ANSWER KEY CPNPP 2019 NRC SRO Exam

1	C	26	B	51	A	76	B
2	C	27	A	52	A	77	C
3	A	28	A	53	B	78	C
4	B	29	A	54	D	79	A
5	B	30	B	55	D	80	C
6	~~C~~ B	31	C	56	A	81	A
7	D	32	B	57	A	82	A
8	C	33	C	58	A	83	C
9	C	34	B	59	B	84	A
10	A	35	C	60	A	85	D
11	B	36	D	~~61~~	~~A~~	86	B
12	B	37	A	62	C	87	D
13	B	38	B	63	A	88	B & D
14	D	39	B	64	C	89	A
15	B	40	D	65	A	90	B
16	B	41	C	66	D	91	B
17	D	42	B	67	D	92	D
18	C	43	B	68	B	93	D
19	A	44	A	69	C	94	D
20	C	45	D	70	C	95	A
21	C	46	B	71	C	96	C
22	B	47	C	72	C	97	D
23	B	48	C	73	A	98	B
24	B	49	B	74	B	99	C
25	B	50	D	75	C	100	C

A – 24
B – 30
C – 28
D – 18

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev: 0		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	003 A4.02	
		Importance Rating	2.9	

K/A Statement: Reactor Coolant Pump: Ability to manually operate and / or monitor in the control room: RCP motor parameters

Question 1

Unit 2 plant conditions:

- 15% power
- Multiple computer alarms on RCP 2-03
- All RCP seal injection flows are 7.5 gpm and stable
- RCP 2-03 parameters are as follows:

SEAL WTR IN TEMP (T0183A)	225°F and stable
MOT UP RDL BRG TEMP (T0453A)	187°F and rising
MOT LOW RDL BRG TEMP (T0455A)	190°F and rising
LOW SEAL WATER BRG TEMP (T0457A)	220°F and stable
MOT STAT WNDG TEMP (T0452A)	304°F and rising

What action is required per ABN-101, Reactor Coolant Pump Trip/Malfunction?

A. ENSURE RCP 2-03 Seal Injection flow at least 13 gpm

B. ENSURE RCP 2-03 Thermal Barrier CCW flow at least 35 gpm

C. TRIP the Reactor, TRIP RCP 2-03, due to high stator winding temperature

D. TRIP the Reactor, TRIP RCP 2-03, due to high seal water inlet temperature

Answer: C

K/A Match:
This question matches the KA by requiring the ability to monitor RCP motor parameters and take appropriate manual actions to address those parameters. In this case, trip the reactor and RCP.

Explanation:

A. Incorrect. Incorrect because seal injection flow is 7.5 gpm and stable with seal water bearing temperatures stable no adjustment of seal injection flow is necessary. Plausible because ABN-101, Section 8.0, Step 2 RNO would require a reactor trip followed by a trip of RCP 2-03, THEN Seal Injection flow must be increased, as necessary.

B. Incorrect. Plausible because Step 3 of ABN-101, Section 8.0, is to verify CCW thermal barrier flow to all RCPs greater than or equal to 35 gpm per pump, however, the Reactor and the RCP should have been tripped on Step 2 and then transition to Section 2.0 of ABN-101 for RCP trip. Step 3 of Section 8.0 of ABN-101 will not be performed and therefore cannot be correct.

C. Correct. With RCP 2-03 Motor Stator Winding Temperature above 300°F a reactor trip followed by a trip of RCP 2-03 is required. ABN-101, Attachment 1, RCP Parameters, lists the upper operating limits for all RCPs, above these limits the RCP must be tripped in accordance with ABN-101, Section 8.0, Step 2 RNO.

D. Incorrect. Plausible because the temperature limit at which an RCP trip is required on High Seal Water Inlet Temperature is 235°F. This is a number often confused with the high temperature limit on the Lower Seal Water Bearing Temperature of 225°F. If a candidate confused these numbers they would believe a reactor trip/RCP 2-03 trip would be required on High Seal Water Inlet Temperature.

Technical Reference(s): ABN-101 Rev 12 PCN-4 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **ANALYZE** the response to an RCP High Temperature or Loss of CCW to any RCP in accordance with ABN-101, Reactor Coolant Pump Trip/Malfunction. (LO21ABN101OB105)

Question Source:
Bank # 2016 NRC Exam Q1
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehensive or Analysis

10CFR Part 55 Content:	**55.41** **55.43**	3

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**	2	
		Group	1	
Level of Difficulty:	2	**K/A**	004A3.04	
		Importance Rating	2.8	

K/A Statement: Chemical and Volume Control System: Ability to monitor automatic operation of the CVCS, including: VCT pressure control.

Question 2

During normal operation, MINIMUM VCT pressure is automatically maintained ABOVE ____(1)____ psig in order to ____(2)____.

A. (1) 10
(2) provide minimum back pressure to the RCP seals

B. (1) 10
(2) maintain adequate NPSH to the CCPs

C. (1) 15
(2) provide minimum back pressure to the RCP seals

D. (1) 15
(2) maintain adequate NPSH to the CCPs

Answer: C

K/A Match:
This question matches the KA by requiring the ability to monitor automatic operation of VCT pressure control and the reason for the minimum pressure.

Explanation:

A. Incorrect: Part 1 is incorrect but plausible because 10 psig is the value at which RCP seal damage will occur during degas operation. Part 2 is correct 15 psig is the minimum pressure to be maintained on the VCT to maintain adequate back pressure to the RCP seals.

B. Incorrect: Part 1 is incorrect but plausible, see A above. Part 2 is incorrect but plausible because design analysis shows that CCP NPSH available from the VCT exceeds requirements if the VCT level is at the outlet nozzle and the tank is at 0 psig pressure.

C. Correct: Part 1 is correct, during normal system operation VCT level is maintained at 15 psig or higher, therefore, providing necessary back pressure to the RCP seals. Part 2 is correct, see A above.

D. Incorrect: Part 1 is correct, see C above. Part 2 is incorrect but plausible, see B above.

Technical Reference(s):	LO21SYSCS1 Study Guide Rev. 4/28/11 LO21SYSCS1 Lesson Plan Rev. 8/18/17 LO21SYSCS1 Power Point Rev. 8/18/17	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

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

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		X
	Comprehensive or Analysis		
10CFR Part 55 Content:	**55.41** 7		
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	004A1.07	
		Importance Rating	2.7	

K/A Statement: Chemical and Volume Control System: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls, including: Maximum specified letdown flow.

Question 3

Unit 2 plant conditions:

- Heatup in progress
- RCS temperature 495°F rising

In accordance with SOP-103B, Chemical and Volume Control System complete the statements below:

When RCS temperature is ≥ 500°F, CVCS Letdown flow indicated on 2-FI-132, LTDN FLO must be maintained below a MAXIMUM flowrate of ____(1)____ gpm.

A. 140

B. 170

C. 195

D. 270

Answer: A

K/A Match: This question matches the KA by requiring the ability to predict/determine a required change in a parameter associated with operating CVCS controls, specifically the maximum allowed Letdown flow for the provided temperature and the maximum allowed charging flow through the Regenerative HX

Explanation:

A. Correct: Correct, per SOP-103B when RCS temperature is ≥ 500°F, Letdown flow is limited to 140 gpm with the 45 gpm orifice and ONE 75 gpm orifice in service.

B. Incorrect: Incorrect but plausible because 170 gpm is the Letdown flow limit when RCS temperature is < 500°F with 1 Mixed Bed Demineralizer in service.

C. Incorrect: Incorrect but plausible because 195 gpm is the Letdown flow limit when RCS temperature is < 500°F with 2 Demineralizers in service.

D. Incorrect: Incorrect but plausible because per SOP-103B, Charging flow through the Regenerative HX is limited to 270 gpm due to indication on 2-FI-121A only going up to 270 gpm.

Technical Reference(s): SOP-103B, Rev. 13, PCN-28
CB-06, FI-121A, CHRG FLO
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

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

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	005 K4.01	
		Importance Rating	3.0	

K/A Statement: Residual Heat Removal System: Knowledge of the RHRS design feature(s) and/or interlock(s) which provide for the following: Overpressure mitigation system

Question 4

Complete the following statements regarding the LTOP System with RHR in the Shutdown Cooling Mode of operation.

1) Two RHR ____(1)____ relief valves are allowed to be substituted for two PRZR PORVs as a means of LTOP protection.

2) With RCS pressure at 458 psig the associated relief valves would be ____(2)____.

A. (1) Pump Suction
(2) closed

B. (1) Pump Suction
(2) open

C. (1) Cold Leg Injection
(2) closed

D. (1) Cold Leg Injection
(2) open

Answer: B

K/A Match: This question matches the KA by requiring knowledge of RHR design features (RHR Suction Relief valve) which provide overpressure mitigation at CPNPP (LTOP).

Explanation:

A. Incorrect. 1st part is correct, the RHR pump suction relief IS allowed to be substituted for a PRZR PORV for LTOP protection on a one for one basis per T.S. LCO 3.4.12.
The 2nd part is incorrect but plausible because the setpoint for the RHR pump suction relief is 450 psig, however, the RHR Cold Leg Injection Relief valves have a setpoint of 600 psig. If the examinee confused the setpoints of the two relief valves this could be chosen as a correct answer.

B. 1st part is correct, see A above. 2nd part is correct, per TS LCO 3.4.12 the RHR Pump Suction Relief valves will have a setpoint of 450 psig, therefore at 453 psig the valves would be open.

C. 1st part is incorrect but plausible because the RHR Cold Leg Injection relief valves are relief valves in the RHR system piping, however, they are on the discharge side of the RHR pumps and cannot be used as a means of LTOP protection. The 2nd part is incorrect but plausible, see A above.

D. 1st part is incorrect but plausible, see C above. 2nd part is correct, see B above.

Technical Reference(s):	LO21SYSRH1 Power Point, Rev. 5/3/08 TS LCO 3.4.12, Amd. 156	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the Residual Heat Removal system including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-260. (LO21.SYS.RH1.OB03)

Question Source:
Bank #
Modified Bank # 2014 NRC Retake Q3 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**	2	
		Group	1	
Level of Difficulty:	2	**K/A**	005 A1.02	
		Importance Rating	3.3	

K/A Statement: Residual Heat Removal System: Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls, including: RHR flow rate

Question 5

With the RCS being cooled by the RHR system, a malfunction causes RCS pressure to rise with a corresponding reduction in RHR pump flow.

As RHR pump flow lowers, at which of the following flowrates will the RHR Pump Miniflow Valves (FCV-610/611) FIRST be open to protect the RHR pump from overheating (assuming they are in AUTO)?

A. 1400 gpm

B. 740 gpm

C. 500 gpm

D. 340 gpm

Answer: B

K/A Match: This question matches the KA by requiring the ability to predict automatic valve action based on system flow in order to prevent exceeding minimum flow limit though the pump.

Explanation:

A. Incorrect: Plausible because approximately 1402 gpm is the value at which the valves will FIRST be closed when flow rises, if the candidate confuses the values at which the valve opens and closes the selection could be chosen

B. Correct: The flow setpoint for the RHR Pump Miniflow valves to open is 746 gpm so at 740 gpm, they will have FIRST opened.

C. Incorrect: Incorrect because the RHR Pump Miniflow valves will be open when flow lowers to 746 gpm. It is plausible because they will be open at 500 gpm and 500 gpm is the minimum design flow required for pump cooling so if the applicant confused the minimum design flow for the Miniflow valve setpoint, they could pick this answer.

D. Incorrect: Incorrect because the RHR Pump Miniflow valves will be open when flow lowers to 746 gpm. It is plausible because they will be open at 340 gpm and RHR pump suction pressure is limited to 400 psig so if the applicant confused the two numbers and was looking for the first value below 400, they could pick this answer.

Technical Reference(s): RHR System Study Guide — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Residual Heat Removal system in accordance with SOP-102 and ABN-104, 108 and 121. (LO21.SYS.RH1.OB05)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	2	**K/A**	006A2.13	
		Importance Rating	3.9	

K/A Statement: Emergency Core Cooling System: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent SIS actuation

Question 6

An inadvertent SI Actuation has occurred on Unit 1.

Which of the following is an action performed in EOS-1.1A, Safety Injection Termination and the reason the action is performed?

A. Secure SI Pumps to prevent motor heating due to running for extended time periods at minimum flow.

B. Restore Instrument Air to containment to allow use of normal PRZR spray valves to control PRZR pressure.

C. Secure CCPs running in Injection Mode to prevent PRZR bubble collapse and risk damaging PRZR safety valves.

D. Secure RCPs to prevent running with inadequate pump seal cooling due to seal water return containment isolation.

Answer: ~~C~~ B

K/A Match: This question matches the KA by requiring the use of EOS-1.1A to mitigate the effects of an inadvertent SI and predict the reason the action is performed due to the impact of the inadvertent SI.

Explanation:

A. Incorrect. Plausible because it could be thought that preventing damage to SI Pumps running at low flows would be the reason to secure the SI pumps, however, the procedure secures the pumps only if they are not required without consideration to damaging the pumps running at shutoff head.

B. ~~Incorrect. Plausible because it could be thought that using Pressurizer Spray Valves could prevent an overpressure condition, however, the Pressurizer would continue to fill and pressurize the RCS until inventory was controlled.~~ Correct. EOS-1.1 step 6 directs establishing instrument air to containment, and step 20 requires the use of normal PRZR spray as necessary to maintain PRZR pressure stable.

C. ~~Correct. The high head Centrifugal Charging Pumps will continue to increase inventory resulting in high pressures up to the PORV setpoint if steps to reduce flow and restore letdown as part of SI Termination are not performed~~ Incorrect. Plausible because it could be thought that all injection needs to be secured, however, EOS-1.1 step 7directs stopping all but one CCP in injection mode, not all CCPs.

D. Incorrect. Plausible because the RCPs would be running with Seal Injection but Seal Return flow would be via the Seal Water Return Relief Valve to the PRT, which could lead to the conclusion that the relief to the PRT would provide inadequate seal cooling.

Technical Reference(s): OPGD 3, Attachment 6, Rev 1/9/19 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Given a procedural step, or sequence of steps from EOS-1.1, Safety Injection Termination, **STATE** the purpose/basis for the step(s).

Question Source: **Bank #** 2014 NRC Exam Q31
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	2	**K/A**	007K1.01	
		Importance Rating	2.9	

K/A Statement: Pressurizer Relief/Quench Tank: Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: Containment System.

Question 7

Complete the statement below regarding the PRT.

The PRT has ____(1)____ rupture disk(s) that will rupture at a MAXIMUM PRT pressure of approximately ____(2)____ psig above containment pressure.

A. (1) One
(2) 106

B. (1) One
(2) 91

C. (1) Two
(2) 106

D. (1) Two
(2) 91

Answer: D

K/A Match: This question matches the KA by requiring the knowledge of the physical connection and interrelationship between the PRT and Containment.

Explanation:

A. Incorrect: Part 1 is incorrect but plausible because there are other components in the plant that have only one rupture disk, for example each LP Turbine has only one rupture disk to protect from overpressure. It is reasonable to believe the PRT would have only one rupture disk. Part 2 is incorrect but plausible because this would be the correct pressure if the question was asking for absolute pressure vice gauge pressure. If the question asked for the value of absolute, 15 psi would be added to the 91 psig to obtain a value of 106 psia.

B. Incorrect: Part 1 is incorrect but plausible, see A above. Part 2 is correct per DBD-ME-0250 the PRT rupture disks will discharge nominally at 91 psig into the reactor containment.

C. Incorrect: Part 1 is correct per DBD-ME-0250 the PRT has two rupture disks to protect the tank against a discharge exceeding a value of 100 psig. Part 2 is incorrect but plausible, see A above.

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

Technical Reference(s): DBD-ME-0250, Rev 60
Main Turbine Study Guide, Rev 5/4/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the RCS including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-0250, Reactor Coolant System.

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

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X (comprehensive due to the extra mental process of converting a memory value of gauge pressure to absolute pressure)

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	008K1.04	
		Importance Rating	3.3	

K/A Statement: Component Cooling Water System (CCWS): Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: RCS, in order to determine source(s) of RCS leakage into the CCWS

Question 8

Given the following conditions:

- Both Units in normal 100% power alignments
- Unit 1 CCW surge tank level is 80% and rising
- 1RE-4510 (CCW168), UNIT 1 COMPONENT COOLING WATER NON-SFGD RADIATION DETECTOR, goes into ALERT
- 1-FI-132, LTDN FLO is 120 gpm and stable
- Unit 1 RCS Tave is 585.4ºF and stable
- Unit 1 RCP seal leakoff is 2.5 gpm per pump

Which of the following Unit 1/Common components is the source of the CCW surge tank level increase?

A. Letdown Heat Exchanger

B. Seal Water Return Heat Exchanger

C. RCP Thermal Barrier Heat Exchanger

D. Spent Fuel Pool Heat Exchanger X-02

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the cause/effect relationship of the CCW and RCS systems in the event of a leak.

Explanation:

A. Incorrect. Plausible because the CCW non-safeguards loop cools CVCS via the Letdown Heat Exchanger which would leak into the CCW system if a leak were to develop in the Letdown Heat Exchanger. However, as letdown flow is normal, no leakage out of the Letdown Heat Exchanger into CCW is indicated.

B. Incorrect. Plausible because the CCW non-safeguards loop cools CVCS via the Seal Water Return Heat Exchanger. The CCW system would leak into CVCS if a leak were to develop in the Seal Water Return Heat Exchanger. However, as RCS temperature is normal, no leakage into CVCS from CCW is indicated as this would appear as a dilution event and RCS temperature would be increasing. Also, given in the stem, CCW Surge Tank level is rising, which eliminates this answer from being correct.

C. Correct. With leakage from a RCP thermal barrier heat exchanger to the CCW system, indications would include a radiation monitor alarm in the non-safeguards CCW loop and rising CCW surge tank level. As leakage would be out of the RCS, automatic makeup would replace the leakage flow with minimal impact on RCS parameters as long as the leak remained small.

D. Incorrect. Plausible because the Unit 1 CCW non-safeguards loop can be aligned to cool Spent Fuel Pool Heat Exchanger X-02, however during normal alignments Unit 1 CCW cools the X-01 Heat Exchanger and Unit 2 cools the X-02 Heat Exchanger.

Technical Reference(s):	LO21SYSCC1 Study Guide, Rev 5/1/11	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the CCW system including interrelations with other systems to include interlocks and control loops as described in DBD-ME-229

Question Source:	**Bank #**	2014 NRC Exam Q52	
	Modified Bank #		(note changes or attach parent)
	New		
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X

10CFR Part 55 Content:	**55.41** **55.43**	5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	4	**K/A**	008K3.01	
		Importance Rating	3.4	

K/A Statement: Component Cooling Water System: Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS.

Question 9

Unit 1 plant conditions:

- Reactor power = 50%
- 1-TV-4646, LTDN HX CCW RET TEMP CTRL VLV is in Automatic
- Airline to 1-TV-4646 severs

Based on the above conditions, complete the following statement:

The Manual/Auto station (1-TK-130, Letdown Heat Exchanger Temperature Controller) located on CB-06 will indicate 1-TV-4646 demanded position being ____(1)____ with actual CCW flow to Letdown HX being ____(2)____.

A. (1) 100%
   (2) at maximum

B. (1) 100%
   (2) at minimum

C. (1) 0%
   (2) at maximum

D. (1) 0%
   (2) at minimum

Answer: C

K/A Match: The question matches the KA by requiring knowledge of how a CCW system malfunction will affect CCW flow to one of the CCW loads (heat exchangers).

Explanation:

The temperature control valve TV-4646 will fail open upon a loss of air so actual flow will increase. The CVCS system will see letdown temperature decrease due to the increased CCW flow and send a signal to cut down on CCW flow to the LD HX in an attempt to return temperature. The demand on the MA station will decrease (this is what the operator will see).

A. 1st part is incorrect because when the valve fails open, cooling will increase causing demand to go down. It is plausible because if it indicated actual valve position, it would be correct. 2nd part is correct. Upon a loss of air, TV-4646 will fail open.

B. 1st part is incorrect but plausible (see A). 2nd part is incorrect because flow will be at maximum. It is plausible because if the TV-4646 failed closed upon losing air, it would be correct.

C. 1st part is correct. As actual cooling flow increases, cooling will increase causing demand for the temperature valve to go low (telling the valve to go closed). 2nd part is correct. Actual CCW flow will go to maximum because the valve fails open.

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

Technical Reference(s): OP51SYSCC1, Rev 1/7/09
CCW Study Guide, Rev 5/1/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the Component Cooling Water system including interrelations with other systems to include interlocks and control loops. (LO21.SYS.CC1.OB03)
EXPLAIN the instrumentation and controls of the Component Cooling Water System and **PREDICT** the system response. (LO21.SYS.CC1.OB04)

Question Source:
Bank # 2014 NRC Retake Exam Q8
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 7

55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	2	
		Group	1	
Level of Difficulty:	2	**K/A**	010K6.04	
		Importance Rating	2.9	

K/A Statement: Pressurizer Pressure Control: Knowledge of the effect that a loss or malfunction of the following will have on the PZR PCS: PRT

Question 10

Since taking the shift, Unit 2 PRT pressure has risen 10 psig. After the change in downstream pressure, the lift setpoint for 2-PCV-456, PRZR PORV is _____ psig.

A. 2335

B. 2345

C. 2460

D. 2470

Answer: A

K/A Match: This question matches the KA by requiring knowledge of how a malfunction of the PRT (increased backpressure) will affect the PORV lift setpoint.

Explanation:

A. Correct: The PORV lift setpoint is based on a signal sent by a PT at a set RCS pressure. Not DP across the disk as in a PRZR safety valve. That being said, backpressure has no effect on PORV lift setpoint.

B. Incorrect: Incorrect because the PORV lift setpoint will remain at 2335 psig. It is plausible because if it were a typical PRZR safety valve that depends on pressure across the disk of the valve, you could assume that a 10 psig rise in downstream pressure would raise the pressure required on the upstream side (RCS press) to cause the valve to lift.

C. Incorrect: Incorrect because this is the PRZR Safety Valve setpoint. It is plausible because it is the Safety Valve lift setpoint.

D. Incorrect: Incorrect because the PORV setpoint will remain the same (2335 psig). It is plausible because this is the PRZR Safety setpoint + 10 psig.

Technical Reference(s):	LO21SYSPP1 Power Point, Rev 4/26/18	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the Pressurizer Pressure Control System including interrelations with other systems to include interlocks and control loops in accordance with the CPNPP FSAR and DBD-ME-250, Reactor Coolant System. (LO21.SYS.PP1.OB03)

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	7	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	007EK2.02	
		Importance Rating	2.6	

K/A Statement: Reactor Trip, Stabilization, Recovery. Knowledge of the interrelations between a reactor trip and the following: Breakers, relays, and disconnects.

Question 11

Unit 1 plant conditions:

- Reactor Trip Breaker testing is in progress on Train "A"
- Reactor Trip Breaker "A" is open
- Reactor Trip Bypass Breaker "A" is closed
- Reactor Trip Breaker “B” is closed
- A transient occurs initiating an AUTOMATIC Reactor Trip signal
- The Reactor does NOT trip from the AUTOMATIC signal

Which of the following describes the condition that has contributed to the Reactor Trip failure?

Reactor Trip...

A. Breaker "B" Undervoltage Trip coil failed to energize.

B. Bypass Breaker "A" Undervoltage Trip coil failed to deenergize.

C. Bypass Breaker "A" Shunt Trip coil failed to energize.

D. Breaker "B" Shunt Trip coil failed to deenergize.

Answer: B

K/A Match: The question matches the KA by requiring knowledge of the breaker actuations that should occur on a reactor trip in order to determine the failed component when it doesn't occur.

Explanation:

A. Incorrect: Plausible because Breaker "B" is equipped with an undervoltage trip coil, however, trip coils are normally energized and deenergize on a trip signal**.**

B. Correct: Given the conditions listed, the Bypass Breaker Undervoltage Trip coil failed to deenergize.

C. Incorrect: Plausible because the Shunt Trip coil is designed to energize and trip open the breaker; the breaker is equipped with a shunt trip coil but it energizes on a Manual Rx trip vice an Auto Rx trip.

D. Incorrect: Plausible because Breaker "B" is equipped with a Shunt Trip coil, however, it energizes to trip.

Technical Reference(s):	LO21SYSES1 Study Guide, Rev 5/4/11	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Reactor Protection and Engineered Safeguard Actuation System in accordance with EOP-0.0A/B, Reactor Trip or Safety Injection and DBD-EE-021, Reactor Protection and NSSS Related Control Systems. (LO21SYSES1OB05)

Question Source:	**Bank #**	2016 NRC Exam Q45	
	Modified Bank #		(note changes or attach parent)
	New		
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	6	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	008AA2.25	
		Importance Rating	2.8	

K/A Statement: Pressurizer Vapor Space Accident: Ability to determine and interpret the following as they apply to the pressurizer vapor space accident: Expected leak rate from an open PORV or code safety.

Question 12

The Unit is at 100% power when a Pressurizer Pressure malfunction results in **BOTH** PORVs opening fully. While the PRZR PORVs are open, the MAXIMUM mass flow rate that could pass through **BOTH** valves is approximately _____ lbm/hr.

A. 210,000

B. 420,000

C. 630,000

D. 840,000

Answer: B

K/A Match: This question matches the KA by requiring the ability to determine the expected leak rate from open PRZR PORVs.

Explanation:

A. Incorrect: This is incorrect because the rated flow through both PORVs is ~ 420,000 lbm/hr. It is plausible because this is the rated flow through one PORV.

B. Correct: The rated flow through each PORV is ~ 210,000 lbm/hr. Therefore, with both PORVs open, you could expect the initial flow rate to be ~ double that or 420,000 lbm/hr.

C. Incorrect: The rated flow through each PORV is ~ 210,000 lbm/hr. Therefore, with both PORVs open, you could expect the initial flow rate to be ~ double that or 420,000 lbm/hr. It is plausible because this would be the approximate flow rate for one PORV and one Safety being open.

D. Incorrect: The rated flow through each PORV is ~ 210,000 lbm/hr. Therefore, with both PORVs open, you could expect the initial flow rate to be ~ double that or 420,000 lbm/hr. It is plausible because this would be the approximate flow rate for two Safety Valves being open

Technical Reference(s): TSB LCO 3.4.10, Rev 76
TSB LCO 3.4.11, Rev 76
PRZR Press & Lvl Control LP, Rev. 4/4/17

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the Pressurizer Pressure Control System including interrelations with other systems to include interlocks and control loops in accordance with the CPNPP FSAR and DBD-ME-250, Reactor Coolant System.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 3
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	4	**K/A**	009G2.4.18	
		Importance Rating	3.3	

K/A Statement: Small Break LOCA: Knowledge of the specific bases for EOPs.

Question 13

Unit 1 plant conditions:

- SBLOCA has occurred
- RCS pressure is 1000 psig and stable
- RCS T_{COLD} is 540°F and lowering
- Containment pressure is 12 psig and rising
- Steam Generator pressures are 1000 psig and lowering

In accordance with EOP-1.0A, Loss of Reactor or Secondary Coolant, which of the following is the reason for stopping the RCPs under these plant conditions?

To prevent...

A. core damage resulting from pump operation under forced two-phase flow conditions

B. core damage resulting from phase separation upon subsequent loss of forced flow

C. pump damage resulting from a loss of Component Cooling Water to the motor

D. pump damage resulting from the adverse Containment environment

Answer: B

K/A Match: This question matches the KA by requiring knowledge of specific bases for EOP steps for a SBLOCA.

Explanation:

A. Incorrect: Incorrect because phase separation during forced circulation is not a concern. It is plausible because phase separation is a concern if/when the pump subsequently secured.

B. Correct: Given the conditions listed, this is the reason the Reactor Coolant Pumps are stopped.

C. Incorrect: Incorrect because the reason for securing the RCPs is for subsequent phase separation if/when RCPs are secured. It is plausible if thought that CCW flow would be lost for the conditions listed.

D. Incorrect: Incorrect because the reason for securing the RCPs is for subsequent phase separation if/when RCPs are secured. It is plausible because adverse containment conditions do exist.

Technical Reference(s): EOP-1.0A, Attachment 4 Bases, Rev 9 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DISCUSS** the baes for all steps, Cautions and Notes included in EOP-1.0 (LO21.ERG.E1A.OB105)

Question Source:
Bank # 2011 NRC Exam Q35
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	011EK2.02	
		Importance Rating	2.6*	

K/A Statement: Large Break LOCA: Knowledge of the interrelations between the Large Break LOCA and the following: Pumps.

Question 14

Unit 2 plant conditions:

- A LBLOCA occurs
- Aligning for Cold Leg Recirculation per EOS-1.3B, Transfer to Cold Leg Recirculation
- Crew just closed 1/2-8806, RWST TO SIP SUCT VLV
- LOOP occurs

Which of the following pumps are running with no suction?

A. CTPs

B. RHRPs

C. SIPs

D. CCPs

Answer: D

K/A Match: This question matches the KA by requiring Knowledge of the relationship between a LBLOCA and ECCS pumps.

Explanation:

A. Incorrect. Plausible to believe that CTPs restarted from the BO Sequencer at a specific point in EOS-1.3B where the suction from the RWST was being swapped to the ECCS Sump.

B. Incorrect. Plausible to believe that RHRPs restarted from the BO Sequencer at a specific point in EOS-1.3B where the suction from the RWST was being swapped to the ECCS Sump.

C. Incorrect. Plausible to believe that SIPs restarted from the BO Sequencer and if they restarted at this specific point in procedure flowpath of EOS-1.3B they would be running with no suction.

D. Correct. CCPs will be started by the BO Sequencer and based on specific identified point in procedure flowpath of EOS-1.3B when they restart they will be running with no suction.

Technical Reference(s):	EOS-1.3B, Rev. 9 PCN-LO21.SYS.ES3, Rev. 6/10/11	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the SSPS and **PREDICT** the system response in accordance with DBD-EE-021, Reactor Protection and NSSS related control systems and Westinghouse drawings.

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	8	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	1	
		Group	1	
Level of Difficulty:	2	**K/A**	015AK3.07	
		Importance Rating	4.1	

K/A Statement: Reactor Coolant Pump Malfunctions: Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump malfunctions (Loss of RC Flow): Ensuring that S/G levels are controlled properly for natural circulation enhancement.

Question 15

Unit 1 crew has initiated an RCS Cooldown to Cold Shutdown in accordance with EOS-0.2A, Natural Circulation Cooldown.

Which of the following states the TARGET NR SG level that is required to be maintained in accordance with EOS-0.2A and the reason for maintaining that level?

A. 67%, limit decay heat removal to prevent exceeding cooldown rate

B. 67%, establish/maintain stable heat sink during cooldown

C. 60%, limit decay heat removal to prevent exceeding cooldown rate

D. 60%, establish/maintain stable heat sink during cooldown

Answer: B

K/A Match: This question matches the KA by requiring knowledge of the reason for maintaining SG levels at a certain band during natural circulation which would only occur after a loss of all RCPs.

Explanation:

A. Incorrect: Part 1 is correct, normal levels are maintained during natural circulation cooldown with a Target level of 67%. Part 2 is incorrect but plausible because the maximum cooldown rate allowed by EOS-0.2A is 50°F/hour. This is a low cooldown rate for the ERGs and reasonable to believe that the required SG level would be designed to prevent from exceeding this rate.

B. Correct: Part 1 is correct, see A above. Part 2 is correct normal levels are maintained in order to ensure an adequate heat sink (this is a target level of 67% for Unit 1).

C. Incorrect: Part 1 is incorrect but plausible because 60% would be the target level range (59% to 69%) for Unit 2. Part 2 is incorrect but plausible, see A above.

D. Incorrect: Part 1 is incorrect but plausible, see C above. Part 2 is correct, see B above.

Technical Reference(s): EOS-0.2A, Rev 9 PCN-
EOS-0.2B, Rev 9 PCN-
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DISCUSS** the ERG background for performing Natural Circulation Cooldown per EOS-0.2, Natural Circulation Cooldown. (LO21.ERG.EO2.OB01)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	022AA1.08	
		Importance Rating	3.4	

K/A Statement: Loss of Reactor Coolant Makeup: Ability to operate and/or monitor the following as they apply to Loss of Reactor Coolant Makeup. VCT level.

Question 16

Unit 1 plant conditions:

- Reactor power = 100%
- ALM-0061A, 1-ALB-6A, Window 3.5 – VCT LVL LO alarms
- 1-LI-185, VCT LVL indicates 15% and lowering slowly
- 1-LI-112A, VCT LVL indicates 48% and stable
- I & C Maintenance technicians have determined that VCT Level Transmitter 1-LT-112A is failed

Based on the current state of the VCT level control system, complete the statement below.

Auto RCS Makeup capability ____(1)____ functional. If 1-LI-185 continues to lower to less than 2% VCT level, the Charging Pump Suction Valves ____(2)____ automatically realign to the RWST.

A. (1) is NOT
(2) will

B. (1) is NOT
(2) will NOT

C. (1) is
(2) will

D. (1) is
(2) will NOT

Answer: B

K/A Match: This question matches the KA by requiring the ability to monitor the RCS Makeup system in the event of a low VCT level and determine that a loss of auto RCS Makeup has occurred due to a failed channel as well as a loss of the capability of the charging pumps to swap to the RWST.

Explanation:

A. Incorrect: Part 1 is correct. VCT Auto Makeup is initiated off of LT-112A only. If the level transmitter is failed to 48% then auto RCS makeup capability has been lost. LT-112A will initiate an auto RCS makeup when VCT level drops below 46% and it will secure the makeup when level rises above 56%. Part 2 is incorrect but plausible if the auto realignment to the RWST is thought to be driven from only one transmitter. In order for the charging pump suction valves to realign to the RWST both transmitters LT-112A and LT-185 must decrease to below 2% VCT level. In this case LT-112A is stuck at 48% therefore, the charging pump suction valves will not realign when actual VCT level decreases to below 2% level.

B. Correct: Part 1 is correct, see A above. Part 2 is correct, in order for the charging pump suction valves to realign to the RWST both transmitters LT-112A and LT-185 must decrease to below 2% VCT level. In this case LT-112A is stuck at 48% therefore, the charging pump suction valves will not realign when actual VCT level decreases to below 2% level.

C. Incorrect: Part 1 is incorrect but plausible because it is a common misconception that a VCT auto makeup will occur when either level transmitter, LT-185 or LT-112A, falls below a value of 46% level. It will only occur when LT-112A falls below 46% and with that transmitter stuck at 48%, auto makeup functionality is lost. Part 2 is incorrect but plausible, see A above.

D. Incorrect: Part 1 is incorrect but plausible, see C above. Part 2 is correct, see B above.

Technical Reference(s): ALM-0061A, Rev 7 PCN-16
CVCS Study Guide, Rev 4/28/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Chemical and Volume Control system in accordance with referenced procedures. (LO21SYSCS1OB04).

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 7

55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	025G2.1.20	
		Importance Rating	4.6	

K/A Statement: Loss of Residual Heat Removal System: Ability to interpret and execute procedure steps.

Question 17

On Unit 1 the US has entered ECA-1.1A, Loss of Emergency Coolant Recirculation due to a loss of BOTH trains of RHR Cold Leg Recirculation

- The crew has reached Steps 9 and 9.a
  - Step 9 "Check If Containment Fan Coolers Should Be Started"
  - Step 9. a "Verify containment pressure – HAS REMAINED LESS THAN 5 PSIG"

Complete the following statements regarding execution of ECA-1.1A, Steps 9 and 9.a.

1) Restoration of Containment cooling via the Containment Fan Coolers is desired ____(1)____.

2) The reason containment pressure is verified to have remained less than 5 psig is because ____(2)____.

A. (1) because they previously tripped on a CVI signal
(2) if the LOCA occurred outside Containment this step should NOT be performed

B. (1) to allow the CS Pumps to be secured in future steps
(2) if the LOCA occurred outside Containment this step should NOT be performed

C. (1) because they previously tripped on a CVI signal
(2) the Containment Fan Coolers and their associated Ventilation Chilled Water cooling piping are not qualified for post accident operation

D. (1) to allow the CS Pumps to be secured in future steps
(2) the Containment Fan Coolers and their associated Ventilation Chilled Water cooling piping are not qualified for post accident operation

Answer: D

K/A Match: This question matches the KA by requiring the ability to interpret and execute procedure steps for a loss of residual heat removal.

Explanation:

A. Incorrect. Part 1 is incorrect but plausible because the Containment Fan Coolers did previously trip, however, they tripped on the Safety Injection signal not the Containment Vent Isolation Signal. Part 2 is incorrect but plausible because ECA-1.1A specifically states this step is not pertinent for recovering from only a LOCA outside containment event since containment conditions would be normal. However, it also states the step should still be performed.

B. Incorrect. Part 1 is correct per ECA-1.1A bases document the intent of Step 9 is to provide containment heat removal capability using containment fan coolers, therefore, containment spray pumps can be stopped in future steps. Part 2 is incorrect but plausible, see A above.

C. Incorrect. Part 1 is incorrect but plausible, see A above. Part 2 is correct per ECA-1.1A bases document the containment fan coolers and its associated vent chill water cooling are not qualified for post accident operation. High temperature conditions inside containment can cause flashing/water hammer to occur and chilled water to containment should not be aligned under those conditions, consequently, the containment fan coolers should not be started under these conditions.

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

Technical Reference(s): ECA-1.1A and bases, Rev 9 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Given a procedural step, or sequence of steps from ECA-1.1, **STATE** the purpose/basis for the step(s)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	026AK3.03	
		Importance Rating	4.0	

K/A Statement: Loss of Component Cooling Water: Knowledge of the reasons for the following responses as they apply to Loss of Component Cooling Water: Guidance actions contained in EOP for Loss of CCW.

Question 18

Unit 1 plant conditions:

- Reactor power = 100%
- Excess Letdown is in service
- A Reactor Trip occurred
- Safety Injection actuated and EOP-0.0A, Reactor Trip or Safety Injection, Attachment 2, Safety Injection Actuation Alignment, in progress

Based on plant conditions, complete the following statements:

1) XS LTDN ISOL VLVs, 1/1-8153 and 1/1-8154 ____(1)____ automatically close on an SI signal.

2) In accordance with EOP-0.0A, Attachment 2, these valves are verified closed ____(2)____.

A. (1) do
(2) because CCW to the XS LTDN HX isolates on a Phase A Signal

B. (1) do
(2) to prevent overpressurizing the VCT

C. (1) do NOT
(2) because CCW to the XS LTDN HX isolates on a Phase A Signal

D. (1) do NOT
(2) to prevent overpressurizing the VCT

Answer: C

K/A Match: This question matches the KA because it requires knowledge of the reason for actions in the EOP due to a loss of CCW (to the XS LTDN HX in this case).

Explanation:

A. Incorrect: 1st part is incorrect because the XS LTDN ISOL VLVs do not isolate automatically on an SI signal. It is plausible because the normal letdown isolation valves do isolate on an SI actuation. 2nd part is correct. CCW supply to the XS LTDN HX isolates on a Phase A Signal so there will be no cooling to the XS LTDN HX.

B. Incorrect: 1st part is incorrect but plausible (see A). 2nd part is incorrect because the reason as stated in the EOP bases is that CCW is lost to the XS LTDN HX. It is plausible because XS LTDN is normally aligned to the VCT and after an SI has occurred the VCT is isolated, therefore if XS LTDN was aligned to the VCT during SI conditions it would over pressurize it.

C. Correct: 1st part is correct. The XS LTDN isolation valves do not automatically isolate on an SI signal. 2nd part is correct (see A).

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

Technical Reference(s): EOP-0.0A, Rev 9
E1-0051, Sheet 40, Rev CP-2
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** the automatic action verifications contained in EOP-0.0, Reactor Trip or Safety Injection. (LO21ERGEE0AOB05)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 5,10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	4	**K/A**	029EA1.09	
		Importance Rating	4.0	

K/A Statement: Anticipated Transient Without Scram: Ability to operate and/or monitor the following as they apply to ATWS: Manual Rod Control.

Question 19

Unit 1 plant conditions:

- Reactor Power = 100%
- Loss of BOTH MFW Pumps occurs

Time = 08:00:00

- A Main Turbine runback to 700 MWe begins
- Reactor fails to trip automatically or manually
- Main Turbine fails to trip automatically or manually
- FRS-0.1A, Response to Nuclear Power Generation / ATWT has been entered
- Control Rods are in AUTOMATIC

Based on the above plant conditions, complete the following statements.

(1) If Control Rods were placed in MANUAL and driven IN, the rods would move at a MAXIMUM speed of ____(1)____ steps per minute.

(2) At Time 08:00:30, per FRS-0.1A and based on the predicted error signal from the trip of BOTH MFW pumps, the RO will initially ____(2)____ to achieve the HIGHEST rod speed.

A. (1) 48
(2) maintain Control Rods in AUTOMATIC and allow them to insert

B. (1) 72
(2) maintain Control Rods in AUTOMATIC and allow them to insert

C. (1) 48
(2) place Control Rods in MANUAL and insert

D. (1) 72
(2) place Control Rods in MANUAL and insert

Answer: A

K/A Match: This question matches the KA by requiring the ability to operate control rods during an ATWS.

Explanation:

A. Correct: 1st part is correct. When the Control Rod Bank Select switch is placed in Manual the control banks will step at a rod speed of 48 steps per minute with normal control bank overlap. 2nd part is correct. IAW FRS-0.1A, the RO is directed to Ensure control rods are inserting greater than or equal to 48 steps per minute, therefore, in this case the rods must be left in AUTOMATIC because the Power Mismatch Error Channel and the Average Temperature Error Channel will be generating an error signal requesting rods to drive IN at 72 steps per minute. The two error channels are summed together to determine the total error.

B. Incorrect: 1st part is incorrect but plausible because 72 steps per minute is the maximum rod speed when rods are in automatic control. 48 steps per minute is the maximum rod speed when rods are in manual. It is reasonable to believe the two numbers could be confused and 72 spm could be selected over 48 spm. 2nd part is incorrect but plausible, see A above.

C. Incorrect: 1st part is correct, see A above. 2nd part is incorrect but plausible because the Average Temperature Error Channel (which is what the RO can see on the Control Boards) will be generating an error signal that corresponds to a value of control rod insertion of less than 48 steps per minute. If the candidate believes the automatic control rod circuitry is processing a signal for rods to drive at less than 48 steps per minute, he/she must take manual control of the rods and drive rods to achieve 48 steps per minute rod speed. However, when the Power Mismatch Error Channel (not visible on the control boards) is added to the Average Temperature Error Channel, the Total Error will correspond to a value of control rod insertion of 72 steps per minute (the maximum) with control rods in automatic.

D. Incorrect: 1st part is incorrect but plausible, see B above. 2nd part is incorrect but plausible, see C above.

Technical Reference(s): FRS-0.1A, Rev 9
LO21SYSCR1 Study Guide, Rev 5/2/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Rod Control System (LO21YSCR1OB05)

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

Question History: **Last NRC Exam**

Question Cognitive Level:	**Memory or Fundamental Knowledge**	
	Comprehensive or Analysis	X

10CFR Part 55 Content:	**55.41**	7
	55.43	

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev: 0		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	038EK1.01	
		Importance Rating	3.1	

K/A Statement: Steam Generator Tube Rupture: Knowledge of the operational implications of the following as they apply to a SGTR: Use of Steam tables.

Question 20

Unit 1 plant conditions:

- A SGTR occurred on SG 1-04
- The reactor has been tripped
- EOP-3.0A, Steam Generator Tube Rupture is in progress
- The crew is evaluating the step to determine if RCS Subcooling requires transfer to ECA-3.1A, SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired
- RCS pressure is 1635 psig
- Containment pressure is 1 psig and stable

Based on plant conditions, which of the following is the MAXIMUM RCS temperature at which the crew would <u>remain</u> in EOP-3.0A?

A. 683ºF

B. 653ºF

C. 563ºF

D. 533ºF

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the Steam Tables in order to apply them during an SGTR event.

Explanation:

The step in EOP-3.0A asks if RCS Subcooling is greater than 45ºF (75ºF) to determine of a transfer to ECA-3.1A is required. With RCS pressure = 1635 psig (1650 psia), the corresponding Tsat = 609 ºF. A maximum RCS temperature of 564 ºF would ensure a minimum of 45ºF subcooling. Added 1ºF to each answer to get off of the exact number.

A. Incorrect. ECA-3.1 transition required. Plausible because 1) it uses a 75 ºF subcooling margin which would be correct if adverse containment conditions existed and 2) it adds the 75 ºF on to Tsat instead of subtracting it which is a common mistake when calculating

subcooling margin.

B. Incorrect. ECA-3.1 transition required. Plausible because it adds the 45 °F on to Tsat instead of subtracting it which is a common mistake when calculating subcooling margin.

C. Correct. This is the correct value in that it's the maximum RCS temperature stated that has at least 45 °F of subcooling.

D. Incorrect. This temperature would require the crew to remain in EOP-3.0A but is not the not the Maximum value stated. Plausible because it uses a 75 °F subcooling margin which would be correct if adverse containment conditions existed.

Technical Reference(s):	EOP-3.0A, Rev 9 Steam tables	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: Steam Tables

Learning Objective:

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	10	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	APE040 AK2.02	
		Importance Rating	2.6*	

K/A Statement: Steam Line Rupture: Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors.

Question 21

Given the following Unit 1 conditions:

- Reactor trip and SI has occurred
- MSLI was verified incomplete in EOP-0.0A, Reactor Trip or Safety Injection [1-HV-2333A, MSIV 1 indicates Open]
- Steam Flow is still indicated on the following:
  - 1-FI-512A, SG 1 STM FLO
  - 1-FI-513A, SG 1 STM FLO
  - 1-FI-532A, SG 3 STM FLO
  - 1-FI-533A, SG 3 STM FLO
- Transition has been made to EOP-2.0A, Faulted Steam Generator Isolation

Subsequently:

- Local operator action closes 1-HV-2333A
- Steam Flow is still indicated on the following:
  - 1-FI-532A, SG 3 STM FLO
  - 1-FI-533A, SG 3 STM FLO
- Steam Generator (SG) pressures are:
  - SG 1-01 400 psig and stable
  - SG 1-02 640 psig and stable
  - SG 1-03 200 psig and lowering
  - SG 1-04 640 psig and stable

Based on plant indications, SG 1-01 is faulted ____(1)____ of the MSIV and SG 1-03 is faulted ____(2)____ of the MSIV.

A. (1) upstream
   (2) upstream

B. (1) upstream
   (2) downstream

C. (1) downstream
   (2) upstream

D. (1) downstream

(2) downstream

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the interrelations between a steam line break and the detectors (pressure/flow) used to determine its location.

Explanation:

A. Incorrect: 1st part incorrect but plausible if incorrectly diagnoses the indications that pressure has stabilized and flow has ceased for SG 1-01 once the MSIV was closed. 2nd part is correct as described in "C" below.

B. Incorrect: 1st part is correct see "A" above. 2nd part is incorrect but plausible if incorrectly diagnoses the indications that the cooldown is continuing to lower SG 1-03 pressure and does not properly regard the steam flow indication.

C. Correct. Based on the SG pressures and indicated steam flows, SG1-01 is faulted downstream of the MSIV which has been isolated once the MSIV was closed. SG 1-03 is faulted upstream of the MSIV as the pressure continues to lower with indicated steam flow.

D. Incorrect. 1st part is correct see "C" above. 2nd part is incorrect but plausible, see "B" above.

Technical Reference(s): LO21ERGE2A Lesson Plan, Rev 10/29/18 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Upon completion of this lesson, the student should be able to discuss plant response, operator actions, and the reasons for the actions contained in EOP-2.0, Faulted Steam Generator Isolation.

Question Source:
Bank # 2015 NRC Exam Q49
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 5, 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev: 4		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	055EK1.01	
		Importance Rating	3.3	

K/A Statement: Station Blackout: Knowledge of the operational implications of the following as they apply to the Station Blackout: Effect of battery discharge rates on capacity.

Question 22

Unit 1 plant conditions:

- A Loss of Offsite and Onsite Power occurred
- The crew entered ECA-0.0A, Loss of All AC Power
- ECA-0.0A, Attachment 2.A, Initial DC Load Shed, has been completed
- ELAP has NOT been declared
- Battery parameters are as follows:
  - Battery BT1ED1 Voltage: 112 VDC
    Current: 75 amps discharging
  - Battery BT1ED2 Voltage: 109 VDC
    Current: 80 amps discharging

Based on current battery discharge rates and capacities, which of the following describes the required action in accordance with ECA-0.0A?

Additional DC load shedding is required to maintain _____.

A. Battery BT1ED1 voltage greater than 110 VDC

B. sufficient voltage to flash the Diesel Generator field

C. all Post Accident Monitoring indications available in the Control Room

D. voltage for subsequent closure of Safeguard Bus supply breakers 1EA1-1 or 1EA1-2

Answer: B

K/A Match: This question matches the KA by requiring knowledge of why (operational implication) load shedding is performed based on battery discharge rates/capacity.

Explanation:

A. Incorrect. Incorrect but plausible because with battery 1ED2 already less than 110 volts it is reasonable to believe additional load shedding would be performed to maintain battery 1ED1 above 110 volts.

B. Correct. Battery voltage needs only to be greater than 105 VDC for required equipment operation. Additional load shedding is performed on the DC Bus that is less than 110 volts. This will conserve DC energy to assist in recovery actions, such as closing the DG output breaker and flashing the Diesel Generator field to restore AC power.

C. Incorrect. Incorrect but plausible because in ECA-0.0A, Attachment 2 (Initial DC Load Shed, not Additional DC Load Shed in the question) step 2, two channels of control room indication are secured to ensure that sufficient battery energy is maintained to facilitate restoring AC power. Vital Control Room instrumentation is powered via the instrumentation buses which receive power from their respective instrumentation inverters. The inverters can operate with DC input as low as 100 VDC, which is below the 105 VDC requirement of ECA-0.0A. In a normal lineup, the inverters will automatically shift to unfiltered transformer power. With a loss of all AC, the inverters will not shift and will continue to supply power to the instrument buses until DC input reaches ~100 VDC.

D. Incorrect. Incorrect but plausible because additional load shedding is performed on the DC Bus that is less than 110 volts (Train B), thereby conserving DC energy to assist in subsequent closures of the associated DG output breaker and/or the Safeguards Bus Supply Breakers for that train. It is plausible that an examinee could believe additional load shedding is performed for both trains whenever one train's Safeguards battery DC voltage falls below 110 volts; which would make subsequent closures Train A Normal and Alternate Safeguard Bus Supply breakers (1EA1-1 and 1EA1-2) correct. However, additional load shedding is only performed on the train with less than 110 volts when the voltage falls below 110 volts making this choice incorrect.

Technical Reference(s): ECA-0.0A, Rev 9 PCN-2 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Given a procedural step, or sequence of steps from ECA-0.0A, **STATE** the purpose/basis for the step(s).

Question Source:
Bank #
Modified Bank # 2015 NRC Exam Q51 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	056EA2.02	
		Importance Rating	3.5	

K/A Statement: Loss of Offsite Power: Ability to determine and interpret the following as they apply to Loss of offsite power: Load sequencer status lights.

Question 23

Unit 2 plant conditions:

- A loss of offsite power occurs
- The reactor trips
- Both Emergency Diesel Generators start and tie to their respective busses
- The Blackout Sequencer actuates
- The Sequencer Operator Lockout Indicating Light (OPL) is illuminated (Red)

Based on the above plant conditions, complete the following statements:

1) When the OPL Light is illuminated, ALL of the Blackout Sequencer Operator Lockouts have ____(1)____.

2) Two minutes after the sequencer started, the Operator Lockouts will be ____(2)____.

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: B

K/A Match: This question matches the KA by requiring the ability to interpret the load sequencing status light to determine the status of related sequencer lockouts.

Explanation:

A. Incorrect: 1st part is correct because when the light is lit, all of the automatic lockouts have energized. 2nd part is incorrect because after 110 seconds, the operator lockouts will clear (de-energize). It is plausible because if it were > 10 seconds earlier, it would be correct.

B. Correct. 1st part is correct (see A). 2nd part is correct. The BOS Operator Lockouts de-energize 110 seconds after the sequencer was started so they will be de-energized.

C. Incorrect: 1st part is incorrect because if the light is illuminated, it means that all operator lockouts have energized. It is plausible because it is reasonable to believe that this light would indicate that the Operator Lockouts have cleared when it is illuminated. 2nd part is incorrect but plausible (see A).

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

Technical Reference(s): SI and BO Sequencer Study Guide, Rev 6/10/11 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Safety Injection and Blackout Sequencers and **PREDICT** the system response in accordance with CP-0575-001, Solid State Safeguards Sequencers and associated Automation Industries drawings. (LO21SYSE3OB06)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	057AA1.05	
		Importance Rating	3.2	

K/A Statement: Loss of Vital AC Instrument Bus: Ability to operate and/or monitor the following as they apply to Loss of Vital AC Instrument bus: Backup instrument indications.

Question 24

Unit 1 plant conditions:

- A loss of 118 VAC Protection Bus 1PC1 occurs
- ABN-603, Loss of Protection or Instrument Bus is entered
- 1PC1 is unable to be recovered

Based on conditions above, complete the statements below.

1) With no Operator Action, the loss of 1PC1 will result in Charging flow ____(1)____.

2) In order to restore Pressurizer level to automatic control, the backup instrument must be selected for control by placing 1/1-LS-459D, PRZR LVL CTRL CHAN SELECT to the ____(2)____ position.

A. (1) rising
(2) 459/460

B. (1) rising
(2) 461/460

C. (1) lowering
(2) 459/460

D. (1) lowering
(2) 461/460

Answer: B

K/A Match: This question matches the KA by requiring the ability to monitor and operate backup instrumentation powered from the Vital AC Instrument Bus

Explanation:

A. Incorrect: Part 1 is correct, if 1PC1 is lost then the controlling channel, level channel 459, would fail to 0% level. With no operator action the Pressurizer Level Control system would then attempt to increase pressurizer level by increasing charging flow. Part 2 is incorrect but plausible because not all instrumentation arrangements have the lowest numbered channel powered by PC1, for example, Containment pressure utilizes 4 channels with Channel 1 (PC1) as channel 937. Channels 2, 3, and 4 are 936, 935, and 934 (PC2, PC3, and PC4) respectively. Therefore, in the case of containment pressure the highest channel is powered from PC1 which makes it reasonable to believe in the case of pressurizer level control that the original controlling channel lost could have been channel 461 (the highest numbered channel) and now the level channel select switch must be placed in the 459/460 position to eliminate channel 461 from the control circuit.

B. Correct: Part 1 is correct, see A above. Part 2 is correct, with a loss of PC1 that is unrecoverable the pressurizer level channel that has failed is channel 459. Therefore, this channel must be removed from the pressurizer level control circuit. Selecting the 461/460 position on 1/1-LS-459D will remove the failed channel (459) from the circuit and allow restoration of normal pressurizer level control.

C. Incorrect: Part 1 is incorrect but plausible because if this were a loss of 1PC2 with no operator action, charging flow would lower. If this were a loss of PC2 the controlling channel (459) would remain energized at the correct value and indicate actual pressurizer level. Letdown would isolate due to a failure of level channel 460 to 0%. Due to the Isolation of Letdown actual pressurizer level would start to rise and the pressurizer level control system would lower charging flow to attempt to lower pressurizer level. Part 2 is incorrect but plausible, see A above.

D. Incorrect: Part 1 is incorrect but plausible, see C above. Part 2 is correct, see B above.

Technical Reference(s): ABN-603, Rev 8 PCN-7
ABN-706, Rev 8 PCN-
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **ANALYZE** the response to a Loss of Instrument Bus in accordance with ABN-603, Loss of Protection or Instrument Bus (LO21ABN603OB02)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**

	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	7	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	058AK3.02	
		Importance Rating	4.0	

K/A Statement: Loss of DC Power: Knowledge of the reasons for the following responses as they apply to Loss of DC power: Actions contained in the EOP for loss of DC power.

Question 25

Unit 1 plant conditions:

- Loss of Offsite and Onsite Power occurred
- Actions of ECA-0.0A, Loss of All AC Power, are being performed
- The crew is executing Step 13 – "Check SG Status"
- Personnel are dispatched locally to the MSIVs

Complete the statement below regarding the reasons for the actions of ECA-0.0A, Step 13.

Personnel are locally dispatched in order to ____(1)____ to ensure the MSIVs remain closed ____(2)____.

A. (1) align alternate power to each MSIV hydraulic oil pump SOV
(2) during performance of DC load shedding in Attachment 2.B, DC Load Shed Following Phase A and CVI Actuation

B. (1) isolate air to the hydraulic oil pump for each MSIV
(2) during performance of DC load shedding in Attachment 2.B, DC Load Shed Following Phase A and CVI Actuation

C. (1) align alternate power to each MSIV hydraulic oil pump SOV
(2) because Battery 1D2 is not credited for continued operation in the SBO coping analyses

D. (1) isolate air to the hydraulic oil pump for each MSIV
(2) because Battery 1D2 is not credited for continued operation in the SBO coping analyses

Answer: B

K/A Match: This question matches the KA by requiring knowledge of the reasons for performing actions in the EOP to maintain MSIVs closed following DC load shedding (loss of DC power).

Explanation:

A. Incorrect. First part is incorrect but plausible because aligning alternate power to the MSIV hydraulic oil pump SOVs is similar to another action performed in the EOPs to ensure the MSIVs remain closed, however, this similar action is performed in EOP-2.0A/B when a Steam Generator is faulted outside containment to ensure that DC power remains available to the MSIV hydraulic oil pump solenoid valves to ensure they remain closed (the valves are energized to close), therefore, preventing the MSIVs from inadvertently opening. Second part is correct, per ECA-0.0A bases for Step 13, air to the MSIVs is isolated to ensure the valves remain closed during DC load shedding of Attachment 2.B.

B. Correct. First part is correct, per ECA-0.0A Step 13 bases air to the MSIVs is isolated to ensure the valves remain closed during DC load shedding of Attachment 2.B. Second part is correct, see A above.

C. Incorrect. First part is incorrect but plausible, see A above. Second part is incorrect but plausible, the MSIV hydraulic oil solenoid valves are powered from non-safeguard battery 1D2, therefore it is plausible/reasonable to believe that this non-safeguard battery would not be included in the SBO coping analyses and would therefore necessitate local operator action to negate any negative consequences from the loss of power to this battery's downstream loads and place the components which may lose power in the safest possible condition. The MSIVs remain closed by isolating air to the solenoid valves that may subsequently fail open if battery power (1D2) to them is lost. If the solenoid valves were to fail open due to a loss of battery power from 1D2 then air would be ported to the MSIV hydraulic oil pumps which would then increase hydraulic oil pressure to cause the MSIVs to come off their closed seat.

D. Incorrect. First part is correct, see B above. Second part is incorrect but plausible, see C above.

Technical Reference(s): ECA-0.0A, Rev 9 PCN-1
EOP-2.0A, Rev 9 PCN-

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** the bases for operator actions, notes and cautions from ECA-0.0A, Loss of All AC Power.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X

Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	2	**K/A**	065AA2.05	
		Importance Rating	3.4	

K/A Statement: Loss of Instrument Air: Ability to determine and interpret the following as they apply to Los of Instrument Air: When to commence a plant shutdown if instrument air pressure is decreasing.

Question 26

Unit 1 plant conditions:

- 100% power
- Instrument Air header pressure is 33 PSIG and lowering rapidly
- All available Air Compressors are running
- ABN-301, Instrument Air System Malfunction, is being implemented

Which of the following describes required actions for this condition?

A. Initiate a Manual Turbine Runback per ABN-302, Feedwater, Condensate, Heater Drain System Malfunction.

B. Trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection.

C. Align the Unit Instrument Air Cross-tie to assist in pressure recovery per ABN-301.

D. When pressure reaches 25 PSIG, trip the Reactor and enter EOP-0.0A, Reactor Trip or Safety Injection.

Answer: B

K/A Match: This question matches the KA by requiring the ability to determine when the plant is required to be shut down (tripped) due to lowering instrument air pressure**.**

Explanation:

A. Incorrect. Plausible because loss of Instrument Air has a significant impact on the secondary plant and it could be thought that rapidly reducing power would minimize the effects, however, ABN-301 requires a Unit trip when pressure reaches 35 PSIG

B. Correct. ABN-301, Instrument Air Malfunction, requires a Unit trip when pressure reaches 35 PSIG. The EOP is entered but the actions of ABN-301, Instrument Air Malfunction are still performed which includes local control of Charging flow due to failed open valves

C. Incorrect. Plausible because if IA pressure were above 35 psig, it could be correct, however, ABN-301 requires a Unit trip when pressure reaches 35 PSIG

D. Incorrect. Plausible if thought that the pressure criteria for tripping was 25 PSIG and the actions to trip at that point would be correct

Technical Reference(s): ABN-301, Rev 13 PCN-4 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EVALUATE** all system limitations and precaution associated with responding to Instrument Air system malfunctions (LO21ABN301O06)

Question Source: **Bank #** 2011 NRC Exam Q54
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	077G.2.4.4	
		Importance Rating	4.5	

K/A Statement: Generator Voltage and Electric Grid Disturbances: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question 27

Unit 1 plant conditions:

- Reactor power = 75%
- Grid voltage & frequency fluctuations occurring
- Plant Computer alarm received for Main Generator Volts/Hz
- ABN-601, Response to a 138/345 KV System Malfunction, in progress
- Grid Frequency as indicated below:

Based on the above plant conditions, complete the following statement:

If grid frequency remains at the indicated value, ABN-601 directs tripping the Reactor _____.

A. immediately

B. within 2 seconds

C. within 30 seconds

D. within 2 minutes

Answer: A

K/A Match: This question matches the KA by requiring knowledge of an entry level abnormal condition required to be taken from ABN-601.

Explanation:

A. Correct. Correct per ABN-601 if Grid Frequency is at or below 57.5 Hz the Reactor must be tripped immediately

B. Incorrect. Incorrect but plausible because a 2-second timer is used in ABN-601 to trip the reactor if frequency is below 58.0 Hz.

C. Incorrect. Incorrect but plausible because a 30-second timer is used in ABN-601 to trip the reactor if frequency is below 58.4 Hz.

D. Incorrect. Incorrect but plausible because a 2-minute timer is used in ABN-601 to divorce the safeguards busses from the grid if frequency is below 59.4 Hz.

Technical Reference(s): ABN-601 Rev 13 PCN-4 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Main Generator (LO21YSMG1OB26)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	1	
Level of Difficulty:	3	**K/A**	W/E11EK2.2	
		Importance Rating	3.9	

K/A Statement: Loss of Emergency Coolant Recirculation: Knowledge of the interrelations between the loss of emergency coolant recirculation 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 28

Unit 1 initial plant conditions:

- Reactor power = 100%
- A Large Break LOCA occurs

Current plant conditions:

- Crew transitions to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION
- Containment pressure = 48 psig
- RWST level = 15%

Based on current plant conditions, complete the following statements:

1. ECA-1.1A directs stopping ____(1)____Containment Spray Pumps.

2. Per ECA-1.1A when 1-ALB-2B, Window 4.6 – RWST EMPTY alarm is received it ____(2)____ permissible to secure ALL Containment Spray Pumps.

A. (1) two
   (2) is

B. (1) zero
   (2) is

C. (1) two
   (2) is NOT

D. (1) zero
   (2) is NOT

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the relationship between emergency coolant and a loss of emergency coolant recirculation including the proper operation of the system.

Explanation:

A. Correct: 1st part is correct. Per ECA-1.1A, step 10, with water in the RWST and containment pressure between 18 and 50 psig, there should be 2 Containment Spray pumps operating. 2nd part is correct, per the bases of ECA-1.1A it is permissible to secure containment spray pumps when the RWST EMPTY alarm is received.

B. Incorrect: 1st part is incorrect because the operator should only secure 2 CS pumps. It is plausible because if containment pressure were > 50 psig (2 psig away), it would be correct. 2nd part is correct, see A above.

C. Incorrect: 1st part is correct, see A above. 2nd part is incorrect but plausible because the containment spray pumps are NOT normally allowed to be secured at an RWST EMPTY alarm (9%), they are normally secured when RWST level reached 0%. However, an exception is made for ECA-1.1A to optimize response and recovery action performance.

D. Incorrect: 1st part is incorrect but plausible, see B above. 2nd part is incorrect but plausible, see C above.

Technical Reference(s): ECA-1.1A, Rev 9 PCN- Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Containment Spray System (LO21YSCT1OB05).

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

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	012K2.01	
		Importance Rating	3.3	

K/A Statement: Reactor Protection: Knowledge of bus power supplies to the following: RPS channels, components and interconnections.

Question 29

Which of the following provides power to the Unit 2 Train 'A' Reactor Trip Breaker shunt trip coil?

A. 2ED1

B. 2ED3

C. 2D1

D. 2D3

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the power supply to the shunt trip portion of RPS.

Explanation:

A. Correct. The Unit 2 Train A Reactor Trip Breaker Shunt Trip Coils are powered from 2ED1.

B. Incorrect. Plausible because 2ED3 is generally considered to be a Train A power supply and it is a DC bus.

C. Incorrect. Plausible because 2D1 is a DC bus. Although it is a non-safeguards DC bus there are other components which could be mistaken as safeguards equipment powered off of non-safeguards busses; such as DC Control Power to the RCPs being supplied from 2D3.

D. Incorrect. Plausible because 2D3 is a DC bus. Although it is a non-safeguards DC bus there are other components which could be mistaken as safeguards equipment powered off of non-safeguards busses; such as DC Control Power to the RCPs being supplied from this buss.

Technical Reference(s): LO21SYSES2 Study Guide, Rev 9/24/04
DC Electrical Distribution Study Guide Rev 5/5/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Solid State Protection System including interrelations with other systems to include interlocks and control loops.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	012G2.1.32	
		Importance Rating	3.8	

K/A Statement: Reactor Protection: Ability to explain and apply system limits and precautions.

Question 30

Unit 1 plant conditions:

- Train ‘A’ SSPS is being tested
- Train ‘A’ Input Error Inhibit Switch is in INHIBIT
- Train ‘A’ Mode Selector Switch is in TEST

Which of the following are capable of being processed by Train ‘A’ SSPS?

A. Manual SI ONLY

B. Manual Reactor Trip ONLY

C. Auto SI and Auto Reactor Trip ONLY

D. Manual SI and Manual Reactor Trip ONLY

Answer: B

K/A Match: This question matches the KA by requiring the ability to explain RPS system limits and precautions.

Explanation:

A. Incorrect. Plausible if confused whether a manual SI or manual Rx Trip is still available in this condition. No SI actuation will occur because the Mode Selector Switch is in the TEST position, however, a manual Reactor Trip is still available.

B. Correct. As stated in the PRECAUTIONS of SOP-711A, Solid State Protection System, having the Mode Selector Switch in TEST while the Input Error Inhibit Switch is in INHIBIT blocks any ESF slave relay actuation such as Safety Injection. A manual Reactor Trip is still available in this condition.

C. Incorrect. Plausible if thought that placing the Input Error Inhibit Switch in INHIBIT would still allow an automatic Reactor Trip, however, the purpose of this switch is to prevent any external signals other than manual from being input into the Solid State Protection System. This condition includes an automatic Safety Injection (SI).

D. Incorrect. Plausible because a manual Reactor Trip is still available in this condition, however, with the Mode Selector Switch in TEST it prevents any Engineered Safety Feature (ESF) slave relay actuations such as SI.

Technical Reference(s): SOP-711A, Rev 10, PCN-
OPT-447A, Rev 10 PCN-5

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Solid State Protection System and **PREDICT** the system response.

Question Source:
Bank #
Modified Bank # 2014 NRC Exam Q35 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	013K4.03	
		Importance Rating	3.9	

K/A Statement: Engineered Safety Features Actuation: Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: Main Steam Isolation System.

Question 31

Which of the following ESFASs is designed to prevent exceeding the Containment pressure limit?

A. Containment Isolation Phase A

B. Containment Isolation Phase B

C. Main Steam Line Isolation

D. Safety Injection Actuation

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the Main Steam Isolation System.

Explanation:

A. Incorrect. Plausible because all non-essential lines in or out of Containment are isolated by Phase A, however, Phase A isolation is designed to prevent radiological release from Containment.

B. Incorrect. Plausible because all CCW support for RCPs in or out of Containment is isolated by Phase B, however, Phase B isolation is designed to prevent radiological release from Containment.

C. Correct. MSLI is designed to minimize excessive cooldown and Containment pressure.

D. Incorrect. Plausible because the SI signal starts Containment Spray and initiates Phase A isolation, however, the SI signal starts pumps and positions valves to ensure the core is cooled during an accident.

Technical Reference(s): FSR Section 7.0, Rev 102
OP51SYSES1 Study Guide, Rev 4/4/08
TSB LCO 3.3.2 ESFAS Instrumentation, Rev 77

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **LIST** the ESF Systems and STATE their functions (OP51SYSES1OB03).
LIST and **EXPLAIN** the coincidence and setpoints associated with the following ESF actuation signals:
Steamline Isolation (OP51SYSES1OB05)

Question Source:
Bank # 2011 NRC Exam Q13
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	4	**K/A**	013K5.02	
		Importance Rating	2.9	

K/A Statement: Engineered Safety Features Actuation: Knowledge of the operational implications of the following concepts as they apply to ESFAS: Safety system logic and reliability.

Question 32

Unit 2 plant conditions:

- 100% power
- Distribution Panel 2PC2 de-energized
- ABN-603, Loss of Protection or Instrument Bus, in progress

Subsequently:

- 2-PT-934, Containment Pressure Channel IV, fails high

Which of the following completes the statements below?

1) An automatic Safety Injection actuation ____(1)____ occur.

2) An automatic Containment Spray actuation ____(2)____ occur.

A. (1) does
   (2) does

B. (1) does
   (2) does NOT

C. (1) does NOT
   (2) does

D. (1) does NOT
   (2) does NOT

Answer: B

K/A Match: This question matches the KA by requiring knowledge of the safety system logic for ESFAS systems.

Explanation:

A. Incorrect. First part is correct, see B below. Second part is incorrect but plausible because the applicant may believe that the CS signal is de-energize to actuate which would cause CS auto actuation.

B. Correct. First part is correct because the containment high pressure SI signal, which is 2/3 coincidence, is de-energize to actuate. With channel II de-energized due to the loss of power to 2PC2, which de-energizes 2-PT-936 (containment pressure channel II), plus the failure of 2-PT-934 (channel IV) the 2/3 coincidence is met causing an auto SI. Second part is correct because the CS signal is 2/4 coincidence and energize to actuate. With channel II de-energized due to the loss of 2PC2 and the failure of 2-PT-934 (channel IV) which is energize to actuate for the CS signal there is only 1/4 coincidence met and Containment Spray will not actuate.

C. Incorrect. First part is incorrect but plausible since one of the two signals is energized to actuate and the other is de-energized to actuate, but SI is de-energized to actuate and CS is energized to actuate, the applicant must recall which is which. Second part is incorrect but plausible, see A above.

D. Incorrect. First part is incorrect but plausible, see C above. Second part is correct, see B above

Technical Reference(s): 208,120,118 VAC Dist Study Guide, Rev 5/9/11
Containment Spray Study Guide, Rev 5/2/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operations of the Reactor Protection and Engineered Safeguard Actuation Systems (LO21SYSES1OB105).

Question Source:
Bank # 2015 NRC Exam Q9
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 7

55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	022G2.2.44	
		Importance Rating	4.2	

K/A Statement: Containment Cooling: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question 33

Unit 1 plant conditions:

- 50% power
- The following Annunciators are in alarm:
  - 1-ALB-02B, Window 1.12 - CNTMT FN CLR 3 & 4 CNDS LVL HI
  - 1-ALB-02B, Window 3.12 - CNTMT FN CLR 3 & 4 CND FILL RATE HI
  - X-ALB-11C, Window 1.11 - CH WTR SRG TK LVL HI-HI / LO-LO is alarming intermittently due to a low level
- 1-HS-5413A, CNTMT FN CLR FN 3, and 1-HS-5417A, CNTMT FN CLR FN 4 were stopped

Based on the conditions above, complete the statement below.

After stopping Containment Fan Coolers 3 & 4, the US will _____.

A. continue plant operation per IPO-003A, Power Operation

B. trip the unit and enter EOP-0.0A, Reactor Trip or Safety Injection

C. prepare for a containment entry to isolate the leak per STA-620, Containment Entry

D. isolate the leak from Ventilation Panel CV-03 per SOP-814, Ventilation Chilled Water

Answer: C

K/A Match: This question matches the KA by requiring the ability to interpret control room indications for the containment cooling system and how taking action will affect the plant.

Explanation:

A. Incorrect. Plausible because continued plant operation is allowed, however, the leak has not been isolated.

B. Incorrect. Plausible if thought that with 2 Containment Fan Coolers out of service the Unit. must be tripped

C. Correct. A Containment entry must be made to locally isolate the leak. The Chill Water Outlet Valves automatically close, however, the Chill Water Inlet valves must be manually closed and are NOT shown on Figure 3.

D. Incorrect. Plausible because the leak is not isolated, however, leak isolation must be performed inside Containment.

Technical Reference(s): 1-ALB-02B, Rev 10 PCN-3
X-ALB-11C, Rev 5 PCN-12
Containment Study Guide, Rev 5/2/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **INDENTIFY** and **DESCRIBE** the main control board/plant computer control, alarms and indications associated with the Ventilation Chilled Water System (OP51SYSCH2OB06)

Question Source:
Bank # 2011 NRC Exam Q14
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 9
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	G2.1.32	
		Importance Rating	3.8	

K/A Statement: Containment Spray: Ability to explain and apply system limits and precautions.

Question 34

In accordance with SOP-204B, Containment Spray System, complete the following statement.

The opening stroke time of 2-HV-4776 and 2-HV-4777, Containment Spray Heat Exchanger Discharge Valves, is based on preventing _____.

A. spray riser damage due to pipe whip

B. pump motor damage due to overcurrent

C. piping system damage due to water hammer

D. heat exchanger u-tube damage due to flow induced vibrations

Answer: B

K/A Match: This question matches the KA by requiring the ability to explain and apply Containment Spray limits and precautions.

Explanation:

A. Incorrect: Incorrect but plausible because the slow opening of the valves could be thought to reduce fluid flow velocities and therefore reduce excessive forces on spray risers; eliminating pipe whip.

B. Correct: Correct, per SOP-204B, the opening stroke time of the HX Outlet valves is based on preventing an overcurrent trip of the CTPs thus preventing motor damage.

C. Incorrect: Incorrect but plausible because this is another P&L listed in SOP-204B for maintaining the risers full of water while the system is in a standby condition.

D. Incorrect: Incorrect but plausible because flow induced vibrations are a concern on the Unit 2 SG U-tubes and the reason a split flow bypass valve is used in the Unit 2 MFW system; the slow opening of the valve could be thought to eliminate this concern in the Containment Spray system as well.

Technical Reference(s): SOP-204B, Rev 6 PCN-8 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Containment Spray system including Technical Specifications and TRM.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**	2	
		Group	1	
Level of Difficulty:	2	**K/A**	039K5.08	
		Importance Rating	3.6	

K/A Statement: Main and Reheat Steam: Knowledge of the operational implications of the following concepts a they apply to MRSS: Effect of steam removal on reactivity.

Question 35

Unit 1 is at 100% when a steam generator Atmospheric Relief Valve (ARV) fails OPEN.

The ARV opening adds ____(1)____ reactivity to the core.

The net reactivity effect is greater at ____(2)____ of Core Life

A. (1) negative
   (2) End

B. (1) negative
   (2) Beginning

C. (1) positive
   (2) End

D. (1) positive
   (2) Beginning

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the effect that steam removal has on reactivity.

Explanation:

A. Incorrect: 1st part is in incorrect because at 100% power, even at BOL, an increase in steam demand will add positive reactivity. It is plausible because at low power at BOL, the moderator temperature coefficient could be positive (steam demand adding negative reactivity. 2nd part is correct because reactivity effects at EOL are greater due to a larger MTC.

B. Incorrect: 1st part is incorrect but plausible, see A above. 2nd part is incorrect because the moderator temperature coefficient is more negative at EOL. It is plausible because it is a common error to think that that reactivity effects at BOL are greater.

C. Correct. 1st part is correct. An increase in steam demand lowers RCS temperature which will, in turn add positive reactivity (even at BOL when at 100% power). 2nd part is correct, see A above.

D. Incorrect: 1st part is correct, see C above. 2nd part is incorrect but plausible, see B above.

Technical Reference(s): Increased Heat Removal Accidents Study Guide, Rev 8/19/11
LO21GFRCOF, Rev 11/29/07
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DISCUSS** the excessive increase in secondary steam flow transient (LO21MCOTA8OB102).

Question Source:
Bank # 2016 NRC Exam Q16
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	059K3.02	
		Importance Rating	3.6	

K/A Statement: Main Feedwater: Knowledge of the effect that a loss or malfunction of the MFWS will have on the following: AFW System.

Question 36

Unit 1 plant conditions:

- 45% power
- The running MFP trips followed by a manual reactor trip
- MDAFWP 1-01 fails to start

Based on the conditions above, complete the statements below.

1) TDAFWP will automatically start when NR level reaches the LOW-LOW setpoint in ____(1)____ SG(s).

2) If TDAFWP failed to start, MDAFWP 1-02 flow shall be limited to a MAXIMUM of ____(2)____ gpm with both trains cross-connected.

A. (1) ONLY ONE
(2) 700

B. (1) ONLY ONE
(2) 800

C. (1) a MINIMUM of TWO
(2) 700

D. (1) a MINIMUM of TWO
(2) 800

Answer: D

K/A Match: This question matches the KA by requiring knowledge of the effect that a loss or malfunction of the MFWS will have on the AFW system.

Explanation:

A. Incorrect. 1st part is incorrect because for the Turbine Driven AFW pump, the start setpoint is 2/4 SGs at the LOW-LOW setpoint. It is plausible because the auto start for the Motor Driven AFW pumps occur when 1 SG is at the LOW-LOW setpoint. 2nd part is incorrect because flow is limited to 800 gpm. It is plausible because the orifice installed downstream of each Feed Regulating Valve is designed to limit flow to 700 gpm to preclude run-out conditions.

B. Incorrect. 1st part is incorrect but plausible, see A above. 2nd part is correct. When cross-connected, flow is limited to 800 gpm to prevent a run-out condition.

C. Incorrect. 1st part is correct. The Auto-Start setpoint for the Turbine Driven AFW pump is LOW-LOW on 2/4 SGs. 2nd part is incorrect but plausible, see A above.

D. Correct. 1st part is correct, see C above. 2nd part is correct, see B above.

Technical Reference(s): ABN-305, Rev 8 PCN-2
Auxiliary Feedwater Study Guide, Rev 5/11/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Auxiliary Feedwater System (LO21SYSAF1OB105)

Question Source:
Bank # 2016 NRC Exam Q24
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**	2	
		Group	1	
Level of Difficulty:	4	**K/A**	061K2.01	
		Importance Rating	3.2	

K/A Statement: Auxiliary / Emergency Feedwater: Knowledge of bus power supplies to the following: AFW system MOVs.

Question 37

Which of the following completes the statement below regarding the power supplies to the Auxiliary Feedwater Motor Operated Isolation Valves for SGs 1-02 and 1-03?

SG 1-02

- 1-HV-2492A, TD AFW PMP DISCH TO SG 1-02 ISOL VLV
- 1-HV-2492B, MD AFW PMP 1-01 DISCH to SG 1-02 ISOL VLV

SG 1-03

- 1-HV-2493A, MD AFW PMP 1-02 DISCH TO SG 1-03 ISOL VLV
- 1-HV-2493B, TD AFW PMP DISCH TO SG 1-03 ISOL VLV

1) The isolation valves associated with each SGs TD and MD AFWP are powered from ____(1)____.

2) The power supplies for 1-HV-2492B and 1-HV-2493B are ____(2)____.

A. (1) opposite trains
   (2) the same

B. (1) opposite trains
   (2) different

C. (1) the same train
   (2) the same

D. (1) the same train
   (2) different

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the power supplies to AFW isolation valves (MOVs).

Explanation:

A. Correct: First part correct, for each SG, the associates MD and TD AFW isolation valves come from opposite train power supplies. For example, 1-HV-2492A, TD AFW PMP DISCH TO SG 1-02 ISOL VLV and 1-HV-2492B, MD AFW PMP 1-01 DISCH to SG 1-02 ISOL VLV are both isolation valves on SG 1-02, however, they are powered from opposite trains (i.e. 2492A powered from 1EB1-1 and 2492B powered from 1EB2-1). Second part is correct, per design drawing E1-0007, Sheet B, the power supply to 1-HV-2492B and 1-HV-2493B is 1EB2-1.

B. Incorrect. First part is correct, see A above. Second part is incorrect but plausible because these valves are on different SGs. 1-HV-2492B is an isolation valve to SG 1-02, which is a train B component. 1-HV-2493B is an isolation valve to SG 1-03, which is a train A component. However, despite these valves belonging to opposite train components they have the exact same power supply, 1EB2-1.

C. Incorrect. First part is incorrect but plausible, because there are multiple examples of valves in the plant designated by their nomenclature as a specific train but they receive power from the opposite train. For example, 1-HV-2480, Service Water to Motor Driven AFW Pump 01 is a Train B valve by its nomenclature (i.e. even numbered components are Train B, odd numbered components are Train A), however, it is powered from 1EB1-1 per design drawing E1-0007, Sheet -. Second part is correct, see A above.

D. Incorrect: First part is incorrect but plausible, see C above. Second part is incorrect but plausible, see B above.

Technical Reference(s): Control Board Layout — Attached w/ Revision # See Comments / Reference
LO21SYSAC2, Rev 10-3-13
E1-0007, Sht -, Rev CP-36
E1-0007, Sht B, Rev CP-37

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the Auxiliary Feedwater system including interrelations with other systems to include interlocks and control loops in accordance with DBD-ME-206.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	061A2.04	
		Importance Rating	3.4	

K/A Statement: Auxiliary / Emergency Feedwater: Ability to (a) predict the impacts of the following malfunctions or operations on the AFWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pump failure or improper operation.

Question 38

Unit 1 plant conditions:

- Reactor Trip from 85% power
- BOP secured the TDAFWP per the Foldout Page of EOP-0.0A, Reactor Trip or Safety Injection

Based on conditions above, complete the statements below.

If MDAFWP 1-02 tripped, …

1) SGs ____(1)____ would be without AFW flow.

2) ABN-305, AFW System Malfunction, directs ____(2)____.

A. (1) 1-01 and 1-02
   (2) starting the TDAFWP and feeding all 4 SGs

B. (1) 1-03 and 1-04
   (2) starting the TDAFWP and feeding all 4 SGs

C. (1) 1-01 and 1-02
   (2) cross-connecting MDAFW trains and feeding all 4 SGs

D. (1) 1-03 and 1-04
   (2) cross-connecting MDAFW trains and feeding all 4 SGs

Answer: B

K/A Match: This question matches the KA by requiring the ability to predict the impact of a tripped MDAFW pump and use the procedure to mitigate the failure.

Explanation:

A. Incorrect. First part is incorrect but plausible with MDAFWP 1-02 tripping it could be thought that SG 1-02 had lost flow, however, since the MDAFWPs do not feed SGs based on designation or train this is incorrect. Second part is correct, ABN-305 directs starting the TDAFWP, if it is available, and feeding all 4 SGs.

B. Correct. First part is correct, MDAFWP 1-02 feeds SGs 1-03 and 1-04. Therefore, if MDAFWP 1-02 trips and the TDAFWP is secured, then SGs 1-03 and 1-04 will lose AFW flow. Second part is correct, see A above.

C. Incorrect. First part is incorrect but plausible, see A above. Second part is incorrect but plausible because this action is required by ABN-305 if the TDAFWP is not available.

D. Incorrect. First part is correct, see B above. Second part is incorrect but plausible, see C above.

Technical Reference(s): ABN-305, Rev 8 PCN-2
AFW Big Book, Rev 7/1/11
SOP-304A, Rev 17 PCN-13
EOP-0.0A, Rev 9 PCN-4

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **ANALYZE** the response to a MDAFWP malfunction in accordance with ABN-305, Auxiliary Feedwater System Malfunction

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	062K3.01	
		Importance Rating	3.5	

K/A Statement: AC Electrical Distribution: Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following: Major system loads.

Question 39

Site conditions:

- Both Units AC distribution systems are in a normal lineup
- A fire in the 138 KV Switchyard occurred and caused a Lockout Relay actuation on the associated Startup Transformer
- The '– 2' breakers on the affected 6.9 KV Safeguards Busses normally supplied by the Locked Out Transformer failed to automatically close

Based on the above plant conditions, complete the statements below.

1) The Startup Transformer affected by the fire in the 138 KV Switchyard was ____(1)____.

2) Unit ____(2)____ 6.9 KV Safeguards busses will momentarily lose power, prior to their associated EDG supply breakers closing.

A. (1) XST1
   (2) 1

B. (1) XST1
   (2) 2

C. (1) XST2
   (2) 1

D. (1) XST2
   (2) 2

Answer: B

K/A Match: This question matches the KA by requiring knowledge of the effect of several startup transformer failures and how these failures will affect downstream essential buses (loads).

Explanation:

A. Incorrect: Part 1 is correct; the 138 KV Switchyard provides power to Startup Transformer XST1. Part 2 is Incorrect but plausible because it would logically seem that XST1 would supply primary power to Unit 1 6.9 KV Safeguards Busses, however, XST1 provides primary power to the Unit 2 6.9 KV Safeguard Busses.

B. Correct: Part 1 is correct, see A above. Part 2 is correct, XST1 is supplied from the 138 KV Switchyard and it provides primary power to the Unit 2 Safeguards Busses, therefore a lockout relay actuation on XST1 will cause the primary power source breakers 2EA1-1 and 2EA2-1 to open. Since the '– 2' breakers on 2EA1 and 2EA2 failed to automatically close and receive power from XST2…these Safeguards Busses will lose power until their associated EDGs start and tie on to the busses.

C. Incorrect. Part 1 is incorrect but plausible because it is a common misconception that XST1 is supplied by the higher voltage (345 KV) switchyard. Part 2 is incorrect but plausible, see A above.

D. Incorrect. Part 1 is incorrect but plausible, see C above. Part 2 is correct, see B above.

Technical Reference(s):	LO21SYSAC2 Study Guide, Rev 4/7/17 Electrical Grip Connection Diagram Big Book Drawing, Rev 11/9/17	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the 6.9KV and 480V Electrical Distribution system and PREDICT the system response in accordance with SOP-603, 604 and BN-602.

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	7	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	062K4.03	
		Importance Rating	2.8*	

K/A Statement: AC Electrical Distribution: Knowledge of AC distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers.

Question 40

Complete the statements below regarding a 6.9 KV bus fast transfer.

1) Only 6.9 KV ____(1)____ busses have the capability to fast transfer.

2) Fast transfer of a 6.9 KV bus to its alternate power supply must occur within a MAXIMUM of ____(2)____ second(s).

A. (1) Safeguards
(2) 1.0

B. (1) Non-Safeguards
(2) 1.0

C. (1) Safeguards
(2) 0.25

D. (1) Non-Safeguards
(2) 0.25

Answer: D

K/A Match: This question matches the KA by requiring knowledge of the AC system interlocks for automatic bus transfer relays.

Explanation:

A. Incorrect: Part 1 is incorrect but plausible because on a fast transfer downstream loads do not lose power, therefore, it is reasonable to believe the Safeguards busses will fast transfer to maintain continuity of power to the downstream vital loads. Part 2 is incorrect but plausible because 1.0 seconds is the maximum amount of time two breakers can be closed on to the same bus (paralleled). After the alternate breaker is shut the normal breaker will open within 1.0 seconds and vice versa.

B. Incorrect: Part 1 is correct, only Non-Safeguards busses have the capability to Fast Transfer. Part 2 is incorrect but plausible, see A above.

C. Incorrect: Part 1 is incorrect but plausible, see A above. Part 2 is correct the transfer of the bus to its alternate power supply must occur within 0.25 seconds before the TDPU times out and opens the contact preventing alternate power supply breaker closure.

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

Technical Reference(s): LO21SYSAC2 Study Guide, Rev 4/7/17 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the 6.9KV and 480V electrical distribution system and PREDICT the response in accordance with SOP-603, 604 and ABN-602.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	063K1.02	
		Importance Rating	2.7	

K/A Statement: DC Electrical Distribution: Knowledge of the physical connections and/or cause-effect relationship between the dc distribution system and the following systems: AC electrical system.

Question 41

Inverter 1PC4 Static Switch will automatically supply panel 1PC4 loads from its unregulated 120 VAC bypass power source when DC input voltage lowers to a setpoint value of _____ VDC.

A. 59

B. 100

C. 105

D. 120

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the cause/effect relationship between the DC and AC electrical distribution systems.

Explanation:

A. Incorrect: Incorrect because the static transfer switch will actuate when DC voltage lowers to 105 VDC. It is plausible because if the question were asking when AC output voltage would cause the static switch to actuate, it would be correct.

B. Incorrect: Incorrect because the static transfer switch will actuate when DC voltage lowers to 105 VDC. It is plausible because for the static switch to actuate, the frequencies have to be within .8 Hz. If the applicant incorrectly associates the 0.8 with voltage requirement (0.8 x 125 VDC = 100 VDC).

C. Correct: This is the setpoint for the static switch to actuate.

D. Incorrect: Incorrect because the static transfer switch will actuate when DC voltage lowers to 105 VDC. It is plausible because if output current rises to 120 % of rated value, the static switch will actuate.

Technical Reference(s):	208, 120 & 118 VAC Study Guide, Rev 4/11/17	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the 208/120 VAC Distribution system including interrelations with other systems to include interlocks and control loops as described in DBD-EE-041.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	4	**K/A**	064K6.08	
		Importance Rating	3.2	

K/A Statement: Emergency Diesel Generator: Knowledge of the effect that a loss or malfunction of the following will have on the EDG: Fuel oil storage tanks.

Question 42

Unit 1 plant conditions:

- Crew just completed a 24 hour run of DG 1-02
- Level in the DG 1-02 FOST is 85,000 gallons

1) In accordance with OWI-104-27, NEO DG 1-02 Operating Logs, the FOST level is ____(1)____ the amount required.

2) DG FOST level is covered by TS LCO ____(2)____.

A. (1) below
(2) 3.8.1, AC Sources – Operating

B. (1) below
(2) 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air

C. (1) above
(2) 3.8.1, AC Sources – Operating

D. (1) above
(2) 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air

Answer: B

K/A Match: This question matches the KA by requiring knowledge of the effect that a fuel oil storage tank malfunction (low level) has on the EDG.

Explanation:

A. Incorrect. First part is correct, the logs require 88,000 gallons of fuel oil in the storage tank. Second part is incorrect but plausible because TS 3.8.1 covers the required level in the fuel oil day tanks vice the fuel oil storage tanks.

B. Correct. First part is correct, see A above. Second part is correct, TS 3.8.3 covers the required level in the fuel oil storage tanks.

C. Incorrect. First part is incorrect but plausible because the required level by logs is 88,000 gallons and the 85,000 gallons provided in the stem is within a reasonable to range to be plausible. Second part is incorrect but plausible, see A above.

D. Incorrect. First part is incorrect but plausible, see C above. Second part is correct, see B above.

Technical Reference(s): TS LCO 3.8.3, Amend 153
TS SR 3.8.3.1, Amend 156
TS LCO 2.8.1, Amend 150
TS SR 3.8.1.4, Amend 156
OWI 104-27, Rev 19

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Emergency Diesel Generator system in accordance with Technical Specifications and TRM.

Question Source: **Bank #**
Modified Bank # 2014 NRC Exam Q44 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	073A1.01	
		Importance Rating	3.2	

K/A Statement: Process Radiation Monitoring: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRMS controls, including: Radiation levels.

Question 43

Given the following conditions:

- X-HS-WM182A, VALVE TO CO-CURRENT WASTE HOLD-UP TANKS, and X-HS-WM183A, VALVE TO LVW POND, are in their normal alignment
- A radioactive spill in the vicinity of SFGD Sump 1-03 causes radiation detector X-RE-5251A (ABP074), LVW/EVAP POND VNT & DRN HDR RADIATION DETECTOR, to go into HIGH alarm

Following the HIGH alarm, X-HS-WM182A is ____(1)____ and X-HS-WM183A is ____(2)____.

A. (1) OPEN
   (2) OPEN

B. (1) OPEN
   (2) CLOSED

C. (1) CLOSED
   (2) OPEN

D. (1) CLOSED
   (2) CLOSED

Answer: B

K/A Match: This question matches the KA by demonstrating the operator's ability to monitor the PRMS system radiation levels and predict the changes that will occur in system controls (valve positions) based on elevated radiation levels.

Explanation:

A. Incorrect. The first part is correct, see B below. The second part is incorrect but plausible, if the misconception exists that the operator is required to reposition the valve in the field as would be required for an ALERT alarm in accordance with RWS-108.

B. Correct. A HIGH alarm from X-RE-5251A automatically redirects flow from the LVW pond to the Co-current waste system. Therefore, X-HS-WM182A which is normally Closed – Opens and X-HS-WM183A which is normally Open – Closes.

C. Incorrect. The first part is incorrect but plausible, if the misconception exists that the valve did not automatically reposition on a HIGH alarm and RWS-108 would require then operator to manually reposition the valve as on an ALERT. The second part is incorrect but plausible, see A above.

D. Incorrect. The first part is incorrect but plausible, see C above. The second part is correct, see B above.

Technical Reference(s): ALM-3200A, Rev 5 PCN-2
RWS-108, Rev 8 PCN-6

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Digital Radiation Monitoring System and **PREDICT** the system response.

Question Source:
Bank #
Modified Bank # 2015 NRC Exam Q18 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 11 and 13
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	076A2.02	
		Importance Rating	2.7	

K/A Statement: Service Water: 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: Service water header pressure.

Question 44

Unit 1 plant conditions:

- The SSW system is in an abnormal lineup with SSW Pump 1-01 RUNNING and SSW Pump 1-02 in STANDBY
- CCW Pump 1-02 is out of service for preventative maintenance

Subsequently, SSW Pump 1-01 TRIPS and SSW Pump 1-02 automatically STARTS.

Based on the conditions above, which of the following describes the SSW Pump AUTO start feature and required actions per ABN-501, Station Service Water System Malfunction?

SSW Pump 1-02 AUTO started when header ____(1)____. The crew will ____(2)____.

A. (1) pressure in Train A SSW lowered to 10 psig
(2) trip reactor and ALL RCPs

B. (1) flow in Train A SSW lowered to 16,456 gpm
(2) trip reactor and ALL RCPs

C. (1) pressure in Train A SSW lowered to 10 psig
(2) ensure CCWP 1-01 running with both Train Safeguards Loop Isolation Valves open.

D. (1) flow in Train A SSW lowered to 16,456 GPM
(2) ensure CCWP 1-01 running with both Train Safeguards Loop Isolation Valves open.

Answer: A

K/A Match: This question matches the KA by requiring the ability to predict the impact of a service water pump trip and use of the procedure to mitigate the consequences of the opposite train component cooling water pump being out of service during the event.

Explanation:

A. Correct: 1st part is correct. The standby SSW pump starts if the affected train header pressure lowers to 10 psig. The 2nd part is correct, IAW ABN-501, if the unaffected train CCW pump will not start (it can't in this case) you are directed to trip the reactor, GO TO EOP-0.0A and trip ALL RCPs.

B. Incorrect. 1st part is incorrect because the auto start feature is based on pressure, not flow. It is plausible because it could be thought that the AUTO start feature was based on a low flow condition. 2nd part is correct, see A above.

C. Incorrect: 1st part is correct, see A above. 2nd part is incorrect because the action is not directed by ABN-501. Plausible because it could be thought that the CCW System cooling is accomplished via flow though both Safeguards Loop Isolation Valves to the header being cooled by the running SSW Pump. This is a recent change to CPNPP procedures and this used to be the correct action.

D. Incorrect: 1st part is incorrect but plausible, see B above. 2nd part is incorrect but plausible, see C above.

Technical Reference(s): ABN-501, Rev 9 PCN-12
ALM-0011A, Rev 10 PCN-7
ABN-501, Old Version
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **ANALYZE** the response to Station Service Water Pump Trip in accordance with ABN-501, Station Service Water System Malfunction.

Question Source: **Bank #**
Modified Bank # 2012 NRC Exam Q24 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	3	**K/A**	078A3.01	
		Importance Rating	3.1	

K/A Statement: Instrument Air: Ability to monitor automatic operation of the IAS, including; Air pressure.

Question 45

Unit 2 plant conditions:

- Per OWI-409, Equipment Rotation Program, IAC 2-01 was placed in LEAD
- The BOP is monitoring 2-PI-3488, INST AIR AFTFILT OUT PRESS in Control Room

Per SOP-509B, Instrument Air System, …

IAC 2-01 should be loading and unloading between _____ and 115 psig.

A. 90

B. 95

C. 100

D. 105

Answer: D

K/A Match: The question matches the K/A as it requires the operator to demonstrate knowledge of the automatic operation of the Instrument Air System including air pressure at which the Lead IAC would load and unload.

Explanation:

A. Incorrect. Incorrect but plausible if confused when the LEAD compressor loaded and thought it loaded at 90 psig (5 psig below the BACKUP compressor load pressure).

B. Incorrect. Incorrect but plausible as the common BACKUP compressor would cycle pressure between 95 and 115 psig if required.

C. Incorrect. Incorrect but plausible as the STANDBY compressor would cycle pressure between 100 and 115 psig if required.

D. Correct. Correct as the LEAD compressor should load and unload between 105 psig and 115 psig.

Technical Reference(s): SOP-509B, Rev 15 PCN-10
OWI-409, Rev 18 PCN-13

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of the Instrument Air System.

Question Source:
Bank #
Modified Bank # 2016 NRC Exam Q27 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	1	
Level of Difficulty:	2	**K/A**	103A4.01	
		Importance Rating	3.2	

K/A Statement: Containment: Ability to manually operate and/or monitor in the control room: Flow control, pressure control, and temperature control valves, including pneumatic valve controller.

Question 46

BOP is performing EOP-0.0A, Attachment 2, Safety Injection Actuation Alignment, Step 3 – "Verify Containment Isolation Phase A"

_____, must be verified closed as a part of the Phase A Containment Isolation.

A. 1-LCV-0459, U1 LTDN ISOL VLV

B. 1-HV-6082, CH WTR RET ISOL VLV

C. 1-HV-5549, CNTMT PRESS RLF ISOL VLV

D. 1-HV-4696, THBR CLR CCW RET ISOL VLV

Answer: B

K/A Match: This question matches the KA by requiring the ability to operate containment isolation valves in order to stop flow into/out of containment during a Phase A isolation signal.

Explanation:

A. Incorrect. Incorrect but plausible because 1-LCV-459 will close, however, the actuation signal is a Safety Injection slave relay vice a Containment Isolation Phase A signal.

B. Correct. The Chill Water Return and Supply Isolation Valves, 1-HV-6082 is isolated on a Phase A Containment Isolation Signal.

C. Incorrect. Incorrect but plausible because the Containment Pressure Relief Valve, 1-HV-5549 will close, however, it is isolated on a Containment Vent Isolation Signal vice a Containment Phase A Signal.

D. Incorrect. Incorrect but plausible because the Thermal Barrier CCW Return Isolation Valve, 1-HV-4696 is isolated on a Phase B Isolation Signal vice a Phase A Isolation Signal.

Technical Reference(s): EOP-0.0A, Rev 9 PCN-3 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Containment System and **PREDICT** the system response

Question Source:
Bank #
Modified Bank # 2014 NRC Exam Q48 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	2	
Level of Difficulty:	3	**K/A**	003AK1.04	
		Importance Rating	3.1	

K/A Statement: Dropped Control Rod: Knowledge of the operational implications of the following concepts as they apply to dropped Control Rod: Effects of power level and control position on flux.

Question 47

Unit 1 plant conditions:

- Initially at 100% power
- ALM-0064A, 1-ALB-6D, Window 3.7 – ANY ROD AT BOT alarms
- DRPI shows Control Bank D Rod D-4 (near Power Range N44) rod bottom light LIT
- US has determined Control Rod will not be restored within one hour

1. Over the next several hours the overall core Quadrant Power Tilt Ratio (QPTR) will ____(1)____.

2. Within one hour, action must be taken to verify or restore ____(2)____.

A. (1) lower
(2) QPTR

B. (1) lower
(2) Shutdown Margin

C. (1) raise
(2) Shutdown Margin

D. (1) raise
(2) QPTR

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the effect of a dropped control rod on flux distribution**.**

Explanation:

A. Incorrect. Part 1 is incorrect, reduction of QPTR value is plausible if the applicant confuses the actual power change in the fuel assembly with the quadrant power tilt. Power produced in the assembly and in that quadrant overall would be lowered. The lowered power produced in that quadrant results in an increase in the value of the quadrant power tilt ratio (i.e., the power in that quadrant compared to the power in the other quadrants). Part 2 is incorrect but plausible since there is a Tech. Spec. associated with QPTR (3.2.4) and it would apply if the

ratio was out of spec. If it did apply, the Completion Time is not within one hour; it is two hours.

B. Incorrect. Part 1 is incorrect but plausible, see A above. Part 2 is correct, see C below.

C. Correct. Part 1 is correct, as a result of the dropped, indication on N-44 will be lower than on the other three power ranges. This will result in an indicated power tilt in that quadrant. Over the next several hours, Xenon will build in at the dropped rod assembly, further suppressing power in that location (and quadrant) and thus raising the severity of the quadrant power tilt. Part 2 is correct, per ABN-712, Section 3, Dropped or Misaligned Rod in Mode 1 or 2, Step 10 a verification of Shutdown Margin is required within 1 hour and if SDM is not within limits it must be restored per TS 3.1.4.

D. Incorrect. Part 1 is correct, see C above. Part 2 is incorrect but plausible, see A above.

Technical Reference(s): ALM-0064A, Rev 6 PCN-23
TS 3.2.4, Amend 150
TS 3.1.4, Amend 150
ABN-712, Rev 11 PCN-

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the operation of the Rod Control system in accordance with IPO-003A/B, ODA-102, and Operations Guideline 3 and evaluate the effects of system malfunctions on pant operation in accordance with ABN-712.

Question Source:
Bank # 2015 NRC Exam Q57
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 8,10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	2	
Level of Difficulty:	3	**K/A**	028AA2.13	
		Importance Rating	2.9	

K/A Statement: Pressurizer (PZR) Level Control Malfunction: Ability to determine and interpret the following as they apply to a pressurizer level control malfunction: The actual PZR level, given uncompensated level with an appropriate graph.

Question 48

Unit 1 plant conditions:

- IPO-001A, Plant Heatup from Cold Shutdown to Hot Standby, in progress
- PRZR temperature 325°F and stable
- 1-FCV-121, CCP CHRG FLO CTRL VLV inadvertently starts opening
- PRZR level 35% and rising
- 1-FK-121, CCP CHRG FLO CTRL placed in MANUAL and flow stabilized
- PRZR level stabilizes at an indicated level of 36%

Based on the above plant conditions, determine the actual PRZR level in accordance with IPO-001A, Attachment 3, Actual Versus Indicated Pressurizer Level.

REFERENCE PROVIDED

A. 30 %

B. 34%

C. 38%

D. 48%

Answer: C

K/A Match: This question matches the KA by requiring the ability to utilize indicated PRZR level to determine actual PRZR level using appropriate graphs.

Explanation:

A. Incorrect: Incorrect because actual level ~ 38%. It is plausible because if you use the Hot Calibrated Curve (to be used if > 450 °F) you would get 30%.

B. Incorrect: Incorrect because actual level ~ 38%. It is plausible because if you incorrectly use the actual/indicated axis on the cold calibrated curve, you would get 34%.

C. Correct: Using the cold calibrated (uncompensated) curve, an indicated level of 36% represents an actual level of 38%.

D. Incorrect: Incorrect because actual level ~ 38%. It is plausible because if you incorrectly use the Hot Calibrated curve and incorrectly use the actual/indicated axis you would get 48%.

Technical Reference(s): IPO-001A, Rev 24 PCN-6 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: IPO-001A, Attachment 3.(2 pages)

Learning Objective: **DISCUSS** IPO-001, Plant Heatup from Cold Shutdown to Hot Standby, to include the following: Applicability; Limitations; Notes; Instructions.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 5
55.43

Examination Outline Cross-Reference:		**Level**	**RO** **SRO**
Rev:1		**Tier**	1
		Group	2
Level of Difficulty:	3	**K/A**	032AK2.01
		Importance Rating	2.7

K/A Statement: Loss of Source Range Nuclear Instrumentation: Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: Power supplies, including proper switch positions.

Question 49

Given the following conditions:

- A Reactor Startup is in progress
- Intermediate Range power indicates 5 x 10^{-11} amps on both channels
- Source Range High Flux trip has NOT been blocked

For the two switch positions shown below, which of the following describes the RPS response to a blown INSTRUMENT POWER fuse on Source Range Channel N-31 per ABN-701, Source Range Instrument Malfunction?

	SR Level Trip Switch: NORMAL	SR Level Trip Switch: BYPASS
A.	NO Reactor Trip	NO Reactor Trip
B.	Reactor Trip	NO Reactor Trip
C.	NO Reactor Trip	Reactor Trip
D.	Reactor Trip	Reactor Trip

Answer: B

K/A Match: This question matches the KA by requiring knowledge of the relationship between the effects of a loss of power and switches positions for SR instrumentation.

Explanation:

A. Incorrect. Plausible because the BYPASS position is correct, however, a Reactor Trip occurs with the switch in NORMAL.

B. Correct. Removing the Source Range Instrument power fuses will result in a Reactor Trip with the switch in NORMAL. It does not occur if the switch is in BYPASS.

C. Incorrect. Plausible because the conditions are reversed, however, a Reactor Trip occurs with the switch in NORMAL and does NOT occur with the switch in BYPASS.

D. Incorrect. Plausible because the NORMAL position is correct, however, a Reactor Trip does NOT occur with the switch in BYPASS.

Technical Reference(s):	ABN-701, Rev 12 PCN-1 LO21SYSEC1 Study Guide, Rev 5/1/11	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Excore Instrumentation System.

Question Source:
Bank # 2012 NRC Exam Q58
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content:
55.41 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	1	
		Group	2	
Level of Difficulty:	3	**K/A**	059EK3.01	
		Importance Rating	3.5	

K/A Statement: Accidental Liquid Radwaste Release: Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Termination of a release of radioactive liquid.

Question 50

Unit 1 plant conditions:

- 100% power
- Circ Water pumps 1-01, 1-02, 1-03 and 1-04 running

Unit 2 plant conditions:

- Mode 6 – Refueling
- ALL Circ Water pumps secured

Radwaste Operator inadvertently lined up to discharge PET X-01 and LHMT X-01 simultaneously to Unit 1 Circ Water system via Outfall 004.

Which of the following will automatically secure the liquid waste release?

A. Trip of Circ Water Pumps 1-02 and 1-04.

B. MAXIMUM discharge flowrate of 100 gpm is exceeded.

C. Low flow detected on X-FT-5288, Liquid Waste Processing Discharge Flow Transmitter.

D. High radiation detected on X-RE-5253, Liquid Waste Processing Discharge Radiation Detector.

Answer: D

K/A Match: This question matches the KA by requiring knowledge of the reason (high rad) for termination of an accidental (simultaneously discharging two tanks at once) Radwaste release.

Explanation:

A. Incorrect: Incorrect but plausible because there is an interlock which will automatically terminate the release if either unit does not have two Circ Water pumps running. In this case Unit 2 has no Circ Water pumps running and with two pumps tripping on Unit1 this is a reasonable selection.

B. Incorrect: Incorrect but plausible because administratively the discharge flowrate is limited to 100 gpm, however, there is no automatic action associated with this limit.

C. Incorrect: Incorrect but plausible because X-FT-5288 is the flow transmitter which provides indication of discharge flowrate for the release, however, there are no automatic actions associated with this flow transmitter.

D. Correct: Correct, high radiation detected by X-RE-5253 will automatically close X-RV-5253 and terminate the release. This will also occur on an Operate Failure of the rad monitor.

Technical Reference(s):	Radwaste Systems Big Book, Rev 3/29/18 LWPS Lesson Notes, Rev 10/14/03	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** the functions, operation and interlocks of the following LWPS components: X-RE-5253, X-RV-5253

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 13
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	2	
Level of Difficulty:	3	**K/A**	067AA2.05	
		Importance Rating	3.2	

K/A Statement: Plant Fire On Site: Ability to determine and interpret the following as they apply to Plant Fire onsite: Ventilation alignment necessary to secure affected area.

Question 51

CPNPP site conditions:

- Both Units are operating at 100% power
- Train A Control Room Ventilation is in service
- Following a report of a fire outside of the Control Building, Air Intake Smoke Alarms are actuated on BOTH Control Room Ventilation air intakes

Which of the following is the appropriate action to be taken per SOP-802, Control Room Ventilation System?

A. Manually initiate Emergency Recirculation, then Manually shift to single train operation and align the train for Isolation Mode.

B. Manually initiate Emergency Recirculation, then Manually shift both trains to Isolation Mode.

C. Verify automatic actuation of Emergency Recirculation, then Manually shift both trains to Isolation Mode.

D. Verify automatic actuation of Emergency Recirculation, then Manually shift to single train operation and align the train for Isolation Mode.

Answer: A

K/A Match: This question matches the KA by requiring the ability to determine the required ventilation alignment in the event of a fire to secure the control room envelope.

Explanation:

A. Correct. 1st part is correct; Emergency Recirculation is not automatically initiated by smoke alarms, a manual initiation to Emergency Recirculation is required. When Emergency Recirculation is manually initiated, with the handswitch on CV-03, both trains of Control Room Ventilation will shift/start to Emergency Recirc Mode of operation (i.e. the running train will shift to Emergency Recirc and the Standby train will start in the Emergency Recirc mode of operation). 2nd part is correct, Per SOP-802, when CR Ventilation is running with both trains in the Emergency Recirc mode, then one train must be shut down as the system is designed to be operated by train only. Once one train has been shut down and the other train is running in emergency Recirc mode of operation, then the running train can be shifted

to Isolation Mode.

B. Incorrect. 1st part is correct, see A above. 2nd part is incorrect but plausible, because when Emergency Recirculation is initiated both trains of Containment Ventilation will be running. However, only one train is required and the system shall be shifted to Isolation Mode with one train in operation per SOP-802, section 5.3.6.

C. Incorrect. 1st part is incorrect but plausible, if believed that Emergency Recirculation is generated automatically from a smoke alarm, however, this is a manual initiation. 2nd part is incorrect but plausible, see B above.

D. Incorrect. 1st part is incorrect but plausible, see C above. 2nd part is correct, see A above.

Technical Reference(s):	SOP-802, Rev 13 PCN-3 LO21SYSHV1 Study Guide, Rev 5/7/11	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Control Room Ventilation System

Question Source:
Bank # 2012 NRC Exam Q61
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehensive or Analysis

10CFR Part 55 Content:
55.41 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	1	
		Group	2	
Level of Difficulty:	2.5	**K/A**	069G2.2.39	
		Importance Rating	3.9	

K/A Statement: Loss of containment Integrity: Knowledge of less than or equal to one-hour Technical specification action statements for systems.

Question 52

Which of the following TS LCOs requires Condition A to be completed within 1 hour (restore to operable status)?

A. 3.6.1 Containment (Containment Inoperable)

B. 3.6.4 Containment Pressure (Containment pressure not within limits)

C. 3.6.5 Containment Air Temperature (Containment average air temperature not within limit)

D. 3.6.6 Containment Spray System (One containment spray train inoperable)

Answer: A

K/A Match: This question matches the KA by requiring knowledge of 1 hour TS requirements related to loss of Containment integrity.

Explanation:

A. Correct: IAW TS 3.6.1, Condition A is a 1 hour TS that requires entry into Condition B if not met.

B. Incorrect: Incorrect because Condition A has an 8 hour time limit. It is plausible because an elevated pressure would reduce the containments ability to mitigate the pressure rise during an event.

C. Incorrect: Incorrect because Condition A has an 8 hour time limit. It is plausible because an elevated temperature would reduce the containments ability to mitigate the pressure rise during an event.

D. Incorrect: Incorrect because if 1 CS train is inoperable, Condition A has a 72-hour requirement. It is plausible to think that an entire train of ECCS equipment would have a 1-hour requirement because of its safety function.

Technical Reference(s): TS LCO 3.6.1, Amend 150
TS LCO 3.6.4, Amend 156

Attached w/ Revision # See Comments / Reference

TS LCO 3.6.5, Amend 156
TS LCO 3.6.6, Amend 156

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the completion time convention used in Technical Specifications (LO21RLSSL2OB03)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7,10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	1	
		Group	2	
Level of Difficulty:	3	**K/A**	074EA1.16	
		Importance Rating	4.4	

K/A Statement: Inadequate Core Cooling: Ability to operate and monitor the following as they apply to Inadequate Core Cooling: RCS in-core thermocouple indicators.

Question 53

Unit 2 plant conditions:

- Reactor Trip and SI has occurred
- RCS pressure is 50 psig
- Crew has transitioned from EOP-0.0B, Reactor Trip or Safety Injection to EOP-1.0B, Loss of Reactor or Secondary Coolant
- EOP-0.0B, Attachment 2, Safety Injection Actuation Alignment is complete

Which of the following would require entry into FRC-0.2B, Response to Degraded Core Cooling?

RVLIS 11" Above Plate light ____(1)____ AND CET temperature of ____(2)____.

A. (1) LIT
   (2) 1150°F

B. (1) DARK
   (2) 1150°F

C. (1) LIT
   (2) 1250°F

D. (1) DARK
   (2) 1250°F

Answer: B

K/A Match: This question matches the KA by requiring the ability to monitor the CETCs and use them to make procedure decisions.

Explanation:

A. Incorrect. Plausible because CET temperature is above 750°F, which is the required temperature for entry in to FRC-0.2B, however, the RVLIS 11" Above Plate light is LIT, therefore, in this case the correct procedure is FRC-0.3B or the ERG in effect.

B. Correct. With RCS Subcooling Margin at 10°F, the RVLIS 11" Above Plate light Dark, and CET temperature greater than 750°F, then the correct procedure is FRC-0.2B.

C. Incorrect. Plausible because CET temperature is above 750°F, which is the required temperature for entry in to FRC-0.2B, however, in this case with CET temperature above 1200°F the correct procedure is FRC-0.1B.

D. Incorrect. Plausible because CET temperature is above 750°F and the RVLIS 11" Above Plate light is DARK, which are the required temperature and RVLIS indications for entry in to FRC-0.2B, however, in this case with CET temperature above 1200°F the correct procedure is FRC-0.1B.

Technical Reference(s): FRC-0.2B, Rev 9 PCN- Attached w/ Revision # See Comments / Reference

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 FRC-0.2.

Question Source: **Bank #**
Modified Bank # ILOT1004 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	1	
		Group	2	
Level of Difficulty:	3	**K/A**	W E13EA1.1	
		Importance Rating	3.1	

K/A Statement: Steam Generator Overpressure: Ability to operate and/or monitor the following as they apply to Steam Generator Overpressure: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and manual and automatic features.

Question 54

Unit 1 crew has entered FRH-0.2A, Response to Steam Generator Overpressure.

In accordance with FRH-0.2A, an "affected" SG has a MAXIMUM of _____ MSSVs open. (Disregard any Accumulation)

A. 1

B. 3

C. 4

D. 5

Answer: D

K/A Match: This question matches the KA by requiring the ability to monitor the automatic features of the Steam Generator Overpressure protection system and how a steam generator overpressure condition relates to the associated ERG.

Explanation:

A. Incorrect. Incorrect but plausible because 1 MSSV would be open, however, it is not the MAXIMUM number of MSSVs open. The MAXIMUM number open would be 5 MSSVs at 1235 psig. An "affected" SG is any SG whose pressure is greater than 1235 psig per FRH-0.2A.

B. Incorrect. Incorrect but plausible because 3 MSSV would be open, however, it is not the MAXIMUM number of MSSVs open. The MAXIMUM number open would be 5 MSSVs at 1235 psig. An "affected" SG is any SG whose pressure is greater than 1235 psig per FRH-0.2A.

C. Incorrect. Incorrect but plausible because 4 MSSV would be open, however, it is not the MAXIMUM number of MSSVs open. The MAXIMUM number open would be 5 MSSVs at 1235 psig. An "affected" SG is any SG whose pressure is greater than 1235 psig per FRH-0.2A.

D. Correct. Correct the MAXIMUM number of MSSVs open at 1235 psig would be 5. An "affected" SG is any SG whose pressure is greater than 1235 psig per FRH-0.2A.

Technical Reference(s): FRH-0.2A, Rev 9, PCN-
Main Steam Study Guide, Rev 6/9/11
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Main Steam System and **PREDICT** the system response.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	1	
		Group	2	
Level of Difficulty:	3	**K/A**	W E15G2.4.6	
		Importance Rating	3.7	

K/A Statement: Containment Flooding: Knowledge of EOP mitigation strategies.

Question 55

Unit 1 crew is performing actions of FRZ-0.2A, Response to Containment Flooding.

In accordance with FRZ-0.2A, water from _____, in the Containment Sump, is considered an emergency source of stored water and NOT a source of flooding.

A. CVCS

B. RMUW

C. the CST

D. the RWST

Answer: D

K/A Match: This question matches the KA by requiring knowledge of EOP mitigation strategies (i.e. possible sources of flood water in the Containment Sump) for Containment Flooding.

Explanation:

A. Incorrect: Per FRZ-0.2A, Step 1, CVCS is listed as a possible source of flooding in containment. The bases for this step states the operator should try and identify unexpected sources of water in containment sump other than the stored emergency sources.

B. Incorrect: Per FRZ-0.2A, Step 1, RMUW is listed as a possible source of flooding in containment. The bases for this step states the operator should try and identify unexpected sources of water in containment sump other than the stored emergency sources.

C. Incorrect: Per FRZ-0.2A, Step 1, MFW and AFW are listed as possible sources of flooding in containment. These systems are supplied water from the CST, therefore, if water from the CST were in the Containment Sump then there must be a leak in the AFW system as MFW will isolate on P-4 with Lo Tave and would not be available as a leakage source. The bases for this step states the operator should try and identify unexpected sources of water in containment sump other than the stored emergency sources.

D. Correct: Per FRZ-0.2A, Step 1, the operator is identifying unexpected sources of water in the Containment Sump other than the stored emergency sources. The RWST is not listed as a possible unexpected water source. In the bases the RWST and the SI Accumulators are

listed as examples of emergency sources of water not considered as flooding that may be in the Containment Sump.

Technical Reference(s): FRZ-0.2A, Rev 9 PCN- Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the operational implications associated with all cautions, notes and actions of the "Response to Containment Flooding" procedure in accordance with FRZ-0.2A/B.

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

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	001K3.02	
		Importance Rating	3.4	

K/A Statement: Control Rod Drive: Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: RCS.

Question 56

Scenario 1
Unit 1 plant conditions:

- 100% power
- Middle of Life
- [Boron] is 772 ppm

Scenario 2
Unit 1 plant conditions:

- 100% power
- End of Life
- [Boron] is 6 ppm

Based on the Scenario conditions provided above, complete the following statement regarding the effects of Control Rod H8 dropping to the bottom of the core with 1/1-RBSS, Control Rod Bank Select in MANUAL.

In BOTH scenarios T_{AVE} will ____(1)____, however, it will ____(2)____ in Scenario 2 than it will in Scenario 1.

A. (1) decrease
   (2) decrease less

B. (1) decrease
   (2) decrease more

C. (1) increase
   (2) increase less

D. (1) increase
   (2) increase more

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the effect that a control rod malfunction will have on the RCS (Tave).

Explanation:

A. Correct. 1st part is correct, Tave will decrease in both cases as the temperature coefficient of reactivity will be negative in both cases. Therefore, when the negative reactivity of the dropped rod is inserted into the core, positive reactivity must counteract that negative

reactivity in the form of a decrease in temperature. 2nd part is correct, in scenario 2 Tave will not drop as far as in Scenario 1 because at EOL the temperature coefficient of reactivity is more negative, therefore it requires less change in temperature to counteract the effects of the dropped rod.

B. Incorrect. 1st part is correct, see A above. 2nd part is incorrect but plausible, because as the core ages the radial flux profile of the core changes over core life, therefore, if the radial flux profile moved to the center of the core then the rod worth of the center rod would increase and more negative reactivity would have been inserted in the core on the rod drop, thus requiring Tave to decrease more at EOL to counteract the negative reactivity inserted into the core.

C. Incorrect. 1st part is incorrect, in both scenarios Tave will decrease. 1st part is plausible because the examinee must know that the temperature coefficient of reactivity is negative at both MOL and EOL. If the examinee thinks the temperature coefficient of reactivity is positive, then temperature would rise in order to offset the negative reactivity of the rod dropping to the bottom of the core. 2nd part is incorrect but plausible, because the examinee must know how the magnitude of the temperature coefficient changes over core life, therefore, if the examinee believes the temperature coefficient of reactivity is positive and that the magnitude of temperature coefficient of reactivity gets smaller over core life then the rise in temperature to counteract the negative reactivity of the dropped rod would be smaller at EOL than MOL, thus it would "increase less."

D. Incorrect. 1st part is incorrect but plausible, see C above. 2nd part is incorrect but plausible, because the examinee must know how the magnitude of the temperature coefficient changes over core life, therefore, if the examinee believes the temperature coefficient of reactivity is positive and that the magnitude of temperature coefficient of reactivity gets larger over core life then the rise in temperature to counteract the negative reactivity of the dropped rod would be larger at EOL than MOL, thus it would "increase more."

Technical Reference(s):	GFES, Reactivity Coefficients Student Guide, Rev 3	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** how the magnitude of the moderator temperature coefficient varies with changes in the following parameters: Moderator Temperature, Boron Concentration, Control Rod position, and Core age.

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

10CFR Part 55 Content: **55.41** 2
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	002K4.03	
		Importance Rating	2.9	

K/A Statement: Reactor Coolant: Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: Venting the RCS.

Question 57

Unit 1 plant conditions:

- FRC-0.1A, Response to Inadequate Core Cooling in progress
- After 1-HS-3607 and 1-HS-3608 are placed in the CLOSE position, the following conditions are observed:
  - 1-HS-3607, RV HEAD VENT VLV – GREEN light LIT and RED light DARK
  - 1-HS-3608, RV HEAD VENT VLV – GREEN light DARK and RED light LIT

Based on the conditions above, complete the statements below.

1) To originally OPEN BOTH of the valves simultaneously, ONE KEY was required for ____(1)____ valve(s).

2) After the handswitches are placed in CLOSE, flow from the Reactor Vessel Head Vent Valves to containment ____(2)____ isolated.

A. (1) EACH
   (2) is

B. (1) EACH
   (2) is NOT

C. (1) BOTH
   (2) is

D. (1) BOTH
   (2) is NOT

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the design features of the RCS Vessel Head Vent Valves**.**

Explanation:

A. Correct. Part 1 is correct as a Key is required to OPEN these valves and when the key is placed in the Key switch and turned to the “OPEN” position the key is now captured. To un-capture the key the key switch must be turned to the “CLOSE” position. Therefore, ONE Key

is required for EACH of the two valves. Part 2 is correct in that the Reactor Vessel Head vent valves are in series and therefore either valve indicating closed would isolate the flow path.

B. Incorrect. Part 1 is correct as described in 'A' above. Part 2 is incorrect but plausible as described in 'D' below.

C. Incorrect. Part 1 is incorrect but plausible if the applicant does not realize that when the key switch is placed in the "OPEN" position the key is now captured. If it were not captured the operator could open the first valve and use the same key to open the second valve. Part 2 is correct as described in 'A' above.

D. Incorrect. Part 1 is incorrect but plausible as described in 'C' above. Part 2 is incorrect but plausible if believed that the Reactor Vessel Head vent valves were parallel flow paths as opposed to in series on a single vent path.

Technical Reference(s): FRC-0.1A, Rev 9 PCN-M1-0250, Rev CP34
E1-0064, Sh. 40, Rev CP5
E1-0064, Sh. 41, Rev CP5
Control Board Pictures

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective:

Question Source: **Bank #**
Modified Bank # 2016 NRC Exam Q6 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	2	**K/A**	011A2.02	
		Importance Rating	3.2	

K/A Statement: Pressurizer Level Control: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Excessive Charging.

Question 58

Unit 2 plant conditions:

- 100% power
- CCP 2-01 is operating
- Instrument Air is lost to 2-FCV-121 CENTRIFUGAL CHARGING PUMP FLOW CONTROL VALVE

Based on conditions above, complete statement below.

1) Charging flow indication in the Control Room will ____(1)____.

2) In accordance with ABN-105, Chemical and Volume Control System Malfunction, 2-HCV-182, RCP SEAL WATER PRESSURE CONTROL VALVE, will be ____(2)____ in order to maintain RCP seal water injection flow between 6-13 gpm.

A. (1) increase
(2) throttled open

B. (1) increase
(2) throttled closed

C. (1) decrease to 55 gpm
(2) throttled open

D. (1) decreased to 55 gpm
(2) throttled closed

Answer: A

K/A Match: This question matches the KA by requiring the ability to predict the impact of a failed controller (excessive charging) and actions required to mitigate it per the correct ABN.

Explanation:

A. Correct. 1st part is correct, per ABN-301 if instrument air is lost to 1-FCV-0121 the valve will fail to the OPEN position; when the valve fails OPEN then charging flow indicated in the

control room will increase. 2nd part is correct, 1-HCV-0182 is used to maintain the correct back flow of charging water to the RCP seals, therefore, it must be throttled open to lower RCP seal flow to within the required band of 6-13 gpm even though overall charging flow has increased due to the failure 1-FCV-0121 in the open position.

B. 1st part is correct, see A above. 2nd part is incorrect because when 2-FCV-121 fails open, flow to the seals will be excessive. With the split injection header, 2-HCV-0182 will have to be opened to reduce seal injection flow. It is plausible because if the RCP Seal Pressure Control valve were on the same piping line as the RCP seals, it would be correct.

C. 1st part is incorrect because 2-FCV-121 will fail open on a loss of instrument air. It is plausible because when the controller has power and is in AUTO, it has a "limiter" which does not allow flow to decrease below 55 gpm. 2nd part is correct, see A above.

D. 1st part is incorrect but plausible, see C above. 2nd part is incorrect but plausible, see B above.

Technical Reference(s):	CVCS Study Guide ABN-105, Rev 8 PCN-1 CVCS Big Book Drawing ABN-301, Rev 13 PCN-4	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam:

Learning Objective: **DESCRIBE** the components of eh Chemical and Volume Control system including interrelations with the other systems to include interlocks and control loops (LO21SYSCS1OB03)

Question Source:	**Bank #**	2014 NRC Retake Exam Q59	
	Modified Bank #		(note changes or attach parent)
	New		
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	5	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	015K2.01	
		Importance Rating	3.3	

K/A Statement: Nuclear Instrumentation: Knowledge of bus power supplies to the following: NIS channels, component, and interconnections.

Question 59

On Unit 2 while recording Critical Data at $1x10^{-8}$ amps, a fault causes a loss of 118 VAC Protection Bus 2PC3

Which of the following is the expected NI plant response?

A. Bistables for N-35 will trip, Reactor will trip.

B. Bistables for N-43 will trip, Reactor will NOT trip.

C. Bistables for N-42 will trip, Reactor will NOT trip.

D. P-6 block is removed re-energizing SR NIs, Reactor will trip.

Answer: B

K/A Match: This question matches the KA by requiring knowledge of the power supplies to Nuclear Instrumentation.

Explanation:

A. Incorrect. Plausible because Critical Data is taken at 1X10-8 amps where the IR High Flux Trip is still active and a loss of power to an IR Channel would cause a trip, however, IR channel N-35 is powered from 2PC1and no IR channel will be lost.

B. Correct. Power Range N-43 is powered from 2PC3 and bistables will trip on loss of power. Since the coincidence for a Reactor Trip is 2 of 4 channels, the Reactor will NOT trip.

C. Incorrect. Plausible because bistables for Power Range N-42 will trip upon a loss of power, however, N-42 is powered from 2PC2 and will not lose power.

D. Incorrect. Plausible because removal of P-6 will re-energize SR NIs and since Critical Data is taken at a power above the SR High Flux setpoint, a Rx Trip would occur, however, the P-6 permissive signal is generated from the IR channels (1 of 2 > 10^{-10} amps). It is automatically removed when 2 of 2 IR channels are < 10^{-10} amps. Since both IR channels remain powered, P-6 remains active.

Technical Reference(s): ABN-603, Rev 8 PCN-7
ABN-701, Rev 12 PCN-1
Immediate Operator Actions List

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Excore Instrumentation system and **PREDICT** the system response.

Question Source:
Bank #
Modified Bank # 2012 NRC Exam Q30 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 7
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	017K6.01	
		Importance Rating	2.7	

K/A Statement: In Core Temperature Monitor: Knowledge of the effect that a loss or malfunction of the following will have on the ITM: Sensors and detectors.

Question 60

Which of the following combinations of CETs, per core quadrant, meets the MINIMUM requirements per OPT-112A-1, Accident Monitoring Instrumentation Check?

A. 2 Train A and 2 Train B

B. 2 Train A or 2 Train B

C. 4 Train A and 4 Train B

D. 4 Train A or 4 Train B

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the effect of failed CETs on the Tech Specs associated with the ITM system.

Explanation:

A. Correct. OPT-112A-1 surveillance requires 2 CHANNELS with 2 CETs per quadrant.

B. Incorrect. Plausible because the CHANNELS required is correct, however, both Trains must be in service.

C. Incorrect. Plausible because both Trains must be in service, however, only 2 CETs each per quadrant are required.

D. Incorrect. Plausible because 4 CETs are required per core quadrant, however, not 4 per Train A and 4 per Train B in each core quadrant.

Technical Reference(s): OPT-112A Form 1, Rev 20 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** the performance and design attributes of the following Core Cooling

Monitor System / RVLIS components, flowpaths and features.
-Core Exit Thermocouples (OP51SYSRC3OB02)

Question Source:	**Bank #**	2011 NRC Exam Q34	
	Modified Bank #		(note changes or attach parent)
	New		
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		X
	Comprehensive or Analysis		
10CFR Part 55 Content:	**55.41**	2	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	4	**K/A**	041K5.05	
		Importance Rating	2.6	

K/A Statement: Steam Dump/Turbine Bypass Control: Knowledge of the operational implications of the following concepts as they apply to SDS: Basis for RCS pressure design limits.

Question 61

In accordance with the Bases for the RCS Pressure Safety Limit, IF the _____ were to fail to operate as designed, THEN the RCS would still be protected from overpressurization by operation of the RCS High Pressure Trip.

A. Steam Dump System

B. Pressurizer Safety Valves

C. Main Steam Safety Valves

D. Pressurizer Pressure Control System

Answer: A

K/A Match: This question matches the KA by requiring knowledge of how the Steam Dumps are not utilized in the analysis for protection of RCS pressure design limits.

Explanation:

A. Correct: Per the Technical Specification Bases for the Safety Analyses of the Reactor High Pressure Trip, the setpoint is specifically set to provide protection against overpressurization. The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices. More specifically, no credit is taken for operation of the following: Pressurizer power operated relief valves (PORVs); Steam Generator Atmospheric Relief Valves; Steam Dump System; Reactor Control System; Pressurizer Level Control System; or Pressurizer spray valves.

B. Incorrect: Plausible because most analyses assume the single worst active component has failed, it is reasonable to assume the PRZR Safety Valves would be excluded from the analysis as an active component failure.

C. Incorrect: Plausible since the MSSVs are NOT a part of the RCS and are NOT included in the RCS pressure boundary, it is reasonable to assume MSSVs would be excluded from the analysis.

D. Incorrect: Plausible because the PRZR level control system and the PORVs are excluded from the analysis, it is reasonable to believe the PRZR pressure control system would also be excluded from the analysis.

DELETED FROM EXAM

Technical Reference(s): TS LCO 2.1.2 Bases, Rev 51 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** the function of the Steam Dump System in accordance with DBD-EE-01, Rector Protection and NSSS Related Control Systems. LO21SUSSD1OB01)

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 5
55.43

DELETED FROM EXAM

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	2	
		Group	2	
Level of Difficulty:	2	**K/A**	045A3.05	
		Importance Rating	2.6	

K/A Statement: Main Turbine Generator: Ability to monitor automatic operation of the MT/G system, including: Electrohydraulic control.

Question 62

Unit 1 at 100% power

What is the status of the LP Turbine Control Valves and the HP Turbine Control Valves?

	LP Turbine Control Valves	HP Turbine Control Valves
A.	FULLY OPEN	FULLY OPEN
B.	NOT FULLY OPEN	FULLY OPEN
C.	FULLY OPEN	NOT FULLY OPEN
D.	NOT FULLY OPEN	NOT FULLY OPEN

Answer: C

K/A Match: This question matches the KA by requiring the ability to monitor automatic operation of the MT/G system in relation to electro-hydraulic control.

Explanation:

A. Incorrect. Plausible if thought that the HP Turbine Control Valves were 100% open at 100% power.

B. Incorrect. Plausible if thought that the HP Turbine Control Valves were 100% open at 100% power.

C. Correct. At 100% power the LP Turbine Control Valves are 100% open and the HP Turbine Control Valves are throttled to maintain turbine load.

D. Incorrect. Plausible if thought that the LP Turbine Control Valves were NOT 100% open at 100% power.

Technical Reference(s):	LO21SYSMT1 Study Guide, Rev 5/4/11	Attached w/ Revision # See Comments / Reference

Unit 1 Benchmark Graph

References to be provided to applicants during exam: None

Learning Objective: **LIST** and **DESCRIBE** the purpose of in-plant and Control Room System controls, indications and alarms for the Main Turbine components.

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**	5	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	055K1.06	
		Importance Rating	2.6	

K/A Statement: Condenser Air Removal: Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: PRM system.

Question 63

Unit 1 plant conditions:

- SG tube leak in progress
- 1-RE-2959 (COG-182), CONDENSER OFF GAS Radiation Monitor in service
- Two CEV Pumps in service
- Rapid plant shutdown in progress

Based on the conditions above, complete the statement below.

Fission products from the condenser are monitored by 1-RE-2959, located in a bypass on the common CEV pump ____(1)____ line, before being transmitted to the ____(2)____.

A. (1) exhaust
   (2) Primary Plant Ventilation System

B. (1) suction
   (2) Primary Plant Ventilation System

C. (1) exhaust
   (2) Waste Gas System

D. (1) suction
   (2) Waste Gas System

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the physical connection between the CARS and PRM system.

Explanation:

A. Correct. First part is correct, a bypass line from the CEV pump common discharge header goes to the Process Monitor. Once sampled at the monitor the bypass line rejoins the common discharge header. Second part is correct, the Primary Plant Ventilation System accepts the flow for a filtered and monitored discharge to the environment.

B. Incorrect. First part is incorrect but plausible because the CEV pump common suction line would also monitor the incoming streams from all condensers and it would also allow the flow to be monitored prior to entry into the CEV pumps themselves to alert the unit of possible CEV pump contamination from the fission products. Second part is correct, see A above.

C. Incorrect. First part is correct, see A above. Second part is incorrect but plausible because the Waste Gas System is used for storage and later discharge of radioactive waste gasses at the site.

D. Incorrect. First part is incorrect but plausible, see B above. Second part is incorrect but plausible, see C above.

Technical Reference(s): LO21SYSCV1 Study Guide, Rev 3/15/17 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause effect relationships between the Condenser Vacuum and Water Box Priming System and the following systems, components or events:
Main Turbine.

Question Source:
Bank #
Modified Bank # 2013 NRC Exam Q37 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 11
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	2	
		Group	2	
Level of Difficulty:	4	**K/A**	072A4.02	
		Importance Rating	2.5*	

K/A Statement: Area Radiation Monitoring: Ability to manually operate and/or monitor in the control room. Major Components:

Question 64

Unit 1 plant conditions:

- Mode 6
- A Containment Purge is in progress
- Cavity fill is in progress per SOP-102A, Residual Heat Removal System Operation
- 1-RE-5502, Containment Particulate Radiation Monitor, is in HI alarm
- 1-RE-5503, Containment Gas Radiation Monitor, has a stable trend
- X-RE-5567A, Plant Vent Gas Monitor, has a stable trend
- X-RE-5570A, Wide Range Gas Monitor Low, has a stable trend
- NEO reports local Refueling Cavity Area Radiation Monitors, 1-RE-6251 and 1-RE-6253, are alarming

Of the following actions, which action is taken FIRST for the conditions listed above, in accordance with ABN-902, Release of Radioactive/Toxic Gas?

A. Secure Cavity Fill.

B. Evacuate Containment.

C. Verify Containment Vent Isolation complete.

D. Place Containment Pre-Access Filtration Units in service.

Answer: C

K/A Match: This question matches the KA by requiring the ability to monitor major components affected by the area radiation monitors.

Explanation:

A. Incorrect. Plausible because could be thought that the cause was the cavity fill and stopping the fill rate would reduce the particulate activity seen by the monitor, however, while stopping the fill would be prudent, an automatic containment isolation has occurred on containment particulate high and must be verified complete.

B. Incorrect. Plausible because an automatic Containment Isolation has occurred on

Containment Particulate High alarm and containment should be evacuated, however, ABN-902 requires a verification of CVI first

C. Correct. Automatic Containment Isolation occurred when 1-RE-5502 went in to High alarm and the first step of ABN-902 requires a verification that CVI is complete.

D. Incorrect. Plausible because starting the Pre-Access Filtration Units is required later by procedure, however, first the automatic Containment Isolation should be verified complete.

Technical Reference(s): ABN-902, Rev 7 PCN-
ABN-908, Rev 5 PCN-2
ABN-909, Rev 8 PCN-8

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Given conditions that warrant a radioactive effluent release, **EVALUATE** and **DETERMINE** the proper methodology for review and authorization of the release permit.

Question Source: **Bank**
Modified Bank # 2011 NRC Exam Q72 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 11
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	2	
		Group	2	
Level of Difficulty:	3	**K/A**	079G2.2.44	
		Importance Rating	4.2	

K/A Statement: Station Air: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question 65

Unit 1 plant conditions:

- 30% power and ramping to 100% after a refueling outage
- A Containment entry is required to tighten a pipe hanger
- The worker will be using an air wrench to complete the task
- The BOP operator must open the valve indicated below to support the use of an air wrench in Containment

Based on the above plant conditions, complete the following statements:

1) On the handswitch (HS) above, the arrow above the word “OPEN” means ____(1)____.

Subsequently, an operator caused an inadvertent SI.

2) Service Air to Containment will be lost due to a ____(2)____.

A. (1) HS will spring return to “AUTO” position when released
(2) Containment Isolation Phase ‘A’ signal

B. (1) HS must be held in “OPEN” position until indicates open or valve will return closed
(2) Containment Isolation Phase ‘A’ signal

C. (1) HS will spring return to “AUTO” position when released
(2) loss of IA to Containment causing 1-HV-3486 to fail closed

D. (1) HS must be held in “OPEN” position until indicates open or valve will return closed
(2) loss of IA to Containment causing 1-HV-3486 to fail closed

Answer: A

K/A Match: This question matches the KA by requiring knowledge of Control Room indications to verify operation of a Service Air system component including how the operator’s actions to manipulate the component may subsequently affect plant conditions.

Explanation:

A. Correct: Part 1 is correct per the station design drawing E1-0048, Sheet 4, the handswitch is a 3 position, spring-return to AUTO handle. Part 2 is correct per the Service Air System Study Guide this valve will isolate Service Air to Containment in the event a Containment Isolation Phase A signal is received.

B. Incorrect: 1st part is incorrect, but plausible because this valve must be held in the OPEN position until the valve’s open limit switch contact is made-up. If the valve handswitch is released prior to the valve in the full open position the valve will travel back closed. 2nd part is correct, see A above.

C. Incorrect: 1st part is correct, see A above. 2nd part is incorrect but plausible because this valve’s control air is supplied by Instrument Air and it would fail to the closed position if Instrument Air to the valve was lost, however, this valve is located outside containment and Instrument Air to the valve would NOT be lost on a Containment Phase A Isolation signal because on a Phase A only Instrument Air to Containment is Isolated.

D. Incorrect: 1st part is incorrect but plausible, see B above. 2nd part is incorrect but plausible, see C above.

Technical Reference(s): Service Air System Study Guide Rev. 1/21/09
E1-0048, Sheet 4 Rev. CP-6
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** how the following signals affect the Service Air System: SI, BO, Containment Isolation Phase A, Increasing/Decreasing Air Pressure, Loss of Power to the Elektronikon Regulator, Compressor Shutdown Warning, Compressor Shutdown Signal, Emergency Shutdown.

Question Source:

Bank #		
Modified Bank #		(note changes or attach parent)
New	X	

Question History:	**Last NRC Exam**	
Question Cognitive Level:	**Memory or Fundamental Knowledge**	
	Comprehensive or Analysis	X
10CFR Part 55 Content:	**55.41** 7	
	55.43	

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	3	
		Group		
Level of Difficulty:	2	**K/A**	2.1.1	
		Importance Rating	3.8	

K/A Statement: Knowledge of conduct of operations requirements.

Question 66

Of the following personnel who desire access to the "At the Controls" area of the control room, the _____ requires specific approval from a member of the currently assigned control room staff.

A. Duty Manager

B. Site Vice President

C. NEOs assigned to the shift

D. PROMPT Team Supervisor

Answer: D

K/A Match: This question matches the KA by requiring knowledge of Conduct of Operations requirements.

Explanation:

A. Incorrect. Plausible to believe that someone, even if it is the Duty Manager, would require permission especially carrying a beverage, but this situation is not prohibited by STA-616.

B. Incorrect. Plausible since some facilities do require anyone not on the immediate operating staff to obtain permission to enter the At the Controls area.

C. Incorrect. Plausible to believe that non-licensed operations personnel would still need to request permission, since they are not licensed individuals; but STA-616 excludes all operations personnel assigned to the control room from the requirement.

D. Correct. Maintenance and engineering personnel, including managers, are not exempt from having to obtain permission to enter the At the Controls area, per STA-616. The PROMPT Team Supervisor is plausible as they work rotating shifts with the same Operations crew. It is plausible that they would be granted additional access.

Technical Reference(s):	STA-616, Rev 8 PCN-4 MDA-101, Rev 9 PCN-	Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** requirements for Conduct of Operations in accordance with OA-102 and Operations Guideline 3 (LO21ADMXA3OB101)

Question Source:
Bank # 2015 NRC Exam Q68
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev: 4		**Tier**	3	
		Group		
Level of Difficulty:	3	**K/A**	2.1.9	
		Importance Rating	2.9*	

K/A Statement: Ability to direct personnel activities inside the control room.

Question 67

A non-licensed operator is enrolled in the ILOT Program and has successfully completed the Systems portion of classroom training.

- Trainee is under instruction of the Unit 2 RO
- Trainee requests to perform a required RCS dilution for training

What is the LOWEST level of authority that can approve the trainee performing the dilution?

A. Shift Operations Manager

B. Reactor Operator

C. Unit Supervisor

D. Shift Manager

Answer: D

K/A Match: This question matches the KA by requiring knowledge of actions directed inside the control room and who can approve those actions.

Explanation:

A. Incorrect: Plausible because per ODA-102, the SOM must approve a trainee performing a Reactor Startup as the Reactor Operator.

B. Incorrect: Plausible because the trainee is under instruction of the RO, if the trainee assumes he has the same authority as the RO (with direct approval of RO for any operation) then this could be thought to be correct.

C. Incorrect: Plausible because the US directs crew operations for the Unit in which he/she is responsible for, therefore could be thought to be correct.

D. Correct: Per ODA-102, Shift Manager approval is required for manipulation of controls by a trainee and the trainee should be directly supervised by a licensed operator. The SM must approve the use of a trainee in any evolution which may affect reactivity.

Technical Reference(s): ODA-102, Rev 27 PCN-5
Operations Admin LP, 7/29/15

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the reactor controls in a manner consistent with reactivity management expectations and station procedures.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	3	
		Group		
Level of Difficulty:	3	**K/A**	2.1.37	
		Importance Rating	4.3	

K/A Statement: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Question 68

During a reactor startup in the Source Range with Reactivity SRO assigned, complete the following statements in accordance with ODA-102, Conduct of Operations.

1) Unit Supervisor ____(1)____ required to be in proximity of the Reactor Operator.

2) Reactivity SRO ____(2)____ allowed to provide peer checks for reactivity manipulations.

A. (1) is NOT
(2) is NOT

B. (1) is NOT
(2) is

C. (1) is
(2) is NOT

D. (1) is
(2) is

Answer: B

K/A Match: This question matches the KA by requiring knowledge of guidelines associated with reactivity management.

Explanation:

A. Incorrect: First part is correct as ODA-102, Attachment 8.B specifies that if a Reactivity SRO has been assigned, the Unit Supervisor is NOT required to be positioned in proximity of the Reactor Operator. Second part is incorrect but plausible because even though ODA-102 allows the “Reactivity SRO” to provide peer checks for reactivity manipulations it also states that peer checks by the “Reactivity SRO” should not routinely be used.

B. Correct: First part is correct, see A above. Second part is correct as ODA-102; Attachment 8.B states the “Reactivity SRO” may perform peer checks on reactivity manipulations.

C. Incorrect: First part is incorrect but plausible as the Reactivity SRO is assigned to assist the Unit Supervisor. The Unit Supervisor does not relinquish responsibility of the unit to the

Reactivity SRO and it is therefore plausible that the Unit Supervisor is to remain in the proximity of the Reactor Operator. Second part is incorrect but plausible, see A above.

D. Incorrect: First part is incorrect but plausible, see C above. Second part is correct, see B above.

Technical Reference(s): ODA-102, Rev. 27 PCN-5
OPGD-3, Att 2, Rev. 01/10/13
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** what steps are taken at CPNPP to avoid the contributing factor for the reactivity events cited in the industry experience covered in this lesson.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**	3	
		Group		
Level of Difficulty:	2	**K/A**	2.2.13	
		Importance Rating	4.1	

K/A Statement: Knowledge of tagging and clearance procedures.

Question 69

As part of a clearance, several normally sealed throttled valves were closed. During restoration, a _____ seal should be reapplied to these valves.

A. Red

B. Green

C. Blue

D. Yellow

Answer: C

K/A Match: This question matches the KA by requiring knowledge of tagging procedures.

Explanation:

A. Incorrect. Plausible as this is the color used to identify normally sealed open or off positions.

B. Incorrect. Plausible as this is the color used to identify normally sealed closed positions.

C. Correct. This is the color used to identify normally sealed throttled positions.

D. Incorrect. Plausible as this is the color used for personal safety.

Technical Reference(s): ODA-403, Rev 8 PCN-2 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Remove plant equipment from service as required for maintenance (OPD1ADMXA1OB09).

Question Source: **Bank #** 2016 NRC Exam Q69
Modified Bank # (note changes or attach parent)

	New		
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		X
	Comprehensive or Analysis		
10CFR Part 55 Content:	**55.41**	10	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	3	
		Group		
Level of Difficulty:	3	**K/A**	2.2.14	
		Importance Rating	3.9	

K/A Statement: Knowledge of the process for controlling equipment configuration or status.

Question 70

In accordance with ODA-410, System Status Control, complete the following statements.

1) When an ODA-410-3, Component Out of Position Log, is initiated, a Control Room operator is required to perform an Open Narrative Log entry in the ____(1)____ Log.

2) ____(2)____ authorization is required to maintain the valve out of position, per the ODA-410-3 form, past the end of shift turnover.

A. (1) Station
   (2) Shift Ops Manager's

B. (1) Unit
   (2) Shift Ops Manager's

C. (1) Unit
   (2) Shift Manager's

D. (1) Station
   (2) Shift Manager's

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the process for controlling equipment status.

Explanation:

A. Incorrect. First part is incorrect but plausible because the ODA-410-3 Form does not specify which Log to make the entry and the question does not specify a unit. Also, the Station Log is a Log where an open narrative log entry could be made. Second part is incorrect but plausible because the Shift Operations Manager is responsible for coordinating operating activities in accordance with ODA-102.

B. Incorrect. First part is correct, in accordance with the ODA-410-3 Form an Open Narrative Log Entry is required when the valve is closed and out of its normal position to remind crew personnel of the out of position component. A unit Log entry would be made vice a station log entry because station log entries are required for major operations that are pertinent to the station. Second part is incorrect but plausible, see A above.

C. Correct. First part is correct, see B above. Second part is correct, per ODA-410, a shift manager clearance is required to be place on the component out of position before the end of shift, however, this requirement may be waived by the shift manager on a case by case basis (shall not be delegated). Therefore, in this case the shift manager will have to waive the requirement for the shift manager clearance before the end of shift because it will not be complete in time.

D. Incorrect. First part is incorrect but plausible, see A above. Second part is correct, see C above.

Technical Reference(s): ODA-410, Rev 16 PCN-4
ODA-410-3 Form, Rev 3
ODA-104, Rev 14 PCN-9
ODA-102, Rev 27 PCN-5

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **INTEPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines and policies.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	3	
		Group		
Level of Difficulty:	2.5	**K/A**	2.2.21	
		Importance Rating	2.9	

K/A Statement: Knowledge of pre and post maintenance operability requirements.

Question 71

In accordance with STA-606, Control of Maintenance and Work Activities, complete the following statements.

1) A test performed to verify the operability of an SSC is called a(n) ____(1)____.

2) The Work Order utilized to verify the operability of an SSC is printed on ____(2)____ paper.

A. (1) Equipment Test
   (2) Pink

B. (1) Equipment Test
   (2) Blue

C. (1) Surveillance Test
   (2) Pink

D. (1) Surveillance Test
   (2) Blue

Answer: C

K/A Match: This question matches the KA by requiring knowledge of operability requirements associated with maintenance (testing).

Explanation:

A. Incorrect: First part is incorrect but plausible because per STA-606 Equipment Tests are performed to verify that work performed was satisfactory, however, it would not apply to technical specification related equipment. Second part is correct, per STI-606.01, Work Control Process, a Surveillance Work Order (SV) will be printed on pink paper to enhance identification of the work order ensuring expeditious reviews are performed.

B. Incorrect: First part is incorrect but plausible, see A above. Second part is incorrect but plausible because per STA-202 unit specific procedures should be designated by the use of colored paper; Unit 1 is on blue paper and Unit 2 is on yellow paper.

C. Correct. First part is correct, per STA-606, Surveillance Tests are performed to verify the Operability of technical specification related SSCs. These tests are performed to verify that surveillance requirements of CPNPP Technical Specifications, the TRM, Part 1 of the ODCM and the IST Plan remain satisfied. Second part is correct, see A above.

D. Incorrect: First part is correct, see C above. Second part is incorrect but plausible, see B above.

Technical Reference(s): STA-606, Rev 34 PCN-
STA-202, Rev 38 PCN-6
STI-606.01, Rev 6 PCN-

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DELINEATE** the role of surveillances into Operability in accordance with the Technical Specifications.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**	3	
		Group		
Level of Difficulty:	2	**K/A**	2.3.4	
		Importance Rating	3.2	

K/A Statement: Knowledge of radiation exposure limits under normal or emergency conditions.

Question 72

A 20-year-old CPNPP employee has 1000 mrem of TEDE exposure for 2019.

Which of the following is the MAXIMUM amount of additional TEDE exposure that may be received prior to exceeding 10CFR20 (NRC) exposure limits?

A. 2000 mrem

B. 3000 mrem

C. 4000 mrem

D. 5000 mrem

Answer: C

K/A Match: This question matches the KA by requiring knowledge of radiation exposure limits.

Explanation:

A. Incorrect. Incorrect but plausible because the new admin limit is 2000 mrem, therefore, if the applicant disregards previously received exposure and answers based on current admin limit this would be the correct answer.

B. Incorrect. Incorrect but plausible because the Admin limit used to be 4000 mrem, therefore, if 3000 mrem were added to the 1000 mrem already received this would be the maximum old admin limit.

C. Correct. Correct 10CFR20 limit is 5000 mrem per year therefore an additional 4000 mrem may be received.

D. Incorrect. Incorrect but plausible because 5000 mrem is the legal limit therefore if the examinee does not add the original 1000 mrem received from the stem of the question this would be the correct answer.

Technical Reference(s): STA-655, Rev 22 PCN-1 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines and policies.

Question Source:
Bank #
Modified Bank # 2013 NRC Exam Q70 (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41** 12
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	3	
		Group		
Level of Difficulty:	2	**K/A**	2.3.14	
		Importance Rating	3.4	

K/A Statement: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question 73

In accordance with STA-656, Radiation Work Control, …

1) if the Glen Rose fire department required access to the RCA, they ____(1)____ require an escort by a qualified Radiation Worker at all times.

2) the requirement to perform a whole body count prior to RCA entry may be waived by the ____(2)____ or designee, as appropriate.

A. (1) would
(2) RP Supervisor

B. (1) would NOT
(2) Shift Manager

C. (1) would NOT
(2) RP Supervisor

D. (1) would
(2) Shift Manager

Answer: A

K/A Match: This question matches the KA by requiring knowledge of procedural guidance on combating radiation/contamination hazards during an emergency condition.

Explanation:

A. Correct. First part is correct, per STA-656, during emergencies personnel from offsite agencies should be allowed immediate access to the RCA as Escorted Radiation Workers. Radiation Protection should provide a qualified Radiation Worker escort for emergency personnel entering the RCA. Second part is correct, per STA-656, The whole body count requirement prior to RCA entry should be waived by the RP Supervisor or designee, as appropriate.

B. Incorrect. First part is incorrect but plausible, during an emergency it is reasonable to believe that offsite responders combating a fire would be allowed access to the RCA without an escort. Second part is incorrect but plausible, it is reasonable to believe the Shift Manager

who becomes the Emergency Coordinator during an E-Plan event would have the authority to waive the requirement for a whole body frisk, however, per STA-656, the RP Supervisor (or designee) retains the authority to waive this requirement.

C. Incorrect. First part is incorrect but plausible, see B above. Second part is correct, see A above.

D. Incorrect. First part is correct, see A above. Second part is incorrect but plausible, see B above.

Technical Reference(s): STA-656, Rev 22 PCN-1 Attached w/ Revision # See Comments / Reference

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.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 12
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	3	
		Group		
Level of Difficulty:	2	**K/A**	2.4.14	
		Importance Rating	3.8	

K/A Statement: Knowledge of general guidelines for EOP usage.

Question 74

Complete the following statement regarding execution of the ERGs in accordance with ODA-407, Operations Department Procedure Use and Adherence.

During implementation of _____ status FRGs, continuous actions and foldout page items previously identified in the ORGs are required to be performed.

A. GREEN

B. YELLOW

C. ORANGE

D. RED

Answer: B

K/A Match: This question meets the KA by requiring knowledge of general guidelines for EOP usage.

Explanation:

A. Incorrect. Incorrect but plausible because GREEN status is used while monitoring CSF Status Trees and a green status would mean conditions have not degraded enough to execute a yellow condition procedure although if question misread could be thought as a correct answer since if Status Trees were green then all previously encountered ORG continuous action steps would still be valid until superseded by an alternate instruction or stated to be no longer applicable.

B. Correct. Correct, per ODA-407, when implementing a YELLOW status FRG, continuous actions and foldout page items of ORGs are monitored and implemented as required.

C. Incorrect. Incorrect but plausible because if a CSF ORANGE condition were being implemented, the ORG in effect is suspended; meaning continuous actions and foldout page items are suspended as well and no longer apply.

D. Incorrect. Incorrect but plausible because if a CSF RED condition were being implemented, the ORG in effect is suspended; meaning continuous actions and foldout page items are suspended as well and no longer apply.

Technical Reference(s): ODA-407, Rev 16 PCN-5 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **INTERPRET** and **ENSURE** Compliance with Plant Administrative and Operational Procedures, Guidelines, and Policies

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41** 10
55.43

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**	3	
		Group		
Level of Difficulty:	2	**K/A**	2.4.3	
		Importance Rating	3.7	

K/A Statement: Ability to identify post accident instrumentation.

Question 75

Post accident instrumentation required by Regulatory Guide 1.97 is identified by a(n) _____ label with white letters.

A. red

B. blue

C. black

D. orange

Answer: C

K/A Match: This question matches the KA by requiring the ability to identify post accident instrumentation.

Explanation:

A. Incorrect: Plausible because red would indicate importance and some fire related equipment is identified this way.

B. Incorrect: Plausible because some indications that are “controlling channel” indications are identified this way.

C. Correct: RG 1.97 instrumentation is identified by a black label with white lettering.

D. Incorrect: Plausible because some instrumentation (including post accident instrumentation) has orange labels identifying the instrumentation number (Ex. LI-322)

Technical Reference(s): Control Board LO21SYSPA1, Rev 5/30/2017 — Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: From memory, **DESCRIBE** the purpose of the Post Accident Instrumentation and EXPLAIN the systems interrelationship with other plant

systems based on approved station documents including but not limited to, DBD, drawings, FSAR vendor manuals procedures or other appropriate documents.

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		X
	Comprehensive or Analysis		
10CFR Part 55 Content:	**55.41**	7	
	55.43		

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		1
		Group		1
Level of Difficulty:	3	**K/A**	025AA2.04	
		Importance Rating	3.6	

K/A Statement: Loss of Residual Heat Removal System: Ability to determine or interpret the following as they apply to a loss of RHR system: Location and isolability of leaks.

Question 76

Unit 1 plant conditions:

- Mode 6
- RHR Train A in service
- RHR Train B in standby
- RCS level starts to lower
- ABN-104, Residual Heat Removal System Malfunction, Section 5.0, Mode 5 or 6 Unexpected Decrease in RCS Level – RCS Not Filled is entered
- CCP 1-01 has just been started

Based on the above plant conditions, complete the following statements:

1. If RCS level continues to lower, ABN-104 directs securing the TRAIN A RHR pump and closing ____(1)____.

2. If RCS level still continues to lower, ABN-104 directs ____(2)____.

A. (1) 1/1-8701A, RHRP 1 HL RECIRC ISOL VLV
   (2) re-starting Train A RHR in accordance with Attachment 7, RHR System Vent/Re-Start Instructions

B. (1) 1/1-8701A, RHRP 1 HL RECIRC ISOL VLV
   (2) starting Train B RHR in accordance with Attachment 18, Standby RHR Train Startup Instruction

C. (1) 1/1-8809A, RHR TO CL 1 & 2 INJ ISOL VLV
   (2) re-starting Train A RHR in accordance with Attachment 7, RHR System Vent/Re-Start Instructions

D. (1) 1/1-8809A, RHR TO CL 1 & 2 INJ ISOL VLV
   (2) starting Train B RHR in accordance with Attachment 18, Standby RHR Train Startup Instruction

Answer: B

K/A Match: This question matches the KA by requiring the ability to isolate an RCS leak while utilizing RHR in the Shutdown Cooling Mode of operation.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of ABN-104 content is required to correctly answer the written test item; specifically: knowledge of when to implement attachments, including how to coordinate the attachment with procedure steps.

Explanation:

A. Incorrect: 1st part is correct. IAW ABN-104, Section 5.0, Step 3 RNO, if RCS level is NOT stable at or trending to required level then the step directs starting one available charging pump and if level continues to fall the RHR Pump is secured and the HL Recirc Isolation Valve, 8701A, is closed. 2nd part is incorrect because ABN-104 directs starting the STBY RHR Train first. This is plausible because 1) since level continues to lower after the Hot Leg Recirc Isolation Valve was closed this was not the source of the leak and 2) if the STBY RHR train (Train B) doesn't start, the procedure directs starting Train A at the next step (Section 5.0, Step 7, RNO b) per Attachment 7.

B. Correct: 1st part is correct, see A above. 2nd part is correct. Per ABN-104, Section 5.0, Step 7, RNO a, the Standby RHR Train (Train B) is directed to be started first per Attachment 18.

C. Incorrect: 1st part is incorrect. ABN-104 directs closing the RHR Pump Suction Valves. It is plausible because the injection lines also connect to the RCS and could be a source of the leak. 2nd part is incorrect but plausible, see A above.

D. Incorrect: 1st part is incorrect but plausible, see C above. 2nd part is correct, see B above.

Technical Reference(s): ABN-104, Rev 9 PCN-11 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **ANALYZE** the response to an unexpected decrease in RCS level in accordance with ABN-104, Residual Heat Removal System Malfunctions. (LO21ABN104OB06)

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

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		1
		Group		1
Level of Difficulty:	3	**K/A**	029EA2.01	
		Importance Rating		4.7

K/A Statement: Anticipated Transient Without Scram: Ability to determine and interpret the following as they apply to the Anticipated Transient Without SCRAM or TRIP. Reactor nuclear instrumentation.

Question 77

Unit 1 plant conditions:

Time = 0800:

- Reactor power = 100%
- RCS pressure = 2100 psig lowering
- The reactor fails to trip manually

Time = 0803:

- FRS-0.1A, Response to Nuclear Power Generation / ATWT is in progress
- Power Range NIs = 4% lowering
- Neutron Flux Wide Range Instruments = 8% lowering
- RCS pressure = 1700 psig lowering
- Containment pressure = 6 psig rising

Time = 0806:

- Power Range NIs = 2% lowering
- Neutron Flux Wide Range Instruments = 4% lowering

Based on the above plant conditions, complete the statements below.

1. The EARLIEST time that the crew may transition out of FRS-0.1A is ___(1)___.

2. The crew will TRANSITION to ___(2)___.

A. (1) 0803
(2) EOP-0.0A, Reactor Trip or Safety Injection

B. (1) 0803
(2) EOP-1.0A, Loss of Reactor or Secondary Coolant

C. (1) 0806
(2) EOP-0.0A, Reactor Trip or Safety Injection

D. (1) 0806
(2) EOP-1.0A, Loss of Reactor or Secondary Coolant

Answer: C

K/A Match: This question matches the KA by requiring the ability to determine the correct procedure path using Nuclear Instrumentation.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because 1) additional knowledge of FRS-0.1A content is required to correctly answer the written test item; specifically: knowledge of diagnostic steps and decision points that involve a transition to event specific emergency procedures; and 2) knowledge of the content of the procedure is required versus the procedure's overall mitigative strategy; specifically: the knowledge contained in the bases of FRS-0.1A stating that when containment pressure is above 5 psig the normal power range excore detectors cannot be used to determine Reactor Power because they are not post-accident qualified, therefore the Neutron Flux Wide Range detectors must be used.

Explanation:

A. Incorrect. 1st part is incorrect but plausible as the US will not transfer out of FRS-0.1A until reactor power is < 5% on the Neutron Flux Wide Range Instruments as Containment pressure is adverse (> 5 psig) and the PRNIs are not post-accident qualified. 2nd part is incorrect when paired with time 0803 but plausible as FRS-0.1A does direct you to perform steps 1-8 of EOP 0.0A if an SI signal exists.

B. Incorrect. 1st part is incorrect because you will not transfer out of FRS-0.1A until reactor power is < 5% on the Neutron Flux Wide Range Instruments as Containment pressure is adverse (> 5 psig) and the PRNIs are not post-accident qualified. 2nd part is incorrect but plausible when paired with time 0803 as FRS-0.1A does direct the crew to transition to procedure and step in effect when conditions are met, in this case with a LOCA in progress it is reasonable to think that EOP-1.0A would be the procedure and step in effect, also It is plausible because after transferring to EOP-0.0A, you will then transition to EOP-1.0A due to the LOCA.

C. Correct. 1st part is correct, when power is < 5% on Neutron Flux Wide Range meters, FRS-0.1A directs the US to Step 18 where RCPs are checked just prior to returning to procedure and step in effect. 2nd part is correct when the crew transitions it will be to EOP-0.0A as it is the procedure and step in effect.

D. Incorrect. 1st part is correct, when power is < 5% on Neutron Flux Wide Range meters, FRS-0.1A directs the US to Step 18 where RCPs are checked just prior to returning to procedure and step in effect. 2nd part is incorrect but plausible when paired with time 0806 as after the transition is made to EOP-0.0A, you will then transition to EOP-1.0A due to the LOCA.

Technical Reference(s): FRS-0.1A, Rev 9 PCN- Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DISCUSS** the operator response to an Anticipated Transient Without Trip (LO21MCOMI5OB02).

Question Source:	**Bank #**	X	
	Modified Bank #		(note changes or attach parent)
	New		
Question History:	**Last NRC Exam**	2017 NRC Exam Q78	
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**		
	55.43	5	

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		1
		Group		1
Level of Difficulty:	4	**K/A**	040G2.2.38	
		Importance Rating		4.5

K/A Statement: Steam Line Rupture – Excessive Heat Transfer: Knowledge of conditions and limitations in the facility license.

Question 78

Unit 1 plant conditions:

- A Steam Line Break has occurred inside Containment
- Reactor tripped on Low-Low SG Water Level

In accordance with STI-214.01, Control of Time Critical Actions, which of the following actions is required and the reason for the action?

Isolate Auxiliary Feedwater to the faulted Steam Generator within a MAXIMUM of…

A. 10 minutes to preclude exceeding 10CFR100 dose limits.

B. 30 minutes to preclude exceeding 10CFR100 dose limits.

C. 10 minutes to preclude exceeding Containment design temperature.

D. 30 minutes to preclude exceeding Containment design temperature.

Answer: C

K/A Match: This question matches the KA by requiring knowledge of conditions of the facility license for a steam line rupture.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because the question requires knowledge of an emergency and the resultant Time Critical Actions associated with that emergency condition that could lead to a hazardous radiation condition which requires a "from memory" reference to the Time Critical Actions associated with the emergency and is outlined in the facility license (DBD/FSAR and summarized in STI-214.01). The Time Critical Actions in this case are performed to ensure radiation limits are not exceeded which therefore is testing 10 CFR 55.43(b)(4).

Explanation:

A. Incorrect. First part is correct, per STI-214.01, Attachment 8.A, TCA-1.2, a Main Steam Line Break inside containment has a 10-minute TCA to perform "Isolate AFW Flow to the Faulted SG." Second part is incorrect but plausible, per STI-214.01, Attachment 8.A, TCA-1.1, the actions to Isolate a Main Feed Line Break, the time critical action is performed in order to ensure 10 CFR 100 dose limits are not exceeded and prevent exceeding containment design

pressure.

B. Incorrect. First part is incorrect but plausible, per STI-214.01, Attachment 8.A, TCA-1.1, the time limit associated with performing the necessary time critical actions of a Main Feed Line Break is 30 minutes after event initiation (Rx trip on low-low SG water level). Second part is incorrect but plausible, see A above.

C. Correct. First part is correct, see A above. Second part is correct, per STI-214.01, Attachment 8.A, TCA-1.2, a Main Steam Line Break inside containment has a 10-minute TCA to perform the action of "Isolating AFW Flow to the Faulted SG," the action is performed to protect Containment from exceeding design temperature and pressure.

D. Incorrect. First part is incorrect but plausible, see B above. Second part is correct, see C above.

Technical Reference(s): STI-214.01, Rev 1 PCN-2 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **STATE** the bases for operator actions, notes and cautions from EOP-2.0A, Faulted Steam Generator Isolation.

Question Source: **Bank #** 2012 NRC Exam Q78
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 4

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		1
		Group		1
Level of Difficulty:	3	**K/A**	055EA2.03	
		Importance Rating		4.7

K/A Statement: Station Blackout: Ability to determine and interpret the following as they apply to the Station Blackout: Actions necessary to restore power.

Question 79

Unit 1 plant conditions:

- LOOP has occurred
- Reactor has tripped
- Train A EDG tripped on overspeed
- Train B EDG failed to start due to DG Air Receiver pressure < 150 psig
- Crew transitioned from EOP-0.0A, Reactor Trip or Safety Injection to ECA-0.0A, Loss of All AC Power

Based on the above plant conditions, complete the following statements:

1. What start method MAY be successful in starting Train B EDG?

2. IF Train B AC Safeguards bus is re-energized, THEN performance of ____(2)____, will be required.

   A. (1) normal
      (2) ABN-601, Response to a 138/345 KV System Malfunction

   B. (1) normal
      (2) ABN-602, Response to a 6900/480V System Malfunction

   C. (1) emergency
      (2) ABN-601, Response to a 138/345 KV System Malfunction

   D. (1) emergency
      (2) ABN-602, Response to a 6900/480V System Malfunction

Answer: A

K/A Match: This question matches the KA by requiring the ability to determine actions necessary to restore power and subsequently, the ABNs that must be performed after the correct ERG procedure transition has been made when power is restored.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of EOP-0.0A and ECA-0.0A content is required to correctly answer the written test item; specifically: knowledge of diagnostic steps and decision points in the EOPs that

involve transitions to event-specific sub-procedures.

Explanation:

A. Correct: First part is correct per SOP-609A, Diesel Generator System, A DG will not accept an emergency start signal unless the pressure in at least one start air receiver is greater than 150 psig. Also, per ECA-0.0A bases, an Emergency start of the DG is blocked when starting air receiver pressure is low. This means that only a Normal start may have a chance of starting the DG as the emergency signal is blocked. Second part is correct, when either of the 6.9 KV Safeguards busses are re-energized in ECA-0.0A a return to procedure and step in effect is required per ECA-0.0A, Step 5.d. The procedure in effect at this time is EOP--0.0A. Per EOP-0.0A, Step 3, RNO b, if Both AC Safeguards Busses are not energized (Train A is de-energized in this case) then power should be restored to the de-energized bus utilizing ABN-601 or ABN-602 as appropriate and as time permits. Per the Applicability Flowchart of ABN-601 with NO Grid Frequency Fluctuations, the unit in Mode 3, NO Offsite power supplying any Safeguards Bus, and Train B Safeguards Bus energized, the crew should perform ABN-601.

B. Incorrect: First part is correct, see A above. Second part is incorrect but plausible, it is a common misconception that that with a Safeguards Bus Energized on the unit ABN-602 is always performed, however, in this case since the Safeguards Bus is energized by a DG and there is no offsite power supplying either Safeguards Bus the correct procedure is ABN-601.

C. Incorrect: First part is incorrect but plausible to believe that an emergency start would be effective or at least more effective at achieving a DG start with low starting air receiver pressure, however, with low pressure in the starting air receiver the emergency start signal is blocked and only a normal start will have a chance at being successful. Also, the emergency start is performed first in ECA-0.0A which lends to the plausibility of an emergency start signal being plausible. Second part is correct, see A above.

D. Incorrect: First part is incorrect but plausible, see C above. Second part is incorrect but plausible, see B above.

Technical Reference(s): ECA-0.0A, Rev 9 PCN-1
EOP-0.0A, Rev 9 PCN-3
ABN-601, Rev 13 PCN-4
ABN-602, Rev 8 PCN-10
SOP-609A, Rev 21 PCN-19

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **IDENTIFY** the proper transitions out of ECA-0.0A, Loss of All AC Power.

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

Question History: **Last NRC Exam**

Question Cognitive Level:	**Memory or Fundamental Knowledge**	
	Comprehensive or Analysis	X

10CFR Part 55 Content:	**55.41**	
	55.43	5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**		1
		Group		1
Level of Difficulty:	4	**K/A**	077G2.2.40	
		Importance Rating		4.7

K/A Statement: Generator Voltage and Electric Grid Disturbances: Ability to apply Technical Specifications for a system.

Question 80

Site conditions:

- At 0800, the Transmission Grid Operator reported the following transmission lines were lost due to a significant grid disturbance:
  - 345 KV Parker #1
  - 345 KV Parker #2
  - 138 KV Decordova

- At 0830, 138 KV Stephenville transmission line was lost

Based on the conditions above, complete the following statements.

1. **At 0800**, per Technical Specification 3.8.1, AC Sources – Operating and OPT-215, Class 1E Electrical Systems Operability, two off site power sources ____(1)____ OPERABLE.

2. **At 0830**, ABN-601, Response to a 138/345 KV System Malfunction, Section ____(2)____ will be used by Unit 2 to recover the affected switchyard.

   A. (1) are NOT
      (2) 5.0, Loss of All Offsite AC Power

   B. (1) are
      (2) 5.0, Loss of All Offsite AC Power

   C. (1) are NOT
      (2) 2.0, Loss of Startup/Station Service Transformers

   D. (1) are
      (2) 2.0, Loss of Startup/Station Service Transformers

Answer: C

K/A Match: This question matches the KA by requiring the ability to apply TSs for the electrical system as a result of an electric grid disturbance.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of surveillance requirements is required in accordance with the rules of application requirements of technical specifications. Also, additional

knowledge is required to perform an assessment of plant conditions and subsequent selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

Explanation:

A. Incorrect: First part is correct, per OPT-215A a loss of 138 KV Decordova combined with a loss of both 345 KV Parker lines restricts the operability pairs of transmission lines (i.e. 138 KV Stephenville must be paired with Both of the 345 KV Parker lines because the 138 KV Stephenville line runs under ALL the other 345 KV lines (other than the Parker lines)). Second part is incorrect but plausible because with a loss of Both 138 KV Transmission lines and two of the 345 KV Transmission lines to CPNPP it is reasonable to believe that a loss of all offsite AC power could have occurred and therefore require execution of ABN-601, Section 5.0, Loss of All Offsite AC Power.

B. Incorrect. First part is incorrect but plausible because if the 138 KV Stephenville line had been lost instead of the 138 KV Decordova line then Both Offsite circuits to CPNPP would remain Operable. Because the Stephenville line runs underneath all the other 345 KV lines except the Parker lines it must be paired with the Stephenville lines for Operability. Second part is incorrect but plausible, see A above.

C. Correct. First part is correct, see A above. Second part is correct, per the ABN-601 the plant indications for entrance into Section 2.0 include a slow transfer of Unit 2 safeguards buses to their alternate supply (which will occur in this situation).

D. Incorrect. 1st part is incorrect, but plausible, see B above. 2nd part is correct, see C above.

Technical Reference(s): ABN-601, Rev 13 PCN-4
OPT-215, Rev 17
TS SR 3.8.1.1, Amd 156
Switchyard Big Book, Rev 9/17/18

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the AC Distribution and Transformers system including Technical Specification, TRM and ODCM

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

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content:
55.41
55.43 2, 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**		1
		Group		1
Level of Difficulty:	3	**K/A**	W E05 G2.4.6	
		Importance Rating		4.7

K/A Statement: Inadequate Heat Transfer-Loss of Secondary Heat Sink: Knowledge of EOP mitigation strategies.

Question 81

Unit 2 plant conditions:

- Reactor Trip occurs due to a LOCA inside Containment
- A Phase-to-Phase Fault on 6.9 KV Safeguards Bus 2EA1 occurs
- RED path condition occurred on Heat Sink
- A transition to FRH-0.1B, Loss of Heat Sink is executed

Subsequently:

- Containment pressure reaches 19 psig and rising
- The crew has reached FRH-0.1B, Step 26 – "Check Containment Spray Status"

Based on the conditions above, complete the statements below.

1. The method used by FRH-0.1B to maintain heat removal capability will be to ____(1)____.

2. Based on containment pressure the US will ____(2)____.

A. (1) initiate RCS bleed and feed
(2) perform FRZ-0.1B, Response to High Containment Pressure, while continuing in FRH-0.1B

B. (1) establish AFW flow to at least one SG
(2) perform FRZ-0.1B, Response to High Containment Pressure, while continuing in FRH-0.1B

C. (1) initiate RCS bleed and feed
(2) exit FRH-0.1B and transition to FRZ-0.1B, Response to High Containment Pressure

D. (1) establish AFW flow to at least one SG
(2) exit FRH-0.1B and transition to FRZ-0.1B, Response to High Containment Pressure

Answer: A

K/A Match: This question matches the KA by requiring knowledge of EOP strategies for a loss of heat sink.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of FRH-0.1A content is required to correctly answer the written test item; specifically: knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures.

Explanation:

A. Correct. First part is correct, the phase to phase ground on 6.9 KV Safeguards Bus 2EA1 caused a loss of Charging Pump (CCP) 2-01 and Safety Injection (SIP) Pump 2-01. Step 2 of FRH-0.1B requires both CCPs available; without both CCPs the US must proceed to initiate Bleed and Feed cooling per Step 13. Second part is correct, per FRH-0.1B, Step 26, if containment pressure is greater than 18 psig, FRZ-0.1B will be performed while continuing in FRH-0.1B.

B. Correct. First part is incorrect but plausible because the phase to phase ground on 6.9 KV Safeguards Bus 2EA1 caused a loss of Charging Pump (CCP) 2-01 and Safety Injection (SIP) 2-01. Step 2 of FRH-0.1B requires both CCPs available; without both CCPs the US must proceed to initiate Bleed and Feed cooling per Step 13. However, after SI is initiated at Step 13, Step 14 requires both SIPs to be running OR 1 SIP and 1 CCP. If SIP 2-02 was not available or out of service and only CCP 2-02 was running, the US would go back to Step 4 per Step 14 RNO. At Step 4 CST level would be checked greater than 10%, then Step 5 would require establishment of AFW flow to at least one SG. Second part is correct, see A above.

C. Incorrect. First part is correct, see A above. Second part is incorrect but plausible because when Containment pressure exceeds 18 psig the Containment Spray system will initiate on the HI-3 setpoint. Normally when HI-3 is reached FRZ-0.1B should be performed to verify Containment Spray actuated properly, however, in this case FRH is a higher priority FRG procedure and the US should not transition to FRZ-0.1B. FRH-0.1B specifically states to perform FRZ-0.1B while continuing in FRH-0.1B.

D. Incorrect. First part is incorrect but plausible, see B above. Second pert is incorrect but plausible, see C above.

Technical Reference(s): ODA-407, Rev 16 PCN-5
FRH-0.1B, Rev 9 PCN-1
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: From memory, DISCUSS the plant response, operator actions, and the reasons for the actions contained in FRH-0.1, Loss of Secondary Heat Sink in accordance with FRH-0.1, Loss of Heat Sink.

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

	New	X
Question History:	**Last NRC Exam**	
Question Cognitive Level:	**Memory or Fundamental Knowledge**	
	Comprehensive or Analysis	X
10CFR Part 55 Content:	**55.41**	
	55.43 5	

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		1
		Group		2
Level of Difficulty:	3	**K/A**	001G2.2.25	
		Importance Rating		4.2

K/A Statement: Continuous Rod Withdrawal: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question 82

Unit 1 plant conditions:

- Plant startup is in progress
- Reactor power = 3%
- An NEO reports that 2 MSSVs on SG-01 are gagged in the closed position
- TS LCO 3.7.1, Main Steam Safety Valves is entered

Based on the above plant conditions, complete the following statements:

1. The bases for the REQUIRED ACTION of TS LCO 3.7.1 CONDITION B – One or more steam generators with two or more MSSVs inoperable, is to preclude RCS overpressurization in the event of a(n) ____(1)____.

2. In accordance with Technical Specifications, the Unit ____(2)____ allowed to proceed to Mode 1.

A. (1) uncontrolled RCCA bank withdrawal
(2) is

B. (1) uncontrolled RCCA bank withdrawal
(2) is NOT

C. (1) main turbine trip
(2) is

D. (1) main turbine trip
(2) is NOT

Answer: A

K/A Match: This question matches the KA by requiring knowledge of continuous rod withdrawal being the bases for TS LCOs

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of Technical Specifications is required to correctly answer the written test item; specifically: knowledge of generic LCO requirements (LCO 3.0.1 through 3.0.7 as well as knowledge of TS bases that is required to analyze TS-required actions and terminology.

Explanation:

A. Correct: 1st part is correct: The basis for "B" Actions is based on preventing RCS overpressurization in the event of a continuous rod withdrawal at power. 2nd part is correct. IAW LCO 3.0.4, you may go up in mode if actions allow continued operation in that mode which it does with power limitations.

B. Incorrect: 1st part is correct (see A). 2nd part is incorrect because you are allowed to go up in Mode per LCO 3.0.4. It is plausible because if the TS required actions did not allow continued operation in that mode then entry into Mode 1 would not be allowed.

C. Incorrect: 1st part is incorrect because TS 3.7.1, Condition B is based on a continuous rod withdrawal event. It is plausible because a turbine trip at power will cause an RCS pressure spike. 2nd part is correct (see A).

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

Technical Reference(s): TS 3.7.1, Amd 150
TS 3.7.1 Bases, Rev 73
LCO 3.0.4, Amd 150

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Main Steam system including Technical Specifications and TRM (LO21YMR1OB06).

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

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content:
55.41
55.43 2

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:2		**Tier**		1
		Group		2
Level of Difficulty:	3	**K/A**	069G2.2.3	
		Importance Rating		3.9

K/A Statement: Loss of Containment Integrity: Knowledge of the design, procedural and operational difference between units.

Question 83

Regarding Personnel Air Locks (PAL), complete the statements below.

1. The Unit ____(1)____ PAL is provided with 2 hydraulic pumps to supply the door locking mechanisms.

2. IF the PAL noted above has an Outer Door Equalization Valve (see Note below) that will NOT cycle during the door CLOSED sequence, THEN entry into TS ____(2)____ is required.

NOTE: uBS-0025, CNTMT PERS AIRLOCK u-01 EXT DOOR AUTO EQUAL VLV 0025

A. (1) 1
   (2) LCO 3.6.2, Containment Air Locks

B. (1) 1
   (2) LCO 3.6.3, Containment Isolation Valves

C. (1) 2
   (2) LCO 3.6.2, Containment Air Locks

D. (1) 2
   (2) LCO 3.6.3, Containment Isolation Valves

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the differences between Units 1 & 2 containment integrity.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of Technical Specifications is required to correctly answer the written test item; specifically: knowledge of TS bases that are required to analyze TS-required actions and terminology.

Explanation:

A. Incorrect: First part is incorrect, Unit 1 has only one pump to supply the door locking mechanisms. Plausible if confused the two units as this is not a well-known unit difference and could be easily reversed. Second part is correct, per ODA-308 (CPNPPs implementation procedure for Technical Specifications), 2BS-0025 is considered an Outer Door Isolation point and requires entry in to TS LCO 3.6.2 Condition A for the Outer Door being INOPERABLE.

B. Incorrect: First part is incorrect but plausible, see A above. Second part is incorrect but plausible because the TS Bases for LCO 3.6.3, Containment Isolation Valves lists the Unit 1 PAL and the Airlock Hydraulic System as type C Containment Isolation Valves (Closed Loop System with 1 CIV). It is reasonable for an applicant to believe that the Unit 2 Air Equalization Valve could be a part of the requirements for TS LCO 3.6.3.

C. Correct: First part is correct, per the CPNPP Unit Difference Logs, the Unit 2 PAL hydraulic system is two separate systems; one for the inner door and one for the outer door each with its own hydraulic pump. Second part is correct, see A above.

D. Incorrect: First part is correct, see C above. Second part is incorrect but plausible, see B above.

Technical Reference(s): TS 3.6.2, Amd 150
TS 3.6.3, Amd 150
TSB 3.6.3, Rev 74
Unit Difference Logs
ODA-308, Rev 16 PCN-4

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Containment System in accordance with Technical Specifications, TRM and ODCM

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 2

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		1
		Group		2
Level of Difficulty:	3	**K/A**	074EA2.1	
		Importance Rating		4.0

K/A Statement: Inadequate Core Cooling: Ability to determine and interpret the following as they apply to the (Saturated or Inadequate Core Cooling): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Question 84

Unit 1 plant conditions:

Time = 0800

- A Large Break LOCA has occurred
- A transition to EOP-1.0A, Loss of Reactor or Secondary Coolant has just been made and EOP-0.0A, Attachment 2, Safety Injection Actuation Alignment is complete
- One RHR train injection line has ruptured inside containment
- The HIGHEST Reading CET Temperatures are as follows:
  - 1205°F
  - 1202°F
  - 1197°F
  - 1195°F
  - 1194°F
- All CET Temperatures are slowly rising

Time = 0810

- RWST level reaches 33% and lowering

Based on the above plant conditions, complete the statements below.

1. At Time = 0800 the US should transition to ____(1)____.

2. At Time = 0810 the US should ____(2)____.

A. (1) FRC-0.1A, Response To Inadequate Core Cooling
   (2) Go to EOS-1.3A, Transfer To Cold Leg Recirculation

B. (1) FRC-0.1A, Response To Inadequate Core Cooling
   (2) Perform EOS-1.3A while continuing with the procedure in effect

C. (1) FRC-0.2A, Response To Degraded Core Cooling
   (2) Go to EOS-1.3A, Transfer To Cold Leg Recirculation

D. (1) FRC-0.2A, Response To Degraded Core Cooling
   (2) Perform EOS-1.3A while continuing with the procedure in effect

Answer: A

K/A Match: This question matches the KA by requiring the ability to interpret plant conditions to determine the appropriate procedure flowpath to mitigate the event. Unable to write a question at the SRO level specifically on procedure selection of the FRC series as the entry conditions for the FRC series procedures are RO knowledge. However, the second part of the question meets the requirements of SRO-only knowledge.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of FRC-0.1A content is required to correctly answer the written test item; specifically: knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures.

Explanation:

A. Correct: First part is correct, per FRC-0.1A and DBD-EE-004 at CPNPP the ERGs utilize a confirmed reading to implement the Core Cooling Red Path procedure based on two CETs being greater than 1200°F. Second part is correct, per FRC-0.1A and FRC-0.2A if RWST level reaches the Lo-Lo value of 33%, then the US must transition to EOS-1.3A. All FRGs in effect will be suspended until the first four steps of EOS-1.3A are completed.

B. Incorrect: First part is correct, see A above. Second part is incorrect, but plausible because there are other instances in the ERGs where another ERG is performed while continuing in the procedure in effect. For example, in FRS-0.1A, if an SI signal exists or occurs, then Steps 1-8 of EOP-0.0A should be performed while continuing with FRS-0.1A.

C. Incorrect: First part is incorrect but plausible, because Westinghouse analysis bases an inadequate core cooling condition on at least 5 thermocouple readings greater than 1200°F. This is contrary to what is required at CPNPP to implement the Core Cooling ERG flowpath. At CPNPP the expectation of the ERGs is to utilize a confirmed reading based on at least two CETs to implement the ERG criteria. Therefore, with only two CETs above 1200°F it is reasonable to believe that an examinee would believe that FRC-0.2A must be implemented due to all 5 of the highest CETs reading greater than 750°F. Second part is correct, see A above.

D. Incorrect: First part is incorrect but plausible, see C above. Second part is incorrect but plausible, see B above.

Technical Reference(s): FRC-0.1A, Rev 9 PCN-
FRC-0.2A, Rev 9 PCN-
DBD-EE-004, Rev 44
EOS-1.3A, Rev 9 PCN-
FRS-0.1A, Rev 9 PCN

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the operational implications associated with all cautions, notes and actions of the Response to Inadequate Core Cooling procedure in

accordance with FRC-0.1A.

Question Source:	**Bank #**		
	Modified Bank #		(note changes or attach parent)
	New	X	
Question History:	**Last NRC Exam**		
Question Cognitive Level:	**Memory or Fundamental Knowledge**		
	Comprehensive or Analysis		X
10CFR Part 55 Content:	**55.41**		
	55.43	5	

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		1
		Group		2
Level of Difficulty:	4	**K/A**	W/E02EA2.2	
		Importance Rating		4.0

K/A Statement: Rediagnosis & **SI Termination**: Ability to determine and interpret the following as they apply to SI Termination: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Question 85

Unit 2 plant conditions:

- A spurious Safety Injection occurred from 100% power

In accordance with STI-214.01, Control of Timed Operator Actions and based on the above plant conditions, complete the following statement.

In order to ____(1)____, ECCS must be terminated in accordance with EOS-1.1B, Safety Injection Termination within a MAXIMUM of ____(2)____ minutes.

A. (1) prevent the Pressurizer from going solid
(2) 22

B. (1) limit the excessive cooldown associated with ECCS injection flow
(2) 22

C. (1) limit the excessive cooldown associated with ECCS injection flow
(2) 14

D. (1) prevent the Pressurizer from going solid
(2) 14

Answer: D

K/A Match: This question matches the KA by requiring the ability to adhere to operation within the limitations of the facility license for terminating Safety Injection following an inadvertent SI.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of SI Termination requirements is required to correctly answer the written test item; specifically: knowledge of administrative procedures that specify coordination of emergency procedures, specifically, STI-214.01, Control of Timed Operator Actions.

Explanation:

A. Incorrect: First part is correct, in accordance with STI-214.01, following a spurious Safety Injection, ECCS must be terminated within 14 minutes to prevent the Pressurizer from going solid. Second part is incorrect but plausible, 22 minutes is also the time limit of a timed operator action listed in STI-214.01, however, the 22-minute time limit is associated with starting the Train A CCW pump from the RSP following a fire in the Control Room/Cable Spreading Room.

B. Incorrect: First part is incorrect but plausible because an SI will cause a Reactor Trip as well as Injection flow from the CCPs at normal operating pressure after the trip. This injection flow will cause an increased cooldown in the RCS, however, this is not the reason to terminate SI within the 14-minute time frame required by the FSAR and STI-214.01. Second part is incorrect but plausible, see A above.

C. Incorrect: First part is incorrect but plausible, see B above. Second part is correct, in accordance with FSAR Chapter 15.5.1.2 and STI-214.01, operator action is required to terminate ECCS flow within 14 minutes after the initiation of the event to prevent the pressurizer from going solid.

D. Correct. First part is correct, see A above. Second part is correct, see C above.

Technical Reference(s): FSAR Chapter 15, Amd 105
STI-214.01, Rev 1 PCN-2
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: From memory, **EXPLAIN** plant response and operator actions, to include the bases, required to terminate or reinitiate safety injection.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 1

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:4		**Tier**		2
		Group		1
Level of Difficulty:	3	**K/A**	004A2.09	
		Importance Rating		3.9

K/A Statement: Chemical and Volume Control: Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use the procedures to correct, control or mitigate the consequences of those malfunctions or operations: High primary and/or secondary activity.

Question 86

Unit 1 is at full power when the following occurs:

- 1-RE-0406, Gross Failed Fuel Monitor alarms
- The US enters abnormal procedure for alarm
- Chemistry reports a sample that is high in Dose Equivalent I-131 specific activity
- Dose Equivalent I-131 specific activity continues to steadily increase

Based on the conditions above, complete the following statements.

1. If Dose Equivalent I-131 specific activity continues to rise, it will FIRST reach a Technical Specification limit at a value of ____(1)____.

2. In accordance with the abnormal procedure in effect, the US will ____(2)____.

A. (1) 0.45 uCi/gm
(2) minimize movement of control rods

B. (1) 0.45 uCi/gm
(2) maximize letdown flow

C. (1) 4.5 uCi/gm
(2) minimize movement of control rods

D. (1) 4.5 uCi/gm
(2) maximize letdown flow

Answer: B

K/A Match: This question matches the KA by requiring the ability to use CVCS procedures and TS SRs predict the impacts of and mitigate high RCS activity.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of Technical Specifications is required to correctly answer the written test item; specifically: knowledge of the application of Technical Specification Surveillance Requirements and the associated Bases.

Explanation:

A. Incorrect: First part is correct, in accordance with TS SR 3.4.16.2, Dose Equivalent I-131 specific activity must be maintained below a maximum value of 0.45 uCi/gm. Second part is incorrect but plausible because if the high RCS activity was due to a CRUD burst isotope this would be a true statement, however, in this case with high fuel failure isotope it is incorrect.

B. Correct: First part is correct, see A above. Second part is correct, per ABN-102 an increase of an RCS fission product isotope during steady state operation may indicate fuel cladding damage (e.g. Xe-133, Kr-85M, Cs-137, Cs-136, Sr-84, Sr-90, Iodine) and letdown should be increased to remove these isotopes.

C. Incorrect: First part is incorrect, but plausible because the limit for TS 3.4.16 Required Action A.1 is 60 uCi/gm, it is a common misconception that TS SR 3.4.16.2 limit is 4.5 uCi/gm (vice 0.45) because of this limit of 60 uCi/gm in Required Action A.1. Second part is incorrect but plausible, see A above.

D. Incorrect: First part is incorrect but plausible, see C above. Second part is correct, see B above.

Technical Reference(s): ABN-102, Rev 8 PCN-
ALM-3200, Rev 5 PCN-2
TS 3.4.16, Amd 156
TSB 3.4.16, Rev 76

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DISCUSS** the response to High Reactor Coolant Activity in accordance with ABN-102, High Reactor Coolant Activity (LO21ABN102OB01).

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 2

Examination Outline Cross-Reference:		**Level**	**RO** **SRO**
Rev:1		**Tier**	2
		Group	1
Level of Difficulty:	3	**K/A**	008 G2.4.8
		Importance Rating	4.5

K/A Statement: Component Cooling Water: Knowledge of how AOPs are used in conjunction with EOPs.

Question 87

Unit 1 plant conditions:

- Reactor power = 100%
- The following annunciators are in alarm
  - 1-ALB-3B, Window 2.4, CCW SRG TK TRN A LVL HI-HI/LO
  - 1-ALB-3B, Window 3.4, CCW SRG TK TRN B LVL HI-HI/LO
  - 1-ALB-3B, Window 2.11, ANY RCP THBR CLR CCW RET TEMP HI

Based on the above plant conditions, complete the following statement.

When LO SEAL WTR BRG TEMP (pump bearing) first reaches ____(1)____°F, the reactor will be tripped and ABN-502 will be ____(2)____.

A. (1) 183
   (2) completed after the crew has exited the ERG network

B. (1) 183
   (2) performed in parallel with EOP-0.0A

C. (1) 226
   (2) completed after the crew has exited the ERG network

D. (1) 226
   (2) performed in parallel with EOP-0.0A

Answer: D

K/A Match: This question matches the KA by requiring knowledge of how the ABN for CCW is performed in conjunction with the EOPs.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations is being evaluated; specifically: assessment of plant conditions and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed.

Explanation:

A. Incorrect: 1st part is incorrect because ABN-502 will not direct tripping the reactor until temperature reaches 225 °F. It is plausible because at 182.5 °F, the RCP thermal barrier cooler CCW return valve closes. 2nd part is incorrect because ABN-502 directs you to continue with this procedure after tripping the reactor. It is plausible because for some ABNs, it would be correct.

B. Incorrect: 1st part is incorrect but plausible (see A). 2nd part is correct. ABN-502 directs you to trip the reactor and GO TO EOP-0.0A while other qualified operators continue with this procedure.

C. Incorrect: 1st part is correct. ABN-502 directs you trip the reactor if pump bearing temperature is not less than or equal to 225°F. 2nd part is incorrect but plausible (see A).

D. Correct: 1st part is correct (see C). 2nd part is correct (see B).

Technical Reference(s): ABN-502, Rev, 9 PCN-2 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DISCUSS** the response to Loss of CCW Flow to the Non-Safeguards Loop in accordance with ABN-502, Component Cooling Water System Malfunction.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		2
		Group		1
Level of Difficulty:	3	**K/A**	012A2.01	
		Importance Rating		3.6

K/A Statement: Reactor Protection: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Faulty bistable operation.

Question 88

Unit 1 plant conditions:

- 1-ALB-6D, Window 1.3 – 1 OF 4 HI SETPT PR FLUX HI, is in alarm
- All PR NIs are 100% and stable
- OVERPOWER TRIP HIGH RANGE light for N-42 is LIT on NIS Panel due to failure of High Setpoint High Flux Trip Bistable
- ABN-703, Power Range Instrument Malfunction, in progress to place channel out-of-service for repairs

Which of the following identifies…

1) the impact on RPS?

2) the required action to comply with Technical Specifications and ABN-703?

A. (1) An OP HI FLUX ROD STOP C-2 is generated that can be bypassed.
(2) Reactor Trip Bistables for Loop 2 must be placed in TRIP within one hour.

B. (1) A Power Range High Flux Trip is generated that can be bypassed.
(2) If Reactor to remain at 100% RTP, QPTR must be determined using Core Power Distribution Measurement information.

C. (1) A Power Range High Flux Trip is generated that can be bypassed.
(2) Reactor Trip Bistables for Loop 2 must be placed in TRIP within one hour.

D. (1) An OP HI FLUX ROD STOP C-2 is generated that can be bypassed.
(2) If Reactor to remain at 100% RTP, QPTR must be determined using Core Power Distribution Measurement information.

Answer: B & D

K/A Match: This question matches the KA by requiring the ability to predict the impact of a bistable failure and use appropriate procedures to mitigate the consequences of the event.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of Technical Specifications is required to correctly answer the written test item; specifically: application of required actions and SRs in accordance with

rules of application requirements.

Explanation:

A. Incorrect. First part is correct, C-2 Rod Stop is generated whenever 1/4 PR NI channels is greater than 103% of full power and the rod stop can be bypassed. Second part is incorrect but plausible, because the N16 Power Channel Defeat Switch must be aligned to Loop 2, however, the bistables need not be placed in trip.

B. ~~Incorrect~~ Correct. ~~First part is incorrect but plausible because a Power Range High Flux Trip will be generated, however, the trip cannot be blocked.~~ First part is correct because a Power Range High Flux Trip will be generated and can be blocked. Second part is correct, per ABN-703, Section 2.0, Step 2, Reactor Power should be "Verified" less than 75% RTP. If power is above 75%, the RNO requires initiation of actions to comply with TS SR 3.2.4.2. SR 3.2.4.2 requires verification that QPTR is within limits using core power distribution measurement information within 12 hours after PR NI channel is inoperable AND power is greater than 75%.

C. Incorrect. First part is ~~incorrect but plausible~~ correct, see B above. Second part is incorrect but plausible, see A above.

D. Correct. First part is correct, see A above. Second part is correct, see B above.

Technical Reference(s): ABN-703, Rev. 9 PCN-1
TS 3.2.4.2, Amd 156

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Nuclear Instrumentation System.

Question Source:
Bank # 2013 NRC Exam Q86
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		2
		Group		1
Level of Difficulty:	4	**K/A**	022G2.2.37	
		Importance Rating		4.6

K/A Statement: Containment Cooling: Ability to determine operability and/or availability of safety related equipment.

Question 89

Unit 2 plant conditions:

- MODE 4
- OPT-205B, Containment Spray System, in progress
- Operator records the following information on Form 5, Train A CSP Test Data Sheet
    - 2-FI-4772-1, DISCH FLOW indicated 3300 gpm
    - 2-FI-4772-2, DISCH FLOW indicated 3325 gpm
    - 2-LI-4752, CHEM ADD TK LVL indicated 90%
    - CSP 1 DIFF PRESS calculated to be 250 psid
    - CSP 3 DIFF PRESS calculated to be 245 psid
    - CSP 1 & 3 TOTAL FLOW calculated to be 6625 gpm

Which of the parameters recorded above require completion of LCO REQUIRED ACTIONS per TS or TRM?

A. Chem Add Tank Level

B. CSP 1 or 3 Disch Flow

C. CSP 1 or 3 Diff Pressure

D. CSP Combined Disch Flow

Answer: A

K/A Match: This question matches the KA by requiring the ability to determine operability requirements the Containment Spray System, specifically requirements from TRM SRs.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of the TRM is required to correctly answer the written test item; specifically: application of required actions and surveillance requirements in accordance with rules of application requirements.

Explanation:

A. Correct: Per OPT-205B, Form 5, indicated Chem Add Tank level must be between 91% to 94%. This information is also contained in TRM SR 13.6.7.2. SR 13.6.7.2 is satisfied by

performance of OPT-205B.

B. Incorrect: Per OPT-205B, Form 5, CSP 1 or 3 Disch Flow is required to be between 3300 gpm and 3350 gpm for performance of the OPT. This information is also contained in TRM 13.6.6. This is plausible because when the students see the CSP Disch flow in the simulator (discharging to Containment) the flows are approximately 3500 gpm per pump, this may lead the applicant to believe that Disch flows of 3300 gpm (CSP 1) or 3325 gpm (CSP 3) are too low. The flows generally reported in the simulator are when the system is discharging to containment. When the OPT is performed the system is aligned to the Test Header, when aligned to the test header per the OPT flows must be between 3300 and 3350 gpm.

C. Incorrect: Per OPT-205B, Form 5, CSP 1 Diff Pressure must be between 245 and 282.7 psid. CSP 3 Diff Pressure must be between 245 and 278.3 psid. TRM 13.6.6 also requires a minimum Diff Pressure of 245 psid per pump. The values of 250 (CSP 1) and 245 (CSP 3) psid are plausible because the applicant may believe 250 psid is the minimum required Diff Pressure of the CS Pumps and may believe that CSP 3 is below the required value.

D. Incorrect: Per OPT-205B, Form 5, CSP 1 & 3 Combined Disch Flow must be greater than or equal to 6600 gpm. The value of 6625 gpm is plausible because the values normally observed in the simulator are approximately 3500 gpm per pump. This would equate to a combined value of 7000 gpm and 6625 is below 7000 gpm (see B for explanation of difference in flow between OPT and actual CS actuation)

Technical Reference(s): OPT-205B, Rev. 15 PCN-6
OPT-205-5, Rev. 19
TRM 13.6.6, Rev. 88
TRM 13.6.7, Rev. 88

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Containment Spray System, including Technical Specifications, TRM and ODCM

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		2
		Group		1
Level of Difficulty:	3	**K/A**	062 G2.2.25	
		Importance Rating		4.2

K/A Statement: AC Electrical Distribution: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question 90

Unit 2 plant conditions:

- MODE 1
- At Time 0800, Inverter IV2PC1 fails

- At Time 0810:
  - Protection Bus 2PC1 re-energized from Alternate Power
  - 480 Volt Safeguards Bus 2EB3 Voltage is 481 VAC

- At Time 0900, Spare Inverter IV2EC1/3 placed in service powering Protection Bus 2PC1

- At Time 1200, Inverter IV2PC1 repaired and placed in service powering Protection Bus 2PC1

Based on conditions above, complete the statement below.

TS LCO ____(1)____ must be entered from Time ____(2)____.

A. (1) 3.8.7, Inverters – Operating
   (2) 0800 to 1200

B. (1) 3.8.7, Inverters – Operating
   (2) 0800 to 0900 ONLY

C. (1) 3.8.9, Distribution Systems – Operating
   (2) 0800 to 1200

D. (1) 3.8.9, Distribution Systems – Operating
   (2) 0800 to 0900 ONLY

Answer: B

K/A Match: This question matches the KA by requiring knowledge of bases of technical specifications for AC electrical distribution (AC Protection Bus and associated Inverters).

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of Technical Specifications is required to correctly answer the written test item; specifically: knowledge of TS bases that is required to analyze

TS-required actions and terminology.

Explanation:

A. Incorrect: Incorrect because per TSB for LCO 3.8.7, the LCO is met when the spare inverter (IV2EC1/3) is powering protection bus 2PC1. Plausible because the failed inverter (IV2PC1) is not operable and not connected to protection bus 2PC1 until 1200.

B. Correct. The spare inverter (IV2EC1/3) is an acceptable substitute (from TSB for LCO 3.8.7) to power protection bus 2PC1 and per TS LCO 3.8.7, the inverter has to be connected to its associated bus to be operable. This occurs at 0900.

C. Incorrect: Incorrect because TS LCO 3.8.9 is only not met from time 0800 to 0810 when protection bus 2PC1 is de-energized. It is plausible because the inverter has to be connected to its associated but to be operable (not the alternate source) but this is not a restriction for operability with the bus itself.

D. Incorrect: Incorrect because TS LCO 3.8.9 is only not met from time 0800 to 0810 when protection bus 2PC1 is de-energized. It is plausible because the inverter has to be connected to its associated but to be operable (not the alternate source) but this is not a restriction for operability with the bus itself.

Technical Reference(s): TS 3.8.7, 3.8.9, Amd 156
TSB 3.8.7, 3.8.9, Rev. 68
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements fo the AC Distribution and Transformers system including Technical Specifications, TRM and ODCM.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 2

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		2
		Group		2
Level of Difficulty:	3	**K/A**	002G2.2.12	
		Importance Rating		4.1

K/A Statement: Reactor Coolant: Knowledge of surveillance procedures.

Question 91

Unit 2 plant conditions:

- RCS Tave = 320°F
- RCS pressure = 340 psig
- No RCPs running

Based on the conditions above, complete the statement below.

Per the applicable TS LCO for RCS Loops, …

1) in order to use an RCS loop as an acceptable alternate to an OPERABLE operating RHR loop, ____(1)____ in that loop.

-AND-

2) SG secondary side water level must be a MINIMUM of ____(2)____.

A. (1) an RCP must be started
(2) 38%

B. (2) an RCP must be started
(2) 10%

C. (1) SG secondary side water temperature must be 50°F above RCS Cold Leg temperature
(2) 38%

D. (1) SG secondary side water temperature must be 50°F above RCS Cold Leg temperature
(2) 10%

Answer: B

K/A Match: This question matches the KA by requiring knowledge of surveillance requirements for the RCS, specifically when in Mode 5 with the RCS loops filled.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of Technical Specifications is required to correctly

answer the written test item; specifically: application of required actions and SRs in accordance with rules of application requirements as well as knowledge of TS bases that is required to analyze TS-required actions and terminology.

Explanation:

A. Incorrect: First part is correct, per TS & TSB of LCO 3.4.6 in order to consider an RCS loop in operation an RCP must be operating in that loop for forced circulation flow. Second part is incorrect but plausible, per SR 3.4.6.2 on Unit 2, the minimum level required on SG secondary side water level for operability of an RCS loop is 10%, however, on Unit 1, 38% level is required.

B. Correct: First part is correct, see A above. Second part is correct, per SR 3.4.6.2 on Unit 2, the minimum level required on SG secondary side water level for operability of an RCS loop is 10%.

C. Incorrect: First part is incorrect but plausible because per TS LCO 3.4.6 this would be the requirement that must be met in order to start an RCP in the loop in order to consider the loop in operation. Second part is incorrect but plausible, see A above.

D. Incorrect: First part is incorrect but plausible, see C above. Second part is correct, see B above.

Technical Reference(s): TS LCO 3.4.6, Amd 156
TSB LCO 3.4.6, Rev. 76

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Reactor Coolant system in accordance with Technical Specifications and TRM.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 2

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		2
		Group		2
Level of Difficulty:	4	**K/A**	011A2.11	
		Importance Rating		3.6

K/A Statement: Pressurizer Level Control: Ability to a) predict the impacts to the following malfunctions or operations on the PZR LCS; and b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Failure of PZR level instrument – low.

Question 92

Unit 2 plant conditions:

- 12% power during startup
- Controlling PRZR level instrument failed LOW
- Multiple alarms annunciated on CB-05 and CB-06

Based on the conditions above, complete the statements below.

1) Failure of the PRZR level instrument would cause ____(1)____.

2) Per ABN-706, Pressurizer Level Instrument Malfunction and TS LCO 3.3.1, RTS Instrumentation, a REQUIRED ACTION for the failure is to place the channel in trip within 72 hours OR ____(2)____ within 78 hours.

A. (1) flashing in letdown piping
   (2) reduce THERMAL POWER to < P-7

B. (1) flashing in letdown piping
   (2) be in MODE 3

C. (1) thermal shock to RCS piping
   (2) be in MODE 3

D. (1) thermal shock to RCS piping
   (2) reduce THERMAL POWER to < P-7

Answer: D

K/A Match: This question matches the KA by requiring the ability to use procedures and TSs to mitigate the failure of a PZR level instrument failing low.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the application of Technical Specifications is required to correctly answer the written test item; specifically: application of required TS actions and SRs in accordance with rules of application requirements.

Explanation:

A. Incorrect. First part is incorrect but plausible because if the PRZR level instrument failed high then charging flow would reduce automatically to maintain PRZR level. When charging flow reduces then letdown temperature in the Regenerative HX will increase and result in flashing in the Letdown piping. Second part is correct, with power greater than P-7 (10% power) then the PRZR level instrument Function 9 (Pressurizer Water Level – High) of TS 3.3.1 is applicable, however, if power is reduced below P-7 the function no longer applies. TS LCO 3.3.1 Condition M allows reduction below P-7 to where the function is not applicable OR to place the channel in trip.

B. Incorrect. First part is incorrect but plausible, see A above. Second part is incorrect but plausible because if this were a PRZR pressure instrument malfunction then TS LCO 3.3.1 Functions 8.a (Low) and 8.b (High) would apply. Function 8.a is only applicable above P-7 (same as Function 9 for level low). Those functions are governed by conditions M and E. Condition M allows reduction below P-7 to where the Function does not apply and Function E allows reduction to Mode 3. Both functions also allow the channels to be placed in trip.

C. Incorrect. First part is correct, when the PRZR level instrument fails LOW, then Letdown will automatically isolate at less than 17% level. As Letdown isolates charging will be automatically increasing due to the channel seeing a low level and attempting to raise the level. With Letdown isolated and charging continuing, the charging flow will not be pre-heated as it enters the RCS and will thermally shock the RCS piping. Second part is incorrect but plausible, see B above.

D. Correct. First part is correct, see C above. Second part is correct, see A above.

Technical Reference(s): ABN-706, Rev. 8 PCN-1
TS LCO 3.3.1, Amd 156 & 150
CVCS Lesson Plan, Rev 8/18/17
Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EVALUATE** plant performance and make operational judgments based on operating characteristics, reactor behavior and instrumentation while responding to a CVCS malfunction.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		2
		Group		2
Level of Difficulty:	3	**K/A**	034A1.02	
		Importance Rating		3.7

K/A Statement: Fuel Handling Equipment: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal.

Question 93

Unit 1 plant conditions:

- Unit 1 is shutdown for refueling with fuel offload in progress
- SFP Panel, Window 4.3 – RFL CAVITY 1 LEVEL LO is in alarm
- Refueling Cavity Level = 857'11" and lowering
- Spent Fuel Pool Level = 858'9" and rising
- ABN-909, Spent Fuel Pool/Refueling Cavity Malfunction is entered

Based on the above plant conditions, complete the following statements.

In accordance with ABN-909, when equalizing building pressures, …

1) the transfer tube gate valve will be ___(1)___.

2) the crew will use SOP-801A, Containment ventilation System, Section ___(2)___.

A. 1) left open
   2) 5.6.1, Containment Purge Supply AND Exhaust System Startup

B. 1) left open
   2) 5.6.5, Containment Pressure Relief System Operation

C. 1) closed
   2) 5.6.1, Containment Purge Supply AND Exhaust System Startup

D. 1) closed
   2) 5.6.5, Containment Pressure Relief System Operation

Answer: D

K/A Match: This question matches the KA by requiring the ability to monitor changes in spent fuel pool level to determine required actions.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and

external effects on core reactivity is being tested; specifically: administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths and/or activities.

Explanation:

A. Incorrect. 1st part is incorrect because the transfer tube gate will be closed when fuel handling is suspended. It is plausible because leaving them open when equalizing pressure would return the levels to normal as pressure was equalized (quicker). 2nd part is incorrect because you are directed to equalize pressure using Containment Pressure Relief OR Alternating primary plant supply and exhaust fans. It is plausible because operating the Containment Purge fans would lower containment pressure.

B. Incorrect. 1st part is incorrect but plausible (see A). 2nd part correct. ABN-909 directs you to use Containment Pressure Relief OR Alternating primary plant supply and exhaust fans.

C. Incorrect. 1st part is correct. The transfer tube is closed when suspending fuel movement and not opened again until pressures are approximately equal. 2nd part is incorrect but plausible (see A).

D. Correct. 1st part is correct (see C). 2nd part is correct (see B).

Technical Reference(s): Spent Fuel Pool Study Guide, Rev. 5/2/11
ABN-909, Rev. 8 PCN-7
SOP-801A, Rev. 14 PCN-5
ALM-0701, Rev. 6 PCN-1

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **DESCRIBE** the components of the Spent Fuel Pool Cooling and Cleanup system including interrelations with other systems to include interlocks and control loops

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

Question History: **Last NRC Exam** 2017 NRC Exam Q92

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content:
55.41
55.43 7

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		3
		Group		
Level of Difficulty:	4	**K/A**	G2.1.34	
		Importance Rating		3.5

K/A Statement: Knowledge of primary and secondary plant chemistry.

Question 94

Unit 2 plant conditions:

- 100% power
- At 1000 on June 16th, Chemistry reports RCS DOSE EQUIVALENT I-131 sample results for the past 4 hours:
  - 0600 – 1.85 µCi/gm
  - 0700 – 12.6 µCi/gm
  - 0800 – 53.2 µCi/gm
  - 0900 – 75.7 µCi/gm

Which of the following states the REQUIRED ACTION based on the Chemistry report?

A. Restore RCS DOSE EQUIVALENT I-131 to within limit by 0900 on June 18th or be in HOT STANDBY by 1500 on June 18th and in COLD SHUTDOWN by 2100 on June 19th

B. Restore RCS DOSE EQUIVALENT I-131 to within limit by 1000 on June 18th or be in HOT STANDBY by 1600 on June 18th and in COLD SHUTDOWN by 2200 on June 19th

C. Be in HOT STANDBY by 1600 on June 16th and in COLD SHUTDOWN by 2200 on June 17th

D. Be in HOT STANDBY by 1500 on June 16th and in COLD SHUTDOWN by 2100 on June 17th

Answer: D

K/A Match: This question matches the KA by requiring knowledge of primary chemistry limits.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions is being tested; specifically: analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.

Explanation:

A. Incorrect. Plausible if thought that 48 hours are available to restore DEI-131 to within limits before required entry into MODE 3, however because at 0900 DEI-131 is greater than 60µCi/gm MODE 3 must be entered in 6 hours and MODE 5 in 36 hours

B. Incorrect. Plausible if thought that 48 hours are available to restore DEI-131 to within limits before required entry into MODE 3, however because at 0900 DEI-131 is greater than 60µCi/gm MODE 3 must be entered in 6 hours. Also the start time for exceeding 60µCi/gm is 0900 not 1000 when reported.

C. Incorrect. Plausible because it could be thought that the clock for action starts at 1000 vice 0900.

D. Correct. Based on sample results at 0900 LCO 3.4.16 Condition C is entered and the unit must be in MODE 3 by 1500 on June 16th and in MODE 5 by 2100 on June 17th.

Technical Reference(s): TS 3.4.16, Amd 156
TS Mode Chart, Amd 150

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Reactor Coolant System including Technical Specifications, TRM and ODCM.

Question Source:
Bank # 2014 NRC Exam Q94
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		3
		Group		
Level of Difficulty:	2	**K/A**	G2.1.36	
		Importance Rating		4.0

K/A Statement: Knowledge of procedures and limitations involved in core alterations.

Question 95

_____ is considered a Core Alteration per RFO-102, Refueling Operation.

A. RCCA Unlatching

B. Upper Internals Installation

C. Incore Thimble Tube Removal

D. Reactor Vessel Head Removal

Answer: A

K/A Match: This question matches the KA by requiring knowledge of the requirements of the administrative procedures governing core alterations.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity is being tested; specifically: administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths and/or activities.

Explanation:

A. Correct. Per RFO-102 RCCA Unlatching is considered a Core Alteration.

B. Incorrect. Plausible because Upper Internals lift is a Core Alteration.

C. Incorrect. Plausible because RCCA latching and unlatching are Core Alterations.

D. Incorrect. Plausible because often thought of as a Core Alteration even though RFO-102 specifically states it is not.

Technical Reference(s): RFO-102, Rev. 13 PCN-20 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of conducting fuel handling

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 7

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		3
		Group		
Level of Difficulty:	3	**K/A**	G2.2.5	
		Importance Rating		3.2

K/A Statement: Knowledge of the process for making design or operating changes to the facility.

Question 96

In accordance with STA-707, 10CFR50.59 and 10CFR72.48 Reviews and STA-716, Modification Process:

1. The purpose of a 10CFR50.59 Evaluation is to determine if the proposed change requires ____(1)____.

2. System Turnover (Operations Acceptance) of a plant modification is complete when the work order status is ____(2)____.

A. (1) prior NRC approval via a license amendment
   (2) CLOSED

B. (1) prior NRC approval via a CoC (Certificate of Compliance) amendment
   (2) CLOSED

C. (1) prior NRC approval via a license amendment
   (2) ACOMP

D. (1) prior NRC approval via a CoC (Certificate of Compliance) amendment
   (2) ACOMP

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the 10CFR 50.59 process for plant modifications.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of facility licensee procedures required to obtain authority for design and operating changes in the facility is being tested; specifically: screening and evaluation process under 10 CFR 50.59, "Changes, Tests and Experiments."

Explanation:

A. Incorrect. 1st part is correct as STA-707 states that the 10CFR50.59 evaluation is required to determine if prior NRC approval via license amendment is required for a proposed change to the facility. 2nd part is incorrect but plausible because it could be thought that the work order for a modification must be closed to complete the turnover process, however once the work order is statused as ACOMP (filed work complete) it is accepted by Operations for use.

B. Incorrect. 1st part is incorrect but plausible because the purpose of a 10CFR72.48 evaluation is to determine if the proposed activity requires prior NRC approval via a CoC (Certificate of Compliance) amendment, however a 10CFR72.48 evaluation applies to dry cask storage only. 2nd part is incorrect (See A).

C. Correct: 1st part is correct (See A). 2nd part is correct (See A).

D. Incorrect. 1st part is incorrect (See B). 2nd part is correct (See A).

Technical Reference(s): STA-707, Rev. 21 PCN-1
STA-716, Rev. 26 PCN-13

Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Given that a plant design change is necessary, **DETAIL** an operator's responsibility in the design change process to include participating as a stakeholder, reviewing a design modification and accepting a design modification in accordance with STA-716.

Question Source:
Bank # 2015 NRC Exam Q95
Modified Bank # (note changes or attach parent)
New

Question History: **Last NRC Exam**

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehensive or Analysis

10CFR Part 55 Content:
55.41
55.43 3

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		3
		Group		
Level of Difficulty:	3	**K/A**	G2.2.17	
		Importance Rating		3.8

K/A Statement: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Question 97

In accordance with STI-606.01, Work Control Process, Figure 7.1, On-Line Prioritization Template, an emergent maintenance item classified as ORANGE requires work to begin _____.

A. within T-2 planning schedule

B. within T-5 planning schedule

C. immediately and work around the clock

D. within 24 hours and continue daily until completed

Answer: D

K/A Match: This question matches the KA by requiring knowledge of the process for performing maintenance at power including coordination with other groups.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the assessment and selection of procedures is required to correctly answer the written test item; specifically: knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Explanation:

A. Incorrect: Plausible because if the emergent maintenance item were classified as Yellow this could be the correct answer if directed by Ops Management.

B. Incorrect: Plausible because if the emergent maintenance item were classified as Yellow this could be the correct answer if directed by Ops Management.

C. Incorrect: Plausible because if the emergent item were classified as a Red emergent item STI-606.01 would require work to begin immediately and continue around the clock until complete.

D. Correct: When the emergent maintenance item is classified as Orange STI-606.01 requires work to begin within 24 hours and continue daily until completed

Technical Reference(s): STI-606.01, Rev. 6 PCN-1 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **EXPLAIN** the requirements for conducting briefs, reviewing and authorizing work documents and permits in accordance with STA-606, STI-606.01, STI-606.04, STI-605.01, STI- 604.03, STI-604.02, STA-696, STA-605 and Technical Specification 5.5.18.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:0		**Tier**		3
		Group		
Level of Difficulty:	3	**K/A**	G2.3.11	
		Importance Rating		4.3

K/A Statement: Ability to control radiation releases.

Question 98

Complete the statements below regarding control of radiation releases.

1) The ____(1)____ is responsible for ensuring all dose, dose rate, concentration and monitor setpoint calculations are performed for each batch pre-release required by ODCM

2) An automatic termination of a planned radiation release requires the permit to be ____(1)____ after the cause of the termination is resolved.

A. (1) Chemistry Manager
(2) revised and re-initiated

B. (1) Chemistry Manager
(2) closed and a new permit initiated

C. (1) RP Manager
(2) revised and re-initiated

D. (1) RP Manager
(2) closed and a new permit initiated

Answer: B

K/A Match: This question matches the KA by requiring the ability to minimize the magnitude of a radiation release.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions is being tested; specifically: process for gaseous/liquid release approvals (i.e. release permits).

Explanation:

A. Incorrect. First part is correct, per STA-603, Control of Radioactive Effluents, ensuring all dose, dose rate, concentration, and monitor setpoint calculations are performed for each batch pre-release is the responsibility of the Chemistry Manager. Second part is incorrect but plausible if thought that wrong setpoints were used in the permit prior to the release

being started. If this was true, the permit could be revised as long as the release had not been initiated. After the release has begun the permit must be closed and a new permit initiated.

B. Correct. First part correct, see A above. Second part is correct, after the release has begun the permit must be closed and a new permit initiated.

C. Incorrect. First part is incorrect but plausible because the RP Manager would normally be in charge of dose, dose rates, and rad monitor setpoint calculations. However, when referring to a release, STA-603 specifically states the Chemistry Manager maintains responsibility of these actions. Second part is incorrect but plausible, see A above.

D. Incorrect. First part is incorrect, see C above. Second part is correct, see B above.

Technical Reference(s): STA-603, Rev. 21 PCN-1 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **APPLY** the administrative requirements of the Liquid Waste systems including Technical Specification, TRM and ODCM.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 4

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		3
		Group		
Level of Difficulty:	3	**K/A**	G2.4.30	
		Importance Rating		4.1

K/A Statement: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, NRC or transmission system operator.

Question 99

Manual reactor trip initiated due to the inadvertent isolation of a FWIV.

In accordance with STA-501, Nonroutine Reporting, what is the maximum amount of time to report this event to the NRC?

A. Immediately

B. 1 hour

C. 4 hours

D. 8 hours

Answer: C

K/A Match: This question matches the KA by requiring knowledge of events that must be reported to outside agencies such as the NRC.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the assessment and selection of procedures is required to correctly answer the written test item; specifically: knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Explanation:

The requirement stated in STA-501 is to notify the NRC within 4 hours.

A. Incorrect: Plausible because there are requirements in STA-501 that require immediate notification.

B. Incorrect: Plausible because there are requirements in STA-501 that are within one hour.

C. Correct: The requirement stated in STA-501 is to notify the NRC within 4 hours.

D. Incorrect: Plausible because there are requirements in STA-501 that are within 8 hours.

Technical Reference(s): STA-501, Rev. 21 PCN-1 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: Given an event related to system operations/status, **CLASSIFY** which events must be reported to external agencies with respect to each category. 1hr, 4 hr, 8 hr.

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge** X
Comprehensive or Analysis

10CFR Part 55 Content: **55.41**
55.43 5

Examination Outline Cross-Reference:		**Level**	**RO**	**SRO**
Rev:1		**Tier**		3
		Group		
Level of Difficulty:	2	**K/A**	G2.4.21	
		Importance Rating		4.6

K/A Statement: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Question 100

The Unit is implementing an Optimal Recovery Guideline with the following conditions:

- Functional Recovery Guidelines can be implemented
- An ORANGE status is encountered on a CSF Status Tree

In accordance with ODA-407, Operations Department Procedure Use and Adherence, Attachment 8.A, ERG Rules of Usage, complete the following statement.

The US is required to assess all remaining CSFSTs and if no RED status exists, then…

A. perform ORG and FRG required by ORANGE status in parallel.

B. complete the ORG and then perform FRG required by ORANGE status.

C. complete the step in progress, suspend ORG and perform FRG required by ORANGE status.

D. immediately suspend ORG, do not complete the step in progress, and perform FRG required by ORANGE status.

Answer: C

K/A Match: This question matches the KA by requiring knowledge of the logic used to assess the status of safety functions and how to implement the function.

SRO Only: The applicant's knowledge is being evaluated at the SRO ONLY level in this question because additional knowledge of the assessment and selection of procedures is required to correctly answer the written test item; specifically: knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Explanation:

A. Incorrect. Plausible because there are other procedures performed in parallel in the ERG Network such as ABNs and Attachments of ERGs.

B. Incorrect. Plausible because while executing ORGs without a RED or ORANGE condition present the US would complete the ORG in progress, unless directed to transition to another ORG by the procedure in progress.

C. Correct. Per ODA-407, Attachment 8.A, if any ORANGE status is encountered, the US is expected to monitor all of the remaining CSFSTs, then if no RED status is encountered, suspend ORG in progress (complete the step in progress) AND perform the FRG required by the ORANGE status.

D. Incorrect. Plausible because if a RED status was encountered the US is required to immediately stop (do not complete the step in progress) any ORG in progress AND perform the FRG required by the RED status.

Technical Reference(s): ODA-407, Rev. 16 PCN-5 Attached w/ Revision # See Comments / Reference

References to be provided to applicants during exam: None

Learning Objective: **INTERPRET** and **ENSURE** Compliance with Plant Administrative and Operational Procedures, Guidelines, and Policies

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

Question History: **Last NRC Exam**

Question Cognitive Level: **Memory or Fundamental Knowledge**
Comprehensive or Analysis X

10CFR Part 55 Content: **55.41**
55.43 5

Comments / Reference: EOP-0.0A	**Revision:** 9

CPNPP EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE N EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 9	PAGE 25 OF 1

ATTACHMENT 2
PAGE 2 OF 8

SAFETY INJECTION ACTUATION ALIGNMENT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2	Verify Safety Injection Pumps RUNNING	☐ Manually start pump(s).
☐ 3	Verify Containment Isolation Phase A - APPROPRIATE MLB LIGHT INDICATION (RED WINDOWS)	☐ Manually actuate Phase A. ☐ IF any penetration NOT isola on at least one side, THEN manually align components as appropriate (Refer to Attachment 4 as necessary). ☐ IF any penetration NOT isola on both sides, THEN manually align components when time permits (Refer to Attachment as necessary).
☐ 4	Verify Containment Ventilation Isolation - APPROPRIATE MLB LIGHT INDICATION (GREEN WINDOWS)	☐ Manually actuate Containment ventilation isolation. ☐ IF any penetration NOT isola on at least one side, THEN manually align components as appropriate (Refer to Attachment 5 as necessary). ☐ IF any penetration NOT isola on both sides, THEN manually align components when time permits (Refer to Attachment as necessary).

ATTACHMENT 2
PAGE 3 OF 8

SAFETY INJECTION ACTUATION ALIGNMENT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5	Verify CCW Pumps - RUNNING	☐ Manually start pump(s).
☐ 6	Verify RHR Pumps - RUNNING	☐ Manually start pump(s).
7	Verify Proper CVCS Alignment:	
	☐ a. Verify CCPs - RUNNING	☐ a. Manually start pump(s).
	b. Verify Letdown Relief Valve isolation:	
	☐ 1) Letdown orifice isolation valves - CLOSED	☐ 1) Manually close valve(s).
	2) Letdown isolation valves - CLOSED	☐ 2) Manually close valve(s).
	☐ • 1/1-LCV-459	
	☐ • 1/1-LCV-460	
8	Verify ECCS flow:	
	☐ a. CCP SI flow indicator - CHECK FOR FLOW	☐ a. Manually align valves as appropriate.
	☐ b. RCS pressure - LESS THAN 1700 PSIG(1800 PSIG FOR ADVERSE CONTAINMENT)	☐ b. Go to Step 9 of this attachment.
	☐ c. SIP discharge flow indicators - CHECK FOR FLOW	☐ c. Manually align valves as appropriate.
	☐ d. RCS pressure - LESS THAN 325 PSIG(425 PSIG FOR ADVERSE CONTAINMENT)	☐ d. Go to Step 9 of this attachment.
	☐ e. RHR TO CL INJ flow indicators - CHECK FOR FLOW	☐ e. Manually align valves as appropriate.

BASES

7. CCPs provide makeup inventory to the RCS for core cooling during accide conditions. Since an SI has actuated, both CCPs have a start signal; therefore, the operator should verify that they are running. Train A S actuation generates a close signal to the letdown isolation valves (LCV 0459 & LCV-0460). The operator is instructed to verify isolation of th letdown isolation valves to ensure an inventory loss pathway is not set up through the letdown relief valve to the PRT.

8. ECCS flow is necessary to make up for the RCS inventory changes due to either RCS shrinkage or inventory losses from the RCS.

9. The main feedwater system is isolated on a FW Isolation signal to preve uncontrolled filling of any steam generator and the associated excessiv RCS cooldown which could aggravate the transient, especially if it were steamline break.

10. Diesel generators provide a source of emergency AC power to safeguards electrical busses in the event of a loss of offsite power and they receive a start signal as result of SI actuation. Since SI has actuated,the operator should verify that they are running.

11. Non-essential loads connected to the safeguards busses are de-energized upon SI actuation. Since SI has actuated, the operator should verify proper load shedding has occurred.

When the Diesel Generator is supplying power to a AC Safeguards bus, lo shedding ensures non-essential equipment is not loaded onto the bus in order to ensure that the Diesel Generator is not placed in an overloade condition. Therefore, if load shedding is not complete or cannot be verified while the DG is supplying power to an AC safeguards bus, actio should be taken to manually or locally complete the required shedding o loads from the affected bus(ses). If the DG is not supplying power to safeguards bus in which load shedding cannot be verified or is not complete, subsequent steps may be performed. However, actions to complete the load shedding should be pursued as time allows.

ATTACHMENT 4
PAGE 1 OF 6

PHASE A ISOLATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	ESFAS TRAIN	MLB LOCATION
☐ CB-01	1-HS-5366(1)	CNTMT DEMIN WTR ISOL VLV	A	1-MLB-1A1/3.4
☐ CB-01	1-HS-5365(1)	CNTMT DEMIN WTR ISOL VLV	B	1-MLB-1B1/3.4
☐ CB-01	1-HS-3487(1)	CNTMT INSTR AIR ISOL VLV	B	1-MLB-4B2/3.6
☐ CB-01	1-HS-3486(1)	CNTMT SERV AIR ISOL VLV	B	1-MLB-4B2/3.6
☐ CB-01	1-HS-5158(1)	CNTMT SMP DRN ISOL VLV	A	1-MLB-4A1/4.9
☐ CB-01	1-HS-5157(1)	CNTMT SMP DRN ISOL VLV	B	1-MLB-4B1/4.9
☐ CB-02	1/1-8823(3)	SI TO CL 1 • 4 TEST ISOL VLV	A	1-MLB-1A2/3.8
☐ CB-02	1/1-8888(1)	ACCUM FILL ISOL VLV	B	1-MLB-1B2/3.7
☐ CB-02	1/1-8881(3)	SI TO HL 2 & 3 TEST ISOL VLV	A	1-MLB-1A2/3.7
☐ CB-02	1/1-8824(3)	SI TO HL 1 & 4 TEST ISOL VLV	A	1-MLB-1A2/4.8
☐ CB-02	1/1-8871(1)	SI TEST HDR RET ISOL VLV	A	1-MLB-1A2/2.7
☐ CB-02	1/1-8964(1)	SI TEST HDR RET ISOL VLV	B	1-MLB-1B2/2.7
☐ CB-02	1-HS-4165A	PSS ISOL VLV	A	(NOTE 1)
☐ CB-02	1-HS-4167A	PSS ISOL VLV	B	(NOTE 1)
☐ CB-03	1-HS-4075C(1)	CNTMT FIRE PROT ISOL VLV	A	1-MLB-1A2/2.5
☐ CB-03	1-HS-4075B(1)	CNTMT FIRE PROT ISOL VLV	B	1-MLB-1B2/2.5
☐ CB-03	1-HS-6082(1)	CH WTR RET ISOL VLV	B	1-MLB-45B/1.5
☐ CB-03	1-HS-6083(1)	CH WTR RET ISOL VLV	A	1-MLB-45A/1.5
☐ CB-03	1-HS-6084(1)	CH WTR SPLY ISOL VLV	B	1-MLB-45B/2.5
☐ CB-03	1-HS-5544(1)	RAD MON IN ISOL VLV	B	1-MLB-45B/2.2
☐ CB-03	1-HS-5545(1)	RAD MON IN ISOL VLV	A	1-MLB-45A/2.2

ATTACHMENT 5
PAGE 1 OF 1

CONTAINMENT VENTILATION ISOLATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	ESFAS TRAIN	MLB LOCATION
☐ CB-03	*1-HS-5537	AIR PRG SPLY ISOL DMPR (IRC)	A	1-MLB-45A/1.1
☐ CB-03	*1-HS-5539	AIR PRG EXH ISOL DMPR (IRC)	A	1-MLB-45A/1.2
☐ CB-03	1-HS-5549	CNTMT PRESS RLF ISOL VLV (IRC)	A	1-MLB-45A/2.1
☐ CB-03	*1-HS-5543	H2 PRG SPLY ISOL DMPR (IRC)	A	1-MLB-45A/4.1
☐ CB-03	*1-HS-5541	H2 PRG EXH ISOL DMPR (IRC)	A	1-MLB-45A/4.2
☐ CB-03	*1-HS-5536	AIR PRG SPLY ISOL DMPR (ORC)	B	1-MLB-45B/1.1
☐ CB-03	*1-HS-5538	AIR PRG EXH ISOL DMPR (ORC)	B	1-MLB-45B/1.2
☐ CB-03	1-HS-5548	CNTMT PRESS RLF ISOL VLV (ORC)	B	1-MLB-45B/2.1
☐ CB-03	*1-HS-5542	H2 PRG SPLY ISOL DMPR (ORC)	B	1-MLB-45B/4.1
☐ CB-03	*1-HS-5540	H2 PRG EXH ISOL DMPR (ORC)	B	1-MLB-45B/4.2
☐ CB-03	*1-HS-5563	H2 PRG SPLY ISOL DMPR (IRC)	B	1-MLB-45B/4.3
☐ CB-03	*1-HS-5562	H2 PRG EXH ISOL DMPR (IRC)	B	1-MLB-45B/4.4

☐ Notify Unit Supervisor attachment instructions complete AND identify Containment Ventilation Isolation status.

* Damper closed and de-energized in Modes 1-4. MLB indication should be used for valve position.

ATTACHMENT 6
PAGE 2 OF 2

CONTAINMENT SPRAY/PHASE B ISOLATION

COMPONENT LOCATION	EQUIPMENT NUMBER	DESCRIPTION	POSITION	ESFAS TRAIN	MLB LOCATION
☐ CB-03	1-HS-4515	SFGD LOOP CCW SPLY VLV	CLOSED	B	1-MLB-4B3/1.8
☐ CB-03	1-HS-4575	CS HX 2 CCW RET VLV	PARTIALLY OPEN, MLB LIT	B	1-MLB-4B3/3.7
☐ CB-03	1-HS-4573	RHR HX 2 CCW RET VLV	PARTIALLY OPEN, MLB LIT	B	1-MLB-4B3/1.7
☐ CB-03	1-HS-4513	SFGD LOOP CCW RET VLV	CLOSED	B	1-MLB-4B3/2.8
☐ CB-03	1-HS-4701	RCP CLR CCW RET ISOL VLV	CLOSED	A	1-MLB-4A3/4.9
☐ CB-03	1-HS-4708	RCP CLR CCW RET ISOL VLV	CLOSED	B	1-MLB-4B3/4.9
☐ CB-03	1-HS-4700	RCP/THBR CLR CCW SPLY ISOL VLV	CLOSED	B	1-MLB-4B3/3.9
☐ CB-03	1-HS-4709	THBR CLR CCW RET ISOL VLV	CLOSED	B	1-MLB-4B3/2.9
☐ CB-03	1-HS-4699	RCP/THBR CLR CCW SPLY ISOL VLV	CLOSED	A	1-MLB-4A3/3.9
☐ ~~CB-03~~	~~1-HS-4696~~	~~THBR CLR CCW RET ISOL VLV~~	~~CLOSED~~	~~A~~	~~1-MLB-4A3/2.9~~

☐ Notify Unit Supervisor attachment instructions complete AND identify Containment Spray/Phase B Isolation alignment status.

Comments / Reference: 2014 NRC Exam Q48	**Revision:**

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 3/3/2014	Tier	2	
Change: 2	Group	1	
	K/A	103 A3.01	
Level of Difficulty: 3	Importance Rating	3.9	

Containment System: Ability to monitor automatic operation of the Containment system, including: Containment isolation

Proposed Question: 48

Given the following conditions:

- Unit 1 is responding to a Large Break Loss of Coolant Accident in accordance with EOP-0.0A, Reactor Trip or Safety Injection.
- While verifying Containment Isolation Phase A, the Balance of Plant Operator determines that automatic and manual actuation of Containment Isolation Phase A will NOT function.

Which of the following valves must be closed as part of Containment Isolation Phase A?

A. 1/1-8153, XS LTDN ISOL VLV.
1/1-8154, XS LTDN ISOL VLV.

B. 1/1-LCV-0459, U1 LTDN ISOL VLV 0459.
1/1-LCV-0460, U1 LTDN ISOL VLV 0460.

C. 1-HV-6082, CH WTR RET ISOL VLV.
1-HV-6084, CH WTR SPLY ISOL VLV.

D. 1-HV-4650, VENT CHLR CCW SPLY & RET VLV.
1-HV-4631, PSC CCW SPLY & RET VLV.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because 1/1-8152 & 1/1-8160, LTDN ISOL VLVs are part of a Phase A Containment Isolation Signal, however, the Excess Letdown Isolation Valves are located inside Containment and are isolated via the Seal Water Return lines.
B. Incorrect. Plausible because these valves will close, however, the actuation signal is a Safety Injection slave relay.
C. Correct. The Chill Water Return and Supply Isolation Valves are isolated on a Phase A Containment Isolation Signal.
D. Incorrect. Plausible because these valves do receive a close signal, however, it is generated from a Safety Injection Actuation Signal.

ES-401 2014 RO Written Exam Worksheet Form ES-401-5

Technical Reference(s) EOP-0.0A, Attachments 2, 4, 7, & 10 Attached w/ Revision: See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Containment System and **PREDICT** the system response.

Question Source: Bank ILOT7344
Modified Bank _____ (Note changes or attach parent)
New _____

Question History: Last NRC Exam _____

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

10 CFR Part 55 Content: 55.41 9
55.43 _____