Modes 4, 5, 6 ONLY.

B. (1) Centrifugal Charging Pumps and Positive Displacement Charging Pump.
(2) All Modes of Operation.

C. (1) Centrifugal Charging Pumps only.
(2) Modes 4, 5, 6 ONLY.

D. (1) Centrifugal Charging Pumps only.
(2) All Modes of Operation.

Answer: C

Explanation:

The Positive Displacement Charging Pump is powered from safeguards bus 1EB1, but is load shed and not restarted on a Safety Injection Signal, as it is not credited for SBLOCA mitigation. Positive Displacement Charging Pump running indication is provided in the control room by breaker position lights on the pump control switch on CB-06. A 52ax relay, which is deenergized when the pump motor breaker is open, provides one of the contact inputs for the LOAD SHED COMPLETE 1EB1 AND 1EB3 window of Monitor Light Box 9 (MLB-9) on CB-10.

Centrifugal Charging Pump running indication is provided in the control room by monitor light box indication on CB-02 and by breaker position lights on the pump control switch on CB-06. The plant computer monitors CCP breaker position.

The Positive Displacement Pump may be credited as a required boration source in both TRM 13.1.31 (Mode 4 only) and 13.1.32. TRM 13.1.31 is applicable in modes 1,2,3,4, but a NOTE

specifies that the PDP may satisfy the requirement for 2 boration injection sources in Mode 4 only.

A is incorrect. Part 1 in wrong because the PDP does not have a pump running indication on CB-02 Monitor Light Box, though its motor status is an input to CB-10 Monitor Light Box window LOAD SHED COMPLETE 1EB1 AND 1EB3. Part 2 is correct, even though the Mode of applicability for TRM 13.1.31 is Modes 1,2,3, and 4. A note at the beginning of the spec allows for the PDP to be credited as boration injection source (vice one CCP) only in Mode 4, with Modes 5 and 6 credited per TRM 13.1.32

B is incorrect. Part 1 is wrong, see A above. Part 2 is incorrect but plausible because the PDP may be used as a boration injection source (vice one CCP) only in Mode 4, even though the Mode of applicability for TRM 13.1.31 is Modes 1,2,3, and 4. In Modes 5 and 6 the PDP can be used as a boration injection source per TRM 13.1.32.

C is correct. Part 1 is correct. While PDP motor status is an input to Monitor Light Box indication on CB-10, only the Centrifugal Charging Pumps have pump running indication on Monitor Light Box on CB-02. Part 2 is correct. See A above

D is incorrect. Part 1 is correct. See C above. Part 2 is incorrect. See B above.

Technical References:
TRM 13.1.31
TRM 13.1.32
CVCS Study Guide, 4-28-11, Pages 33 and 36.

References to be provided to applicants during exam: None.

Learning Objective: DISCUSS the Technical Specifications and Bases associated with the following specifications: 1) TS 3.5.2, “ECCS-Operating”; 2) TS 3.5.3, “ECCS-Shutdown”; 3) TR 13.1.31, “Boration Injection System-Operating”; 4) TR 13.1.32, “Boration Injection System-Shutdown”. (LO21.SST.CS1.OB08)

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:		
	55.43(b)2	

13.1 REACTIVITY CONTROL SYSTEMS

TR 13.1.31 Boration Injection System - Operating

TR LCO 13.1.31 Two boration injection subsystems shall be OPERABLE:

- NOTES -

1. TR LCO 13.0.4.c is applicable for entry into MODES 3 and 4 for the charging pump declared inoperable pursuant to TS 3.4.12 provided the charging pump is restored to OPERABLE status within 4 hours after entering MODE 3 or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

2. In MODE 4 the positive displacement pump may be used in lieu of one of the required centrifugal charging pumps.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One boration injection subsystem inoperable (except due to required charging pump inoperable).	A.1 Restore boration injection subsystem to OPERABLE status.	72 hours
B. One charging pump inoperable.	B.1 Restore charging pump to OPERABLE status.	7 days
C. Two boration injection subsystems inoperable. <u>OR</u> Required Actions and associated Completion Times not met.	C.1 Initiate action to restore at least one boration injection subsystem to OPERABLE status. <u>AND</u> C.2 Initiate Engineering Evaluation to identify compensatory actions to be completed in a timely manner.	Immediately Immediately

13.1 REACTIVITY CONTROL SYSTEMS

TR 13.1.32 Boration Injection System - Shutdown

TR LCO 13.1.32 One boration injection subsystem shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 5 and 6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required boration injection subsystem inoperable. OR Required boration injection subsystem not capable of being powered from an OPERABLE emergency power source.	A.1 Suspend all operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately

SURVEILLANCE REQUIREMENTS

- NOTES -

1. TRS 13.1.32.2, TRS 13.1.32.3, TRS 13.1.32.4, and TRS 13.1.32.5 are only required to be met when the boric acid storage tank is the required borated water source.
2. TRS 13.1.32.1, TRS 13.1.32.6, and TRS 13.1.32.7 are only required to be met when the RWST is the required borated water source.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TRS 13.1.32.9	Demonstrate the required positive displacement charging pump is OPERABLE by verifying that the flow path from the boric acid storage tanks via a boric acid transfer pump or gravity feed connection to the Reactor Coolant System is capable of delivering at least 30 gpm to the RCS.	18 months
TRS 13.1.32.10	Demonstrate the required centrifugal charging pump is OPERABLE by verifying, on recirculation flow, that a differential pressure across the pump of greater than or equal to 2370 psid is developed.	In accordance with the Inservice Testing Program
TRS 13.1.32.11	Demonstrate the required safety injection pump is operable by verifying, on recirculation flow, the requirements of the IST Program are met for developed head.	In accordance with the Inservice Testing Program

BASES

LCO (continued)

b. A borated water source from either a Boric Acid Tank (BAT) or RWST must be available.

c. An injection flow path must be available from either:

1. The required BAT via a boric acid transfer pump or gravity feed connection through a charging pump to the RCS. Charging suction must also be aligned to the VCT or RWST when discharge is aligned to the High Head Injection flow path, or,

2. The RWST via either a centrifugal charging pump, positive displacement pump or a safety injection pump (only in MODE 6 with the head removed) to the RCS.

An OPERABLE Boration Injection System is required to be capable of being manually aligned when boration is required, and in all cases within 15 minutes, to establish boration flow. When local manual control is being credited, administrative controls are utilized to ensure 1) appropriate personnel are aware of the status of the Boration Injection System, and 2) specified individuals are designated and readily available to perform the valve alignment necessary to establish boration flow.

Examination Outline Cross-Reference	Level	RO
(W E11) Loss of Emergency Coolant Recirculation / 4	Tier #	1
	Group #	1
	K/A #	EA2.1
Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation) EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	Rating	4.2
	QREV	6

Question 78

Unit 1 experienced a LOCA and is currently in EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT."

- As the US, you are evaluating Step 11(a) to "Verify cold leg recirculation capability."
- Current RCS pressure is 450psig and slightly decreasing.
- Containment pressure is 13 psig and slightly decreasing.

You have the following ECCS equipment available/running:

- RHR pump A
- CCW to RHR pump A
- 1/1-8811B, CNTMT SMP TO RHRP 2 SUCT ISOL VLV
- SI pump A
- 1/1-8804B, RHRP 2 TO SIP SUCT VLV

The following ECCS equipment is NOT available/operable:

- All of Train B ECCS pumps
- 1/1-8804A, RHRP 1 TO CCP SUCT VLV
- 1/1-8811A, CNTMT SMP TO RHRP 1 SUCT ISOL VLV
- CCP pump A
- 1/1-8807A, SI TO CHRG SUCT HDR XTIE VLV
- 1/1-8807B, SI TO CHRG SUCT HDR XTIE VLV

What is your next action as the US?

A. Transition to EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION

B. Transition to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION

C. Stay in EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT

D. Transition to ECA-1.3A, CONTAINMENT RECIRCULATION SUMP BLOCKAGE

Answer: B

Explanation:
A is wrong because the next step is the RNO for step 11 which is to transition to ECA-1.1A because you don't have all pumps and valve combinations to complete a defined train

available for this EOP. Plausible because there is a transition out of this procedure to EOS-1.2A but the conditions are not met in this step and you are not at that step yet either
B is correct because the RNO states that “IF at least one train of cold leg recirculation capability can NOT be verified, THEN go to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.These are the conditions given-neither train will work as given and will need some help from ECA-1.1A to get alignments correct.
C is wrong because you don’t have the cross connect valves to stay in EOP-1.0A.. if you don’t realize you need a set of pumps and valves to swap to the sump and if you don’t have them-the next place to get them is in the ECA1.1A procedure (unless you have confirmed sump blockage, which is not given in this stem).
D is wrong because there are no indications of sump blockage in the stem but plausible because it does have alternate lineups and uses CS to get some core cooling going, but there is no transition yet to this procedure.

Technical References:
EOP-1.0A, rev 9, page 12-13.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.C11.OB03.

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:	55.43(b)5	

Examination Outline Cross-Reference	Level	SRO
	Tier #	1
022 Loss of Reactor Coolant Makeup	Group #	1
	K/A #	AA2.02
Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: AA2.02 Charging pump problems	Rating	3.7
	QREV	6

Question 79

Given the following conditions:

- Unit 1 is at 100% power.
- Centrifugal Charging Pump (CCP) 1-01 is in service.
- Volume Control Tank (VCT) level is 50%.
- 1-FI-121A, CHRG FLO is stable at 130 gpm.
- 1-FI-132, LTDN FLO is stable at 120 gpm.
- 1/1-LCV-112B, VCT TO CHRG PMP SUCT VLV spuriously closes.
- CCP 1-01 discharge pressure is beginning to oscillate.

The following alarms are received;

- 1-ALB-6A, Window 1.4, REGEN HX LTDN OUT TEMP HI
- 1-ALB-6A, Window 3.4, CHRG FLO HI/LO

What action(s) must be taken and the procedure used?

A. Open 1-1-LCV-112C, VCT TO CHRG PMP SUCT VLV per ABN-103, Excessive Reactor Coolant System Leakage.

B. Open 1-1-LCV-112C, VCT TO CHRG PMP SUCT VLV per ABN-105, Chemical and Volume Control System Malfunctions.

C. Stop Centrifugal Charging Pump 1-01 and then isolate letdown per ABN-103, Excessive Reactor Coolant System Leakage.

D. Stop Centrifugal Charging Pump 1-01 and then isolate letdown per ABN-105, Chemical and Volume Control System Malfunction.

Answer: D

Explanation:
A is wrong because ABN-103 contains the step this will not help since the valves are in series. Plausible if an applicant is confused on the design of the system. Also, if an applicant knows there are no step in ABN-105 to open VCT valves they may want to go to this procedure because it might save the day. Also, not initial steps of the procedure (why it is not an RO question).
B is wrong because ABN-105 does not contain this step.
C is wrong because this is not the correct procedure.
D is correct because the stem would lead to pump cavitation requiring the use of ABN-105

May need to add some false indication for an RCS leak to make ABN-103 more plausible.
The original question referenced an SOP. Changed for better plausibility.

Technical References:

ABN-105, Chemical and Volume Control System Malfunction, p. 27

References to be provided to applicants during exam: None.

Learning Objective: DISCUSS ABN-105, "Chemical and Volume Control System Malfunctions", to include the following: 2) Symptoms
(LO21.SST.CS1.OB06)

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	 X (2013-04, Q77)
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	Level	RO
000027 (APE 27) Pressurizer Pressure	Tier #	1
Control System Malfunction / 3	Group #	1
	K/A #	G2.1.12
2.1.12 Ability to apply technical specifications	Rating	4.0
for a system	QREV	6

Question 80

Unit 1 is at 85% power when the following conditions occurred during post-outage Turbine Digital Control Testing:

- During a 3% per minute power reduction Pressurizer Spray Valve PCV-455C failed partially open.
- The US entered ABN-705, Pressurizer Pressure Malfunction, and crew actions are in progress.
- Pressurizer pressure lowered to 2175 psig during the transient and is now recovering slowly.
- All four (4) Pressurizer pressure channels are OPERABLE.

1. What action in the ABN is taken to close the spray valve?

2. What Technical Specification ACTION should be taken during the restoration?

A. 1) Remove the Pressurizer Pressure Control driver card for PCV-455C
 2) Restore Pressurizer pressure to meet DNB limits within two hours

B. 1) Remove the fuses for the solenoid control valve for PCV-455C
 2) Maintain THERMAL POWER less than 85% RTP immediately

C. 1) Remove the fuses for the solenoid control valve for PCV-455C
 2) Restore Pressurizer pressure to meet DNB limits within two hours

D. 1) Remove the Pressurizer Pressure Control driver card for PCV-455C
 2) Maintain THERMAL POWER less than 85% RTP immediately

Answer: A

Explanation:
A is correct per ABN-705, pull the driver card to fail closed the spray valve and the TS and COLR require that pressure be restored to at least 2220 psig within 2 hrs.
B is wrong because fuses are used for PORVs, not the spray valve and the 85% RTP is just below the LCO action for 2 hrs in this section, but is for flow DNB limits, which gives its plausibility.
C is wrong because driver card not fuses are used and the second part is correct.
D is wrong because although the first part is correct the second part is not as discussed above in B. …

Technical References:

ABN-705, rev 13, page 9 and TS 3.4.1(amendment 150), page 3.4-1 and 3.4-2. and U1 COLR for current cycle.

References to be provided to applicants during exam: None.

Learning Objective: ABN.705.OB01.006.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	33041
	New	
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	RO
000026 (APE 26) Loss of Component Cooling Water / 8	Tier #	1
	Group #	1
	K/A #	G2.4.18
2.4.18 Knowledge of the specific bases for EOPs.	Rating	4.0
	QREV	6

Question 81

- Unit 1 has experienced a Design Basis Loss of Coolant Accident.
- The crew has transitioned to EOS-1.3A, "TRANSFER TO COLD LEG RECIRCULATION."
- While working thru this procedure, CCW is suddenly lost and the CCW temperature starts to heat up.

What is the maximum post LOCA limit for CCW and the basis for this limit?

A. Once CCW temperature reaches 135 ^{0}F, the containment sump fluid temperature will start to increase with minimal heat transfer from the fuel.

B. Once CCW temperature reaches 120 ^{0}F, Containment Spray pump failure could occur if pumping sump fluid with CCW at this temperature at the CCW heat exchangers.

C. Once CCW temperature reaches 110 ^{0}F, RHR pump failure could occur if pumping sump fluid with CCW at this temperature at the CCW heat exchangers.

D. Once CCW temperature reaches 100 ^{0}F, Charging pump failure could occur if pumping sump fluid with CCW at this temperature at the CCW heat exchangers.

Answer: A

Explanation:
A is correct-see EOP bases for step 16, on page 54. 135F is the max CCW temperature in the EOP bases where no more heat transfer can occur during a LOCA event.
B is wrong because 150F is the temperature in the EOP bases where CS spray pumps can fail.
C is wrong because RHR pump failure could occur once the sump fluid temp reaches 120F for these pumps (not 100F), and this is not the max CCW temperature either.
D is wrong because the Charging pumps are not mentioned in the bases as a concern for max CCW temperature (safety chillers are) and this is not the max CCW temperature either, so this is incorrect as well.

Technical References:
EOS-1.3A (Bases), rev 9, pages 54, pages 38-39.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.E13.OB04.

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:	55.43(b)1	

Examination Outline Cross-Reference		Level	SRO
076 RCS Leak		Tier #	1
		Group #	2
G2.2.22 Knowledge of limiting conditions for operations and safety limits.		K/A #	G2.2.22
		Rating	4.7
		QREV	6

Question 82

Unit 1 Reactor power is 75%.
RCS leak rate data is as follows:

- Total RCS leakage rate is 9.1 gpm.
- Leakage to PRT is 7.0 gpm.
- Leakage to the Reactor Coolant Drain Tank is 1.3 gpm.
- Total primary to secondary leakage is 0.36 gpm.

- SG 1 - 0.09 gpm
- SG 2 - 0.19 gpm
- SG 3 - 0.07 gpm
- SG 4 - 0.01 gpm

Which of the following describes the RCS leakage limit that is being exceeded, and the required Technical Specification ACTION?

A. Unidentified Leakage; Reduce leakage to within limits in next 4 hours

B. Unidentified Leakage; Be in MODE 3 in the next 6 hours

C. Primary to Secondary Leakage; Be in MODE 3 in the next 6 hours

D. Primary to Secondary Leakage; Reduce leakage to within limits in next 4 hours

Answer: C

Explanation:
A is wrong because wrong leakage, wrong action. Plausible because this is 3.4.13 LCO ‘A’ action time and there is unidentified leakage but it does not exceed 1 gpm.
B is wrong because wrong leakage but correct action. Plausible because there is unidentified leakage but it does not exceed 1 gpm.
C is correct because correct leakage and correct action. Per TS 3.4.13B, if prim-sec leakage not within limits of 150 gpd for one SG, then be in mode 3 in the next six hours. SG 2 is 0.19 gpm, or 273.6 gpd, so it exceeds the limit.
D is WRONG because correct leakage but wrong action. Plausible because this is 3.4.13 LCO ‘A’ action time.

Technical References:
REF: TS 3.4.13, amend 156, page 3.4-31, L021.SYS.RC1.pdf, rev 8/18/2017, page rr.
References to be provided to applicants during exam: None.

Learning Objective: SYS.RC1.OB06.008

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	35102
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
000024 (APE 24) Emergency Boration / 1	Tier #	1
Ability to determine and interpret the following as they apply to the Emergency Boration: A2.05 amount of boron to add to achieve required SDM	Group #	2
	K/A #	A2.05
	Rating	3.9
	QREV	6

Question 83

Unit 1 is at full power when an ATWS occurs with the following conditions:

- Three (3) control rods are stuck completely out.
- The US is concurrently in FRS-0.1A and ABN-107, “Emergency Boration”
- Boric Acid Storage Tank X-01 is at 7000 ppm, 80% level.
- RWST concentration is 2500 ppm, 98% level.

Several equipment failures have forced the US to use Attachment 3, Emergency Boration Through the Manual Emergency Borate Valve, 1CS-8439.

What is the MINIMUM amount of boron that is required to achieve an adequate shutdown margin for the GIVEN plant conditions?

A. 2700 gallons

B. 3600 gallons

C. 5400 gallons

D. 15,120 gallons

Answer: C

Explanation:
Using the thumb-rule for the BAT at 7000 ppm for one stuck rod, you need 1800 gallons for one rod with the BAT (which is at 7000 ppm). 3 rods X 1800 gal/rod = 5400 gallons needed.

A is wrong but plausible if you use the wrong correction of 900 gal/rod.
B is wrong but plausible because it is the minimum to borate for loss of DRPI.
C is correct (see discussion above for BAT)
D is wrong because this is the correct amount if you use the RWST to borate because the actual value of 15120 is calculated from the RWST for 3 rods at its given concentration in the stem.
.

Technical References:
ABN-107, rev 9, page 23.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ABN.105.OB06.

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:	55.43(b)2 and 5	

Examination Outline Cross-Reference	Level	RO
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7	Tier #	1
	Group #	2
	K/A #	AA2.11
Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation AA2.11 Loss of compensating voltage	Rating	3.4
	QREV	6

Question 84

Unit 2 is performing a plant startup per IPO-002B, PLANT STARTUP FROM HOT STANDBY. The following steps are complete from this procedure:

- 5.2.21 verified proper overlap between source range and intermediate range instruments
- 5.2.22 SR RX TRIP BLK PERM P-6 is ON, and both SR RX TRIP RESET/BLK switches have been placed in BLOCK

Suddenly, you receive the alarm

- IR CHAN 1 CMPNSATING VOLT FAIL (6D-3.2)

As the US what is the CORRECT Technical Specification ACTION regarding the startup?

A. Reduce THERMAL POWER to < P-6 within 12 hours OR
Raise THERMAL POWER TO > P-10 within 12 hours

B. Reduce THERMAL POWER to < P-6 within 24 hours OR
Raise THERMAL POWER TO > P-10 within 24 hours

C. Suspend all reactivity additions and reduce THERMAL POWER to < P-6 within 2 hours

D. Place the channel in trip within 72 hours or be in Mode 3 in within 78 hours

Answer: B

Explanation:
A is wrong because TS 3.4.1F for loss of one IR channel is answer B. The time is incorrect in this distracter but plausible because other LCO's in this table have 12 hr completion times....
B is correct, TS 3.4.1F states for one IR failure to reduce power less than P-6 or raise it greater than P-10 within 24 hours.
C is wrong because this is the LCO two IR failures (TS 3.4.1G)
D is wrong because this is the action for loss of 1 PR channel above P-6.

Technical References:
TS 3.4.1F, Amendment 150, page 3.3-4 and 3.3-5, IPO-002B, rev 10, page 36, and ABN-702, rev 10, page 3.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.EC1.OB06.

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:	55.43(b)5	

Examination Outline Cross-Reference	Level	SRO
W E08 RCS Overcooling-Pressurized Thermal Shock	Tier #	1
	Group #	2
	K/A #	G2.2.44
G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	Rating	4.4
	QREV	6

Question 85

Given the following conditions on Unit 1:

- RCS Tcold is 240°F and lowering slowly.
- Reactor Coolant System pressure is 30 psig and lowering slowly.
- Containment pressure is 30 psig and lowering slowly.
- The Pressurizer is empty.
- RVLIS indicates below 11 inches above the Core Plate.
- Reactor Coolant System subcooling is 0°F.
- All Engineered Safety Feature Actuations were as expected.

Which of the following describes the challenge to Pressurized Thermal Shock and the actions that are required?

A. ECCS flow has caused RCS cooldown to exceed the entry criteria for FRP-0.2A, Response to Anticipated Pressurized Thermal Shock Condition. Enter FRP-0.2A and reduce RCS cooldown by throttling ECCS flow.

B. RCS pressure and temperature are to the right of the Limit A curve so no challenge exists to Pressurized Thermal Shock. Voids are indicated in the vessel and entry into FRI-0.3A, Response to voids in the Reactor Vessel should be entered to perform Reactor Head venting.

C. RCS cooldown has exceeded the entry criteria for FRP-0.2A Response to Anticipated Pressurized Thermal Shock Condition. Enter FRP-0.2A and place Low Temperature Overpressure Protection in service.

D. RCS cooldown has exceeded the entry criteria for FRP-0.1A, Response to Imminent Pressurized Thermal Shock Condition. Enter FRP-0.1A and verify RCS pressure is less than RHR Pump shutoff head.

Answer: D

Explanation:

A. Incorrect. Plausible because ECCS flow has caused the cooldown but the criteria to reduce ECCS flow does not exist.

B. Incorrect. Plausible because RCS pressure and temperature are to the right of the curve but that doesn't mean a PTS challenge does not exist. FRI-0.3A does not take actions if ECCS is inservice.
C. Incorrect. Plausible because placing LTOP in service would be performed if ECCS was not required.
D. Correct. An ORANGE PTS challenge exists but ECCS flow due to a Large Break LOCA is the cause and re-pressurizing is unlikely. The actions are to ensure the RHR Pumps are preserved.

Technical References:
FRP-0.1A, Revision 9

References to be provided to applicants during exam: None.

Learning Objective: LO21.ERG.FP1.OB03.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	ON 2009 EXAM
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	Level	SRO
064 (SF6 EDG) Emergency Diesel Generator	Tier #	2
	Group #	1
2.1.32 Ability to explain and apply system limits and precautions.	K/A #	G2.1.32
	Rating	4.0
	QREV	6

Question 86

Given the following Unit 1 conditions:

- Mode 1
- Control Room is performing OPT-515A, Diesel Generator Fuel Oil Transfer System, Section 8.3, Fuel Oil Transfer System Automatic Operation Test
- PRA is now YELLOW
- 1DO-0030, DG 1-01 FO DAY TK 1-01 DRN VLV has been opened to begin the test
- No other work activities are in progress

Which of the following completes the statement below?

The Technical Specification limit for day tank volume for an emergency diesel generator is located in ____1____, and is based on _____2_____?

A. 1) TS 3.8.1 surveillance requirements
   2) the requirement to be able to run for 1 hour fully loaded with a slight margin.

B. 1) TS 3.8.1 surveillance requirements
   2) the requirement to be able to run for 1 day fully loaded with a slight margin.

C. 1) TS 3.8.3 surveillance requirements
   2) the requirement to be able to run for 1 hour fully loaded with a slight margin.

D. 1) TS 3.8.3 surveillance requirements
   2) the requirement to be able to run for 1 day fully loaded with a slight margin.

Answer: A

Explanation:
A is correct because this spec is located in SR3.8.1.4 of 3.8.1 and the basis for 1440 gallons in the day tank is the requirement to be able to run for 1 hour fully loaded with a margin of 10% plus 589 gallons (ie a slight margin) that is credited in the 3.8.3 section of the TS.
B is wrong because the 24 hrs or 1 day time is way over the limit of approx. 2-3 hrs but is plausible because of the name of the day tank. This is not the basis for the limit.
C is wrong because the TS aspect is wrong (this is the fuel oil/lube oil section of TS and so it is plausible) and the second part is correct (see A above).
D is wrong because the first and second parts are wrong but plausible (see B and C discussions above).

Technical References:

TS Amendment 156, section SR3.8.1.4, page 3.8-7, TSB, Rev 72, page B 3.8-17, LO21.SYS.ED1.pptx, rev 5/16/2017, slide 12.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.ED1.OB24

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
010 (SF3 PZR PCS) Pressurizer Pressure Control	Tier #	2
	Group #	1
A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PORV failures (CFR: 41.5 / 43.5 / 45.3 / 45.13)	K/A #	A2.03
	Rating	4.2
	QREV	7

Question 87

- A Steam Generator Tube Rupture has occurred on a single SG, resulting in an automatic trip on **Low Pressurizer Pressure**.
- All other Steam Generators are intact and being controlled at 50% NR.
- The crew is performing the actions of EOP-3.0A, Steam Generator Tube Rupture.
- Core Exit Thermocouple temperature has been lowered below the required target temperature.

Subsequently, PORV PCV-456 fails full open.

(1) Which malfunction(s) caused the PORV to open for the GIVEN conditions above?

(2) If neither the PORV nor its associated block valve can be isolated, what next procedural transition is required by the US?

A. (1) A failure of Pressurizer Pressure Transmitter 456 ONLY
(2) Transition to ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED

B. (1) A failure of Pressurizer Pressure Transmitters 456 and 457
(2) Transition to ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED

C. (1) A failure of Pressurizer Pressure Transmitters 456 and 457
(2) Transition to ECA-3.2A, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED

D. (1) A failure of Pressurizer Pressure Transmitter 456 ONLY
(2) Transition to ECA-3.2A, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED

Answer: B
Explanation:

Under normal operating conditions, a single failure of PT-456 to high output will cause PORV PCV-456 to open, until pressure sensed by PT-457 drops below 2185#, at which point PCV-456 will be disarmed and close. However because the SGTR cause a low pressure reactor

trip (1880 psig), actual reactor pressure is below the arming setpoint, and a single failure of PT-456 will not cause the PORV to open. Both PT-456 and PT-457 would have to fail high in order to cause PORV PCV-456 to open under the given plant conditions.

EOP-3.0A Step 8.RNO directs a transition to ECA-3.1A if both the PORV and Block Valve are stuck open. There are no direct transitions to ECA-3.2A in EOP-3.0A; ECA-3.1A may direct a subsequent transition to ECA-3.2A.

A is wrong because a single PT failure cannot cause the PORV to open at this pressure. Procedure is correct.
B is correct.
C is wrong because there is no direct transition to ECA-3.2A from EOP-3.0A. Malfunction is correct.
D is wrong because both the malfunction and procedure are incorrect. See above.

Technical References:
EOP-3.0A, Rev 9, Step 8
Pressurizer Pressure and Level Control Study Guide, 5-5-2011.

References to be provided to applicants during exam: None.

Learning Objective: IDENTIFY the symptoms for the entry conditions of ECA-3.1. (LO21.ERG.C31.OB03)

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	
	55.43(5)	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: If any PRZR PORV opens because of high PRZR pressure, Step 8b should be repeated after pressure decreases to less than the PORV setpoint.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 8	Check PRZR PORVs And Block Valves:	
	a. Power to block valves - AVAILABLE	a. Locally restore power to block valve(s).
	b. PORVs - CLOSED	b. IF PRZR pressure less than PORV open setpoint (2335 psig OR PORV LTOP Setpoint), THEN manually close PORV(s). IF any valve can NOT be closed, THEN manually close its block valve. IF block valve can NOT be closed, THEN go to ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED, Step 1
	c. Block valves - AT LEAST ONE OPEN	c. Manually open one block valve unless it was closed to isolate an open PORV.

Examination Outline Cross-Reference	Level	SRO
022 (SF5 CCS) Containment Cooling	Tier #	2
	Group #	1
2.2.37 Ability to determine operability and/or availability of safety related equipment.	K/A #	G2.2.37
	Rating	4.6
	QREV	6

Question 88

What is the 120ºF limit for temperature inside containment and the associated LCO's that would be entered for operability evaluation if Containment temperature rose to this value after securing all four containment coolers?

A. Containment average air temperature and the LCOs would be 3.6.5 for "Containment Air Temperature" AND TS 3.4.15 for "RCS Leakage Detection"

B. Containment general area temperature and the LCOs would be TR 13.7.36 for "Area Temperature Monitoring" AND TS 3.4.15 for "RCS Leakage Detection"

C. Containment average air temperature and the LCO would be 3.6.5 for "Containment Air Temperature" ONLY

D. Containment general air temperature and the LCO would be TR 13.7.36 for "Area Temperature Monitoring" ONLY

Answer: A
Explanation:
A is correct because per TS 3.6.5 at 120 ºF this part is met and it is based on the average of four containment temp instruments. Also, when four coolers are shutoff it impacts the TS 3.4.15 per SOP-801A.
B is wrong because this value of 120 ºF is incorrect for the temperature given in this choice (ie general area vs average temp). General area temp limit is 129 ºF and its correct LCO is 13.7.36 so the temp is incorrect.
C is wrong because it is missing the leak detection TS which is applicable.
D is wrong because the temp is wrong for the area temp limit and because the other TS is missing.

Technical References:
TS 3.6.5, page 3.6-15, Amendment 156, SOP-801A, page 6, rev 14.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.CL1.OB04.

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:	55.43(b)2	

Examination Outline Cross-Reference	Level SRO
103 (SF5 CNT) Containment	Tier # 2
Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.05 Emergency containment entry	Group # 1 K/A # A2.05 Rating 3.9 QREV 6

Question 89

Unit 1 is operating in Mode 1.

- A confirmed report of sabotage has occurred rendering the Personnel Air Lock INNER door INOPERABLE.
- The Personnel Air Lock OUTER door remains OPERABLE.
- No further sabotage has occurred. The Security threat has ended.
- An Emergency Containment Entry has been approved

(1) Per STA-620, CONTAINMENT ENTRY, which of the following is required to be completed PRIOR to emergency containment entry?

(2) Per Technical Specification Bases 3.6.2, Containment Air Locks, Containment Entry via the Personnel Air Lock outer door _______________ to perform repairs on the INNER door.

A. (1) Containment briefing by Shift Manager
(2) IS NOT permitted

B. (1) RWP issuance
(2) IS permitted

C. (1) Containment briefing by Shift Manager
(2) IS permitted

D. (1) RWP issuance
(2) IS NOT permitted

Answer: C

Explanation:

NOTE:	During emergencies, an RWP need NOT be completed or issued prior to entry. The required paperwork is expected to be completed in a timely manner consistent with related emergency response activities.

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact

(continued)

ACTIONS (continued)

(during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A is wrong because TSB 3.6.2 explicitly makes an allowance to enter an affected air lock via the operable outer door.

B is wrong because STA-620 allows for the RWP to be completed after entry has occurred in a timely manner consistent with related emergency response activities. TSB 3.6.2 does allow for the outer door of the affected air lock to be briefly opened then closed if necessary to access the affected inner door.

C is correct. STA-620 states that "A Containment briefing should be held by the Shift Manager. Each RWO entering into Containment should ensure that all questions on the checklist have been reviewed and that all entry members understand their responsibilities." There is no NOTE stating that this need not occur. Also, TSB 3.6.2 explicitly makes an allowance to enter an affected air lock via the operable outer door.

D is wrong because STA-620 allows for the RWP to be completed after entry has occurred in a timely manner consistent with related emergency response activities.

Technical References:
Technical Specification Bases 3.6.2, Containment Air Locks
STA-620, CONTAINMENT ENTRY

References to be provided to applicants during exam: None.

Learning Objective: APPLY the administrative requirements of the Containment

system including Technical Specifications, TRM and ODCM: 3) Containment Air Locks 3.6.2 (LO21.SYS.CY1.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 55.43(b)(5) / (b)(2)	

Definitions:

Emergency Entry - Entry into Containment to prevent actual or potential personnel injury or damage to equipment and facilities or as requested by the Shift Manager to address plant operational concerns.

General Entry - Includes but are not limited to Operator tours, inspection and system manipulations, RP or Safety surveys, or initial entries to evaluate equipment performance.

4.12 Scheduled Entry - Entry into Containment for the purpose of conducting routine tasks necessary to support unit operations.

4.13 Unplanned Entry - Entry into Containment that has NOT been included in the POD/Outage schedule.

Examination Outline Cross-Reference	Level	SRO
076 (SF4S SW) Service Water	Tier #	2
	Group #	1
Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 Service water header pressure	K/A #	A2.02
	Rating	3.1
	QREV	6

Question 90

Given the following:

- Unit 2 is at 100% power.
- The SSWP TRN A/B HDR PRESS LO alarm is received for the 'B' train pump.
- The reactor operator reports that the discharge pressure for the 'B' train pump has stabilized out at 22 psig as read on 2-PI-4253A, SSWP 2 DISCH PRESS in the main control room.
- Both standby pumps for SSW and CCW auto-started.

Which of the following describes the procedural action to be taken by the Unit Supervisor?

A. Enter the TS LCO for ONLY the SSW Pump 'B' because the supported system(s) are only inoperable because SSW is inoperable per LCO 3.0.6.

B. Enter the TS LCO for ONLY the SSW Pump 'B' and EDG 'B' because the EDG is an exception to LCO 3.0.6.

C. Enter the TS LCO for ONLY the SSW Pump 'B,' the RHR 'B' train, and EDG 'B' because they are specifically listed in the SSW LCOs.

D. Enter the TS LCO for SSW Pump 'B' and the entire ECCS 'B' train because these systems are cooled by this pump.

Answer: B

Explanation:
A is wrong because it is not only the pump that is inoperable in this case, IAW TS 3.7.8 for loss of a SSW train LCO 3.0.6 is not applicable because they are specifically mentioned (EDGs TS 3.8.1, and RHR-Mode 4 TS 3.4.6). However, RHR doesn't apply since the plant is not in Mode 4 currently.
B is correct because per TS 3.7.8 condition B, the pump is inoperable and the EDG is inoperable.
C is wrong because RHR is not applicable since the plant is not in mode 4.
D is wrong because the entire train is not mentioned in TS 3.7.8 therefore this is incorrect
.
Technical References:
TS 3.7.8, amendment 156, page 3.7-21, LCO B.

References to be provided to applicants during exam: None.

Learning Objective: LO21.SYS.SW1.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(b)2	

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
072 Area Radiation Monitoring	Group #	2
	K/A #	2.4.8
2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	Rating	4.5
	QREV	6

Question 91

Unit 1 has just transitioned to Mode 4 in preparation for a refueling outage. RCS temp is 345°F. The RO has just validated increasing containment radiation, temperature, pressure, and humidity. Coincident with the validated containment readings, an electrical fault has tripped the running RHR pump.

As the US you are expected to initially enter which of the following procedures?

A. ABN-108, Shutdown Loss of Coolant

B. ABN-104, Residual Heat Removal System Malfunction

C. ABN-103, Excessive Reactor Coolant Leakage

D. EOP 0.0, Reactor Trip or Safety Injection

Answer: A

Explanation:
A is correct because since you've just entered Mode 4 according to ABN-108 this is the correct procedure. Also, the loops would be assumed to be filled due to not being at reduced inventory yet.
B is wrong because this would be the procedure if in Mode 5 and RCS loops not filled
C is wrong because this is the correct procedure if in Modes 1, 2, 3 and RCS >1000 psig
D is wrong because see A. This is plausible if an applicant thinks they're going to heat up into Mode 3 they may need to enter this initially to allow an entry into EOP-1, Loss of Reactor or Secondary Coolant. EOP are allowed to be used outside of Mode 1, 2, and 3 if a line by line analysis is done while performing the procedure. Not needed in this case because the ABN should have sufficient guidance.

Technical References:
ABN-108, Shutdown Loss of Coolant, Rev 4, p. 2

References to be provided to applicants during exam: None.

Learning Objective: DETERMINE the appropriate procedural section of ABN-104, Residual Heat Removal System Malfunction or ABN-108, Shutdown Loss of Coolant for an RCS or RHR malfunction. (LO21.ABN.104.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	2
10CFR Part 55 Content:	55.43(b)(5)	

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)______.

A. (1) Hydrogen Purge Supply and Exhaust
(2) EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT

B. (1) Hydrogen Purge Supply and Exhaust
(2) EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION

C. (1) Containment Spray
(2) EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT

D. (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:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)5	

Examination Outline Cross-Reference	Level	SRO
	Tier #	2
017 In-Core Temperature Monitor	Group #	2
	K/A #	A2.02
Ability to a) predict the impacts of the following malfunctions or operations on the ITM system; and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations	Rating	4.1
	QREV	6

A2.02 Core Damage

Question 93

Unit 1 has experienced some localized core damage as indicated on the CET maps read off of the plant computer. The US is currently in FRC-0.1A, RESPONDING TO INADEQUATE CORE COOLING, and trying to assess core temperatures now that some flow has been established into the core.

The US should _____1______ and should use _____2____.

A. 1. transition to SACRG-1, Severe Accident Control Room Guideline Initial Response
   2. at least one CET near the center of the core

B. 1. transition to SACRG-1, Severe Accident Control Room Guideline Initial Response
   2. at least one CET from one of the outer two rows of assemblies

C. 1. remain in FRC-0.1A, Responding to Inadequate Core Cooling
   2. at least one CET near the center of the core

D. 1. remain in FRC-0.1A, Responding to Inadequate Core Cooling
   2. at least one CET from one of the outer two rows of assemblies

Answer: C

Explanation:

A is wrong because you don’t transition to SAMGs when you are still cooling the core in the FRC and making progress. The second part is correct.
B is wrong because (see A above) and the second part is also incorrect. The outermost two rows are not to be used for CET temps per the FRC bases document.
C is correct because this is the correct procedure and location and this is a correct CET to use per FRC-0.1A.
D is wrong because the second part is incorrect. The outermost two rows are not to be used for CET temps per the FRC bases document.

Technical References:
FRC-0.1A, rev 9, page 42.

References to be provided to applicants during exam: NONE

Learning Objective:		
Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	
	New	X
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	
	Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)5	

Examination Outline Cross-Reference	Level	RO
Tier 3	Tier #	3
	Group #	
2.1.29 Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	K/A #	G2.1.29
	Rating	4.0
	QREV	6

Question 94

During power operations
- A normally locked open valve in a safety related system is discovered with the lock removed and the valve partially closed.
- As a result, the US needs a component/system verification done and must determine if tampering is involved.

What procedure encompasses BOTH verification activities as well as tampering?

A. OPGD-10, “Equipment Readiness”

B. ODA-410, “System Status Control”

C. ODA-102, “Conduct of Operations”

D. ABN-915, “Security Event”

Answer: B

Explanation:
A is wrong because this procedure does not include tampering in it.
B is correct because this procedure covers both verification and tampering activities.
C is wrong because this procedure does not contain both of these activities in it.
D is wrong because this procedure does not contain both of these activities in it.

Technical References:
ODA-410, rev 16, section 6.3 and section 6.4, pages 12 and 14.

References to be provided to applicants during exam: None.

Learning Objective: LO22.ADM.XA1.OB01.

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:	55.43(b)5	

Examination Outline Cross-Reference	Level	RO
Tier 3	Tier #	3
	Group #	
G2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	K/A #	G2.3.15
	Rating	3.1
	QREV	6

Question 95

Given the following conditions:

Both Units are in MODE 1.
Radiation Monitor X-RE-5895A, North Control Room Air intake fails LOW.
Radiation Monitor X-RE-5895B, North Control Room Air intake is operating normally.
Radiation Monitor X-RE-5896A, South Control Room Air Intake is operating normally.
Radiation Monitor X-RE-5896B, South Control Room Air Intake is operating normally.

Which of the following identifies the Technical Specification requirements placed on the Control Room Heating, Ventilation, and Air Conditioning (HVAC) System?

A. Immediately place both Control Room HVAC Trains in the Emergency Recirculation Mode.

B. Secure the Control Room Makeup Air Supply Fan from the North Air Intake within 7 days.

C. Restore the affected Control Room Emergency Filtration/Pressurization System Train to OPERABLE status within 7 days.

D. Restore the affected Control Room Air Conditioning System Train to OPERABLE status within 30 days.

Answer: B

Explanation:
A is wrong. Plausible because this action would be required if both Trains of Control Room Emergency Filtration System (CREFS) Actuation instrumentation were INOPERABLE but only on one Train per Technical Specification LCO 3.3.7.B....
B is correct. With one Air Intake Radiation Monitor INOPERABLE, place the associated CREFS Train in the Emergency Recirculation Mode or secure the affected Intake Makeup Air Supply Fan within 7 days per Technical Specification LCO 3.3.7.A.
C is wrong. Plausible because it could be thought that the Radiation Monitor failure affected the CREFS per Technical Specification LCO 3.7.10.
D is wrong. Plausible because it could be thought that the Radiation Monitor failure affected the Control Room Air Conditioning System per Technical Specification LCO 3.7.11.

Technical References:

TS 3.3.7, Amendment 156, 3.3-53 (also TS 3.7.10 & 3.7.11)

References to be provided to applicants during exam: None.

Learning Objective: SYS.HV1.OB06.004.

Question Source: (note changes; attach parent)	Bank # Modified Bank # New	X (38984)
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)2 and 4	

Examination Outline Cross-Reference	Level	SRO
Tier 3	Tier #	3
	Group #	
2.2.11 Knowledge of the process for controlling temporary design changes.	K/A #	G2.2.11
	Rating	3.3
	QREV	6

Question 96

In accordance with STA-602, Temporary Modifications, who is responsible for performing walk-downs of active Temporary Modifications?

A. Shift Manager

B. Unit Supervisor

C. Work Control Operations Supervisor

D. System Engineer

Answer: D

Explanation:
A is wrong (see D below) Credible because SM usually approves all of the clearances and does walk-downs prior to return to service.
B is wrong (see D below) Credible because the US must perform a review of all Temp Mod packages on a quarterly basis
C is wrong (see D below). Credible because WCS would have responsibility for par of the process.
D is correct per STA-602, page 10.
Note: the mod to the original question was to remove the time requirement to do the walk-down so that it would be a balanced 1X4 question (as seen above).

Technical References:
STA-602, Rev 18, page 10.

References to be provided to applicants during exam: None.

Learning Objective: LO21.ADM.XA1.OB14.

Question Source:	Bank #	
(note changes; attach parent)	Modified Bank #	X (2007-04, Q97)
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	2
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)3	

Examination Outline Cross-Reference	Level	SRO
Tier 3	Tier #	3
	Group #	
2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations	K/A #	2.4.22
	Rating	4.4
	QREV	6

Question 97

Given the following conditions:

- The crew is responding to a large break LOCA
- A CORE COOLING status tree ORANGE path causes a transition to FRC-0.2, Response to Degraded Core Cooling.
- During performance of FRC-0.2, the CORE COOLING status tree changes from ORANGE to YELLOW.
- An ORANGE path exists on the CONTAINMENT status tree.
- FRZ-0.1, Response to High Containment Pressure, is the procedure referenced by the CONTAINMENT status tree.

Which of the following is the required action?

A. Complete FRC-0.2 after completing FRZ-0.1, since the CORE COOLING status tree had been in an ORANGE path

B. Perform FRC-0.2 and FRZ-0.1 together, since FR procedures of the same priority can be executed together.

C. Complete FRC-0.2 and then go to FRZ-0.1 since CONTAINMENT is a lower priority path than CORE COOLING

D. Go to FRZ-0.1 since an ORANGE path has a higher priority than a YELLOW path.

Answer: C

Explanation:
A is wrong. Must complete the core cooling first, even though its color did change to lower status after you entered it.
B is wrong. You must finish the core cooling FRG first.
C is correct. Per the ODA-407 you must complete the core cooling FRG and then go back to containment since it is a lower priority
D is wrong. Orange on containment does not take priority over orange (when you entered it) for core cooling.

Technical References:
ERG.XD2.LP.pdf, rev 8/24/2017, ODA-407, rev 16, pages 30-32.

References to be provided to applicants during exam: None.

Learning Objective: ERG.XD2.OB15.001

Question Source:	Bank #	X (23165)
(note changes; attach parent)	Modified Bank #	
	New	
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	Level	SRO
Tier 3	Tier #	3
	Group #	
	K/A #	G2.1.1
2.1.1 Knowledge of conduct of operations requirements	Rating	4.2
	QREV	6

Question 98

Which of the following individuals is responsible for approving the use of a trainee during a Reactor Startup?

A. Shift Operations Manager

B. Plant Manager

C. Shift Manager

D. Unit Supervisor

Answer: A

Explanation:
A is correct because As outlined in ODA-102, “Conduct of Operations”, page 31.
B is wrong because (see A above). Plausible if thought that the highest level of management was required, however, for a Reactor Startup the Shift Operations Manager approval is needed.
C is wrong because(see A above). Plausible because the Shift Manager must approve the use of a trainee in any evolution that affects reactivity; however, for a Reactor Startup the Shift Operations Manager approval is required.
D is wrong because (see A above). Plausible because the Unit Supervisor is in charge in the Control Room, however, for a Reactor Startup the Shift Operations Manager approval is required.

Technical References:
ODA-102, rev 27, page 31.

References to be provided to applicants during exam: None.

Learning Objective: ADM.XA1.OB01.046.

Question Source: (note changes; attach parent)	Bank #	X (38766)
	Modified Bank #	
	New	
Question History:	Last NRC Exam	No
Question Cognitive Level:	Memory/Fundamental	3
	Comprehensive/Analysis	
10CFR Part 55 Content:	55.43(b)6	

Examination Outline Cross-Reference	Level	SRO
Tier 3	Tier #	3
	Group #	
Knowledge of the process for conducting special or infrequent tests	K/A #	2.2.7
	Rating	3.6
	QREV	6

Question 99

Per OWI-107, Operations Department Turnover and Briefing Instructions...

Which of the following individuals is expected to be “in charge of High Risk, Heightened Level of Awareness, and Infrequent Evolutions”?

A. Plant Manager

B. Director, Operations

C. Shift Operations Manager

D. Unit Supervisor

Answer: D

Explanation:
A is wrong. Plausible because Plant Manager is the senior manager responsible for the site as prescribed by CPNPP Technical Specifications, however, it is the Unit Supervisor who acts as the SRO in charge.
B is wrong Plausible because the Director of Operations is the department lead of licensed personnel, however, not in charge of these activities.
C is wrong. Plausible because the Shift Operations Manager is the Senior License holder prescribed in CPNPP Technical Specifications, however, it is the Unit Supervisor who is the SRO in charge.
D Correct. As prescribed in OWI-107, Operations Department Turnover and Briefing Instructions

Technical References:
OWI-107 rev 8

References to be provided to applicants during exam: None.

Learning Objective:

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

Question Cognitive Level:	Memory/Fundamental Comprehensive/Analysis	3
10CFR Part 55 Content:	55.43(b)3	

Examination Outline Cross-Reference		Level	SRO
Tier 3		Tier #	3
		Group #	
Knowledge of the operational implications of		K/A #	2.4.20
EOP warnings, cautions, and notes.		Rating	4.3
		QREV	6

Question 100

Complete the following statements regarding ERG Steps, Notes and Cautions.

1) ERG NOTES or CAUTIONS that do not apply under the current plant conditions ____1____ be N/A'd by the US.

2) When the US directs another individual to complete a task in the ERGs and the task is not immediately completed, the US ____2____.

A. 1) may NOT
2) must wait until the task is complete before continuing to the next step

B. 1) may NOT
2) should circle the task and continue to the next step

C. 1) may
2) must wait until the task is complete before continuing to the next step

D. 1) may
2) should circle the task and continue to the next step

Answer: D

Explanation:
A is incorrect. Part 1 is incorrect but plausible because ERG CAUTIONS or NOTES are generally thought of in the same manner as ERG Steps, and ERG Steps cannot be N/A'd unless they contain a conditional statement (e.g. IF/THEN). Part 2 is incorrect but plausible because one might believe that in a procedure as important as the ERGs the US must verify the step has been completed before continuing to the next step.
B is incorrect. Part 1 is incorrect but plausible, see A above. Part 2 is correct. Per ODA-407, Attachment 8.A, ERG Rules of Usage the US may continue to the next after applying appropriate place keeping to the procedure to indicate the step is in progress.
C is incorrect. Part 1 is correct. Per ODA-407, Attachment 8.A, ERG Rules of Usage Cautions or Notes that do not apply to current plant conditions may be N/A'd and are not required to be read aloud. Part 2 is incorrect but plausible, See A above.
D is correct. Part 1 is correct, see C above. Part 2 is correct, see B above

Technical References:
ODA-407, Attachment 8.A, rev 16, pages 36 and 40

References to be provided to applicants during exam: None.

Learning Objective: LO21.ADM.XA3.OB02

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:	55.43(b)5	