Exelon Generation

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VISION Report

(Test Key) 2020 ILT NRC RO EXAM

EXELON Nuclear
CCNPP Operations NRC Examinations

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Test	2020 ILT NRC RO EXAM
VISION ID	329627
Status	

Question 1	ID: 2113876	Points: 1.00

Unit-1 and Unit-2 are at 100% power:

At time 10:00:

- An issue with the grid occurs
- Unit-1 experiences a loss of Main Feed ONLY
- Unit-2 experiences a loss of Main Feed and the Reactor Coolant Pumps

Which one of the following describes:

(1) Which Unit will require the highest amount of Auxiliary Feedwater flow to maintain steady Steam Generator levels 5 minutes following the reactor trip based on total heat load?
And,
(2) At what time will the highest amount of Auxiliary Feedwater flow be required to maintain steady Steam Generator levels?

A. (1) Unit-1
(2) At time 10:05

B. (1) Unit-2
(2) At time 10:05

C. (1) Unit-2
(2) At time 11:00

D. (1) Unit-1
(2) At time 11:00

Answer	A

Answer Explanation

A. Correct. (1) Correct per USFAR, a loss of just Main Feed will require more AFW flow to stabilize Steam Generator levels based on decay heat and RCP heat. (2) Correct since the highest decay heat will be 5 minutes after the reactor trip and will continue to lower afterward.

B. Incorrect. (1) Incorrect but plausible if the operator thinks that a loss of MFW and RCPs will cause a significant rise in RCS Temperature as natural circulation is developed and there will be more heat load in the S/G requiring more AFW Flow. (2) Correct as stated above.

C. Incorrect. (1) Incorrect but plausible as stated above. (2) Incorrect but plausible since the operator may conclude that the additional time necessary for natural circulation to develop will result in a higher RCS T_{HOT} temperature creating additional steam flow from the Steam Generators to maintain RCS T_{COLD} constant.

D. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated above.

Question Information

Topic	Emergency Feedwater flow vs decay heat		
User ID	Q2113876		
System ID	2113876	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only

Question Type	New	**Difficulty**	3
Technical Reference and Revision #	UFSAR, Chapter 14.6 Rev 49		
Training Objective	State the purpose of the AFW system.		
Previous NRC Exam Use	None		

K/A Links

EPE.007.EK1.06	Safety Function 1	Tier 1	Group 1	RO Imp: 3.7	SRO Imp: 4.1
Knowledge of the operational implications of the following concepts as they apply to the reactor trip: (CFR 41.8 / 41.10 / 45.3) Relationship of emergency feedwater flow to S/G and decay heat removal following reactor trip					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.41.04, CFR: 41.4	Secondary coolant and auxiliary systems that affect the facility.

Question 2	ID: 2113887	Points: 1.00

Unit-2 is at 30% power MOC when a load rejection occurs.

- RCS pressure rises to 2420 PSIA

The following conditions exist:

- Acoustic Monitor indicates flow through a PORV
- RCS pressure is 2185 PSIA and lowering
- Pressurizer level is 190 inches and rising

Which of the following lists the actions directed by EOP-0 for regaining control of Pressurizer level and pressure?

A. Lower RCS Pressure to less than 1800 PSIA;
Place PORV Override handswitches in the "Override To Close" position.

B. Shut PORV Block valves;
Start all available Charging pumps.

C. Lower RCS Pressure to less than 1800 PSIA;
Start all available Charging pumps.

D. Shut PORV Block valves;
Place PORV Override handswitches in the "Override To Close" position.

Answer	D

Answer Explanation

A. Incorrect. 1st part is Incorrect since lowering RCS pressure to 1800 PSIA would be a deviation to EOP-0, Post Trip Immediate Actions. Plausible because this is an action, directed by EOP-5 Loss of Coolant Accident, designed to reseat a leaking Pressurizer Safety valve. 2nd Part is correct as directed by EOP-0.

B. Incorrect. 1st part is correct as directed by EOP-0. 2nd part is Incorrect but plausible since starting all available Charging Pumps with Pressurizer level at 190 inches would help offset the inventory loss due to the leaking PORV but would be a deviation to EOP-0, Post Trip Immediate Actions.

C. Incorrect. 1st part is Incorrect since lowering RCS pressure to 1800 PSIA would be a deviation to EOP-0, Post Trip Immediate Actions. Plausible because this is an action, directed by EOP-5 Loss of Coolant Accident, designed to reseat a leaking Pressurizer Safety valve. 2nd part is Incorrect but plausible since starting all available Charging Pumps with Pressurizer level at 190 inches would help offset the inventory loss due to the leaking PORV but would be a deviation to EOP-0, Post Trip Immediate Actions.

D. Correct. 1st and 2nd parts are Correct. With pressure lowering due to PORV leakage, the PORV block MOV must be verified closed, and the PORV Override HS must be placed in “Override to Close”.

Question Information

Topic	Controlling PZR Level with a leaking PORV		
User ID	Q2113887		
System ID	2113887	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-00-2, Rev 01400		

Training Objective	For each EOP-0 Safety Function, analyze and determine correct alternate actions to take.
Previous NRC Exam Use	2010 NRC RO EXAM

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

APE.008.AA1.06	Safety Function 3	Tier 1	Group 1	RO Imp: 3.6	SRO Imp: 3.6
Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: (CFR 41.7 / 45.5 / 45.6) Control of PZR level					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 3	ID: 2121104	Points: 1.00

Given the following conditions on Unit-1:

- A Loss of Coolant Accident has occurred
- EOP-5, LOCA has been initiated
- 11A and 12B RCPs are running
- RCS depressurization is in progress

Which one of the following describes:

(1) The conditions that must be met before HPSI flow can be throttled?
And,
(2) Why?

A. (1) Pressurizer level > 101 inches
(2) Both Steam Generators remain available for heat removal

B. (1) RCS subcooling > 25°F
(2) Both Steam Generators remain available for heat removal

C. (1) Pressurizer level > 101 inches
(2) Pressurizer heaters remain available for pressure control

D. (1) RCS subcooling > 25°F
(2) Pressurizer heaters remain available for pressure control

Answer	C

Answer Explanation

A. INCORRECT - (1) Correct Level as stated below. (2) Incorrect but plausible since at least one Steam Generator is required to be available in order to meet HPSI throttling criteria per EOP-5 Block Step L.

B. INCORRECT - (1) Incorrect but plausible since the operator will recall that this value is correct in EOP-6 which is the correct procedure for a different type of RCS leakage. (2) Incorrect but plausible as stated above.

C. CORRECT - (1) Correct since 101" is correct value for PZR level as stated in EOP-5 Block Step L. (2) Correct reason per EOP-5 basis which states in Block Step L this value is based on the low level heater trip to ensure that the heaters remain available for pressure control. HPSI throttle criteria in EOP-5 are at least 30 degrees subcooling, pressurizer level >101", at least one S/G available for heat removal. RCS level above hot leg

D. INCORRECT - (1) Incorrect but plausible as stated above. (2) Correct as stated above.

Question Information

Topic	Criteria and reason for HPSI throttling criteria		
User ID	Q2121104		
System ID	2121104	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-5-TB Rev 03300		
Training Objective	Given RCS parameters identify appropriate Safety Injection System response.		
Previous NRC Exam Use	None		

References Provided	None

K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

EPE.009.EK3.24	Safety Function 3	Tier 1	Group 1	RO Imp: 4.1	SRO Imp: 4.6
Knowledge of the reasons for the following responses as the apply to the small break LOCA: (CFR 41.5 / 41.10 / 45.6 / 45.13) ECCS throttling or termination criteria					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 4	ID: 2121162	Points: 1.00

At time 10:00:
- A Design Basis Large Break LOCA occurs on Unit-1

At time 11:00:
- Containment Pressure is 5 psig and slowly lowering
- RWT Level is 42 inches actual (23 inches indicated) and slowly lowering

Which one of the following completes the statements below?

At time 11:00:
Low Pressure Safety Injection (LPSI) Pumps are ___(1)___,
And
Containment Spray (CS) Pumps are ___(2)___.

A. (1) running
(2) running

B. (1) **NOT** running
(2) running

C. (1) **NOT** running
(2) **NOT** running

D. (1) running
(2) **NOT** running

Answer	B

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the operator may misinterpret the actions taken prior to RAS and may conclude that the HPSI Pumps are secured instead. (2) Correct as stated below.

B. Correct. (1) Correct per EOP Attachment 6, Both LPSI pumps go to OFF for RAS. Per EOP-5, Block Step S, the LPSI Pumps are placed in pull to lock prior to RAS at an

RWT level of 63 inches. (2) Correct per EOP-5 Block Step S which only directs the CS Pumps to be secured if CSAS has not actuated. The question stem gives a Containment Pressure of 5 psig which exceeds the CSAS setpoint of 4.25 psig so CSAS will not be able to be reset under the provided conditions.

C. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible since EOP-5 Block Step directs to place both CS Pumps in pull to lock if CSAS has not actuated. The operator may not recall the correct CSAS setpoint or may not recall when CSAS can be reset and would conclude that the CS Pumps are no longer running. Also, plausible if the operator misinterprets which pumps are secured upon a RAS actuation and may conclude the CS Pumps are secured by the RAS actuation.

D. Incorrect. (1) Incorrect but plausible as stated above. (2) Incorrect but plausible as stated above.

Question Information

Topic	Identify the effect of RAS on the LPSI and CS Pumps		
User ID	Q2121162		
System ID	2121162	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-Attachments, Rev 20 EOP-5-1 Rev 03001 1C08-ALM, Rev 03600		
Training Objective	Given a set of conditions recall how the ECCS would respond during the first hour of a small or large break LOCA.		

Previous NRC Exam Use	None

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

EPE.011.EK2.02	Safety Function 3	Tier 1	Group 1	RO Imp: 2.6*	SRO Imp: 2.7*
Knowledge of the interrelations between the and the following Large Break LOCA: (CFR 41.7 / 45.7) Pumps					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 5	ID: 2113902	Points: 1.00

Unit 1 is operating at 100% power when the following occurs:

- A fault occurs on P-13000-1

During the performance of reactivity control in EOP-0, Post Trip Immediate Actions, the RO notes that Q-power is reading ~45%.

Which of the following correctly describes the cause of this Q-power reading?

A. The TBVs remain closed due to a loss of vacuum resulting in higher RCS temperatures, which allows for greater neutron leakage. This larger neutron leakage is interpreted by RPS as a higher Q-power.

B. The TBVs fully open due to the loss of P-13000-1 resulting in lower RCS temperatures which adds positive reactivity. This positive reactivity is interpreted by RPS as a higher Q-power.

C. The difference between Th and Tc will grow as natural circulation is established and will cause delta-T power to rise. This difference in temperatures is interpreted in RPS as a higher Q-power.

D. The loss of P-13000-1 has resulted in a loss of power to the Th and Tc instruments which will result in a higher than normal delta-T power. This loss of power is interpreted in RPS has a higher Q-power.

Answer	C

Answer Explanation

A. Incorrect. Plausible since the TBVs will not open on a loss of vacuum and will result in slightly higher RCS temperatures but this rise is not great enough to cause nuclear power to rise to ~40%. A loss of vacuum is plausible on a P-13000-1 loss since power will be lost to the CARs and CWPs and eventually vacuum would lower.

B. Incorrect. Plausible since the operator will conclude on a loss of power that the TBV quick open signal will occur and not clear upon the reactor trip and will fully open the TBVs and cause RCS temperature to lower and add positive reactivity enough to cause nuclear power to rise to ~40%.

C. Correct. Q = M Cp (Th - Tc) = U A (Tc - Tsg); As natural circulation is established, delta-T must rise to accommodate the heat transfer from the RCS to SG and establish natural circulation flow. RPS will translate the rise in delta-T as a rise in delta-T power. Q-power is the auctioneered largest value of NI or delta-T power. NI power has already dropped off to the intermediate range; therefore, delta-T power will be seen as Q-power.

D. Incorrect. Plausible since on a loss of power to the Tc and Th instruments they will read incorrectly which would impact the delta-T power indications, but they have not lost power in this situation.

Question Information

Topic	Operational implications of Natural Circulation		
User ID	Q2113902		
System ID	2113902	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	The question matches the K/A since it provides the operational implication of a higher Q-power reading caused by natural circulation.
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only

Question Type	New	**Difficulty**	3
Technical Reference and Revision #	USFAR Section 7.2 (page 7.2-7, figure 7-21)		
Training Objective	From memory, recall the Automatic RPS trips, including their basis, setpoints, indications, and any associated interlocks, in accordance with the UFSAR, Tech Specs, and 1C05 Alarm Manual		
Previous NRC Exam Use	None		

K/A Links

APE.015/017.AK1.01	Safety Function 4	Tier 1	Group 1	RO Imp: 4.4	SRO Imp: 4.6
Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): (CFR 41.8 / 41.10 / 45.3) Natural circulation in a nuclear reactor power plant					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 6	ID: 2113926	Points: 1.00

Unit 2 is operating at 100% power when the following occurs:

- A charging header leak of 50 gpm develops downstream of the charging pumps
- Only one Charging Pump is running

Which one of the following describes the impact on:

(1) Charging pressure?
And,
(2) Charging flow?

A. (1) Charging header pressure is greater than RCS Pressure
(2) Charging flow increases

B. (1) Charging header pressure is greater than RCS Pressure
(2) Charging flow remains the same

C. (1) Charging header pressure is less than RCS Pressure
(2) Charging flow increases

D. (1) Charging header pressure is less than RCS Pressure
(2) Charging flow remains the same

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible if 50gpm is misinterpreted to be within the capacity of a charging pump since RCS pressure in this situation would be unchanged. (2) Plausible if charging flow is misinterpreted to increase as the pressure differential between charging and RCS changes since this is how most of the other pumps in the plant operate.

B. Incorrect. (1) See above. (2) Correct.

C. Incorrect. (1) Correct. (2) See above.

D. Correct. (1) Per AOP-2A, a leak (greater than the capacity of a charging pump) on the charging header will cause Charging header pressure to be less than RCS Pressure. (2) Charging pumps are positive displacement pumps and the change in D/P between charging and RCS will not impact flow.

Question Information

Topic	Charging Flow changes after a charging header leak		
User ID	Q2113926		
System ID	2113926	**Point Value**	1.00
Status	Active	**Time to Complete**	0
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	AOP-2A-2, Rev 02400		
Training Objective	From memory, describe the operation of the following in accordance with the Operating Instructions, Alarm Manuals,		

	EOPs and AOPs: Charging Pumps.
Previous NRC Exam Use	None

K/A Links

APE.022.AK1.02	Safety Function 2	Tier 1	Group 1	RO Imp: 2.7	SRO Imp: 3.1
Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Pump Makeup: CFR 41.8 / 41.10 / 45.3) Relationship of charging flow to pressure differential between charging and RCS					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 7	ID: 2113947	Points: 1.00

Unit-2 is in Mode 4 at 230°F with Shutdown Cooling in Service when the following occurs:

- RCS level is lowering
- AOP-3B, Abnormal Shutdown Cooling Conditions, is being implemented
- Containment Temperature is 85°F and steady
- Containment Humidity is 38% and steady

AOP-3B Block Step F, Attempt to Isolate the Leak, is performed: both LPSI Pumps are stopped and 2-SI-651-MOV and 2-SI-652-MOV are shut.

- RCS level stops lowering
- Containment parameters remain unchanged

Which of the following describes:

(1) Is the leak in containment?
And
(2) Is the leak on the SDC system?

A. (1) Yes
(2) Yes

B. (1) Yes
(2) No

C. (1) No
(2) Yes

D. (1) No
(2) No

Answer	C

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the operator may misinterpret the unchanged containment parameters to be caused by an open Containment Outage Door and/or an open Personnel Air Lock door. (2) Correct as stated below.

B. Incorrect. (1) Incorrect but plausible as stated above. (2) Incorrect but plausible since the operator may interpret the leak to be on the RCS or Safety Injection headers and that stopping the LPSI pumps has stopped the flowpath through the headers where the leak may exist.

C. Correct. (1) Correct since the leak cannot be inside Containment with unchanging Containment temperature and humidity with a starting RCS temperature of 230°F in Mode 4. (2) Correct since the RCS initially lowered until 2-SI-651-MOV and 2-SI-652-MOV are shut and then the RCS level stops lowering.

D. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated above.

Question Information

Topic	Loss of SDC due to a leak		
User ID	Q2113947		
System ID	2113947	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	3
Technical Reference and Revision #	AOP-3B-2, Rev 02800 SL-074, 64311, Rev 14		
Training Objective	Given plant conditions resulting in a loss of shutdown cooling, determine the required actions to maintain plant parameters within desired limits IAW with AOP-3B.		
Previous NRC Exam Use	None		

K/A Links

APE.025.AA2.04	Safety Function 4	Tier 1	Group 1	RO Imp: 3.3*	SRO Imp: 3.6
Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: (CFR: 43.5 / 45.13) Location and isolability of leaks					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 8	ID: 2113950	Points: 1.00

Unit-1 is operating in Mode 5:

- SDC is in service
- SDC purification is in service

The following events occur in sequence:

- A CCW component develops a leak
- RE-3819 (component cooling radiation detector) alarms on 1C22
- Operators subsequently isolate the leaking CCW component

Which one of the following statements correctly describes:

(1) The location of the leak in the CCW system?

And

(2) The level response in the CCW head tank?

A. (1) The SDC heat exchanger
(2) CC head tank level rises

B. (1) The SDC heat exchanger
(2) CC head tank level lowers

C. (1) The letdown heat exchanger
(2) CC head tank level rises

D. (1) The letdown heat exchanger
(2) CC head tank level lowers

Answer	A

Answer Explanation

A. Correct. (1) Correct per the current plant lineup with SDC purification in service the cause of the high rad alarm will be a leaking SDC heat exchanger. (2) Correct since CCW pressure is ~ 75 psig. SDC pressure is ~ 150 psia. Initial RCS leakage from the SDC HXs will be into, not out of the CCW system. The CCW head tank will fill and overflow.

B. Incorrect. (1) Correct. (2) See Below.

C. Incorrect. (1) See Below. (2) Correct.

D. Incorrect. (1) Plausible if the SDC purification lineup is misinterpreted to use the letdown heat exchanger which is at a higher pressure than CCW. (2) Plausible if the operator misinterprets the requirements of RCS pressure at atmospheric pressure in order to place SDC purification in service. Then, the operator will conclude that CCW pressure is ~ 75 psig and SDC pressure is ~ 15 psia and the leakage will be from CC into the HXs. The CCW head tank level will decrease.

Question Information

Topic	Determine the most likely source of leakage into the CCWs		
User ID	Q2113950		
System ID	2113950	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-3B-1, Rev 03001 1C13, Rev 05500 1C22, Rev 04400		

Training Objective	Given a set of plant conditions related to the CCW system, identify the most likely cause and select the correct actions in accordance with the Alarm manual, Operating Instructions, an Abnormal Operating Procedures.
Previous NRC Exam Use	2006 NRC RO Exam

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

K/A Links

APE.026.AA2.01	Safety Function 8	Tier 1	Group 1	RO Imp: 2.9	SRO Imp: 3.5
Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: (CFR: 43.5 / 45.13) Location of a leak in the CCWS					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 9	ID: 2113952	Points: 1.00

Unit 1 is at 100% power when the following occurs:

- 11 4Kv Bus Normal Feeder breaker trips open
- The 11 4KV Bus is repowered with its associated EDG
- No operator actions are performed

Then, the in-service Pressurizer Pressure Controller (PIC-100) fails, resulting in:

- Controller output rising to 75%
- Main Spray Valves opening

Which of the following describes the Proportional Heaters Response to this event without operator action?

A. Bank 1 auto-energizes
Bank 2 output goes to minimum

B. Bank 1 remains deenergized
Bank 2 output goes to minimum

C. Bank 1 auto-energizes
Bank 2 output goes to maximum

D. Bank 1 remains deenergized
Bank 2 output goes to maximum

Answer	B

Answer Explanation

A. Incorrect. (1) See below (2) Correct.

B. Correct. (1) #11 Proportional Heater will remain de-energized due to the momentary loss of 11 4Kv Bus, until its handswitch is manually reset. (2) As PIC output goes to 75%, #12 Proportional Heater output will go to zero since its handswitch has remained in its normal position of Auto.

C. Incorrect. (1) Plausible if the proportional heater control circuit is thought to auto start when heaters are required due to lowering pressurizer pressure. When the 4KV bus is re-powered from the EDG the handswitch for the proportional heater has to be manually reset. (2) Plausible since normally when pressurizer pressure lowers the proportional heaters will go to maximum output but not for this failure mode

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Proportional Heater Response		
User ID	Q2113952		
System ID	2113952	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	USFAR, Chapter 7 (figure 7-13), Rev 50 AOP-7I, Rev 03400		

Training Objective	From memory, recall the operation and basis of the following in relation to the RCS in accordance with the Tech Specs, UFSAR and Operating Procedures: PZR Level/Pressure Control.
Previous NRC Exam Use	None

K/A Links

APE.027.AA1.02	Safety Function 3	Tier 1	Group 1	RO Imp: 3.1*	SRO Imp: 3.0
Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: (CFR 41.7 / 45.5 / 45.6) SCR-controlled heaters in manual mode					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 10	ID: 2113959	Points: 1.00

Unit 1 is at 100% power when an ATWS occurs.

What action is required per EOP-0 Post-Trip Immediate Actions?

A. Manually open the trip circuit breakers in the Cable Spreading Room.

B. Manually insert all CEAs using "Manual Sequential" mode.

C. Borate the RCS to at least 2000 ppm.

D. De-energize 12A and 13A 480 volt buses.

Answer	D

Answer Explanation

A. Incorrect. Plausible since this action would trip the reactor and mitigate the ATWS condition but it is not the correct action per the procedure.

B. Incorrect. Plausible since this action would shutdown the reactor and mitigate the ATWS condition but it is not the correct action per the procedure.

C. Incorrect. Plausible since borating is directed in EOP-0 when more than one CEA fails to fully insert into the reactor core but to 2300ppm not 2000ppm (which is the required boron for the spent fuel pool). The operator may conclude that an ATWS represents a condition where more than one CEA fails to insert.

D. Correct. Per EOP-0, de-energizing 12A and 13A 480 volt buses is the alternate action if the reactor does not trip as expected. This removes power from the CEDM Motor Generator Sets and is independent of RPS actuation.

Question Information

Topic	Action required upon recognition of ATWS		
User ID	Q2113959		
System ID	2113959	Point Value	1.00
Status	Active	Time to Complete	1
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	2
Technical Reference and Revision #	EOP-0-1, Rev 01300		
Training Objective	For each EOP-0 Safety Function analyze and determine correct alternate actions to take.		
Previous NRC Exam Use	2002 RO NRC Exam		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

K/A Links

P2.4.6	Safety Function 3	Tier 3	Group	RO Imp: 3.7	SRO Imp: 4.7
Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)					
GE.4.0.EPE.029	Safety Function 1	Tier 1	Group 1	RO Imp:	SRO Imp:
Anticipated Transient Without Scram (ATWS)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 11	ID: 2113965	Points: 1.00

Unit 1 is at 100% power when the following occurs:

- S/G Tube Rupture
- All RCPs have been tripped

Which one of the following is the reason for maintaining RCS subcooling as low as possible during plant cooldown and depressurization?

A. Prevents exceeding reactor vessel pressurized thermal shock (PTS) limits.

B. Reduces the differential pressure between the RCS and S/G thus minimizing the leak rate.

C. Prevents drawing a bubble in the reactor vessel head during the plant cooldown to isolate the affected SG.

D. Enhances natural circulation flow when RCPs are secured during plant cooldown to isolate the affected SG.

Answer	B

Answer Explanation

A. Incorrect. Plausible since this is the basis for the high limit of the subcooling band and the candidate misinterprets this information.

B. Correct. Per EOP-6 technical basis, subcooling greater than 25°F will minimize the loss of primary fluid to the secondary side.

C. Incorrect. Plausible since the operator will recall that natural circulation flow can be interrupted by a bubble in the reactor vessel head. The operator may misinterpret maintaining low subcooling to help with preventing drawing a head bubble, in fact, it will allow a head bubble to form earlier than maintaining Subcooling high in the band.

D. Incorrect. Plausible if maintaining low subcooling is misinterpreted to enhance natural circ since it will have higher temperatures which could aid in the temperature differential needed for natural circulation.

Question Information

Topic	EOP-6; reason for maintaining SCM low in band during C/D & depressurization		
User ID	Q2113965		
System ID	2113965	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	2
Technical Reference and Revision #	EOP-6-TB, Rev 02100		
Training Objective	From memory, recall the strategy and the basis for the major actions performed in EOP-6, SGTR, and what actions are required if safety functions are in jeopardy of being lost.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A

Additional Information	No additional information

K/A Links

EPE.038.EK3.06	Safety Function 3	Tier 1	Group 1	RO Imp: 4.2	SRO Imp: 4.5
Knowledge of the reasons for the following responses as the apply to the SGTR: (CFR 41.5 / 41.10 / 45.6 / 45.13) Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: **NRC-10 CFR 55.41, 43, and 45 Links**
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 12	ID: 2113969	Points: 1.00

Post trip, what sensor is used to differentiate between an Excess Steam Demand (ESDE) and a Loss of Coolant Accident (LOCA)?

A. Containment Pressure

B. Pressurizer Pressure

C. Pressurizer Level

D. Subcooling Margin

Answer	D

Answer Explanation

A. Incorrect. Plausible if the ESDE event is thought to occur outside of the Containment then containment pressure could be used to differentiate the two casualties.

B. Incorrect. Plausible if the candidate recalls that Pressurizer Pressure can be used to compare a LOCA or ESDE of very different severity levels rather than to differentiate between the two casualties.

C. Incorrect. Plausible if the candidate recalls that Pressurizer Level can be used to compare a LOCA or ESDE of very different severity levels rather than to differentiate between the two casualties.

D. Correct. Subcooled Margin rises for an ESDE, and lowers for a LOCA.

Question Information

Topic	ESDE impacts on sensors		
User ID	Q2113969		
System ID	2113969	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	2
Technical Reference and Revision #	EOP-0-1, Rev 01300		
Training Objective	Given a set of plant conditions, discriminate between those conditions indicative of an ESDE and those indicative of some other event.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

K/A Links

APE.040.AK2.02	Safety Function 4	Tier 1	Group 1	RO Imp: 2.6*	SRO Imp: 2.6
Knowledge of the interrelations between the Steam Line Rupture and the following: (CFR 41.7 / 45.7) Sensors and detectors					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 13	ID: 2114003	Points: 1.00

Given the following conditions on Unit-1:

- A Station Blackout occurred 30 minutes ago.
- The crew is performing EOP-7, Station Blackout.

The following conditions are observed:

- RCS pressure is 2150 psia and stable
- CET temperatures are consistent with T_{HOT} and lowering slowly
- T_{COLD} is 526°F and stable

Which of the following describes the status of Natural Circulation in accordance with EOP-7?

A. Natural Circulation exists; currently maintained by ADVs.

B. Natural Circulation exists; currently maintained by SG Safety Valves.

C. Natural Circulation does NOT exist; must be enhanced by raising AFW flow.

D. Natural Circulation does NOT exist; must be enhanced by lowering ADV setpoint.

Answer	A

Answer Explanation

A. Correct. Per EOP-7, the above conditions verify that natural circulation exists and ADV are the source of maintaining natural circulation.

B. Incorrect. Plausible if the candidate misinterprets how the loss of power impacts the ADVs and determines only the safety valves are available to remove heat from the S/Gs.

C. Incorrect. Plausible if the candidate misinterprets the criteria for natural circulation of less than 50°F delta T being between T_{CETS} and T_{COLD}. Plausible since this action would raise the cooldown rate and help with natural circulation.

D. Incorrect. Plausible if the candidate misinterprets the criteria for natural circulation of less than 50°F delta T being between T_{CETS} and T_{COLD}. Plausible since opening the ADVs would help with natural circulation and the candidate misinterprets how the ADV controller operates with the setpoint is lowered.

Question Information

Topic	Determine the status of natural circulation		
User ID	Q2114003		
System ID	2114003	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	EOP-07-1, Rev 02002		
Training Objective	Given plant conditions, determine the status of the core in regard to heat removal, including if superheat exists, as described in EOP-7.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	N/A
Additional Information	No additional information

K/A Links

EPE.055.EA2.02	Safety Function 6	Tier 1	Group 1	RO Imp: 4.4	SRO Imp: 4.6
Ability to determine or interpret the following as they apply to a Station Blackout: (CFR 43.5 / 45.13) RCS core cooling through natural circulation cooling to S/G cooling					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 14	ID: 2114009	Points: 1.00

Unit-1 is at 100% power when a plant transient occurs resulting in an automatic trip.

- Boration in progress with 12 Charging Pump, via MOV-514, due to loss of CEA indications.
- 11 4KV Bus is de-energized.
- PRZR level is 120 inches and rising.
- PRZR pressure is 1930 PSIA and rising.
- RCPs are de-energized.
- T_{COLD} is 540°F and rising.
- 11 and 12 S/G pressures are 960 PSIA and rising.
- 11 and 12 S/G levels are -130 inches and rising.
- Containment press is 0.1 PSIG and steady.
- Containment temp is 100°F and steady.
- S/G Blowdown is isolated due to loss of power effects.

What is the correct Optimal Recovery Procedure for managing this event?

A. EOP-1.

B. EOP-2.

C. EOP-6.

D. EOP-7.

Answer	B

Answer Explanation

A.Incorrect. Plausible if the candidate misinterprets the requirement for RCPs in operation for EOP-1 implementation since all the parameters for each safety function are met.

B.Correct. 11 4KV bus and the RCPs are de-energized indicating a Loss of Offsite

Power/Loss of Forced Circulation. 12 charging pump in operation indicates 14 4KV bus is energized. The correct diagnosis is EOP-2.

C.Incorrect. Plausible if candidate misinterprets S/G Blowdown being secured and lower than normal S/G levels to mean that both SG have a Tube Rupture.

D.Incorrect - Plausible if candidate misinterprets RCPs being deenergized with Tcold higher than expected to mean that a station blackout is occurring.

Question Information

Topic	Use indications to diagnose EOP-2		
User ID	Q2114009		
System ID	2114009	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	EOP-0-1, Rev 01300		
Training Objective	Given plant conditions and the EOP-0 diagnostic flowchart, determine the correct procedure to implement following the Safety Function Assessment of EOP-0.		
Previous NRC Exam Use	None		

References Provided	None

K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

K/A Links

P2.4.31	Safety Function 6	Tier 3	Group	RO Imp: 4.2	SRO Imp: 4.1
Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)					
GE.4.0.APE.056	Safety Function 6	Tier 1	Group 1	RO Imp:	SRO Imp:
Loss of Offsite Power					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 15	ID: 2114011	Points: 1.00

Unit-1 is operating at 100% when the following occurs:

At 10:00:
- A loss of 21 125V DC Bus
- AOP-7J, Loss of 120 Volt Vital AC or 125 Volt Vital DC Power, is implemented

At 10:05:
- The reactor is tripped
- EOP-0 is implemented

During performance of Core and RCS heat removal, which of the following describes:

(1) Will the 11 and 12 SGFP speeds automatically adjust to the post trip alignment of 3400 RPM?
And
(2) The action required to control S/G levels?

A. (1) Yes
(2) Place one SGFP in standby speed between 2600 and 2900 RPM

B. (1) Yes
(2) Start 13 AFW Pump and secure Main Feedwater lineup

C. (1) No
(2) Start 13 AFW Pump and secure Main Feedwater lineup

D. (1) No
(2) Place one SGFP in standby speed between 2600 and 2900 RPM

Answer	C

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the normally expected SGFP response is to lower speed to 3400 RPM. The operator may misinterpret which Vital AC Bus will be lost during a loss of 21 DC Bus and will conclude that the SGFPs speed will realign to the post trip conditions. (2) Plausible if the operator recalls that AOP-7J Section XIII states that 12 SGFP has lost all protective trips and remote trip functions. The operator will then conclude if a SGFP has lost overspeed protection and the ability to remotely trip the SGFP, then the action is to lower SGFP speed to the standby conditions.

B. Incorrect. (1) Incorrect but plausible as stated above. (2) Correct as stated below.

C. Correct. (1) Correct per AOP-7J Section XIII, Main Feedwater will NOT reconfigure to post trip state (Speed of 3400 RPM) on a Reactor trip. (2) Correct per AOP-7J Section XIII Step A.15 which states to initiate motor train AFW flow by starting 13 AFW Pump and then secure Main Feedwater system lineup.

D. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated above.

Question Information

Topic	Feedwater pump speed indications on a loss of 1Y02		
User ID	Q2114011		
System ID	2114011	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only

Question Type	New	**Difficulty**	3
Technical Reference and Revision #	AOP-7J-1, Rev 02400		
Training Objective	Identify the response of the MFRV, BFRV and SGFP to a reactor trip/turbine trip from any power level, with or without a vital instrument bus loss.		
Previous NRC Exam Use	None		

K/A Links

APE.057.AA1.03	Safety Function 6	Tier 1	Group 1	RO Imp: 3.6*	SRO Imp: 3.6
Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: (CFR 41.7 / 45.5 / 45.6) Feedwater pump speed to control pressure and level in S/G					

Associated Objective(s)

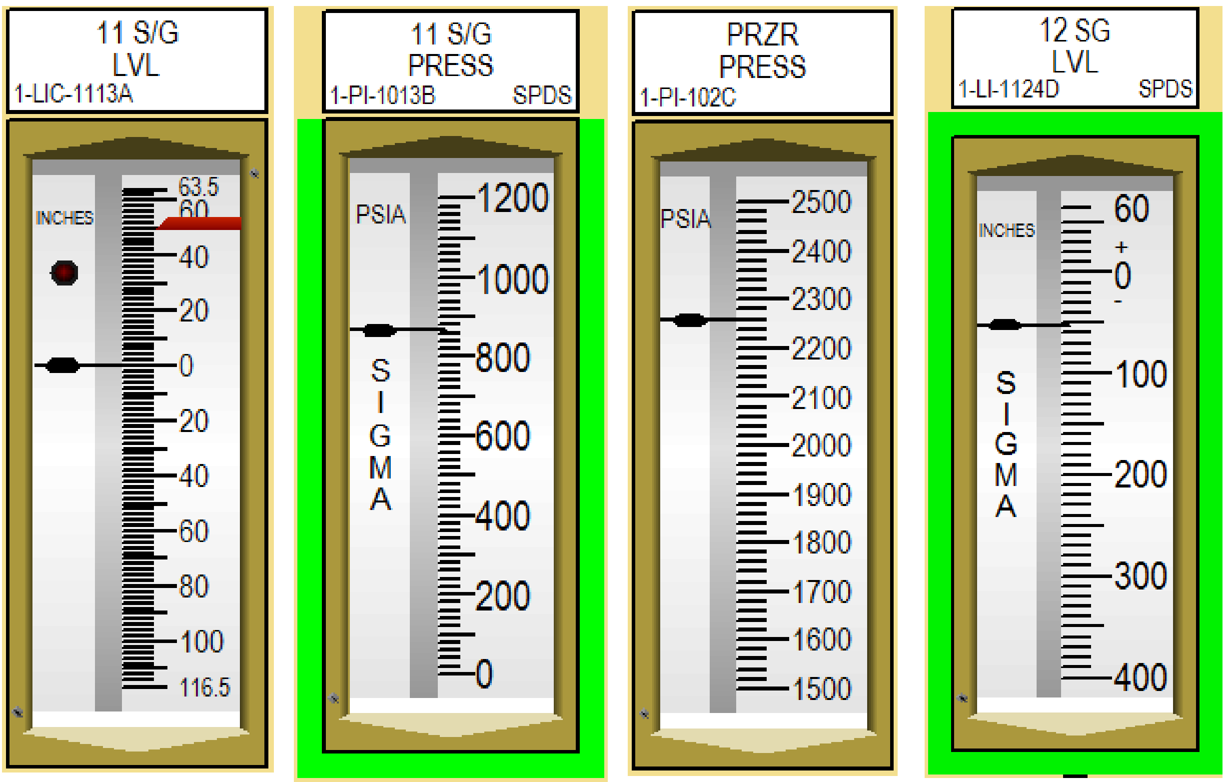
RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 16	ID: 2121168	Points: 1.00

Unit-1 is operating at 100% with the following indications:



Subsequently, 21 DC Bus is lost

Which one of the above indications is a Post-Accident Monitoring System (PAMS) Instrumentation **AND** will be inoperable due to the bus loss?

A. 11 S/G Level, 1-LIC-1113A

B. 11 S/G Pressure, 1-PI-1013B

C. Pressurizer Pressure, 1-PI-102C

D. 12 S/G Level, 1-LI-1124D

Answer	B

Answer Explanation

A. Incorrect. Plausible since Steam Generator Levels are PAMS indicators and the operator may not correctly recall which channels are impacted by 21 DC Bus.

B. Correct. Per AOP-7J-1, Section VI for a Loss of 1Y02 states that 11 and 12 SG Channel B pressure and level indicators fail low. A Loss of 21 DC Bus also causes a Loss of 1Y02 per AOP-7J-1.

C. Incorrect. Plausible since Pressurizer Pressure indications are PAMS instruments and the operator may not correctly recall which channels are impacted by 21 DC Bus.

D. Incorrect. Plausible since 1-LI-1124D is a PAMS instrument and the operator may not correctly recall which channels are impacted by 21 DC Bus.

Question Information

Topic	PAMS with a loss of 21 DC Bus		
User ID	Q2121168		
System ID	2121168	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	Embedded Reference
K/A Justification	No additional information
SRO-Only Justification	N/A

Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	AOP-7J-1, Rev 02400 TS 3.3.10, Amend No. 253 STP O-63-1 Rev 03501		
Training Objective	Identify the PAMS' inputs and outputs utilized for plant operations without error.		
Previous NRC Exam Use	None		

K/A Links

P2.4.3	Safety Function 6	Tier 3	Group	RO Imp: 3.7	SRO Imp: 3.9
Ability to identify post-accident instrumentation. (CFR: 41.6 / 45.4)					
GE.4.0.APE.058	Safety Function 6	Tier 1	Group 1	RO Imp:	SRO Imp:
Loss of DC Power					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 17	ID: 2114053	Points: 1.00

Given the following conditions:

- 13 Saltwater Pump is aligned to the 14 4KV Bus

(1) Following a SIAS actuation, how will 13 SW Pump respond?
And,
(2) Why is the starting circuitry designed to ensure only one SW Pump is running on each SW header?

A. Will start 1 second later **ONLY if** 12 SW pump fails to start;
To ensure minimum flow requirements for the running SW pump.

B. Will start 1 second later **ONLY if** 12 SW pump fails to start;
To ensure the Component Cooling HX tube integrity.

C. Will start 1 second later **if Either** 11 **Or** 12 SW pumps fail to start;
To ensure minimum flow requirements for the running SW pump.

D. Will start 1 second later **if Either** 11 **Or** 12 SW pumps fail to start;
To ensure the Component Cooling HX tube integrity.

Answer	A

Answer Explanation

A. Correct. (1) Correct per 60620SH0007, 13 SW pump will receive a start signal 1 second after a SIAS if the pump on the same power supply fails to start. (2) Correct per STP O-073A, this is to ensure minimum flow requirements are maintained for the running SW Pump.

B. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated below.

C. Incorrect. (1) Incorrect but plausible as stated below. (2) Correct as stated above.

D. Incorrect. (1) Incorrect but plausible since the operator may conclude that the swing pump will start when either pump fails due to the importance of SW flow need during a SIAS. (2) Incorrect but plausible since the operator may recall OI-16 contains the precaution that 2 or more CC Pumps running through a single CCHX will lead to fatigue of the CCHX tubes and potential tube failure. Calvert Cliffs has experienced this issue and this precaution is basis captured as a result.

Question Information

Topic	ESFAS interaction with SW system and why		
User ID	Q2114053		
System ID	2114053	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	STP O-073A1-1, Rev 00300 60620SH0007, Rev 7		
Training Objective	From memory, describe the design features and interrelationship between the Saltwater system and the following system without error: ESFAS.		

Previous NRC Exam Use	None

K/A Links

APE.062.AK3.02	Safety Function 4	Tier 1	Group 1	RO Imp: 3.6	SRO Imp: 3.9
Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: (CFR 41.4, 41.8 / 45.7) The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.04, CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

Question 18	ID: 2114214	Points: 1.00

Unit 1 is at 100% power when the following occurs:

At time 10:00:
- Loss of ALL MFW
- Unit Shutdown and EOP-0 entered
- No AFW capability exists

At time 10:10:
- EOP-0 Exited
- EOP-3 in progress

At time 10:25:
- LOOP
- A fault on the 11 4KV bus occurs

Which one of the following completes the statement below?

The Crew will _______ to ensure a long term successful heat removal path exists.

A. initiate OTCC and tie MCC-114 to MCC-104

B. dispatch an operator to operate disconnect 189-1106, 0C DG to 11 4KV Bus and start an available AFW pump

C. cooldown to 465° F and initiate Condensate Booster Pump injection

D. initiate OTCC and align disconnects for 12 HPSI pump and 13 Charging pump to B train power supplies

Answer	A

Answer Explanation

A. Correct. Once MCC-104 is tied to MCC-114, power is provided to both PORVs and blocking valves ensuring successful OTCC per EOP Att. 17.

B. Incorrect. Plausible since 13 AFW pump is powered from 11 4kv Bus and this pump is the number 1 preferred pump to use in the case of a loss of all feed situation.

C. Incorrect. Plausible is the operator misinterprets which buses have power after the LOOP and determines that CBP injection equipment is available.

D. Incorrect. Plausible because the action is taken with 13 Charging pump to have more equipment for OTCC, but without the PORVs becoming powered OTCC will not be successful. Also, the operator will recall that 12 HPSI pump can be aligned to discharge to either the Main HPSI header (B train) or the Aux HPSI header (A train) and the operator may then incorrectly conclude that 12 HPSI pump can be electrically aligned to either train as well.

Question Information

Topic	Actions in EOP-3 for heat removal		
User ID	Q2114214		
System ID	2114214	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	EOP-3-1, Rev 02301 EOP-ATT, Rev 02001		
Training Objective	Given availability of plant equipment, identify the EOP-3 actions needed to cool the core.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

K/A Links

4.4.E06.EK2.2	Safety Function 4	Tier 1	Group 1	RO Imp: 3.5	SRO Imp: 4.0
Knowledge of the interrelations between the (Loss of Feedwater) and the following: (CFR: 41.7 / 45.7) 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.					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 19	ID: 2114291	Points: 1.00

Unit-2 is operating at 75% power with a reactor startup in progress.

- Regulating Group #5 CEAs are at 100 inches being withdrawn
- CEA motion control switch is released
- CEDS Control System is turned off
- Regulating Group #5 CEA number 1 continues to withdraw

Which one of the following correctly completes the statements below?

As Regulating Group #5 CEA number 1 continues to withdraw, Integral Rod Worth available for insertion ___(1)___.
And,
Per AOP-1B CEA Malfunction, the Reactor Operator is required to ___(2)___.

A. (1) decreases
(2) realign all unaffected Regulating Group #5 CEAs

B. (1) decreases
(2) trip the reactor

C. (1) increases
(2) trip the reactor

D. (1) increases
(2) realign all unaffected Regulating Group #5 CEAs

Answer	C

Answer Explanation

A. Incorrect. (1) Plausible since the candidate may easily misinterpret Integral Rod Worth with Differential Rod Worth which is expected to decrease as CEA position passes the midpoint of the reactor core and continues to be withdrawn. (2) Plausible since AOP-1B Section IV Preliminary Step A.5 directs the attempt to realign CEAs if trip criteria is not met. The candidate may not recognize reaching reactor trip criteria and will conclude that CEA realignment is required.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Correct. (1) Integral Rod Worth is the amount of reactivity available to be inserted in the reactor core based on current rod position which increases as the CEA is withdrawn. (2) AOP-1B Section IV Preliminary Step A.1.1 states to trip the reactor is CEAs continue to move without operation action.

D. Incorrect. (1) Correct as stated above. (2) Incorrect as stated above.

Question Information

Topic	Integral Rod Worth		
User ID	Q2114291		
System ID	2114291	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	

Technical Reference and Revision #	AOP-1B Rev 03005 NEOP-23, Rev 035
Training Objective	Recognize the conditions which would require a reactor trip during the implementation of AOP-1B.
Previous NRC Exam Use	None

K/A Links

APE.001.AK1.21	Safety Function 1	Tier 1	Group 2	RO Imp: 2.9	SRO Imp: 3.2
Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: (CFR 41.8 / 41.10 / 45.3) Integral rod worth					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.01, CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Question 20	ID: 2114304	Points: 1.00

Unit-1 is at 100% power EOC when the following occurs:

- A regulating CEA dropped to the bottom of the core.

What are the initial effects of the fuel temperature and moderator temperature coefficients on reactivity when the CEA drops into the core?

A. Fuel temperature coefficient added positive reactivity
AND
Moderator temperature coefficient added positive reactivity

B. Fuel temperature coefficient added negative reactivity
AND
Moderator temperature coefficient added negative reactivity

C. Fuel temperature coefficient added negative reactivity
AND
Moderator temperature coefficient added positive reactivity

D. Fuel temperature coefficient added positive reactivity
AND
Moderator temperature coefficient added negative reactivity

Answer	A

Answer Explanation

A. Correct. Per USFAR, both MTC and FTC are negative at EOC and a dropped CEA will initially cause lower RCS and Fuel temperature.

B. Incorrect. Plausible if candidate misinterprets the relationship between the coefficients and reactivity and since both coefficients are negative determines the reactivity is also negative.

C. Incorrect. Plausible if candidate misinterprets how FTC changes over core life and determines that FTC is positive at EOL due to less fuel to burn.

D. Incorrect. Plausible if candidate misinterprets the fact that the reactor will not overpower on a dropped CEA so determines that MTC would have to add negative reactivity to ensure this is the case.

Question Information

Topic	Dropped CEA and FTC		
User ID	Q2114304		
System ID	2114304	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not Applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	USFAR, Chapter 14 (section 14.11)		
Training Objective	Given a set of plant conditions, respond to a CEA malfunction in accordance with AOP-1B.		

Previous NRC Exam Use	None

K/A Links

APE.003.AK1.17	Safety Function 1	Tier 1	Group 2	RO Imp: 2.9	SRO Imp: 3.1

Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: (CFR 41.8 / 41.10 / 45.3)

Fuel temperature coefficient

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.01, CFR: 41.1 Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Question 21	ID: 2114313	Points: 1.00

Consider each condition separately from each other:

Which one of the following correctly completes the statements below?

(1) Given a loss of 1Y03: with the Unit-2 Wide Range NI Channel C selected at 2C43, the WRNI power indication is _______.
And,
(2) Given a loss of 1Y04: with the Unit-1 Wide Range NI Channel D selected at 1C43, the WRNI power indication is _______.

A. (1) available
(2) not available

B. (1) available
(2) available

C. (1) not available
(2) not available

D. (1) not available
(2) available

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible since the candidate may easily misinterpret the power supplies for the Wide Range Nuclear Instruments. 1Y03 is a Unit-1 power supply and the candidate may conclude that the WRNI indications for Unit-2 are powered from 2Y03 and therefore not impacted. (2) Plausible since the candidate may recall that a single power supply does power the WRNI indications for both Unit-1 and Unit-2. The candidate may incorrectly recall that 1Y04 powers both Unit-1 and Unit-2 WRNIs.

B. Incorrect. (1) Incorrect as stated above. (2) Correct as stated below.

C. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

D. Correct. (1) Per AOP-7J-2 Section X for a loss of 1Y03, the Wide Range NI Channel C power indication will be lost. (2) Per AOP-7J-1 and AOP-7J-2, a loss of 1Y04 does not impact the Wide Range NI Channel D indications for Unit-1. It is 2Y04 that supplies the power for Unit-1 Channel D WRNIs.

Question Information

Topic	Wide Range Nuclear Instrumentation		
User ID	Q2114313		
System ID	2114313	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	AOP-7J-1 Rev 02400 AOP-7J-2 Rev 01900		
Training Objective	Recall the power supplies for the safety channels of nuclear instrumentation.		

Previous NRC Exam Use	None

K/A Links

APE.032.AK2.01	Safety Function 7	Tier 1	Group 2	RO Imp: 2.7*	SRO Imp: 3.1
Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: (CFR 41.7 / 45.7) Power supplies, including proper switch positions					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 22	ID: 2114312	Points: 1.00

Unit-1 is at 100% power when a loss of condenser vacuum occurs:

- The reactor is tripped and EOP-0, Post trip immediate actions, is entered.
- Vacuum is currently 21" Hg and steady.
- Core and RCS heat removal is being performed.

Which of the following describes:

(1) How is RCS T_{COLD} maintained?
And
(2) The source of feed to the steam generators?

A. (1) TBVs
(2) Main Feed

B. (1) TBVs
(2) Auxiliary Feed

C. (1) ADVs
(2) Main Feed

D. (1) ADVs
(2) Auxiliary Feed

Answer	C

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the TBVs would be the equipment used on Unit 2. (2) Correct as stated below.

B. Incorrect. (1) Incorrect but plausible if candidate misinterprets the 2 different setpoints for loss of condenser vacuum between Unit-1 and Unit-2 and determines that the TBVs are available. (2) Incorrect but plausible if candidate misinterpret the vacuum setpoint that trips the SGFPs and will then conclude that Aux Feedwater must be used.

C. Correct. (1) Correct per AOP-7G-1, TBVs are inoperable at 22.5". (2) Correct per AOP-7G-1, SGFPs will trip at 20".

D. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible if candidate misinterprets the setpoint for a loss of TBVs and SGFPs to be 22.5 for both since the setpoint is the same for both on Unit 2.

Question Information

Topic	Loss of Condenser vacuum post trip actions		
User ID	Q2114312		
System ID	2114312	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	AOP-7G-1, Rev 00400		
Training Objective	Given a loss of condenser vacuum and/or plant conditions and parameters,		

	determine the correct operator response(s).
Previous NRC Exam Use	None

K/A Links

P2.2.44	Safety Function 7	Tier 3	Group	RO Imp: 4.2	SRO Imp: 4.4
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. (CFR: 41.5 / 43.5 / 45.12)					
GE.4.0.APE.051	Safety Function 4	Tier 1	Group 2	RO Imp:	SRO Imp:
Loss of Condenser Vacuum					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 23	ID: 2114316	Points: 1.00

Given a core offload in progress:

- An irradiated fuel bundle is dropped in Containment

Which one of the following radiation monitors will be used to verify a CRS has actuated?

A. Containment Area (1-RI-5316A)

B. Containment Area High Range (1-RI-5317A)

C. Control Room Ventilation Supply (0-RI-5350)

D. Containment Atmosphere Particulate (1-RE-5280)

Answer	A

Answer Explanation

A. Correct. Per 1C08-ALM, 1-RI-5316A is the RMS that gives a CRS signal.

B. Incorrect. Plausible since CRS is based off containment radiation readings and if the CRS signal is thought to be at a high accident level reading.

C. Incorrect. Plausible since this RMS has automatic actions when high radiation readings are identified in the CR and the CRS signal is misinterpreted to be for CR protection.

D. Incorrect. Plausible since CRS is based off containment radiation readings and if CRS is thought to be in operation at all modes and at a lower reading than the high range rad monitors could detect.

Question Information

Topic	Area Rad Monitor that initiates a CRS		
User ID	Q2114316		
System ID	2114316	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	1C08-ALM, Rev 03600		
Training Objective	From memory, identify the radiation monitors that have a control interface with another system and recall their control functions.		
Previous NRC Exam Use	None		

K/A Links

APE.061.AA1.01	Safety Function 7	Tier 1	Group 2	RO Imp: 3.6	SRO Imp: 3.6
Ability to operate and / or monitor the following as they apply to the Area Radiation Monitoring (ARM)System Alarms: (CFR 41.7 / 45.5 / 45.6) Automatic actuation					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.41.12, CFR: 41.12	Radiological safety principles and procedures.

Question 24	ID: 2114462	Points: 1.00

Given the following conditions on Unit-2:

- An RCS Cold Leg Rupture has occurred

Which one of the following is an indication of Inadequate Core Cooling as described in EOP-5 Loss of Coolant Accident and its Technical Basis?

A. RCS T_{COLD} Loop temperatures rising. Pressurizer level lowering when HPSI flow is throttled.

B. Loss of RCS Subcooling. RVLMS indicating the lowest detector in the reactor vessel is uncovered.

C. Pressurizer pressure lowering as RVLMS indicates level rising in the upper reactor vessel.

D. Pressurizer level lowering when HPSI flow is throttled and CET temperatures rising.

Answer	B

Answer Explanation

A. Incorrect. Plausible since RCS Loop temperatures are rising combined with Pressurizer level lowering which the candidate will conclude as being indications of inadequate core cooling.

B. Correct. EOP-5 Technical Basis section IV.O states that core uncovery indicates an advanced phase in the approach to inadequate core cooling. Tech Spec 3.3.10 Basis states that the Subcooled Monitor, CETs, and RVLMS are the three components of inadequate core cooling instrumentation.

C. Incorrect. Plausible since the candidate may conclude the Pressurizer pressure lowering has caused the Pressurizer bubble to shift to the Reactor Vessel head and will conclude that natural circulation flow can be disrupted.

D. Incorrect. Plausible since the candidate may interpret the Pressurizer level lowering and CET temperatures rising as a challenge to subcooling and would conclude that inadequate core cooling conditions are being approached.

Question Information

Topic	Inadequate Core Cooling		
User ID	Q2114462		
System ID	2114462	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	EOP-5-TB Technical Basis Rev 03300 Tech Spec 3.3.10 Basis Rev 2		
Training Objective	Diagnose a LOCA based on the response of plant parameters.		
Previous NRC Exam Use	None		

K/A Links

EPE.074.EA2.07	Safety Function 4	Tier 1	Group 2	RO Imp: 4.1	SRO Imp: 4.7
Ability to determine or interpret the following as they apply to an Inadequate Core Cooling: (CFR 43.5 / 45.13) The difference between a LOCA and inadequate core cooling, from trends and indicators					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 25	ID: 2114458	Points: 1.00

Unit-1 is operating at 100% power when the following occurs:

- An excess steam demand event occurs
- The reactor is tripped and EOP-0, Post Trip Immediate Actions, is implemented
- Pressurizer Pressure is 1900 PSIA
- Pressurizer Level is 140 inches
- 11 SG Pressure is 680 PSIA
- 11 SG Level is -190 inches
- 12 SG Pressure is 820 PSIA
- 12 SG Level is -70 inches
- Containment Pressure is 0.7 PSIA

What ESFAS actuations have occurred?

A. **ONLY** SGIS and AFAS

B. **ONLY** AFAS and AFAS Block

C. **ONLY** SGIS and AFAS Block

D. **ONLY** SGIS, AFAS, and AFAS Block

Answer	D

Answer Explanation

A. Incorrect. Plausible if candidate misinterprets the requirement for AFAS to feed both generators during an ESDE in order to prevent dry out so the AFAS Block does not occur.

B. Incorrect. Plausible if candidate determines that 12 SG pressure is above the SGIS setpoint and they misinterpret that you need <703 PSIA on BOTH S/Gs.

C. Incorrect. Plausible if candidate determines that AFAS had not occurred based on 12

SG Level and they misinterpret that you need <-170" on BOTH S/Gs.

D. Correct. Per 1C03, 1C04, and 1C08, SGIS is at 703 PSIA, AFAS is at -170", and AFAS Block is at 115 PSID.

Question Information

Topic	ESDE - ESFAS/AFAS actuations		
User ID	Q2114458		
System ID	2114458	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	1C03-ALM, 05700 1C08-ALM, 03600 1C04-ALM, 04204		
Training Objective	Given an ESDE, plant conditions and/or parameters, predict the response of the following without error: ESFAS actuations.		

Previous NRC Exam Use	None

K/A Links

4.4.A11.AK2.1	Safety Function 4	Tier 1	Group 2	RO Imp: 3.2	SRO Imp: 3.4
Knowledge of the interrelations between the (RCS Overcooling) and the following: (CFR: 41.7 / 45.7) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 26	ID: 2114467	Points: 1.00

Unit-1 was operating at 100% power when the following occurs:

- A S/G tube rupture.
- EOP-0 is complete and the applicable EOP has been entered.
- All RCPs have been tripped.
- Thot is currently 510°F.

Which one of the following describes:

(1) The maximum RCS cooldown rate in the Optimal EOP?

And

(2) The reason for this cooldown rate?

A. (1) 35°F/Hr
(2) Prevents lifting a main steam safety valve

B. (1) 35°F/Hr
(2) Ensures the two S/Gs remain thermodynamically coupled

C. (1) 100°F/Hr
(2) Prevents lifting a main steam safety valve

D. (1) 100°F/Hr
(2) Ensures the two S/Gs remain thermodynamically coupled

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) Per EOP-6, with RCPs tripped the cooldown rate is 35°F/Hr. (2) Per EOP-6, this rate ensures the two S/Gs remain thermodynamically coupled.

C. Incorrect. (1) Plausible since this is the rate commonly used for most EOPs and the rate used in this EOP until 515 is reached. (2) Plausible since this is the reason why a rapid cooldown is commenced in EOP-6 but not the reason for the rate.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Natural Circulation Cooldown and Basis		
User ID	Q2114467		
System ID	2114467	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	N/A
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-6-TB, Rev 02100		
Training Objective	From memory, recall the strategy and the basis for the major actions performed in EOP-6, SGTR.		

Previous NRC Exam Use	None

K/A Links

4.4.A13.AK3.1	Safety Function 4	Tier 1	Group 2	RO Imp: 3.4	SRO Imp: 3.7
Knowledge of the reasons for the following responses as they apply to the (Natural Circulation Operations) (CFR: 41.5 / 41.10, 45.6, 45.13 Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 27	ID: 2114466	Points: 1.00

The following conditions exist on Unit-1:

- A Small Break LOCA combined with Steam Line Rupture inside Containment occurs
- Containment Pressure is 8 psig and steady
- EOP-8 Functional Recovery Procedure is implemented

Which one of the following correctly completes the statements below?

EOP-8 contains the step to place the 1-IA-2080-MOV CIS OVERRIDE keyswitch, 1-HS-2080A, in the ___(1)___ position to allow the restoration of ___(2)___.

A. (1) NORMAL
(2) Letdown

B. (1) OVERRIDE
(2) Letdown

C. (1) OVERRIDE
(2) Aux Spray

D. (1) NORMAL
(2) Aux Spray

Answer	C

Answer Explanation

A. Incorrect. (1) Plausible since NORMAL is the other available position for the 1-IA-2080 CIS OVERRIDE keyswitch and is the normally found position during the implementation of Emergency Operating Procedures. (2) Plausible since the operator may conclude that the restoration of Letdown will allow the control of Pressurizer Level after a Steam Leak is isolated or will support control during HPSI throttling.

B. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

C. Correct. (1) EOP-8 Pressure and Inventory Control safety function Block Step B.2

states to place the 1-IA-2080 CIS OVERRIDE keyswitch in the OVERRIDE position. (2) The purpose of the step in EOP-8 is to open 1-IA-2080-MOV followed by the step to open the Aux Spray valve 1-CVC-517-CV.

D. Incorrect. (1) Incorrect as stated above. (2) Correct as stated above.

Question Information

Topic	EOP-8 Action and Reason		
User ID	Q2114466		
System ID	2114466	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	EOP-8-1 Rev 04101		
Training Objective	Given a set of plant conditions, demonstrate an understanding of the strategy and basis in EOP-8.		

Previous NRC Exam Use	None

K/A Links

4.4.E09.EK3.3	Safety Function 4	Tier 1	Group 2	RO Imp: 3.7	SRO Imp: 3.9
Knowledge of the reasons for the following responses as they apply to the (Functional Recovery) (CFR: 41.5 / 41.10, 45.6, 45.13) Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 28	ID: 2115026	Points: 1.00

Unit-1 is at Normal Operating Pressure and Temperature.

For each case identify the status of the RCP seal:

CASE	Inlet to Middle Seal Pressure (PSIA)	Inlet to Upper Seal Pressure (PSIA)	Vapor Seal VCT Pressure (PSIG)
Case 1	1150	1150	35
Case 2	1150	50	35

A. Case 1 - Lower Seal is failed
Case 2 - Upper Seal is failed

B. Case 1 - Lower Seal is failed
Case 2 - Vapor Seal is failed

C. Case 1 - Middle Seal is failed
Case 2 - Upper Seal is failed

D. Case 1 - Middle Seal is failed
Case 2 - Vapor Seal is failed

Answer	C

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Incorrect. (1) Plausible if the candidate misinterprets that for Case 1 the middle & upper Seal pressures being equal indicates a lower seal failure. (2) Plausible if the candidate misinterprets that for Case 2 the upper & vapor Seal pressures being similar indicates an upper seal failure.

C. Correct. Per OI-1A and 1C06-ALM, (1) A Middle Seal Pressure of 1150 PSIA indicates pressure breakdown across the Lower Seal. An Upper Seal pressure of 1150 PSIA indicates failure of the Middle seal. (2) The Middle Seal pressure indicates ~ 50% breakdown of RCS pressure meaning an RCP seal has failed. An Upper Seal pressure approximating VCT pressure indicates the Middle Seal is performing the RCS Pressure Breakdown form 50% to VCT pressure indicating the Upper Seal has failed.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Given a set of RCP seal indications, determine the status of the seal.		
User ID	Q2115026		
System ID	2115026	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	1C06-ALM Rev 05300 OI-1A Rev 04400		
Training Objective	Given a set of RCP seal indications, determine the status of the seal.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF4.003.K1.03	Safety Function 4	Tier 2	Group 1	RO Imp: 3.3	SRO Imp: 3.6
Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) RCP seal system					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.41.03, CFR: 41.3	Mechanical components and design features of the reactor primary system.

Question 29	ID: 2115027	Points: 1.00

Unit-1 is at 30% power when the following occurs:

At time 10:00

- 12A RCP trips
- Reactor trips
- EOP-0 is implemented

Which of the following describes:

At time 10:10

12 S/G steam flow is ___(1)___ than 11 S/G,
And
12 S/G level is ___(2)___ than 11 S/G.

A. (1) higher
(2) lower

B. (1) higher
(2) higher

C. (1) lower
(2) higher

D. (1) lower
(2) lower

Answer	C

Answer Explanation

A. Incorrect. (1) Incorrect but plausible as stated below. (2) Incorrect but plausible as stated below.

B. Incorrect. (1) Incorrect but plausible since the operator will correctly recall that 12 S/G will develop a higher temperature difference between T_{HOT} and T_{COLD} due to the lower RCP flow. Then, the operator may misinterpret this difference to cause a higher THOT and higher pressure in 12 S/G causing a higher steam flow compared to 11 S/G. (2) Correct as stated below.

C. Correct. (1) Correct since the Steam Generator with the tripped Reactor Coolant Pump has a lower steaming rate due to the lower heat transfer into the S/G. (2) Correct since steam flow will be lower in 12 S/G causing the level to be higher in 12 S/G.

D. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible since the operator may conclude that a lower steam flow will cause a lower S/G level due to shrink.

Question Information

Topic	RCP trip impact on secondary system		
User ID	Q2115027		
System ID	2115027	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3

Technical Reference and Revision #	EOP-00-TB, Rev 02100 TS 3.4.6 Basis, Rev 19
Training Objective	State from memory the purpose of the reactor coolant pumps.
Previous NRC Exam Use	None

K/A Links

SF4.003.K5.04	Safety Function 4	Tier 2	Group 1	RO Imp: 3.2	SRO Imp: 3.5
Knowledge of the operational implications of the following concepts as they apply to the RCPS: (CFR: 41.5 / 45.7) Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 30	ID: 2115032	Points: 1.00

What is the effect of raising Hydrogen over pressure in the VCT on (1) the RCS stress corrosion rate and (2) why is the rate affected?

A. (1) lowers corrosion rate
(2) because it lowers Oxygen concentration

B. (1) lowers corrosion rate
(2) because it raises Oxygen concentration

C. (1) raises corrosion rate
(2) because it lowers Oxygen concentration

D. (1) raises corrosion rate
(2) because it raises Oxygen concentration

Answer	A

Answer Explanation

A. Correct. Raising the Hydrogen overpressure will lower the Oxygen concentration lowering the corrosion rate.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible if thought that a higher hydrogen concentration will raise the corrosion rate in the RCS since most chemicals need to be kept low to minimize corrosion rates. (2) Plausible if the operator interprets the mechanism of hydrolysis of water within the reactor core to raise the hydrogen peroxide concentration which will also raise the oxygen concentration and raise the corrosion rate.

Question Information

Topic	Importance of Oxygen Control		
User ID	Q2115032		
System ID	2115032	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	AOP-6A-TB Rev 01002 (page 9, section 5)		
Training Objective	Understand and be able to discuss the impact of RCS Chemistry on plant performance.		
Previous NRC Exam Use	None		

K/A Links

SF2.004.K5.01	Safety Function 2	Tier 2	Group 1	RO Imp: 2.7	SRO Imp: 3.3
Knowledge of the operational implications of the following concepts as they apply to the CVCS: (CFR: 41.5/ 45.7) Importance of oxygen control in RCS					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 31	ID: 2115033	Points: 1.00

Unit-1 is in Mode 5:

- 11 SDC Heat Exchanger and 11 LPSI Pump are in service providing Shutdown Cooling flow.
- A header rupture on the Instrument Air system results in loss of Instrument Air to the Auxiliary Building.

Which of the following describes the long term effect of the loss of Instrument Air on the Shutdown Cooling System,
AND
The impact on RCS temperature with no operator action?

A. 1-SI-306-CV, LPSI Flow Control, fails open; RCS temperature lowers.

B. 1-SI-306-CV, LPSI Flow Control, fails closed; RCS temperature rises.

C. 1-SI-657-CV, SD HX Outlet Temperature Control, fails open; RCS temperature lowers.

D. 1-SI-657-CV, SD HX Outlet Temperature Control, fails closed; RCS temperature rises.

Answer	D

Answer Explanation

A. Incorrect. Plausible since this is how 306 will fail and if the candidate misinterprets the location of 306 in the SDC system of being directly in the cooling lineup thus providing more cooling and RCS temperature to lower.

B. Incorrect. Plausible since 306 failing closed would cause RCS temperature to rise but this is not the failure mode on a loss of IA.

C. Incorrect. Plausible if the failure mode of 657 is misinterpreted to be open thinking the fail safe position would be maximum cooling which would cause RCS temperature to lower.

D. Correct. Per AOP-7D, 657 TCV fails closed, resulting in less flow through HX and rising temperatures

Question Information

Topic	Loss of SDC HX impact on RCS		
User ID	Q2115033		
System ID	2115033	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-7D Rev 01600		
Training Objective	Given plant conditions resulting in a loss of shutdown cooling, determine the required actions to maintain plant parameters within desired limits IAW with AOP-3B		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	The question matches the K/A by describing how a loss of SDC flow to the SDC Heat Exchangers effect the Residual

	Heat Removal System.
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF4.005.K6.03	Safety Function 4	Tier 2	Group 1	RO Imp: 2.5	SRO Imp: 2.6
Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: (CFR: 41.7 / 45.7) RHR heat exchanger					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 32	ID: 2115307	Points: 1.00

STP O-14A-1, 'A' Train Safety Injection, Containment Spray, and SDC Gas Accumulation Test, uses ultrasonic testing to verify voids are not forming in the ECCS system.

Which of the following describes:

(1) The consequence of having voids in the ECCS system?

And,

(2) The corrective action to take per STP O-14A-1 if voids have formed in the ECCS system?

A. (1) Water Hammer
(2) Vent the affected location

B. (1) Water Hammer
(2) Raise pressure at the affected location

C. (1) Flow Accelerated Corrosion
(2) Vent the affected location

D. (1) Flow Accelerated Corrosion
(2) Raise pressure at the affected location

Answer	A

Answer Explanation

A. Correct. (1) Per TS Basis 3.5.2, preventing and managing gas intrusion and accumulation is necessary for proper operation of the ECCS and to also prevent water hammer. (2) Per STP O-14A-1, when voids are present the corrective action is to vent the affected location.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since this may be a consequence of a steam water mixture in piping and the candidate misinterprets voids in the ECCS piping to mean a two phase flow is occurring. (2) Plausible since the candidate will recall that a common strategy to collapse voids is to raise system pressure and/or raise the cooldown rate.

Question Information

Topic	STP-O-14 use to prevent water hammer		
User ID	Q2115307		
System ID	2115307	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	STP O-14A-1, Rev 001 TS 3.5.2 Bases, Rev 056		
Training Objective	Given a set of plant conditions regarding ECCS determine the cause and correct actions in accordance with procedures.		

Previous NRC Exam Use	None

K/A Links

SF2.006.A2.06	Safety Function 2	Tier 2	Group 1	RO Imp: 3.3	SRO Imp: 3.5
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: (CFR: 41.5 / 45.5) Water hammer					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 33	ID: 2115309	Points: 1.00

Why is Quench Tank pressure maintained less than 1.5 PSIG while drawing a Pressurizer bubble, per OP-7, Shutdown Operations?

A. Prevents Pressurizer Vent SVs from leaking by.

B. Prevents Pressurizer Safety Valves from unseating.

C. Prevents Reactor Vessel Vent SVs from leaking by.

D. Prevents Power Operated Relief Valves from unseating.

Answer	D

Answer Explanation

A.Incorrect - Plausible since these vents also vent to the quench tank and the candidate thinks that pressure in the quench tank could cause them to leak.

B.Incorrect - Plausible since these safety valves also vent to the quench tank and the candidate thinks that pressure in the quench tank could cause them to unseat.

C.Incorrect - Plausible since these vents also vent to the quench tank and the candidate thinks that pressure in the quench tank could cause them to leak.

D.Correct – Per OP-7, Sect 6.1.2 Prepare RCS for Drawing Pressurizer Bubble contains a note that states maintaining Quench Tank pressure less than 1.5 PSIG helps prevent PORVs from leaking. Per OI-1B (Quench Tank Operations), Section 6.11 (Quench Tank Lineup for Plant Startup at Low RCS Pressure), Quench Tank pressure is maintained less than 1.5 PSIG to help prevent the PORVs from leaking.

Question Information

Topic	Quench Tank parameters for drawing a bubble.		
User ID	Q2115309		
System ID	2115309	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	OI-1B, Rev 01705		
Training Objective	From memory, recall the operation and basis of the following in relation to the RCS in accordance with the Tech Specs, UFSAR and Operating Procedures: Quench Tank.		
Previous NRC Exam Use	2010 NRC RO Exam		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF5.007.K1.03	Safety Function 5	Tier 2	Group 1	RO Imp: 3.0	SRO Imp: 3.2
Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) RCS					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 34	ID: 2115312	Points: 1.00

A loss of load transient resulted in a plant trip with the PORVs lifting

- The PORVs fail to reseat.

What would indicate that the quench tank rupture disk has ruptured?

A. RCS pressure lowers slowly.

B. RCS pressure lowers rapidly.

C. "QUENCH TK TEMP LVL PRESS" alarm clears.

D. "CNTMT NORMAL SUMP LVL HI" alarm actuates.

Answer	D

Answer Explanation

A. Incorrect. Plausible if the candidate misinterprets the impact of the downstream back pressure of the Quench tank and its impact on the leak rate of the PORVs and that backpressure will slowly seat the PORVs causing RCS pressure to lower slowly.

B. Incorrect. Plausible if the candidate misinterprets the relationship of PORV flow rate compared to backpressure and determines that with less backpressure the PORV leak rate will increase causing the RCS pressure to rapidly lower.

C. Incorrect. Plausible since pressure in the quench tank will rapidly lower and clear the alarm setpoint but temperature and possibly level would remain above the alarm setpoint.

D. Correct. Per 1C10, the sump alarm along with quench tank pressure lowering are indications that the rupture disk has ruptured.

Question Information

Topic	Identify indications of a ruptured quench tank rupture disk.		
User ID	Q2115312		
System ID	2115312	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	1C10-ALM, Rev 05102 1C06-ALM, Rev 05300		
Training Objective	From memory, recall the operation and basis of the following in relation to the RCS in accordance with the operating procedures: PORVs.		
Previous NRC Exam Use	2008 NRC RO Exam		

References Provided	None
K/A Justification	The question matches the K/A since it establishes how the quench tank rupture disk provides a direct path of a leaking PORV into Containment filling the normal sump.
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF5.007.K3.01	Safety Function 5	Tier 2	Group 1	RO Imp: 3.3	SRO Imp: 3.6
Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: (CFR: 41.7 / 45.6) Containment					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 35	ID: 2115346	Points: 1.00

Which of the following completes the statements below:

During a total loss of Component Cooling, OI-3A Safety Injection and Containment Spray states:

RCS/RWT temperature is required to be less than ___(1)___ for HPSI pumps to be operated.
And,
RCS/RWT temperature is required to be less than ___(2)___ for LPSI pumps to be operated.

A. (1) 170°F
(2) 170°F

B. (1) 170°F
(2) 300°F

C. (1) 300°F
(2) 300°F

D. (1) 300°F
(2) 170°F

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) Per OI-3A, HPSI pumps may be operated with NO CCW aligned if the RCS/RWT is less than 170°F. (2) Per OI-3A, LPSI pumps may be operated with NO CCW aligned if the RCS/RWT is less than 300°F.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible if candidate misinterprets the HPSI pumps being able to pump under higher temperatures since they are the ones used at higher pressures and temperatures. (2) Plausible if candidate misinterprets the LPSI pumps needing cooler temperatures since they are usually used at lower pressures and temperatures.

Question Information

Topic	Impact of loss of CCW on Safety Injection		
User ID	Q2115346		
System ID	2115346	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	OI-3A, Rev 03800		
Training Objective	From memory, describe the physical connections and interrelationships with the CCW system and the following without error: Safety Injection System.		

Previous NRC Exam Use	None

K/A Links

SF8.008.K3.01	Safety Function 8	Tier 2	Group 1	RO Imp: 3.4	SRO Imp: 3.5
Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 36	ID: 2115347	Points: 1.00

Unit-2 is in Mode 3:

- RCS Pressure is 2250 psig
- RCS Temperature is 530°F
- The controlling Pressurizer Pressure Control Channel Process Variable fails low

Which one of the following identifies the first level of RCS pressure protection when the process variable fails low and NO operator action is taken?

A. Heaters turn off.

B. PZR Spray valves open.

C. A PZR PORV opens.

D. A PZR Code Safety lifts.

Answer	C

Answer Explanation

A. Incorrect. Plausible if the candidate misinterprets how the PV failing low will impact the operation of the controller since the normal first level of protection is for the heaters to turn off.

B. Incorrect. Plausible since spray is what is usually relied on for pressure protection in a transient situation, but the controller failure will also impact the spray controller and not give the spray valves the correct open signal.

C. Correct. Per 2C06, the PORVs will open at 2400 PSIA and since the heaters and the spray valve controllers are not working correctly this will be the first level of over pressure protection.

D. Incorrect. Plausible if the controller failure is misinterpreted to also impact the PORVs making them not open, then the Code Safeties would be the first level of over pressure protection.

Question Information

Topic	Over pressure control design feature		
User ID	Q2115347		
System ID	2115347	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	2C06-ALM, Rev 05200		
Training Objective	From memory, recall the operation and basis of the following in relation to the RCS in accordance with operating procedures: PORVs and PZR Code Safeties.		
Previous NRC Exam Use	None		

K/A Links

SF3.010.K4.03	Safety Function 3	Tier 2	Group 1	RO Imp: 3.8	SRO Imp: 4.1
Knowledge of PZR PCS design feature(s) and/or inter-lock(s) which provide for the following: (CFR: 41.7) Over pressure control					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 37	ID: 2115348	Points: 1.00

Given a Reactor trip on High Pressurizer Pressure:

Which one of the following 1C05 alarms specifically identify that the Diverse Scram System (DSS) has actuated to automatically trip the reactor?

A. Window D-05: “Prot Ch Trip"

B. Window D-46: “MG Set No Output"

C. Window D-16: “Pzr Press Hi Ch Pre-Trip"

D. Window D-45: “Reactor Trip Bus U/V Relay Trip"

Answer	B

Answer Explanation

A. Incorrect. Plausible since this alarm occurs when any one of the ten RPS trip units reaches the trip setpoint and would annunciate in the event of DSS tripping the reactor. Post-trip conditions can result in receipt of this alarm due to normal plant response to a reactor trip (low S/G level, TM/LP, etc.). DSS is monitored by ESFAS and provides a “DSS TRIP” alarm on 1C05 which is not provided in question stem.

B. Correct. Whenever DSS actuates each CEDM MG set main load contactor (3M) is opened and this annunciator window alarms along with “DSS TRIP” alarm which is not provided in question stem.

C. Incorrect. Plausible since examinee may assume this alarm occurs when PZR pressure reaches 2335 PSIA, which is significantly below DSS trip setpoint (2435 to 2460 PSIA). ESFAS sensor channel trips (if not already in alarm) would alert operator of impending DSS condition.

D. Incorrect. Plausible since this alarm, by itself, would not identify a DSS trip. It can occur as the result of any one of the following conditions: RPS generated trip; Manual Rx trip; DSS generated Rx trip; A single faulted UV relay.

Question Information

Topic	Determining when DSS has actuated		
User ID	Q2115348		
System ID	2115348	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	1C05-ALM, Rev 04101 OI-34, Rev 00000		
Training Objective	Identify the cause and effect of the following alarms on Control Element Drive System (CEDS): MG Set No Output.		
Previous NRC Exam Use	2012 NRC RO Exam		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable

Additional Information	No additional information

K/A Links

SF7.012.A4.07	Safety Function 7	Tier 2	Group 1	RO Imp: 3.9*	SRO Imp: 3.9*
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) M/G set breakers					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 38	ID: 2115349	Points: 1.00

Which of the following describes:

(1) The actuation signal(s) that trip the 11 and 13 pressurizer backup heaters?

And,

(2) After the actuation signal(s) has/have been reset and with no operator action will the 11 and 13 pressurizer backup heaters re-energize?

A. (1) UV ONLY
(2) Yes

B. (1) UV ONLY
(2) No

C. (1) UV and SIAS
(2) No

D. (1) UV and SIAS
(2) Yes

Answer	C

Answer Explanation

A. Incorrect. (1) Plausible since most SIAS actuations occur after the heaters would have tripped on low pressurizer level and the candidate determines the SIAS signal would not need to trip the heaters. (2) Plausible if the candidate determines that the heaters being safety related are on the LOCI or SDS sequencer and would re-energize.

B. Incorrect. (1) See above. (2) Correct.

C. Correct. (1) Per EOP-6 note on page 25, Pressurizer Backup heater banks 11 and 13 trip on U/V and SIAS. (2) Per 61075SH0024B, the heaters have to be manually closed at the local breaker.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	SIAS/UV impact of PZR Heaters		
User ID	Q2115349		
System ID	2115349	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-06, Rev 01900 61075SH0024B, Rev 04		
Training Objective	Given any ESFAS alarm condition, assess the impact on plant operation.		
Previous NRC Exam Use	None		

K/A Links

GS.3.0.SF2.013	Safety Function 2	Tier 2	Group 1	RO Imp:	SRO Imp:
Engineered Safety Features Actuation System (ESFAS)					
P2.4.20	Safety Function 2	Tier 3	Group	RO Imp: 3.8	SRO Imp: 4.3
Knowledge of the operational implications of EOP warnings, cautions, and notes. (CFR: 41.10 / 43.5 / 45.13)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 39	ID: 2115587	Points: 1.00

An Excess Steam Demand Event inside Containment has occurred

- The Containment Air Cooler (CAC) fans failed to automatically shift from high speed to low speed on the SIAS actuation.

Which one of the following is a potential impact to the CACs given the provided conditions?

A. Electrical grounding of the motors

B. Fan motor overcurrent/overloading

C. Excessive heat load on the Saltwater system

D. Entrainment of spray droplets in suction of fans

Answer	B

Answer Explanation

A. Incorrect. Plausible since the CAC is being operating in a steam-water atmosphere and electrical grounding of motors can occur in this type of environment.

B. Correct. Per the EOP-4 technical basis, the CACs are shifted to low speed due to the overloading concern with operation in a steam-water atmosphere which would cause the unit to trip on overcurrent.

C. Incorrect. Plausible since the basis for why the CAC SRW valves are throttled on a SIAS is to avoid excessive heat load on service water and saltwater systems and the candidate misinterprets this basis to be the reason why the CACs are shifted to low speed.

D. Incorrect. Plausible since most cooling fans would not be able to operate in a steam-water environment and the candidate misinterprets shifting the fan to low to help with the entrainment of water.

Question Information

Topic	The CAC fans shift to low speed on a SIAS		
User ID	Q2115587		
System ID	2115587	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	EOP-4-TB, Rev 01900		
Training Objective	From memory, state the basis for shifting CACs to slow speed during a Steam Line Break or LOCA in the containment, without error.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF5.022.K3.01	Safety Function 5	Tier 2	Group 1	RO Imp: 2.9*	SRO Imp: 3.2*
Knowledge of the effect that a loss or malfunction of the CCS will have on the following: (CFR: 41.7 / 45.6) Containment equipment subject to damage by high or low temperature, humidity, and pressure					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 40	ID: 2115590	Points: 1.00

Unit-2 is operating at 100% power with the following conditions:

- 22 Containment Spray Pump is safety tagged out of service for maintenance

Then, the following occurs:

- A Design Basis LOCA occurs inside Containment
- The LOCA destroys the ductwork for 21 and 22 Containment Air Coolers rendering them Inoperable

Which one of the following correctly describes:

Whether the current containment cooling capacity is sufficient?

And,

The required components to prevent exceeding Containment pressure and temperature design limits?

A. Containment cooling capacity is NOT sufficient.
Two Containment Spray Pumps and four Containment Air Coolers are required to protect Containment.

B. Containment cooling capacity is NOT sufficient.
Three Containment Air Coolers are required to protect Containment.

C. Containment cooling capacity is sufficient.
One Containment Spray Pump and two Containment Air Coolers are sufficient to protect Containment.

D. Containment cooling capacity is sufficient.
One Containment Spray Pump alone is sufficient to protect Containment.

Answer	C

Answer Explanation

A. Incorrect. Plausible since the candidate will recall that Tech Spec 3.6.6 requires two Containment Spray trains and two Containment Cooling trains to be operable in Modes 1 and 2. The candidate may then incorrectly conclude that the capacity required to be operable per Tech Spec 3.6.6 is what is required to prevent exceeding Containment pressure and temperature limits.

B. Incorrect. Plausible since the candidate may misinterpret the Containment Air Coolers that have become inoperable to represent both trains and not recognize that each train consists of two fans. The candidate will then incorrectly conclude that the cooling capacity is not sufficient.

C. Correct. Tech Spec Basis 3.6.6 states that each Containment Spray train supplies 50% of the design cooling requirement. Each Containment Air Cooling train supplies 67% of the design cooling requirement. So one Containment Spray train combined with one Containment Air Cooling train supplies greater than 100% of the design cooling capacity required.

D. Incorrect. Plausible since the first part is correct and that the cooling capacity is sufficient. The second part is plausible since the candidate may recognize that only one Containment Spray Pump is required for 100% of the Iodine removal design bases but may misinterpret the need to combine one Containment Spray Pump with at least two Containment Air Coolers for pressure and temperature design limits.

Question Information

Topic	Containment Cooling Capacity		
User ID	Q2115590		
System ID	2115590	**Point Value**	1.00
Status	Active	**Time to Complete**	4
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	Tech Spec 3.6.6 Rev 326/304 Tech Spec Basis 3.6.6 Rev 2		
Training Objective	Given a design basis accident LOCA, identify the minimum operating combinations of Containment Spray Pumps and/or Containment Air Coolers that ensures the design pressure and temperature of containment is not exceeded.		
Previous NRC Exam Use	None		

K/A Links

SF5.022.K4.05	Safety Function 5	Tier 2	Group 1	RO Imp: 2.6*	SRO Imp: 2.7
Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) Containment cooling after LOCA destroys ventilation ducts					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.41.08, CFR: 41.8	Components, capacity, and functions of emergency systems.

Question 41	ID: 2115591	Points: 1.00

Unit-1 is operating at 100% when the following occurs:

- LOCA
- Containment Pressure is 5 PSIG

The following alarm failed to clear:

11 CS PP **DISCH PRESS** **HI**

Which of the following describes:

(1) The cause of this alarm?

And,

(2) How to correct this condition per the alarm manual?

A. (1) RAS 'A' Failure to actuate 1-SI-659-MOV, Mini Flow return to RWT Isolation
(2) Open 1-SI-659-MOV

B. (1) RAS 'A' Failure to actuate 1-SI-659-MOV, Mini Flow return to RWT Isolation
(2) Shut 1-SI-659-MOV

C. (1) CSAS 'A' Failure to actuate 1-SI-4150-CV, 11 CS HDR Isolation
(2) Shut 1-SI-4150-CV

D. (1) CSAS 'A' Failure to actuate 1-SI-4150-CV, 11 CS HDR Isolation
(2) Open 1-SI-4150-CV

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible since a possible cause of this alarm is the CS pump running at minimum flow conditions but RAS would normally shut this mini flow valve, but the candidate could misinterpret how RAS impacts this valve. (2) Plausible since the alarm manual directs ensuring open both mini flow valves if this was the cause of the alarm.

B. Incorrect. (1) See above. (2) Plausible since this is the action that would be required on a valid RAS that failed to actuate.

C. Incorrect. (1) Correct. (2) Plausible if the candidate misinterprets the setpoint for CSAS and is assuming the pump is running but spray is not required so the discharge valve should be shut.

D. Correct. (1) Per 1C09-ALM, the possible cause of this alarm is the discharge flow path isolated. (2) 1C09-ALM directs ensuring open 1-SI-4150-CV.

Question Information

Topic	Indication of a loss of a CS Pump		
User ID	Q2115591		
System ID	2115591	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	Embedded Reference
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	1C09-ALM, Rev 03700		
Training Objective	Given any of the following alarms and plant conditions, identify likely causes for the alarms and determine the required actions in accordance with the Alarm manual: Containment Spray Pump discharge pressure high.		
Previous NRC Exam Use	None		

K/A Links

SF5.026.A2.04	Safety Function 5	Tier 2	Group 1	RO Imp: 3.9	SRO Imp: 4.2
Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Failure of spray pump					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 42	ID: 2115593	Points: 1.00

Which one of the following correctly describes:

(1) How do the Turbine Bypass Valves (TBVs) fail on a Loss of Instrument Air?
And,
(2) What is the mechanism used to locally operate the TBVs?

A. (1) TBVs fail shut
(2) Chain Operator

B. (1) TBVs fail shut
(2) Hydraulic Pump Lever

C. (1) TBVs fail open
(2) Chain Operator

D. (1) TBVs fail open
(2) Hydraulic Pump Lever

Answer	B

Answer Explanation

A. Incorrect. (1) Correct as stated below. (2) Incorrect but plausible as stated below.

B. Correct. (1) AOP-7D Loss of Instrument Air Attachment 1 states that the Turbine Bypass Valves fail shut. (2) OI-8C-1 Section 6.10 Manual Operation of a Turbine Bypass Valve describes the use of a hydraulic pump lever to locally operate the TBVs.

C. Incorrect. (1) Plausible since the candidate will recall that numerous plant valves fail open upon a Loss of Instrument Air. The candidate may incorrectly conclude that an open TBV is required to maintain heat removal from the Reactor Coolant System and supports a vital safety function. (2) Plausible since a chain operator is the local control mechanism used for the Atmospheric Dump Valves which are also components associated with the steam dump system.

D. Incorrect. (1) Incorrect but plausible as stated above. (2) Correct as stated above.

Question Information

Topic	TBV failure mode and local operation		
User ID	Q2115593		
System ID	2115593	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	OI-8C-1 Rev 03800 AOP-7D-1 Rev 01600		
Training Objective	Evaluate TBV operation for the following conditions: Loss of Instrument Air.		
Previous NRC Exam Use	None		

K/A Links

SF4.039.A2.04	Safety Function 4	Tier 2	Group 1	RO Imp: 3.4	SRO Imp: 3.7
Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Malfunctioning steam dump					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 43	ID: 2115605	Points: 1.00

Given the following conditions:

- Unit-1 is at 100% power
- Lowering Hydraulic Pressure in 11 MSIV is being performed per OI-8E, MSIV Actuator System
- The CRO momentarily places the 11 MSIV PARTIAL STROKE TEST Switch, 1-HS-4044, in the "CH A" position
- A system failure results in 11 MSIV shutting completely within a few seconds

Which one of the following correctly describes how the plant will respond in this situation?

A. RCS pressure will rise due to less steam demand, a High Pressure reactor trip will occur and the PORVs will open.

B. 12 S/G level will lower due to the steam demand imbalance, causing a Low S/G level reactor trip.

C. 11 S/G pressure will rise, 12 S/G pressure will lower due to the steam demand imbalance, resulting in an ASGT reactor trip.

D. RCS pressure will lower due to the increased steam demand from 12 S/G, resulting in a TM/LP reactor trip.

Answer	C

Answer Explanation

A. Incorrect. Plausible since this is a possible RPS trip which could result from this event, however, ASGT is designed specifically for this event. The candidate may incorrectly conclude that a high RCS pressure trip will occur which is expected if both MSIVs were to shut while operating at 100% reactor power.

B. Incorrect. Plausible since this is a possible RPS trip which could result from this event, however, ASGT is designed specifically for this event. The candidate may conclude that 12 Steam Generator level will rapidly lower since it is the only S/G providing steam flow capability under the listed conditions.

C. Correct. The UFSAR analysis of an Asymmetric Loss of Load (14.12.2.4) states: “The primary trip for the most adverse Asymmetric SG event is the ASGPT” and goes on to state ”The ASGPT will initiate a reactor trip to terminate the event when the absolute differential SG pressure (i.e., Psg1-Psg2) exceeds a preselected analysis setpoint value”. Per OI-8E caution on page 32, a plant trip caused by the MSIV traveling past the partial stroke limit may occur if a stroke test longer than 120 seconds in duration is performed.

D. Incorrect. Plausible since the candidate may conclude that a lowering pressure in 12 Steam Generator will provide a cooldown of the Reactor Coolant System resulting in a reactor trip.

Question Information

Topic	Effect of an MSIV fully closing at 100% power		
User ID	Q2115605		
System ID	2115605	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	OI-8E-1 Rev 02700 UFSAR Chapter 14 Rev 50		
Training Objective	Evaluate the operations of the MSIVs for the following actions/conditions: Receipt of the close signal with a partial stroke test in progress.		

Previous NRC Exam Use	NRC RO Exam - 12-2008

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

GS.3.0.SF4.SEC.039	Safety Function 4	Tier 2	Group 1	RO Imp:	SRO Imp:
Main and Reheat Steam System (MRSS)					
P2.1.32	Safety Function 4	Tier 3	Group	RO Imp: 3.8	SRO Imp: 4.0
Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 44	ID: 2115604	Points: 1.00

Unit-1 startup is in progress following a mid-cycle forced outage:

- Reactor power is at approximately 50%.
- 11 SGFP in operation.
- 12 SGFP is out of service for maintenance.

Which of the following conditions on 11 SGFP would support a decision to raise reactor power to 60% in accordance with OP-3, "Normal Power Operations"?

A. Suction flow: 16,200 GPM;
Turbine speed: 5160 RPM;
Suction pressure: 272 PSIG

B. Suction flow: 17,200 GPM;
Turbine speed: 5360 RPM;
Suction pressure: 262 PSIG

C. Suction flow: 15,200 GPM;
Turbine speed: 5060 RPM;
Suction pressure: 242 PSIG

D. Suction flow: 18,200 GPM;
Turbine speed: 5260 RPM;
Suction pressure: 252 PSIG

Answer	A

Answer Explanation

A.Correct. From AOP-3G: If **ALL** the following conditions are maintained, then one SGFP operation above 440 MWE is permitted:
•SGFP suction flow rate is below 18,000 GPM
•SGFP suction pressure is above 250 PSIG
•SGFP speed is below 5350 RPM

B.Incorrect. Plausible if candidate misinterprets the unit 1 and unit 2 turbine speed values

C.Incorrect. Plausible if candidate misinterprets the suction pressure limit to be less than 250 since the other 2 limits are less than values.

D.Incorrect. Plausible if candidate misinterprets the suction flow limit to be a greater than limit like suction pressure is.

Question Information

Topic	U1 SGFP operating limitations		
User ID	Q2115604		
System ID	2115604	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-3G, Rev 01300		
Training Objective	Given plant conditions and/or parameters, determine if SGFP operating limitations are exceeded in accordance with procedures from memory.		

Previous NRC Exam Use	2008 NRC RO Exam

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF4.059.A1.03	Safety Function 4	Tier 2	Group 1	RO Imp: 2.7*	SRO Imp: 2.9*
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: (CFR: 41.5/45.5) Power level restrictions for operation of MFW pumps and valves.					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 45	ID: 2115608	Points: 1.00

With a Loss of Offsite Power, what MAXIMUM flow can be obtained **OR** is allowed for each of the following pump combinations?

(Consider each case in column A separately - flow rates in column B may be used more than once)

COLUMN A	COLUMN B
___ a. The normally lined up Turbine AFW pump to Unit 2	1. 300 GPM
___ b. 23 AFW alone to Unit 1	2. 575 GPM
___ c. "a" and "b" above combined	3. 600 GPM

A. a. 1
b. 2
c. 1

B. a. 1
b. 2
c. 3

C. a. 3
b. 1
c. 1

D. a. 3
b. 1
c. 3

Answer	D

Answer Explanation

A. Incorrect. Plausible if the candidate misinterprets the setpoints for AFAS flow of 150

GPM per loop to also be the flow limits for the system. 23 AFW flow limit is plausible if the candidate misinterprets the limits for 13 and 23 AFW pump. Plausible if candidate misinterprets why the flow limit for 23 AFW Pump is 300 GPM and thinks it is for CST cavitation issues which would then mean that 300 GPM is the limit anytime 23 AFW pump is running.

B. Incorrect. Plausible if the candidate misinterprets the setpoints for AFAS flow of 150 GPM per loop to also be the flow limits for the system. 23 AFW flow limit is plausible if the candidate misinterprets the limits for 13 and 23 AFW pump.

C. Incorrect. Plausible if candidate misinterprets why the flow limit for 23 AFW Pump is 300 GPM and thinks it is for CST cavitation issues which would then mean that 300 GPM is the limit anytime 23 AFW pump is running.

D. Correct. Per OI-32A, total AFW flow for one unit should be less than 600 GPM. Per EOP-2, 23 AFW Pump flow limit is 300 GPM when power is supplied from an EDG. Per OI-32A, total AFW flow should be less than 600 GPM when feeding both Units from a single AFW System.

Question Information

Topic	AFW limits during cross tie operation		
User ID	Q2115608		
System ID	2115608	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable

Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	OI-32A, Rev 04000 EOP-2-2, Rev 01700		
Training Objective	From memory, identify the following with respect to plant equipment operation in EOP-2: 13(23) AFW pump flow limits and whether they are being met.		
Previous NRC Exam Use	None		

K/A Links

SF4.061.A1.03	Safety Function 4	Tier 2	Group 1	RO Imp: 3.1*	SRO Imp: 3.6*
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: (CFR: 41.5/45.5) Interactions when multi unit systems are cross tied					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.41.10, CFR: 41.10	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 46	ID: 2115614	Points: 1.00

What is the power supply for 23 AFW Pump?

A. 21 4KV Bus

B. 22 4KV Bus

C. 23 4KV Bus

D. 24 4KV Bus

Answer	D

Answer Explanation

A. Incorrect. Plausible since the motor driven AFW pump on Unit 1 is 'A' train.

B. Incorrect. Plausible if the candidate misinterprets that 23 AFW pump is a safety related pump and powered from 'A' train like on Unit 1.

C. Incorrect. Plausible since many other pumps in the plant are powered from the same number bus (i.e. 23 CP powered from 23 4KV Bus).

D. Correct. Per OI-27C, 23 AFW Pump is powered from 24 4KV Bus.

Question Information

Topic	23 AFW Pump Power supply		
User ID	Q2115614		
System ID	2115614	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	OI-27C, Rev 02802		
Training Objective	Given a loss of 4KV, determine the effect on the following systems IAW plant procedures: Auxiliary Feedwater System.		
Previous NRC Exam Use	None		

K/A Links

SF4.061.K2.02	Safety Function 4	Tier 2	Group 1	RO Imp: 3.7*	SRO Imp: 3.7
Knowledge of bus power supplies to the following: (CFR: 41.7) AFW electric drive pumps					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.04, CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

Question 47	ID: 2115615	Points: 1.00

Which one of the following statements correctly describes the relationship of 120 VAC power and 125 VDC to the Plant Computer?

A. The static inverter feeding the computer has both a 120 VAC and a 125 VDC power supply and it has to be manually lined up to 125 VDC when the 120 VAC is lost.

B. The static inverter is a backup to the regular 120 VAC line and automatically switches to 125 VDC when 120 VAC is lost.

C. The static inverter feeding the computer has both a 120 VAC and a 125 VDC power supply and it automatically switches on a loss of the 125 VDC.

D. The static inverter supplies the #11 120 VAC inverter bus and has to be manually lined up to 120 VAC when the 125 VDC is lost.

Answer	C

Answer Explanation

A. Incorrect. Manual transfer is not required. Plausible since the candidate will recall that the 120 Vital AC Buses contain dual inverters that must be manually shifted when necessary and do not have an automatic transfer switch.

B. Incorrect. The Static Inverter is the normal power supply, and it is fed from 1Y10 and 12 DC Bus. Plausible since the candidate may recall that 1(2)Y05 is the power supply designation for the Plant Computer and may incorrectly conclude that the static inverter is a separate backup supply to 1(2)Y05.

C. Correct. Drawing 61023 shows that the Process Plant Computer receives power from 1(2)Y05 which is normally supplied from the 12(22) 125V DC Bus and contains an automatic static transfer switch in the event that power is lost.

D. Incorrect. Plausible since the candidate may misinterpret the alignment to match the Inverter Backup Bus which does have to be manually lined up to supply 120 VAC power.

Question Information

Topic	Relationship between 125VDC/120VAC and the Plant Computer		
User ID	Q2115615		
System ID	2115615	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	Drawing 61023 Rev 7		
Training Objective	Recall the power supplies for the Process Plant Computer and DAS.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF6.062.A3.04	Safety Function 6	Tier 2	Group 1	RO Imp: 2.7	SRO Imp: 2.9
Ability to monitor automatic operation of the ac distribution system, including: (CFR: 41.7 / 45.5) Operation of inverter (e.g., precharging synchronizing light, static transfer)					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 48	ID: 2115617	Points: 1.00

Unit-2 is at 100% power when a transient occurs:

- Channel 'C' RPS cabinet is dark
- 4 TCBs trip open

The above indications indicate a loss of which Bus?

A. 1Y02

B. 1Y03

C. 12 DC Bus

D. 22 DC Bus

Answer	C

Answer Explanation

A. Incorrect. Plausible since unit 1 VAC does power some of the RPS cabinets on Unit 2.

B. Incorrect. Plausible since 1Y03 does power parts of the 'C' RPS cabinet but does not power the entire cabinet.

C. Correct. Per AOP-7J-2, a loss of 12 DC bus causes 4 TCBs to trip open and a loss of Channel 'C' RPS cabinet.

D. Incorrect. Plausible since 22 DC bus does cause a trip of 4 TCBs and a loss of an RPS cabinet and the candidate misinterprets this loss to be channel C vice channel D.

Question Information

Topic	Loss of 12 DC Bus Automatic actions		
User ID	Q2115617		
System ID	2115617	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	3
Technical Reference and Revision #	AOP-7J-1, Rev 02400 AOP-7J-2, Rev 01900		
Training Objective	Given plant conditions, evaluate control room indications to determine if a loss of a 125 Volt DC bus has occurred per AOP-7J guidance.		
Previous NRC Exam Use	None		

K/A Links

SF6.063.A3.01	Safety Function 6	Tier 2	Group 1	RO Imp: 2.7	SRO Imp: 3.1
Ability to monitor automatic operation of the DC electrical system, including: (CFR: 41.7 / 45.5) Meters, annunciators, dials, recorders, and indicating lights					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 49	ID: 2115618	Points: 1.00

Given the following conditions on the 1B EDG:

- An undervoltage condition occurred on 14 4KV Bus and the EDG started and then tripped on an engine fault condition
- The engine fault condition has been found and corrected

Which one of the following statements reflects the status of the 1B EDG after the engine fault is corrected with the Undervoltage signal still present on 14 4KV Bus?

A. EDG immediately restarts within 10 seconds and repowers the bus.

B. EDG remains shutdown for 60 seconds then EDG restarts and powers the bus.

C. EDG remains shutdown until Control Room stop pushbutton is momentarily depressed;
60 seconds later the EDG restarts and powers the bus.

D. EDG remains shutdown until the local alarm reset pushbutton is momentarily depressed;
60 seconds later the EDG restarts and powers the bus.

Answer	D

Answer Explanation

A. Incorrect. This will not occur until local alarm reset PB is depressed to allow stop relay timer to deenergize 60 seconds later and the DG automatically restarts and repowers the bus. Plausible since the candidate will recall that an EDG will perform a fast speed start and is expected to repower a safety related 4KV bus within 10 seconds of a UV actuation.

B. Incorrect. Must depress local alarm reset to deenergize shutdown relay then after 60 seconds the stop relay timer deenergizes and the DG automatically restarts and repowers the bus. Plausible since the candidate may recall that the shutdown relay has a 60 second timer associated with its operation.

C. Incorrect. Since DG tripped on an engine fault the shutdown relay has remained energized to keep stop relay timer energized. The local alarm reset pushbutton must be momentarily depressed to deenergize shutdown relay then 60 seconds later the stop relay timer will deenergize to allow DG to restart and repower the bus. Depressing CR stop PB will not allow the stop relay timer to deenergize after 60 seconds. Plausible since this is action taken during a normal shutdown of the DG.

D. Correct. Per OI-21B-1 Section 6.5 it states 60 seconds after depressing the Local Alarm Reset Pushbutton, the stop relay timer deenergizes and 1B DG restarts to repower the bus since UV signal still present.

Question Information

Topic	Restart of 1B EDG with automatic start signal present		
User ID	Q2115618		
System ID	2115618	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	
Technical Reference and Revision #	OI-21B-1 Rev 02501		
Training Objective	Given the 1B, 2A, or 2B DG tripped on engine fault and an automatic start signal present, determine the required action(s) to restart the DG after engine fault has been cleared.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF6.064.A4.01	Safety Function 6	Tier 2	Group 1	RO Imp: 4.0	SRO Imp: 4.3
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Local and remote operation of the ED/G					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 50	ID: 2115625	Points: 1.00

Given the following conditions:

- 2A DG is paralleled to 21 4KV Bus and loaded to 3000 KW (3.0 MW).
- Annunciator windows “2A DG CONTR BOARD 1C20A” and “2A DG” alarm on control room panels 1C20 and 1C20A respectively.
- The operator monitoring 2A DG local operation reports annunciator window “FUEL-OIL LEVEL LOW IN DAY TANK" is in alarm and the fuel oil transfer pump will not run (verified breaker closed on local DG control panel).

Which one of the following defines the maximum time the 2A Diesel Generator will continue to operate in this condition?

A. 30 to 45 minutes

B. 1 to 2 hours

C. 4 to 5 hours

D. 7 to 8 days

Answer	B

Answer Explanation

A. Incorrect. Per UFSAR the minimum time by design is one hour at a load of 3500 KW. Plausible since the candidate may interpret the 1 hour design rating to be at expected post accident loads of approximately 2 MW and would then conclude that operation at a higher load of 3 MW will be 30 to 45 minutes.

B. Correct. Tech Spec Basis 3.8.3 and UFSAR Section 8.4.1.2 states if DG loaded to 3500 KW and day tank at low level enough fuel remains to operate for 1 hour. Since DG load is 3000 KW it will operate for 1 to 2 hours (275 Gallons / 3.85 GPM = 71 minutes).

C. Incorrect. Plausible since the candidate may recall that 4 hours represent the coping design time for a station blackout event and would conclude that the Diesel Generator will run longer than the design coping time requirement.

D. Incorrect. Plausible if examinee misinterprets the designed run time based on available Fuel Oil Storage Tank level rather than the Fuel Oil Day Tank limits. UFSAR Section 8.4.1.2 states that the DG fuel oil system is based on a fuel oil capacity of seven days following a design basis accident.

Question Information

Topic	Operating time limit of EDG with fuel oil day tank at low level alarm		
User ID	Q2115625		
System ID	2115625	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	Tech Spec Basis 3.8.3 Rev 47 UFSAR Chapter 8 Section 8.4.1.2 Rev 50		
Training Objective	Given a support system malfunction, evaluate the effect on continued DG operability: DG Fuel Oil Supply.		
Previous NRC Exam Use	NRC RO Exam - 2002		

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF6.064.K6.08	Safety Function 6	Tier 2	Group 1	RO Imp: 3.2	SRO Imp: 3.3
Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: (CFR: 41.7 / 45.7) Fuel oil storage tanks					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 51	ID: 2115628	Points: 1.00

The Wide Range Noble Gas Monitor (RIC-5415) would provide **early** detection of radioactivity during which one of the following events?

A. 40 GPM Leak on the VCT

B. 40 GPM RCS cold leg leak

C. 40 GPM Letdown HX to CC leak

D. 40 GPM Spent Fuel Pool HX to SRW leak

Answer	A

Answer Explanation

A. Correct. A leak on the VCT would be monitored by the WRNGM since the Auxiliary Building Ventilation System discharges thru the WRNGM.

B. Incorrect. An RCS Cold Leg leak would be confined to the Containment, therefore not monitored by the WRNGM.

C. Incorrect. Early indication of a Letdown HX tube leak would be provided by the CCW system inline RMS. Eventually, the WRNGM may detect radioactivity as a result of the CC Head tank being vented to the Auxiliary Building Atmosphere that is maintained by the Exhaust ventilation system discharging to the Main Vent that is monitored by the WRNGM.

D. Incorrect. Early indication of a SFP HX leak would be provided by the SRW system inline RMS. Eventually, the WRNGM may detect radioactivity as a result of the SRW Head tank being vented to the Auxiliary Building Atmosphere that is maintained by the Exhaust ventilation system discharging to the Main Vent that is monitored by the WRNGM.

Question Information

Topic	WRNGM (RIC-5415) provides early detection of radioactivity during which accident		
User ID	Q2115628		
System ID	2115628	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	OI-48 Rev 01301 1C10-ALM, Rev 05102		
Training Objective	Recall the flow paths of the Wide Range Noble Gas Monitor.		
Previous NRC Exam Use	NRC RO Exam - 2014		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF7.073.A1.01	Safety Function 7	Tier 2	Group 1	RO Imp: 3.2	SRO Imp: 3.5
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: (CFR: 41.5 / 45.7) Radiation levels					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 52	ID: 2115630	Points: 1.00

The SRW Rad monitor, RMS-1595, has alarmed and an increase in SRW head tank level has been noted.

Which of the following could be the source of the problem?

A. CVCS Letdown HX

B. 11 SG Blowdown HX

C. 11 Containment Air Cooler

D. 11 Spent fuel pool cooling system

Answer	D

Answer Explanation

A. Incorrect. Plausible if candidate misinterprets the cooling medium to letdown and determines that due to a pressure differential the leak from letdown would be into the SRW system.

B. Incorrect. Plausible since a leak from a blowdown HX could cause an RMS alarm but SRW does not cool 11 BDHX (only 12 BDHX).

C. Incorrect. Plausible since CACs are cooled by SRW but a leak would be into the containment vice into SRW due to the pressure differential.

D. Correct. Per 1C22-ALM, SFP coolers are cooled by SRW. SRW is throttled such that SFP water is at a higher pressure that SRW. Therefore a leak would be into the SRW system.

Question Information

Topic	SRW Rad monitor has alarmed what is most likely source?		
User ID	Q2115630		
System ID	2115630	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	1C22-ALM, Rev 04400		
Training Objective	Identify the feature of the SRW system that provides the capability to monitor for radioactivity, and the alarms associated with that feature.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF4.076.K1.17	Safety Function 4	Tier 2	Group 1	RO Imp: 3.6*	SRO Imp: 2.7
Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) PRMS					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.11, CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Question 53	ID: 2115629	Points: 1.00

Unit-1 is operating at 100% power:

- 12 Instrument Air Compressor is in service
- 11 Plant Air Compressor is in service

Then the following occurs:

- Loss of Offsite Power
- Reactor Trips
- 1B EDG fails to start

Assuming no operator actions, which one of the following describes the status of the U-1 Instrument Air System?

A. ONLY 12 Instrument Air Compressor continues to run supplying U-1 Instrument Air.

B. ONLY 11 Instrument Air Compressor has started supplying U-1 Instrument Air.

C. ONLY 11 Plant Air Compressor continues to run supplying U-1 Instrument Air.

D. BOTH 11 and 12 Instrument Air Compressors are running supplying U-1 Instrument Air.

Answer	B

Answer Explanation

A. Incorrect. Plausible if the candidate misinterprets the power supply to 12 IAC to be from a non-safety related source that did not receive a UV actuation. 12 IAC powered from 14 480V bus, which will be deenergized by the UV. Offsite power will not automatically re-energize.

B. Correct. 11 IAC powered by 11 480V bus. Which was energized by 1A EDG. Backup IAC normally in AUTO, so will start as pressure lowers to 93 psi.

C. Incorrect. Plausible since the candidate misinterprets the operation of the Shutdown Sequencer compared to the LOCI Sequencer. The IACs do not restart upon a LOCI Sequencer actuation. The candidate then concludes that only the Plant Compressors would be available.

D. Incorrect. Plausible if the candidate misinterprets the power supply to 12 IAC to be the same as 11 IAC and would then conclude that both IACs are running. 14 480V bus remains deenergized without operator action.

Question Information

Topic	Instrument air compressor power supplies		
User ID	Q2115629		
System ID	2115629	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	
Technical Reference and Revision #	61058ASH0001 Rev 55 OI-27D-2 Rev 02200		
Training Objective	List the power supplies to the Instrument Air Compressors, Plant Air Compressors, and SWACs.		
Previous NRC Exam Use	None		

References Provided	None

K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF8.078.K2.01	Safety Function 8	Tier 2	Group 1	RO Imp: 2.7	SRO Imp: 2.9
Knowledge of bus power supplies to the following: (CFR: 41.7) Instrument air compressor					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.41.08, CFR: 41.8	Components, capacity, and functions of emergency systems.

Question 54	ID: 2115632	Points: 1.00

Which one of the following describes the ESFAS actuation signals that can be **BOTH** actuated **AND** reset from a Control Room pushbutton?

A. SGIS and CSAS

B. SGIS and AFAS Block

C. CIS and AFAS Block

D. CIS and CSAS

Answer	D

Answer Explanation

A. Incorrect. Plausible since SGIS can be both blocked and reset from the Control Room panels but not actuated. The candidate may misinterpret the operation of the unique keyswitch on 1C03 and conclude that SGIS can be both actuated and reset. The second part is correct since CSAS can be actuated and reset from the Control Room panels.

B. Incorrect. SGIS is plausible as described above. AFAS Block is plausible since the candidate will recall that AFAS actuations have unique large square reset pushbuttons on 1C04 and the candidate may misinterpret the operation of these reset pushbuttons. The candidate will also recall that the AFAS Block valve handswitches on 1C04 have three positions including shut, auto, and open. The candidate would then conclude that AFAS Block can be actuated by manually shutting the block valves or overridden by opening the block valves.

C. Incorrect. CIS is correct as stated below. AFAS Block is incorrect but plausible as stated above.

D. Correct. Drawing 61059 shows the manual actuation pushbuttons for both CIS and CSAS on the Control Room panels. Drawing 61059A shows the manual reset pushbuttons for both CIS and CSAS on the Control Room panels.

Question Information

Topic	ESFAS Reset Pushbuttons		
User ID	Q2115632		
System ID	2115632	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	Drawing 61059A Rev 15 Drawing 61059 Rev 25		
Training Objective	Recall the operation of ESFAS that includes reset signals.		
Previous NRC Exam Use	None		

K/A Links

SF5.103.A4.04	Safety Function 5	Tier 2	Group 1	RO Imp: 3.5*	SRO Imp: 3.5*
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Phase A and phase B resets					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.08, CFR: 41.8 Components, capacity, and functions of emergency systems.

Question 55	ID: 2115756	Points: 1.00

Unit-1 is operating at 100% power:

- Containment Cooling System is in a normal lineup with ALL Containment Air Coolers (CACs) available
- 11, 12, and 13 CAC Fans are operating in FAST speed
- 11 CAC Emergency SRW Outlet valve is open

An event occurs resulting in a Reactor Trip with the following conditions:

- All equipment functions as designed upon the trip
- RCS pressure is 1910 PSIA and slowly lowering
- Containment pressure 0.7 PSIG and rising
- Containment humidity for Dome and Rx Cavity are respectively 38% and 52% and both rising
- Containment temperature is 110 °F and rising

Which of the following describes the required operator action, if any, with the CACs for the Containment Environment Safety Function during EOP-0?

A. Start 14 CAC in FAST and ensure open **ALL** CAC Emergency SRW Outlet valves.

B. Start 14 CAC in FAST and ensure open **ALL** CAC Normal SRW Outlet valves.

C. Open the Emergency SRW Outlet valves on **ONLY** 12 and 13 CACs.

D. No additional manipulation of the CACs is required.

Answer	A

Answer Explanation

A. Correct -Since containment pressure is not less than .7psig, alternate actions of EOP-0 require that ALL CACs be started and the Emergency SRW Outlet valves opened.

B. Incorrect - Plausible since the first part is a required action and the candidate misinterprets that only normal SRW cooling is required for the CACs since no ESFAS signal setpoints have been reached.

C. Incorrect - Plausible since EOP-0 does give direction to open Emergency SRW outlet valves for all CACs vice just the ones running.

D. Incorrect - Plausible if the EOP-0 containment pressure value is misinterpreted to equal the CIS setpoint and then no action would be required.

Question Information

Topic	Verify Cntmt Environment Safety Function actions		
User ID	Q2115756		
System ID	2115756	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-00-1, Rev 01300		
Training Objective	For each EOP-0 safety function, analyze and determine correct alternate actions to take.		
Previous NRC Exam Use	2012 NRC RO Exam		

References Provided	NONE
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

GS.3.0.SF5.103	Safety Function 5	Tier 2	Group 1	RO Imp:	SRO Imp:
Containment System					
P2.1.23	Safety Function 5	Tier 3	Group	RO Imp: 4.3	SRO Imp: 4.4
Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 56	ID: 2115845	Points: 1.00

Unit-1 is operating at 100% power:

- The controlling channel of pressurizer level control indication (process variable) fails high

Which of the following describes the automatic actions that occur (1) with letdown, and (2) with pressurizer heaters?

A. (1) Letdown goes to minimum
(2) Backup heaters energize

B. (1) Letdown goes to minimum
(2) Backup heaters de-energize

C. (1) Letdown goes to maximum
(2) Backup heaters energize

D. (1) Letdown goes to maximum
(2) Backup heaters de-energize

Answer	C

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Incorrect. (1) Plausible if the candidate misinterprets the failure of the indication to be impacting the setpoint value of the controller. (2) Plausible is the candidate misinterprets the action on a level deviation of +13 to be to de-energize the heaters.

C. Correct. (1) Per UFSAR 7.4.4.1, a high level indication functions to increase letdown flow. (2) Per UFSAR 7.4.4.2, a high level signal from the controlling channel energizes the backup heaters.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Pressurizer level controller PV fails high		
User ID	Q2115845		
System ID	2115845	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	3
Technical Reference and Revision #	UFSAR 7.4.4, Rev 47		
Training Objective	Given conditions and/or parameters associated with RCS determine if pressurizer heaters and spray are operating properly.		
Previous NRC Exam Use	None		

K/A Links

SF2.011.K6.04	Safety Function 2	Tier 2	Group 2	RO Imp: 3.1	SRO Imp: 3.1
Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: (CFR: 41.7 / 45.7)					
Operation of PZR level controllers					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 57	ID: 2115848	Points: 1.00

U-1 is performing a normal plant shutdown, group 5 CEAs are being inserted.

At what group 5 CEA position should group 4 CEAs start to be inserted using manual sequential operation?

A. As group 5 CEAs approach 90 inches core height, group 4 CEAs will begin insertion sequence.

B. As group 5 CEAs approach 86 inches core height, group 4 CEAs will begin insertion sequence.

C. As group 5 CEAs approach 54.4 inches core height, group 4 CEAs will begin insertion sequence.

D. As group 5 CEAs approach 43.5 inches core height, group 4 CEAs will begin insertion sequence.

Answer	D

Answer Explanation

A. Incorrect. Plausible if the candidate recognizes that the overlap requirement is 90" and misinterprets this to mean once group 5 CEAs approach 90 inches group 4 CEAs will start to move.

B. Incorrect. Plausible since this represents the Transient Insertion Limit of COLR Figure 3.1.6 and if the candidate misinterprets this number for being when group 4 CEAs would begin to move.

C. Incorrect. Plausible since this is the number of the short term steady state insertion limit and if the candidate misinterprets this number for being when group 4 CEAs would begin to move.

D. Correct. Per OP-2, the overlap maintained is 90 inches. Group 4 would be at 133.5" at the start of a downpower. 133.5 - 90 = 43.5.

Question Information

Topic	Sequence of actions that occur during manual sequential insertion of CEAs		
User ID	Q2115848		
System ID	2115848	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	OP-2-1, Rev 05000		
Training Objective	Maintain proper overlap during a reactor startup.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF1.014.A4.01	Safety Function 1	Tier 2	Group 2	RO Imp: 3.3	SRO Imp: 3.1
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Rod selection control					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.02, CFR: 41.2 General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Question 58	ID: 2115853	Points: 1.00

A failure occurs in 1-PIC-100X, Pressurizer Pressure Controller causing the controller to fail low.

As a direct result of this failure, the RPS Pressurizer Pressure trip unit protection circuit will immediately _______.

A. be affected, causing the associated channel to trip.

B. be affected, preventing the associated channel from tripping.

C. NOT be affected, due to separation between instruments.

D. NOT be affected, due to the use of optical isolators.

Answer	C

Answer Explanation

A. Incorrect. Plausible if the candidate misinterprets the failure of 1-PIC-100X and the impact on RPS and determines the unit would trip on TMLP.

B. Incorrect. Plausible if the candidate misinterprets the pressurizer pressure inputs into the RPS system and which parameter feeds the TMLP calculator and that this failure would prevent an active trip from occurring on TMLP.

C. Correct. Per 1C06-ALM and OP-CA-103-102-0200, these systems are fed from different instruments. RPS uses PT-102A through PT-102D instead of PT-100X and PT-100Y that the PZR Pressure controllers use.

D. Incorrect. Plausible since the operator may recall that RPS uses optical isolators in the Wide Range NI circuitry.

Question Information

Topic	Separation of RRS and RPS instrumentation		
User ID	Q2115853		
System ID	2115853	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	OP-CA-103-102-0200, Rev 000 1C06-ALM, Rev 05300		
Training Objective	From memory, identify which temperature, pressure, and level detectors, by design, provide an input to the following systems or controls: Reactor regulating system Reactor protection system		
Previous NRC Exam Use	None		

K/A Links

SF7.016.K5.01	Safety Function 7	Tier 2	Group 2	RO Imp: 2.7*	SRO Imp: 2.8*
Knowledge of the operational implication of the following concepts as they apply to the NNIS: (CFR: 41.5 / 45.7) Separation of control and protection circuits					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 59	ID: 2115856	Points: 1.00

Given the following conditions:

- Unit-2 is in Mode 6 with Core Reload in progress
- Containment Purge has been established
- The Containment Outage Door (COD) is shut
- One of the PAL doors is shut
- A momentary loss of power results in tripping the operating Main Exhaust Fan

Which one of the following correctly describes the effect on personnel entry into containment?

A. Containment pressure rises 2 to 3 PSIG challenging a CIS actuation.

B. Containment and Aux Building differential pressure changes challenging equalization and opening of the PAL doors.

C. Containment temperature rises between 3°F to 5°F challenging personnel heat stress.

D. Containment area radiation monitors (RI-5316A, B, C, and D) counts rise challenging personnel dose rates.

Answer	B

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets the level of impact that Containment Purge ventilation will have on containment pressure. Pressure may change, however, will not be observable on available board indications to this magnitude. The operator may then recall that a CIS actuation will occur at 2.8 psig.

B. Correct. Since the transfer tube isolation valve is open for refueling. OI-36, precaution F, specifically states “IF the Transfer Tube Isolation Valve is open, THEN MONITOR Refueling Pool and Spent Fuel Pool levels closely when performing any evolution that may create or change differential pressure between the Containment and

Auxiliary Buildings. Unanticipated changes in Refueling Pool and Spent Fuel Pool water levels can occur following such evolutions."

C. Incorrect. Plausible since the candidate will recognize that ventilation is a primary means of controlling Containment temperature and will conclude that a heat stress condition will occur. But, any temperature change would be small as Containment Air Cooling is running and temperature will not rise as a result.

D. Incorrect. Plausible since the candidate may conclude that these monitors are affected by a loss of ventilation to the containment. Incorrect, as only a fuel handling incident occurring (radiation levels changing) or respective loss of power to instruments would affect these monitors.

Question Information

Topic	Effect following loss of the Main Exhaust Fan on Cntmt Purge		
User ID	Q2115856		
System ID	2115856	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Modified	**Difficulty**	
Technical Reference and Revision #	OI-36 Rev 03000		
Training Objective	Identify the systems that require a Main Vent Exhaust fan running.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF8.029.K3.02	Safety Function 8	Tier 2	Group 2	RO Imp: 2.9*	SRO Imp: 3.5*
Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following: (CFR: 41.7 / 45.6) Containment entry					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.07, CFR: 41.7 Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question 60	ID: 2115858	Points: 1.00

Unit 1 is in MODE 6

- Refueling Pool level is 63 feet in preparation for Upper Guide Structure (UGS) removal
- A control circuit malfunction causes 1-SI-652-MOV (SDC HDR RETURN ISOL) to shut
- **ALL** attempts to reopen 1-SI-652-MOV are unsuccessful

Which of the following completes the statement below?

The optimal strategy to restore decay heat removal is to ______.

A. Use a SFP Cooling Pump to cool the Refueling Pool

B. Initiate Once-Thru-Core-Cooling using a HPSI Pump or LPSI Pump

C. Open the SFP Transfer Gate Valve and cool with the SFP cooling system

D. Fill the Refueling Pool using a LPSI Pump and makeup as inventory is lost to boil-off

Answer	A

Answer Explanation

A. Correct - Per AOP-3B, Section VII, for common mode loss of SDC with the RFP that is or can be filled a SFP pump is the optimal recovery method.

B. Incorrect - Plausible since these actions are in AOP-3B when the RFP is not available and the candidate misinterprets this.

C. Incorrect - Plausible if the candidate misinterprets the need for extra water in the RFP and determines that the SFP Transfer Gate should be opened.

D. Incorrect - Plausible since Boil-off of the Refueling Pool inventory is an option in AOP-3B and the candidate misinterprets that this is the only method available with the current plant conditions.

Question Information

Topic	A complete loss of SDC occurs w/RFP level at 63 Ft.		
User ID	Q2115858		
System ID	2115858	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	AOP-3B-1, Rev 03001		
Training Objective	From memory, describe the physical connection and recall the relationship that exists between the Spent Fuel Pool Cooling system and the following plant components or systems, without error: Shutdown cooling.		
Previous NRC Exam Use	2014 NRC RO Exam		

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF8.033.K1.02	Safety Function 8	Tier 2	Group 2	RO Imp: 2.5	SRO Imp: 2.7
Knowledge of the physical connections and/or cause- effect relationships between the Spent Fuel Pool Cooling System and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) RHRS					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 61	ID: 2115961	Points: 1.00

Unit-1 is operating at 100% power:

- The 11 Main Feedwater Regulating Valve (MFRV) starts to fail open due to a controller issue, causing the level in the 11 SG to rise.
- The 11 Main Feedwater Regulating Bypass Valve (BFRV) remains shut.
- The 12 MFRV position, 12 BFRV position, and 12 SG level have not changed.
- The appropriate procedure for this condition is implemented.

Which of the following is the FIRST action the crew should perform to restore 11 SG level to normal?

A. Trip one of the SGFPs and attempt to restore normal SG level.

B. Shift 11 SGFP speed controller, 1-HIC-4516/4518, to Manual and lower pump speed to control 11 SG level.

C. Ensure 11 MFRV controller, 1-FIC-1111, has shifted to manual and adjust the FIC to restore SG level to normal.

D. Ensure 11 BFRV controller, 1-FIC-1105, has shifted to manual and adjust the FIC to restore SG level to normal.

Answer	C

Answer Explanation

A. Incorrect. Plausible if candidate misinterprets the failure of the MFRV to only impact one SGFP and understands that level is getting to high and needs to take action to lower steam generator level.

B. Incorrect. Plausible since this an action in AOP-3G if level continues to rise after attempting to control the MFRV controller.

C. Correct. Per AOP-3G, the actions for this failure are to ensure 1-FIC-1111 has shifted to manual and take control of the FIC to restore SG level.

D. Incorrect. Plausible since on certain failures of the MFRV controller the BFRV would need to be used to restore SG level.

Question Information

Topic	12 MFRV starts opening: identify actions to restore SG level		
User ID	Q2115961		
System ID	2115961	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-3G-1, Rev 01300		
Training Objective	Given a set of plant conditions, demonstrate an understanding of the strategy and basis of AOP-3G and the 1C03 Alarm Manual for the following situations: Malfunction of the feedwater regulating system.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information

SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

SF4.035.A3.01	Safety Function 4	Tier 2	Group 2	RO Imp: 4.0	SRO Imp: 3.9
Ability to monitor automatic operation of the S/G including: (CFR: 41.7 / 45.5) S/ G water level control					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 62	ID: 2115862	Points: 1.00

Unit-1 was operating at 100% power when the following occurs:

- Main Condenser Vacuum lowers to 21" Hg
- RPS fails to trip the Reactor
- Automatic actions associated with the Main Turbine, if any, have occurred

Which of the following correctly describes the effect on secondary and primary parameters, **prior** to manually tripping the Reactor?

A. SG pressure rises, RCS temperature rises, pressurizer pressure rises.

B. SG pressure rises, RCS temperature lowers, pressurizer pressure lowers.

C. SG pressure lowers, RCS temperature rises, pressurizer pressure rises.

D. SG pressure lowers, RCS temperature lowers, pressurizer pressure lowers.

Answer	A

Answer Explanation

A is correct: At the given vacuum of 21" Hg, a main turbine trip occurs on Unit-1. With a failure of the reactor to trip, SG pressure will rise due to the heat removal from the main turbine being removed. The RCS temperature and pressure also rise due to less heat removal from the SGs.

B is incorrect: Plausible since SG pressure increases when Turbine stop valves close which the operator may conclude as causing a SG safety valve to lift which will lower RCS temperature and lower pressurizer pressure.

C is incorrect: Plausible since the operator may misinterpret the given vacuum of 21" Hg to cause the SGFPs to trip and a loss of main feedwater. The operator may then conclude a lowering inventory and makeup volume in the SGs will cause the SG pressure to lower. Then the operator will interpret the loss of feedwater to cause RCS temperature to rise as well as pressurizer pressure to rise.

D is incorrect: Plausible since the operator may misinterpret the given vacuum of 21" Hg to cause the SGFPs to trip and a loss of main feedwater. The operator may then conclude a lowering inventory and makeup volume in the SGs will cause SG pressure to lower which will then lower RCS temperature and pressurizer pressure.

Question Information

Topic	Turbine-Reactor trip effect on primary and secondary parameters		
User ID	Q2115862		
System ID	2115862	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Modified	**Difficulty**	3
Technical Reference and Revision #	EOP-0 Technical Basis Rev 02100 USFAR Chapter 14.5, Rev 49		
Training Objective	Identify the relationship between the Main Turbine and the following: Reactor Coolant System.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No Additional
SRO-Only Justification	N/A

Additional Information	N/A

K/A Links

SF4.045.A1.05	Safety Function 4	Tier 2	Group 2	RO Imp: 3.8	SRO Imp: 4.1
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: (CFR: 41.5 / 45.5) Expected response of primary plant parameters (temperature and pressure) following T/G trip					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 63	ID: 2116006	Points: 1.00

Unit-1 is operating at 100% power:

- A liquid waste discharge is in progress
- The following alarm on 1C22 is received:

LIQUID WASTE DISCH

Which of the following describes:

(1) The possible cause of this alarm?

And,

(2) If the alarm is valid and the liquid waste discharge valves, 0-MSW-2201 and 0-MSW-2202, fail to shut, what is the required action?

A. (1) High Activity
(2) Stop the running RCW Pump.

B. (1) High Activity
(2) Throttle 0-RCW-344, RCW Effluent Discharge Header to CW Discharge B/U Valve.

C. (1) High Flow Rate
(2) Stop the running RCW Pump.

D. (1) High Flow Rate
(2) Throttle 0-RCW-344, RCW Effluent Discharge Header to CW Discharge B/U Valve.

Answer	A

Answer Explanation

A. Correct. (1) Per 1C22-ALM, this alarm is based on RMS readings. (2) Per AOP-06B, if RIC-2201 alarms during a discharge and the valves fail to shut then stop the running RCWRT Pump.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since there are plant computer alarm setpoints for discharge flow rates that will alarm if the flow is too high and the candidate misinterprets those alarms to feed this alarm window. (2) Plausible since this action would be taken on a high flow rate condition and also if the candidate misinterprets that high activity can be mitigated by lower the flow rate to increase the dilution effect.

Question Information

Topic	Failure of Automatic isolation of RIC-2201		
User ID	Q2116006		
System ID	2116006	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	Embedded Reference
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	1C22-ALM, Rev 04400 AOP-6B, Rev 01200 OI-17C-4, Rev 01200		
Training Objective	Determine the correct actions to terminate any liquid release from the Waste Processing System.		
Previous NRC Exam Use	None		

K/A Links

SF9.068.A2.04	Safety Function 9	Tier 2	Group 2	RO Imp: 3.3	SRO Imp: 3.3
Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Failure of automatic isolation					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.11, CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Question 64	ID: 2120932	Points: 1.00

- The Resource Assessment Table of EOP-8, Functional Recovery Procedure, is being evaluated
- HR-2, Steam Generator Heat Sink with Safety Injection System in Operation, resource conditions apply
- RCS Pressure is 500 PSIA

To determine whether the HR-2 Acceptance Criteria of less than 50°F superheated is met or not met, which one of the following describes:

(1) What temperature parameters are used?
And,
(2) What is the highest temperature that will meet the HR-2 Acceptance Criteria?

A. (1) T_{AVE} temperatures
(2) 491°F

B. (1) T_{AVE} temperatures
(2) 516°F

C. (1) CET temperatures
(2) 516°F

D. (1) CET temperatures
(2) 491°F

Answer	C

Answer Explanation

A. Incorrect. (1) Incorrect but plausible since the operator may incorrectly recall that it is TAVE that is used in EOP-8 HR-2 to verify Natural Circulation is adequate. Also the operator may believe that TAVE is used in EOP-8 HR-2 as the trigger for isolation of a Steam Generator. (2) Incorrect but plausible since 491°F represents the highest temperature that is within a limit of 25°F from the saturation temperature of 467°F at 500 PSIA. The operator will recall that a limit of 25°F is used throughout EOP-8 including HR-2 for the RCS subcooling band, temperature difference during a Steam Generator blowdown event, as well as for verification of Natural Circulation.

B. Incorrect. (1) Incorrect but plausible as stated above. (2) Correct as stated below.

C. Correct. (1) Correct per EOP-8 HR-2 Resource Assessment Table which states that CET temperatures be less than 50°F superheated. (2) Correct per EOP-8 HR-2 Acceptance Criteria which states a limit of 50°F superheated. At an RCS Pressure of 500 PSIA, the saturation temperature is 467°F as derived from the steam tables. The highest CET temperatures that will still meet the Acceptance Criteria must be less than 517°F (467°F plus 50°F).

D. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated above.

Question Information

Topic	Incore Temperature Parameters		
User ID	Q2120932		
System ID	2120932	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only

Question Type	New	**Difficulty**	
Technical Reference and Revision #	EOP-8-1 Rev 04101		
Training Objective	Given a set of plant conditions, identify the success paths for the Safety Functions of EOP-8.		
Previous NRC Exam Use	None		

K/A Links

GS.3.0.SF7.017	Safety Function 7	Tier 2	Group 2	RO Imp:	SRO Imp:
In-Core Temperature Monitor System (ITM)					
P2.4.21	Safety Function 7	Tier 3	Group	RO Imp: 4.0	SRO Imp: 4.6
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. (CFR: 41.7 / 43.5 / 45.12)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 65	ID: 2115966	Points: 1.00

Unit-1 is at 100% power:

- The Fire system is in a normal lineup

Which of the following describes:

(1) The pump that maintains system pressure in the normal band?

And,

(2) The fire pump that will start if system pressure drops to 94 psig?

A. (1) Main Pressurizer Pump
(2) Electric Fire Pump

B. (1) Main Pressurizer Pump
(2) Diesel Fire Pump

C. (1) Booster Jockey Pump
(2) Electric Fire Pump

D. (1) Booster Jockey Pump
(2) Diesel Fire Pump

Answer	A

Answer Explanation

A. Correct. (1) Per OI-20, the main pressurizer pump will run as necessary to maintain system pressure. (2) Per OI-20, the electric fire pump with start if system pressure drops to between 92 and 94 PSIG.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since this pump is used to maintain pressure when the fire system is being used for non fire system operation but is normally keep in Stop. (2) Plausible since both the diesel and electric fire pumps will start as system pressure lowers and the candidate misinterprets which pump starts first.

Question Information

Topic	Maintenance of fire header pressure		
User ID	Q2115966		
System ID	2115966	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	OI-20, Rev 03500		
Training Objective	From memory, state the basic function and location of the system components including: 11 Fire Pump, 12 Fire Pump, Main Pressurizer Pump		

Previous NRC Exam Use	None

K/A Links

SF8.086.K4.02	Safety Function 8	Tier 2	Group 2	RO Imp: 3.0	SRO Imp: 3.4
Knowledge of design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7) Maintenance of fire header pressure					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.04, CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.

Question 66	ID: 2116008	Points: 1.00

Which of the following describes the technical specification requirements when moving irradiated fuel assemblies inside containment:

(1) the Containment outage door ______.

And,

(2) both doors in the ______ air lock can be open as long as one door is capable of being closed.

A. (1) must be closed
(2) emergency

B. (1) must be closed
(2) personnel

C. (1) can be open but must be capable of being closed
(2) emergency

D. (1) can be open but must be capable of being closed
(2) personnel

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible if the candidate misinterprets the requirements for the equipment hatch and the containment outage door. (2) Plausible if the candidate misinterprets the closure requirements for the two access doors.

B. Incorrect. (1) See above. (2) Correct.

C. Incorrect. (1) Correct. (2) See above.

D. Correct. (1) Per TS 3.9.3, the COD must be capable of being closed this is checked by the Refueling Reactor Operator per the checklist in OP-7, Shutdown Operations. (2) Per TS 3.9.3, both personnel air lock doors must be capable of being closed this is checked by the Refueling Reactor Operator per the checklist in OP-7, Shutdown Operations.

Question Information

Topic	Containment penetrations during fuel moves		
User ID	Q2116008		
System ID	2116008	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	TS 3.9.3, Amend 281		
Training Objective	Establish and maintain administrative and plant conditions to perform refueling operations.		
Previous NRC Exam Use	None		

K/A Links

P2.1.42	Safety Function 8	Tier 3	Group	RO Imp: 2.5	SRO Imp: 3.4
Knowledge of new and spent fuel movement procedures. (CFR: 41.10 / 43.7 / 45.13)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 67	ID: 2116009	Points: 1.00

Unit-2 is operating at 100% power:

- An uncomplicated reactor trip occurs
- EOP-0, Post-Trip Immediate Actions, is implemented
- The CRO is performing Core and RCS Heat Removal

Which of the following completes the statements below:

Per EOP-0, the CRO will verify that T_{HOT} minus T_{COLD} is ___(1)___ °F, which validates that ___(2)___.

A. (1) less than 50
(2) proper flow is being obtained from the operating RCPs

B. (1) less than 50
(2) pressurizer level indication as being representative of total RCS inventory

C. (1) less than 10
(2) proper flow is being obtained from the operating RCPs

D. (1) less than 10
(2) pressurizer level indication as being representative of total RCS inventory

Answer	C

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Incorrect. (1) Incorrect but plausible since this is the maximum temperature value used in EOPs for the verification of natural circulation flow. The operator will recognize this value as being used in EOPs. (2) Incorrect but plausible since per EOP-0-TB and with the understanding of the relationship of pressurizer level and subcooling, subcooling can then be used to validate pressurizer level.

C. Correct. (1) Per EOP-0, the CRO will verify T_{HOT} minus T_{COLD} is less than 10°F. (2)

Per EOP-0-TB and with the understanding of the relationship of the RCS temperature differential and forced circulation, this ensures that proper flow is being obtained from the operating RCPs.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	Validation of parameters		
User ID	Q2116009		
System ID	2116009	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EOP-00-2, Rev 01400 EOP-00-TB, Rev 02100		
Training Objective	For each EOP-0 Safety Function, recall the basis for EOP-0 desired parameters and conditions which make the safety		

	function met.
Previous NRC Exam Use	None

K/A Links

P2.1.45	Safety Function 8	Tier 3	Group	RO Imp: 4.3	SRO Imp: 4.3
Ability to identify and interpret diverse indications to validate the response of another indication. (CFR: 41.7 / 43.5 / 45.4)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 68	ID: 2116010	Points: 1.00

Which of the following meets the MINIMUM requirement for Source Range Nuclear Instrumentation prior to commencing core off-load in MODE 6?

A. 2 Visual in CR
0 Audible in CR and 1 Audible in CTMT

B. 2 Visual in CR
1 Audible in CR and 1 Audible in CTMT

C. 4 Visual in CR
0 Audible in CR and 1 Audible in CTMT

D. 4 Visual in CR
1 Audible in CR and 1 Audible in CTMT

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) This is the minimum requirement per TS 3.9.2 this is checked by the Refueling Reactor Operator per the checklist in OP-7, Shutdown Operations. (2) This is the minimum requirement per OP-7 Mode 6 checklist.

C. Incorrect. (1) Plausible since the normal required number of Source Range NIs when at power is 4 channels. (2) Plausible since the CR is required to ensure the containment watchstanders can hear the audible count rate and the candidate misinterprets this to mean that the audible is only required in containment.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Minimum SR channel requirements to begin fuel movement		
User ID	Q2116010		
System ID	2116010	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	TS 3.9.2, Amend 279 OP-7, Rev 05600		
Training Objective	Given conditions for the following parameters, discriminate whether refueling may be commenced or continued in accordance with refueling procedures: Wide Range NIs.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

P2.1.44	Safety Function 8	Tier 3	Group	RO Imp: 3.9	SRO Imp: 3.8
Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation. (CFR: 41.10 / 43.7 / 45.12)					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.41.10, CFR: 41.10	Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 69	ID: 2116012	Points: 1.00

When preparing a tagout, which of the following sets of values independently require the use of Double valve isolation?

(1) Fluid Temperature

And,

(2) System Pressure

A. (1) > 200 °F
(2) > 200 PSIG

B. (1) > 200 °F
(2) > 500 PSIG

C. (1) > 120 °F
(2) > 200 PSIG

D. (1) > 120 °F
(2) > 500 PSIG

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) Per OP-AA-109-101, double valve isolation should be provided when temperatures are greater 200 °F. (2) Per OP-AA-109-101, double valve isolation should be provided when pressures are greater than 500 PSIG.

C. Incorrect. (1) Plausible since this temperature is the definition of hazardous energy and requires the use of tagout and the candidate misinterprets this to also be the setpoint for double isolation. (2) Plausible if the candidate misinterprets the pressure limit to be equal to the actual temperature limit.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	Knowledge of Safety Tagging procedures		
User ID	Q2116012		
System ID	2116012	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	2
Technical Reference and Revision #	OP-AA-109-101, Rev 14		
Training Objective	Apply the requirements of OP-AA-109-101, Personnel and Equipment Tagout Process.		
Previous NRC Exam Use	2014 NRC RO Exam		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

P2.2.13	Safety Function 8	Tier 3	Group	RO Imp: 4.1	SRO Imp: 4.3
Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Question 70	ID: 2123086	Points: 1.00

Unit-2 is at 100% power:

- An uncomplicated reactor and turbine trip occur

Which of the following is directed by EOP-0, Post Trip Immediate Actions, to prevent an uncontrolled cooldown on Unit-2?

A. Ensure the Main Steam to MSR Second Stage Control Valves shut.

B. Ensure MSR 2nd Stage Steam Source MOVs shut.

C. Shut Upstream Drain MOVs.

D. Trip the S/G Feed Pumps.

Answer	A

Answer Explanation

A. Correct - Per U-2 EOP-0 step B.3 basis, by verifying steam isolated to the MSRs, the operator prevents uncontrolled cooldown due to MSR failures.

B. Incorrect - Plausible since this verification is directed when performing EOP-0, Post Trip Immediate Actions, on Unit-1.

C. Incorrect - Plausible since this is a mitigating strategy in EOP-1 for controlling cooldown. There is no direction to shut upstream drain valves in EOP-0 and leaking drain valves will have a small effect on RCS temperature immediately after a trip.

D. Incorrect – Plausible if the candidate misinterprets that tripping the SGFPs will help to prevent an uncontrolled cooldown since it will help to alleviate any excessive feeding which can sometimes cause a cooldown but this is not the right action for an uncomplicated reactor trip.

Question Information

Topic	Preventing an uncontrolled cooldown on a U-2 Rx trip		
User ID	Q2123086		
System ID	2123086	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	EOP-0, Rev 01400 EOP-0-TB, Rev 02100		
Training Objective	Recall how a Unit 1 and Unit 2 Turbine trip are verified.		
Previous NRC Exam Use	2008 RO NRC Exam		

References Provided	None
K/A Justification	This question asks about the procedural differences when performing the steps for turbine trip between Unit-1 and Unit-2.
SRO-Only Justification	N/A
Additional Information	N/A

K/A Links

P2.2.4	Safety Function 8	Tier 3	Group	RO Imp: 3.6	SRO Imp: 3.6
(multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility. (CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13)					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.04, CFR: 41.4 Secondary coolant and auxiliary systems that affect the facility.
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Question 71	ID: 2116033	Points: 1.00

Unit-1 is operating at 100%:

- A S/G tube leak of approximately 60 GPM develops.

Given the above condition, which of the following steps describe:

The action directed by AOP-2A, Excessive Reactor Coolant Leakage, to minimize the release of radiation?

A. Borate to reduce T_{AVE} < 537 °F.

B. Reduce turbine load to maintain T_{COLD} on program.

C. Ensure no leakage into Component Cooling subsystem.

D. Direct Rad Safety to survey the Water Treatment Area.

Answer	A

Answer Explanation

A. Correct. The condition provided is Steam Generator tube leakage greater than the capacity of a single Charging Pump and the stated action is directed per Section VI Block Step D. AOP-2A basis states: “the intent of this step is to reduce T_{AVE} to less than 537 °F if possible before tripping the Reactor to minimize the possibility of lifting a Steam Generator Safety Valve. The setpoint of the lowest lifting main steam safety valve corresponds to a saturation temperature of approximately 537 °F.

B. Incorrect. Plausible since this is the normal action on most rapid downpowers to ensure that the reactor and turbine are coupled preventing unnecessary lifting of PORVs. For a SGTR, Tcold on the downpower is controlled differently to help prevent the SG safeties from lifting.

C. Incorrect. Plausible since this step would prevent radiation releases from an RCS leak into the Component Cooling system and is directed in Block Step E of AOP-2A Section VI but is not correct for the condition of Steam Generator tube leakage.

D. Incorrect. Plausible since this step is directed by AOP-2A Section V Block Step D for Steam Generator tube leakage less than the capacity of a single Charging Pump. The step states to direct begin taking surveys in order to establish radiological controlled areas. This action, although incorrect for the conditions provided in the stem, allows Rad Safety to monitor and control the plant areas necessary to prevent the spread of contamination.

Question Information

Topic	Control of radiation releases during a SGTR Event		
User ID	Q2116033		
System ID	2116033	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-2A-1, Rev 02701 AOP-2A-TB, Rev 02301		
Training Objective	From memory and given a set of plant conditions, demonstrate an understanding of the strategy and basis of AOP-2A.		
Previous NRC Exam Use	2014 NRC RO Exam		

References Provided	None

K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

P2.3.11	Safety Function 8	Tier 3	Group	RO Imp: 3.8	SRO Imp: 4.3
Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)					

Associated Objective(s)

RO NRC Test **User (Sys) ID** N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.05, CFR: 41.5 Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
10CFR55.41.12, CFR: 41.12 Radiological safety principles and procedures.

Question 72	ID: 2116042	Points: 1.00

Which conditions below would cause alarm window, F-21, "RAD MON LVL HI" to alarm on 1C07?

A. A 10 gpm RCS Leak.

B. A CRUD Burst in the RCS.

C. A 100 gpm SG Tube Rupture.

D. A Main Steam Line Rupture in containment.

Answer	B

Answer Explanation

A. Incorrect. Plausible since an RCS leak would cause radiation levels to rise in the plant. The operator may misinterpret how the given alarm from the CVCS Letdown Rad Monitor may detect an increase in area radiation rather than the increase detected by a process Rad Monitor.

B. Correct. Per 1C07-ALM, a crud burst is a possible cause of this alarm.

C. Incorrect. Plausible since numerous Rad Monitor alarms will actuate on this accident in the plant. The operator may misinterpret how the given alarm from the CVCS Letdown Rad Monitor may detect an increase in area radiation rather than the increase detected by a process Rad Monitor. Also plausible if the operator misinterpret which process Rad Monitor causes the given alarm and may conclude that a SGTR will cause the alarm.

D. Incorrect. Plausible if thought that a MSLR in containment would cause fuel damage. Although plausible that fuel damage may occur from a large steam line break, SIAS or required operator action will isolate Letdown quickly on this casualty.

Question Information

Topic	Cause of alarm window, F-21, "RAD MON LVL HI" alarm		
User ID	Q2116042		
System ID	2116042	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	2
Technical Reference and Revision #	1C07-ALM, Rev 03708		
Training Objective	From memory, state the purpose and functions of CVCS and its components, without error.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

P2.3.14	Safety Function 8	Tier 3	Group	RO Imp: 3.4	SRO Imp: 3.8
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.11, CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Question 73	ID: 2116043	Points: 1.00

Which one of the following radiation monitors must be clear of alarms in order to meet the Radiation Levels External to Containment safety function during EOP-0?

A. S/G B/D RMS (1-RI-4014).

B. Main Steam Line Monitor (1-RE-5421).

C. Control Room Ventilation Monitor (1-RE-5350).

D. Containment Hi-Range Monitor (1-RE-5317A/B).

Answer	A

Answer Explanation

A. Correct. Per EOP-0, alarms must be clear on 1-RI-4014.

B. Incorrect. Plausible since this is one of 3 monitors that indicates a SGTR and the other 2 monitors are monitored in RLEC.

C. Incorrect. Plausible since this is a rad monitor external to containment and the candidate misinterprets that is required for RLEC assessment.

D. Incorrect. Plausible since this monitor is used in the EOP-0 safety function assessments just not for RLEC.

Question Information

Topic	Which rad monitors must be clear of alarms in order to satisfy RLEC in EOP-0?		
User ID	Q2116043		
System ID	2116043	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	2
Technical Reference and Revision #	EOP-0, Rev 01300		
Training Objective	For each EOP-0 Safety Function, analyze and determine correct actions to take.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

P2.3.5	Safety Function 8	Tier 3	Group	RO Imp: 2.9	SRO Imp: 2.9
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.11, CFR: 41.11 Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Question 74	ID: 2116150	Points: 1.00

Per TRM 15.7.5, Fire Suppression Water System, which of the following completes the statement below:

___(1)___ fire pumps and a minimum water volume of ___(2)___ gallons available in **EACH** of the Pretreated Water Storage Tanks are required.

A. (1) Three
(2) 300,000

B. (1) Three
(2) 400,000

C. (1) Two
(2) 300,000

D. (1) Two
(2) 400,000

Answer	C

Answer Explanation

A. Incorrect. (1) Plausible since the operator will recall that a third normally running pressurizer pump maintains system pressure and may then conclude it is also one of the TRM required pumps. (2) Correct as stated below.

B. Incorrect. (1) See above. (2) See below.

C. Correct. (1) Per TRM 15.7.5, 2 fire pumps are required. (2) Per TRM 15.7.5, a total of 600,000 gallons is required with each tank needing 300,000 gallons.

D. Incorrect. (1) Correct. (2) Incorrect but plausible since the operator may incorrectly recall the required volume for a similarly sized tank surveilled by the Outside Operator.

Question Information

Topic	TRM requirements for Pumps and PWST		
User ID	Q2116150		
System ID	2116150	Point Value	1.00
Status	Active	Time to Complete	0
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	TRM 15.7.5, Rev 00300		
Training Objective	Identify the impact of system operability on Technical Requirements Manual.		
Previous NRC Exam Use	None		

K/A Links

P2.4.26	Safety Function 8	Tier 3	Group	RO Imp: 3.1	SRO Imp: 3.6
Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage. (CFR: 41.10 / 43.5 / 45.12)					

Associated Objective(s)

RO NRC Test

User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.08, CFR: 41.8 Components, capacity, and functions of emergency systems.

Question 75	ID: 2116153	Points: 1.00

Which of the following completes the statement below:

The purpose of AOP-9A, Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire, is to ___(1)___.

And,

AOP-9A directs establishing ___(2)___.

A. (1) achieve and maintain Hot Standby
(2) AFW flow within 30 minutes

B. (1) achieve and maintain Hot Standby
(2) Charging within 30 minutes

C. (1) achieve and maintain Cold Shutdown
(2) AFW flow within 30 minutes

D. (1) achieve and maintain Cold Shutdown
(2) Charging within 30 minutes

Answer	A

Answer Explanation

A. Correct. (1) Per AOP-9A, the purpose is to place the plant in Hot Standby. (2) Per AOP-9A notes, it is important to establish AFW flow within 30 minutes.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since this was the requirement for the last revision of AOP-9A, and ultimately the plant will transition to cold shutdown, but this is not a purpose of AOP-9A. (2) Plausible since establishing Charging within a specified time was a requirement for the last revision of AOP-9A, and Charging will be established in Block Step BP.

Question Information

Topic	AOP-9A purpose and timed actions		
User ID	Q2116153		
System ID	2116153	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	RO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-9A, Rev 01901		
Training Objective	From memory, summarize the time critical actions associated with AOP-9A.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	Not applicable
Additional Information	No additional information

K/A Links

P2.4.27	Safety Function 8	Tier 3	Group	RO Imp: 3.4	SRO Imp: 3.9
Knowledge of "fire in the plant" procedures. (CFR: 41.10 / 43.5 / 45.13)					

Associated Objective(s)

RO NRC Test
User (Sys) ID N/A (1527461)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.41.10, CFR: 41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Exelon Generation

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VISION Report

(Test Key) 2020 ILT NRC SRO EXAM

EXELON Nuclear
CCNPP Operations NRC Examinations

Generated January 14, 2020

Test	2020 ILT NRC SRO EXAM
VISION ID	329628
Status	

Question 76	ID: 2114043	Points: 1.00

SRO ONLY

Given the following conditions on Unit-1:

- 30 minutes ago, an uncomplicated Reactor Trip from 100% power occurred
- All EOP-0 Safety Functions were reported Met
- Implementation of EOP-1 is currently in progress

Then, the following alarms are received:

- PZR CH 100 PRESS, 1C06 Alarm Window E-29
- PZR CH X LVL, 1C06 Alarm Window E-33
- PZR CH Y LVL, 1C06 Alarm Window E-34
- PZR HTR CUTOFF, 1C06 Alarm Window E-35
- MAIN STM EFFL RAD MON 2C26, 1C03 Alarm Window C-43
- UNIT 1 S/G B/D, 1C22 Alarm Window A-4.2

Assuming a procedure deviation is not declared, which one of the following actions will the Unit Supervisor direct the crew to perform?

A. Remain in the current EOP for up to 60 minutes

B. Remain in the current EOP and concurrently implement the appropriate AOP for up to 45 minutes

C. Transition to the appropriate Optimal Recovery EOP within 30 minutes

D. Transition to the Functional Recovery EOP within 15 minutes

Answer	C

Answer Explanation

A. Incorrect. Plausible since the candidate will recognize that all four Pressurizer alarms are consistent with and expected during a Reactor Trip. The candidate may conclude that implementation of EOP-1 will continue until the Final Safety Function Status

Checks are completed. If the Main Steam Effluent Rad Monitor alarm is recognized as not consistent with plant conditions, the candidate may conclude that a procedure deviation is only required after 60 minutes.

B. Incorrect. Plausible since the candidate may recall that the E-33, E-34, E-35, and C-43 alarms all state to reference AOP-2A Excessive Reactor Coolant Leakage. The candidate may also conclude that the total time before a procedure deviation is required is 30 minutes of EOP-1 implementation combined with an additional 15 minutes of AOP-2A implementation.

C. Correct. The six listed alarms are not expected or consistent 30 minutes after an uncomplicated Reactor Trip. The four PZR alarms received upon the Reactor Trip will all have cleared prior to the given time in the question stem. All alarms indicate that a Steam Generator Tube Rupture has occurred and the transition to EOP-6 will be required within 30 minutes unless a procedure deviation is declared. The EOP-1 Safety Function Status Checks are no longer met for Radiation Levels External to Containment and for RCS Pressure and Inventory.

D. Incorrect. Plausible since the candidate may misinterpret the alarms received to have been caused by a LOCA and a Steam Generator Tube Rupture and conclude that the Functional Recovery EOP-8 is required.

Question Information

Topic	SRO-Alarms Consistent with Plant Conditions		
User ID	Q2114043		
System ID	2114043	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	The question matches the K/A since it requires knowledge of alarms that are consistent with EOP-1, Reactor Trip.

SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	OP-CA-103-102-1001 Rev 006 EOP-1-1 Rev 01500 1C06 Alarm Manual Rev 05300		
Training Objective	Determine plant conditions requiring EOP-1 be exited.		
Previous NRC Exam Use	None		

K/A Links

P2.4.46	Safety Function	Tier 3	Group	RO Imp: 4.2	SRO Imp: 4.2
Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)					
GE.4.0.EPE.007	Safety Function 1	Tier 1	Group 1	RO Imp:	SRO Imp:
Reactor Trip					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 77	ID: 2114253	Points: 1.00

SRO ONLY

Using provided references:

Unit-1 is operating at 100% power with the following indications:

- An RCS Leak of 15 gpm exists
- Visual inspection confirms a through wall leak of an RCS pipe is the source
- The Wide Range Noble Gas Monitor is the only rising RMS indication

Which one of the following correctly describes:

(1) What is the location of the RCS Leakage?
And,
(2) If the leak is not isolated, what action is required per the Technical Specifications?

A. (1) Charging line downstream of the Regenerative Heat Exchanger
(2) Be in Mode 3 within 6 hours, and be in Mode 5 within 36 hours

B. (1) Charging line downstream of the Regenerative Heat Exchanger
(2) Reduce Leakage to within limits within 4 hours

C. (1) Charging line in the 27' West Penetration Room
(2) Reduce Leakage to within limits within 4 hours

D. (1) Charging line in the 27' West Penetration Room
(2) Be in Mode 3 within 6 hours, and be in Mode 5 within 36 hours

Answer	D

Answer Explanation

A. Incorrect. (1) Incorrect as stated below. (2) Correct as stated below.

B. Incorrect. (1) Plausible since the candidate will recall that the Charging Pump discharge flowpath is through the Regenerative Heat Exchanger. The candidate may misinterpret the ventilation flowpath that would cause a rising indication on the Wide Range Noble Gas Monitor. (2) Plausible since Tech Spec 3.4.13.A has the required action to reduce leakage to within limits within 4 hours. The candidate may misinterpret the description of a welded pipe joint to be a mechanical connection and would then conclude that pressure boundary leakage does not exist.

C. Incorrect. (1) Correct as stated below. (2) Incorrect as stated above.

D. Correct. (1) The Wide Range Noble Gas Monitor will only experience a rising indication for leaks within the Aux Building, such as the 27' West Penetration Room, and not for leaks inside Containment. (2) A through wall leak of an RCS pipe is pressure boundary leakage so Tech Spec 3.4.13.B applies and has the required actions to be in Mode 3 within 6 hours and be in Mode 5 within 36 hours.

Question Information

Topic	SRO-References: Charging Line Leakage		
User ID	Q2114253		
System ID	2114253	**Point Value**	1.00
Status	Active	**Time to Complete**	4
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	Tech Spec 3.4.13, pages 3.4.13-1 thru 3.4.13-3 and 3.4.14
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only

Question Type	New	**Difficulty**	
Technical Reference and Revision #	Tech Spec 3.4.13 Rev 278/255 1C10-ALM, Rev 05102		
Training Objective	Given a known leak rate value, determine the required actions in accordance with Technical Specifications.		
Previous NRC Exam Use	None		

K/A Links

APE.022.AA2.01	Safety Function 2	Tier 1	Group 1	RO Imp: 3.2	SRO Imp: 3.8
Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Pump Makeup: (CFR: 43.5 / 45.13) Whether charging line leak exists					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.02, CFR: 43.2	Facility operating limitations in the technical specifications and their bases.

Question 78	ID: 2114215	Points: 1.00

SRO ONLY

Unit-1 is in MODE 3 preparing for the reactor startup.

- A pressure control malfunction occurs and RCS pressure rises to 2760 PSIA

(1) What are the required actions and (2) basis for the required actions?

A. (1) Restore RCS pressure to within Safety Limit values within 5 minutes:
(2) Exceeding RCS pressure Safety Limit is more severe in lower modes.

B. (1) Restore RCS pressure to within Safety Limit values within 2 hours;
(2) RCS pressure exceeds 125% of design pressure, system integrity is challenged

C. (1) Restore RCS pressure to within Safety Limit values within 5 minutes
(2) RCS pressure exceeds 125% of design pressure, system integrity is challenged

D. (1) Restore RCS pressure to within Safety Limit values within 2 hours;
(2) Exceeding RCS pressure Safety Limit is more severe in lower modes.

Answer	A

Answer Explanation

A. Correct. Per technical specifications 2.2.2.2, In MODE 3, 4, or 5, restore compliance within 5 minutes; Lower mode pressure transients are potentially more severe and require a faster response time to restore, than in Mode 1 or 2.

B. Incorrect. (1) Plausible if the candidate misinterprets the initial conditions to meet Tech Spec 3.4.1 for RCS Pressure exceeding DNB limits. 2 hours is the correct completion time if DNB pressure is violated. (2) Plausible since system integrity is challenges at a % of design pressure and if the candidate misinterprets this value to be 125%.

C. Incorrect. (1) Correct. (2) See above.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	SRO-Safety Limit compliance 2.1.2		
User ID	Q2114215		
System ID	2114215	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	3
Technical Reference and Revision #	Tech Spec 2.1.2, Amend 297 TS Basis 2.1.2, Rev 2		
Training Objective	Given a set of plant or system conditions, evaluate the conditions and apply the appropriate actions in accordance with the Tech Specs and/or TRM.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information

Additional Information	No additional information

K/A Links

P2.2.22	Safety Function 2	Tier 3	Group	RO Imp: 4.0	SRO Imp: 4.7
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)					
GE.4.0.APE.027	Safety Function 3	Tier 1	Group 1	RO Imp:	SRO Imp:
Pressurizer Pressure Control System (PZR PCS) Malfunction					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.01, CFR: 43.1	Conditions and limitations in the facility license.

Question 79	ID: 2114239	Points: 1.00

SRO ONLY

Unit-1 is at 100% power when the following occurs:

The following alarms are received on control panel 1C34:

- Window U-13: 11, 12 125V DC BUS U/V
- Window U-15: 11, 12, 23, 24 125V BATT CHGR FAILURE
- DC Bus 11 voltage indication on panel 1C24 is 122 VDC and lowering slowly

Which one of the following describes:

(1) The failure that has occurred?

And

(2) The operability of DC Bus 11 in accordance with Technical Specifications?

A. (1) 11 and 23 Battery Chargers have failed;
(2) DC Bus 11 operability will be restored when bus voltage is restored to ≥ 125 VDC by BOTH Battery Chargers being restored to the bus within 4 hours.

B. (1) 11 and 24 Battery Chargers have failed;
(2) DC Bus 11 operability will be restored when bus voltage is restored to ≥ 125 VDC by BOTH Battery Chargers being restored to the bus within 4 hours.

C. (1) 11 and 23 Battery Chargers have failed;
(2) DC Bus 11 operability will be restored when bus voltage is restored to ≥ 125 VDC by EITHER Battery Charger being restored to the bus within 2 hours.

D. (1) 11 and 24 Battery Chargers have failed;
(2) DC Bus 11 operability will be restored when bus voltage is restored to ≥ 125 VDC by EITHER Battery Charger being restored to the bus within 2 hours.

Answer	C

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Incorrect. (1) Plausible is the candidate remembers that 11 and 24 buses are safety related and misinterprets that to also apply to the battery chargers and determines that 11 DC bus is supplied from 11 and 24 battery chargers. (2) Plausible if thought that both chargers are required for operability since in most cases all of the pieces of equipment are needed to satisfy operability.

C. Correct. (1) Listed battery chargers are those that normally supply the Bus. (2) Per T.S. 3.8.4, 11 DC Bus is operable when EITHER battery charger is restored to the DC bus and voltage is >= 125 VDC within 2 hours.

D. Incorrect. (1) See above. (2) Correct. Neither battery charger supplies 11 DC Bus therefore, its operability will NOT be restored using either battery charger 12 and 24.

Question Information

Topic	SRO-Action needed to restore DC Bus 11 operability with Battery Chargers		
User ID	Q2114239		
System ID	2114239	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	Tech Spec 3.8.4, Amend 326 1C34-ALM, 04002		
Training Objective	Given plant conditions, determine if 125 VDC buses are operable per appropriate tech specs.		

Previous NRC Exam Use	2012 SRO NRC Exam

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Links

APE.058.AA2.02	Safety Function 6	Tier 1	Group 1	RO Imp: 3.3*	SRO Imp: 3.6
Ability to determine and interpret the following as they apply to the Loss of DC Power: (CFR: 43.5 / 45.13) 125V dc bus voltage, low/critical low, alarm					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.02, CFR: 43.2	Facility operating limitations in the technical specifications and their bases.

Question 80	ID: 2114255	Points: 1.00

SRO ONLY

Unit-1 is at 100% power when the following occurs:

- 11 SRW HEAD TK LVL alarm is received and momentarily clears
- 11 SRW Head Tank Level Control Valve is open and tank level lowering slowly
- CNTMT NORMAL SUMP LVL HI alarm is received
- The crew confirms the source of the leak is the 11 SRW header
- The crew enters the appropriate procedure to address the condition

Which of the following describes:

(1) The source of the leak?
And
(2) What are the required procedure actions and Technical Specifications, if any?

A. 1) 11 or 12 CNTMT CLRs;
2) Isolate one cooler at a time and determine Head Tank level response;
Leave the leaking cooler isolated;
NO Technical Specification action is required if only 1 CNTMT CLR is isolated.

B. 1) 11 or 12 CNTMT CLRs;
2) Isolate BOTH coolers concurrently and verify Head Tank level stabilizes;
Determine the leaking cooler by placing them in service one at a time and isolate the leaking cooler;
ONE Train of CNTMT Cooling must be declared inoperable.

C. 1) 13 or 14 CNTMT CLRs;
2) Isolate one cooler at a time and determine Head Tank level response,
Leave the leaking cooler isolated;
NO Technical Specification action is required if only 1 CNTMT CLR is isolated.

D. 1) 13 or 14 CNTMT CLRs;
2) Isolate BOTH coolers concurrently and verify Head Tank level stabilizes;
Determine the leaking cooler by placing them in service one at a time and isolate the leaking cooler;
ONE Train of CNTMT Cooling must be declared inoperable.

Answer	**B**

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) Per AOP-7B, 11 SRW Head Tank Level is affected which is associated with 11 SRW Header which supplies 11 / 12 Cntmt Clrs. (2) AOP-7B directs isolation of the coolers as a pair, unisolating one at a time. Isolation of one cooler removes a T.S. Train from service.

C. Incorrect. (1) Plausible if thought that the placement of the CACs in containment will impact the draining frequency of the containment sump and with this high frequency it is from 13 and 14 CACs due to them being lower in containment. (2) Plausible because isolation procedure is the backup to the actual isolation procedure. Incorrect - A train of cntmt cooling requires 2 coolers. 1 inoperable makes the train inoperable. Minimum safety function for containment cooling is still met, but 1 train must be declared inoperable.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	SRO-Diagnosis of and response to a CAC SRW leak.		
User ID	Q2114255		
System ID	2114255	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3

Technical Reference and Revision #	TS 3.6.6 Basis, Rev 2 AOP-7B-1, Rev 1306
Training Objective	Given plant conditions in modes 1 or 2 determine the proper actions per AOP-7B.
Previous NRC Exam Use	2008 SRO NRC Retake Exam

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Links

P2.1.30	Safety Function 6	Tier 3	Group	RO Imp: 4.4	SRO Imp: 4.0
Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)					
GE.4.0.APE.062	Safety Function 4	Tier 1	Group 1	RO Imp:	SRO Imp:
Loss of Nuclear Service Water					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 81	ID: 2114284	Points: 1.00

SRO ONLY

Unit-2 is at 100% power with the following conditions:

- 21 SW Header is OOS due to maintenance

Then the following occurs:

- "INSTR AIR SYS MALFUNCTION" alarm on 2C13 actuates
- 22 SWAC fails to start and is declared inoperable
- Instrument Air Pressure is 75 PSIG and steady

What direction will the Unit Supervisor give the Reactor Operator?

A. Commence a plant shutdown per AOP-7D, Loss of Instrument Air.

B. Commence a plant shutdown per Technical Specification 3.0.3.

C. Trip the reactor per AOP-7A, Loss of Saltwater Cooling.

D. Trip the reactor per AOP-7D, Loss of Instrument Air.

Answer	B

Answer Explanation

A. Incorrect. Plausible if the candidate misinterprets the automatic actions at 75 psig IA Header Pressure that isolates IA to Containment which would also impact both trains of Containment Spray CVs requiring a plant shutdown.

B. Correct. Per TS 3.7.7, 2 SW headers are required to be in service. Per AOP-7D, both 21 and 22 SWACs should have been started and the stem says 22 SWAC failed to start. the failure of 22 SWAC takes 22 SW header OOS requiring a shutdown per TS 3.0.3. 21 SWAC can not be cross connected to 22 SW header.

C. Incorrect. Plausible if the candidate misinterprets the failure of the SWAC to mean that the SW valves will begin to fail leading to high temperatures in SW cooled components.

D. Incorrect. Plausible if the candidate misinterprets the automatic actions at 75 psig IA Header Pressure that isolates IA to Containment which would also impact Pressurizer Spray CVs requiring a reactor trip.

Question Information

Topic	SRO-Plant shutdown requirement for lowering IA pressure		
User ID	Q2114284		
System ID	2114284	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	The question includes assessment of plant conditions that resulted from the IA pressure, and the selection of corresponding actions the SRO would direct.
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	3

Technical Reference and Revision #	AOP-7D-2, Rev 01400 AOP-7A-2, Rev 01301 TS 3.7.7, Amend 304 TS 3.0.3, Amend 262
Training Objective	Identify which components receive air from the SWACs.
Previous NRC Exam Use	None

K/A Links

APE.065.AA2.05	Safety Function 8	Tier 1	Group 1	RO Imp: 3.4*	SRO Imp: 4.1
Ability to determine and interpret the following as they apply to the Loss of Instrument Air: (CFR: 43.5 / 45.13) When to commence plant shutdown if instrument air pressure is decreasing					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 82	ID: 2114470	Points: 1.00

SRO ONLY

A Unit-1 startup is in progress at 13% power.

- A review of post-maintenance documentation indicates that there is no reasonable assurance that Regulating Control Element Assembly 35 is trippable.

Which one of the following describes:

(1) The Limiting Condition for Operation (LCO) that contains the required actions to be performed?

And

(2) What is the reason for those actions?

A. (1) LCO 3.1.4, "Control Element Assembly Alignment."
(2) The potential effects of a CEA ejection accident are not limited to acceptable limits.

B. (1) LCO 3.1.4, "Control Element Assembly Alignment."
(2) Plant is outside of the safety analysis because one additional rod is assumed to stick on a reactor trip.

C. (1) LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits"
(2) The potential effects of a CEA ejection accident are not limited to acceptable limits.

D. (1) LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits"
(2) Plant is outside of the safety analysis because one additional rod is assumed to stick on a reactor trip.

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) An untrippable CEA is covered by LCO 3.1.4. (2) Bases 3.1.4 states: "If a CEA is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable CEA, meeting the insertion limits of LCOs 3.1.5 and 3.1.6 does not ensure that adequate SDM exists. Condition F must be entered."

C. Incorrect. (1) Plausible if the operator misinterprets the MISH and MIRG circuitry of the Regulating and Shutdown CEAs and concludes that the Shutdown CEA movement is disabled. (2) Plausible since this is the basis for the Shutdown CEA insertion limits and an untrippable CEA is misinterpreted to not be at the right height.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	SRO-LCO 3.1.4 and Bases		
User ID	Q2114470		
System ID	2114470	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	TS 3.1.4, Amend 227 TS 3.1.4 Basis, Rev 2		
Training Objective	Given a set of plant or system conditions, evaluate the conditions and apply the appropriate actions in accordance with the Tech Specs.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Links

P2.2.40	Safety Function 8	Tier 3	Group	RO Imp: 3.4	SRO Imp: 4.7
Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)					
GE.4.0.APE.005	Safety Function 1	Tier 1	Group 2	RO Imp:	SRO Imp:
Inoperable/Stuck Control Rod					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.02, CFR: 43.2	Facility operating limitations in the technical specifications and their bases.

Question 83	ID: 2114474	Points: 1.00

SRO ONLY

Unit-1 is operating at 100% power when the following conditions exist:

- An Excess Steam Demand Event inside Containment occurs
- Containment Pressure is 40 psig and slowly lowering
- Containment Temperature is 275°F and slowly lowering
- RCS Pressure is 1200 psia and steady
- Safety Injection flow has refilled the Pressurizer to an indicated level of 180 inches

Which one of the following correctly describes:

(1) What is the expected effect, if any, on Pressurizer level indication due to the degraded Containment conditions?
And,
(2) What direction will the Unit Supervisor provide the crew to control Pressurizer level per EOP-4, Excess Steam Demand Event?

A. (1) Indicated PZR level (LI-110X/Y) will equal Actual PZR level
(2) Secure and/or Throttle HPSI flow

B. (1) Indicated PZR level (LI-110X/Y) will be higher than Actual PZR level
(2) Secure and/or Throttle HPSI flow

C. (1) Indicated PZR level (LI-110X/Y) will be higher than Actual PZR level
(2) Restore Letdown flow

D. (1) Indicated PZR level (LI-110X/Y) will equal Actual PZR level
(2) Restore Letdown flow

Answer	B

Answer Explanation

A. Incorrect. (1) Incorrect but plausible as stated below. (2) Correct as stated below.

B. Correct. (1) The Pressurizer level instruments, LI-110X/Y use condensing pots on the reference leg which will experience a lower differential pressure and higher indicated level during the harsh Containment conditions provided. (2) EOP-4 Block Step K provides the direction to secure and/or throttle HPSI flow to control Pressurizer level.

C. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated below.

D. Incorrect. (1) Incorrect but plausible since the candidate may misinterpret the Containment temperature effect on indicated Pressurizer level. If the operation of a condensing pot on the Pressurizer reference leg is misunderstood, then the candidate will conclude that indicated and actual Pressurizer levels will be equal. (2) Incorrect but plausible since Block Step V of EOP-4 provides the direction to restore letdown flow. The candidate may misinterpret the operating circuitry of the Letdown isolation valves and may conclude that the valves can be opened during the given plant conditions.

Question Information

Topic	SRO-Pressurizer Level Control Malfunction		
User ID	Q2114474		
System ID	2114474	**Point Value**	1.00
Status	Active	**Time to Complete**	4
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	The question matches the K/A regarding reflux boiling since at an RCS pressure of 1200 psia, voids can form which would cause reflux boiling.
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	
Technical Reference and Revision #	EOP-4-1 Rev 01901 60906SH0004 Rev 13		
Training Objective	Given an ESDE, plant conditions and/or parameters, predict the response of the following: Pressurizer level.		
Previous NRC Exam Use	None		

K/A Links

APE.028.AA2.14	Safety Function 2	Tier 1	Group 2	RO Imp: 2.6	SRO Imp: 2.8
Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: (CFR: 43.5 / 45.13) The effect on indicated PZR levels, given a change in ambient pressure and temperature of reflux boiling					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 84	ID: 2114497	Points: 1.00

SRO ONLY

Unit-1 and Unit-2 are at 100% power when the following occurs:

- A fuel bundle is damaged in the spent fuel pool
- The following alarm on 1C22 actuates

CONTR RM **VENT**

Which of the following describes:

(1) The actions required by the RO?

And,

(2) The basis for this action?

A. (1) Ensure the post-LOCI discharge dampers are shut.
(2) Protect the Control Room Ventilation Filters from high radiation.

B. (1) Ensure the post-LOCI discharge dampers are shut.
(2) Places the Control Room Ventilation in complete recirculation mode.

C. (1) Ensure the kitchen Exhaust Fan is secured.
(2) Protect the Control Room Ventilation Filters from high radiation.

D. (1) Ensure the kitchen Exhaust Fan is secured.
(2) Places the Control Room Ventilation in complete recirculation mode.

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible since most control room dampers are shut to place the control room in complete recirculation mode. (2) Plausible since the Unit Supervisor is required to direct actions pertaining to filters in the High Radiation Attachment 19 of OP-CA-108-3.0.

B. Incorrect. (1) See above. (2) Correct.

C. Incorrect. (1) Correct. (2) See above.

D. Correct. (1) Per 1C22-ALM the kitchen exhaust fan outside of the control room is secured. (2) Per AOP-6C-TB, this action places the control room into complete recirculation mode.

Question Information

Topic	SRO-RO tasks during fuel handling incident		
User ID	Q2114497		
System ID	2114497	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	Embedded Reference
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only

Question Type	New	**Difficulty**	3
Technical Reference and Revision #	1C22-ALM, Rev 04400 AOP-6C-TB, Rev 01200		
Training Objective	Given any of the following events, identify from memory the required actions for ventilation systems: Fuel handling incident has occurred inside/outside the containment.		
Previous NRC Exam Use	None		

K/A Links

P2.4.34	Safety Function 2	Tier 3	Group	RO Imp: 4.2	SRO Imp: 4.1
Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)					
GE.4.0.APE.036	Safety Function 8	Tier 1	Group 2	RO Imp:	SRO Imp:
Fuel Handling Incidents					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.07, CFR: 43.7	Fuel handling facilities and procedures.

Question 85	ID: 2114610	Points: 1.00

SRO ONLY

Unit-1 is operating at 100% power when the following occurs:

- Three Charging Pumps are running.
- Letdown flow is indicating 15 gpm.
- VCT level is 90 inches and lowering at 2 inches per minute.
- Pressurizer level is 210 inches and lowering slowly.
- CNTMT NORMAL SUMP LVL HI, annunciator on 1C10 is LIT.

Which of the following describes:

(1) The event in progress?
And,
(2) The required actions?

A. (1) Charging line leak upstream of 1-CVC-183, Regenerative Heat Exchanger Charging Inlet.
(2) Isolate letdown per AOP-2A, Excessive Reactor Coolant Leakage.

B. (1) Charging line leak upstream of 1-CVC-183, Regenerative Heat Exchanger Charging Inlet.
(2) Reduce power to < 50% in one hour per OP-3, Normal Power Operation.

C. (1) Letdown line leak between the Regenerative Heat Exchanger and Containment.
(2) Isolate letdown per AOP-2A, Excessive Reactor Coolant Leakage.

D. (1) Letdown line leak between the Regenerative Heat Exchanger and Containment.
(2) Reduce power to < 50% in one hour per OP-3, Normal Power Operation.

Answer	C

Answer Explanation

A. Incorrect. (1) See below (2) Correct

B. Incorrect. (1) Plausible if the operator misinterprets the physical location of 1-CVC-183 and determines it is inside Containment. (2) Plausible since these actions are taken when the leakage is within the capacity of one charging pump, however the section that would be used is for exceeding the capacity of a charging pump in MODES 1 and 2.

C. Correct. (1) Letdown flow is below minimum of 30 gpm, the leak is inside Containment. (2) Per AOP-2A, this leakage exceeds the capacity of one charging pump and the required action is to isolate letdown.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	SRO-AOP-2A diagnosis and actions		
User ID	Q2114610		
System ID	2114610	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	Difficulty	2
Technical Reference and Revision #	AOP-2A, Rev 02701		
Training Objective	Determine the plant parameter criteria evaluated to identify the following RCS leaks: RCS leak greater than the capacity of one charging pump.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Links

4.4.A16.AA2.1	Safety Function 2	Tier 1	Group 2	RO Imp: 2.7	SRO Imp: 3.5
Ability to determine and interpret the following as they apply to the (Excess RCS Leakage) (CFR: 43.5 / 45.13) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.					

Associated Objective(s)

SRO NRC Test **User (Sys) ID** N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 86	ID: 2115813	Points: 1.00

SRO ONLY

Using provided references:

The following conditions exist on Unit-2:

- Unit-2 is in Mode 3 preparing for a plant startup
- RWT water volume is 405,000 gallons
- RWT boron concentration is 2200 ppm
- RWT water temperature is 105°F

Which one of the following describes the RWT Tech Spec LCO(s) that apply and the required actions?

A. Enter 3.5.4.A for Boron Concentration and restore within 8 hours; Enter 3.5.4.B for Water Volume and restore within 1 hour.

B. Enter 3.5.4.A for Water Temperature and restore within 8 hours; Enter 3.5.4.B for Water Volume and restore within 1 hour.

C. **ONLY** Enter 3.5.4.A for Boron Concentration and restore within 8 hours.

D. Enter 3.5.4.A for Water Temperature and restore within 8 hours; Enter 3.5.4.A for Boron Concentration and restore within 8 hours; Enter 3.5.4.B for Water Volume and restore within 1 hour.

Answer	A

Answer Explanation

A. Correct. Tech Spec LCO 3.5.4.A applies due to boron concentration less than 2300 ppm and the restoration time is 8 hours. LCO 3.5.4.B also applies due to water volume less than 412,350 gallons and the restoration time is 1 hour.

B. Incorrect. Plausible since water temperature is greater than the 100°F limit of surveillance requirement 3.5.4.2 and the mode of applicable for Tech Spec 3.5.4 is

Modes 1-4. The candidate may easily misinterpret the note for surveillance requirement 3.5.4.2 which clarifies which Mode is applicable for the required water temperature. The second part is correct for water volume as stated above.

C. Incorrect. Plausible since LCO 3.5.4.A is correct for boron concentration as stated above. The candidate may misinterpret the water volume given as being greater than the required value since 400,000 gallons had been the Tech Spec required volume until a recent plant modification had been installed during the two most recent refueling outages.

D. Incorrect. Water Temperature is plausible as stated above. The second and third parts for boron concentration and water volume are correct as stated above.

Question Information

Topic	SRO-References: RWT Parameters and Tech Specs		
User ID	Q2115813		
System ID	2115813	Point Value	1.00
Status	Active	Time to Complete	4
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	Tech Spec 3.5.4, Pages 3.5.4-1 to 3.5.4-2 and 3.5.2
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only

Question Type	New	**Difficulty**	
Technical Reference and Revision #	Tech Spec 3.5.4 Rev 227/201		
Training Objective	Given a set of plant conditions and the applicable references, correctly apply the requirements of the Tech Specs and TRM.		
Previous NRC Exam Use	None		

K/A Links

SF1.004.A2.13	Safety Function 1	Tier 2	Group 1	RO Imp: 3.6	SRO Imp: 3.9
Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5/ 43/5 / 45/3 / 45/5) Low RWST					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.02, CFR: 43.2	Facility operating limitations in the technical specifications and their bases.

Question 87	ID: 2115819	Points: 1.00

SRO ONLY

Unit-1 is operating at 100%:

- An audit of past completed STPs has determined that a surveillance was missed on 1-CC-3824-CV, 11 CC HX CC OUT
- Per the Surveillance Frequency Control Program the frequency of this missed surveillance is 31 days

When would TS 3.7.5, Component Cooling System, be required to be entered (assuming any required risk evaluations are completed satisfactorily)?

A. Immediately

B. in 36 hours

C. in 72 hours

D. in 31 days

Answer	D

Answer Explanation

A. Incorrect. Plausible if the operator misinterprets the requirements of SR 3.0.3 and determines the LCO needs to be declared immediately.

B. Incorrect. Plausible since the operator may conclude that one CC train is declared inoperable due to the missed surveillance. Then, since the required actions of TS 3.7.5.A have not been performed, the operator may conclude that LCO 3.7.5.B is required with the action to be in Mode 5 within 36 hours.

C. Incorrect. Plausible since this is the time requirement to restore from a loss of 1 CC loop.

D. Correct. Per SR 3.0.3, the requirement to declare the LCO not met may be delayed, from the time of discovery up to 24 hours, or up to the limit of the specified Frequency, whichever is greater.

Question Information

Topic	SRO-CC tech spec implementation		
User ID	Q2115819		
System ID	2115819	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	Tech Spec 3.0, Amend 322		
Training Objective	Given the applicable surveillance requirements, apply the frequency requirements in accordance with Tech Spec section 3.0 and the Surveillance Frequency Control Program.		

Previous NRC Exam Use	None

K/A Links

GS.3.0.SF8.008	Safety Function 8	Tier 2	Group 1	RO Imp:	SRO Imp:
Component Cooling Water System (CCWS)					
P2.2.42	Safety Function 8	Tier 3	Group	RO Imp: 3.9	SRO Imp: 4.6
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)					

Associated Objective(s)

SRO NRC Test **User (Sys) ID** N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.02, CFR: 43.2	Facility operating limitations in the technical specifications and their bases.

Question 88	ID: 2115823	Points: 1.00

SRO ONLY

Using provided references:

Unit-2 is operating at 100% power:

- A loss of load event occurs
- Pressurizer Pressure is above the PORV setpoint and the PORVs did not open
- The reactor did not trip

Then, the following alarm is received:

DSS
TRIP

- WRNI power indicates 1×10^{-3}% and lowering

Which of the following describes:

(1) Pressurizer Pressure at the time the DSS Trip Alarm is received?

And,

(2) The Emergency Action Classification for this event?

A. (1) 2400 PSIA
(2) Unusual Event

B. (1) 2400 PSIA
(2) Alert

C. (1) 2450 PSIA
(2) Unusual Event

D. (1) 2450 PSIA
(2) Alert

Answer	C

Answer Explanation

A. Incorrect. (1) See below. (2) Correct.

B. Incorrect. (1) Plausible if candidate misinterprets the PORV setpoint and the DSS setpoint. (2) Plausible since the operator may recall that EOP-1 final safety function acceptance criteria is WRNI power less than $1x10^{-4}$% and may then conclude that actions taken in the Control Room has not established a shutdown reactor. In the most recent past revision of the EAL, the conditions described in this question would have been an Alert.

C. Correct. (1) Per 2C05-ALM, the DSS setpoint is 2450 PSIA. (2) Per EP-AA-1011 Addendum 3, and ATWS condition where DSS trips the reactor is an unusual event.

D. Incorrect. (1) Correct. (2) See above.

Question Information

Topic	SRO-References: DSS EAL Call		
User ID	Q2115823		
System ID	2115823	Point Value	1.00
Status	Active	Time to Complete	0
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	EP-AA-1011 Addendum 3, Rev 5 Embedded Reference
K/A Justification	No additional information

SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	EP-AA-1011 Addendum 3, Rev 5 2C05-ALM, Rev 04101		
Training Objective	Given a set of plant conditions, implement the Emergency Plan in accordance with EP-AA-1011, CCNPP Emergency Action Levels.		
Previous NRC Exam Use	None		

K/A Links

GS.3.0.SF3.010	Safety Function 3	Tier 2	Group 1	RO Imp:	SRO Imp:
Pressurizer Pressure Control System (PZR PCS)					
P2.4.46	Safety Function 3	Tier 3	Group	RO Imp: 4.2	SRO Imp: 4.2
Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: **NRC-10 CFR 55.41, 43, and 45 Links**
10CFR55.43.01, CFR: 43.1 Conditions and limitations in the facility license.

Question 89	ID: 2115826	Points: 1.00

SRO ONLY

Given the following conditions on Unit-2:

- Reactor power is 100%
- A Loss of Instrument Bus 2Y01 has occurred

Which one of the following correctly describes:

(1) Which one of the following component responses is observed?
And,
(2) What actions are required of the Unit Supervisor?

A. (1) Two TCBs open and ESFAS Channel ZA Sensor Cabinet is deenergized;
(2) De-energize RPS Channel A and ESFAS Channel A Actuation Cabinet prior to power restoration, IAW AOP-7J-2.

B. (1) Two TCBs open and ESFAS Channel ZA Sensor Cabinet is deenergized;
(2) De-energize RPS Channel D and ESFAS Channel D Actuation Cabinet prior to power restoration, IAW AOP-7J-2.

C. (1) Four TCBs open and ESFAS Channel ZD Sensor Cabinet is deenergized;
(2) De-energize RPS Channel A and ESFAS Channel A Actuation Cabinet prior to power restoration, IAW AOP-7J-2.

D. (1) Four TCBs open and ESFAS Channel ZD Sensor Cabinet is deenergized;
(2) De-energize RPS Channel D and ESFAS Channel D Actuation Cabinet prior to power restoration, IAW AOP-7J-2.

Answer	C

Answer Explanation

A. Incorrect. (1) Incorrect but plausible as stated below. (2) Correct as stated below.

B. Incorrect. (1) Incorrect but plausible since the candidate will recall that a loss of 2Y01 will de-energize one of the four K relays in the RPS circuitry and will conclude that two of the eight Trip Circuit Breakers will open. ESFAS Channel ZA Sensor Cabinet being de-energized is plausible since the candidate may incorrectly identify the confusing nomenclature of the ESFAS Logic and Sensor Cabinets. 2Y01 is a Channel A power supply and ESFAS Channel ZA appears to be the logical but incorrect choice. (2) Incorrect but plausible since the candidate may incorrectly identify the confusing nomenclature of RPS and ESFAS cabinets. The candidate will recall that ESFAS ZD is impacted by the Channel A power supply and will conclude that Channel D Actuation Cabinet is the logical choice.

C. Correct. This is response observed in the control room. Four TCBs will open due to this condition. Alarm manual would be referenced as part of crew response directing them to AOP-7J which provides direction for de-energizing the affected RPS/ESFAS channel.

D. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated above.

Question Information

Topic	SRO-Loss of Vital AC Inst Bus 2Y01 effect on ESFAS		
User ID	Q2115826		
System ID	2115826	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	AOP-7J-2 Rev 01900		

Training Objective	Given plant conditions, determine how an ESFAS sensor or logic cabinet is affected on a loss of its power supply.
Previous NRC Exam Use	None

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Links

SF2.013.A2.04	Safety Function 2	Tier 2	Group 1	RO Imp: 3.6	SRO Imp: 4.2
Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; (CFR: 41.5 / 43.5 / 45.3 / 45.13) Loss of instrument bus					

Associated Objective(s)

SRO NRC Test

User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 90	ID: 2115841	Points: 1.00

SRO ONLY

Unit 1 is operating at 100%:

- 1-PDT-111A, RCS flow, fails low
- The Unit Supervisor directs the CRO to bypass Trip Unit 3, RCS Flow
- Due to a miscommunication issue the CRO bypasses Trip Unit 3 on Channel B
- No further operator actions are taken

Which of the following describes:

(1) What is the remaining RPS Trip Logic that will cause a reactor trip?

And,

(2) How long can RPS stay in this lineup with no additional actions (assuming RICT cannot be applied)?

A. (1) 1 out of 2
(2) 48 hours

B. (1) 1 out of 2
(2) indefinitely

C. (1) 1 out of 3
(2) 48 hours

D. (1) 1 out of 3
(2) indefinitely

Answer	A

Answer Explanation

A. Correct. (1) Per the RPS Drawing, 1 trip unit in trip reduces the logic to 1 out of 3 and then once another channel is taken to bypass the logic is reduced to 1 out of 2. (2) Per

TS 3.3.1, RPS can stay in this lineup for 48 hours.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible if candidate misinterprets how bypasses the incorrect channel will impact the logic and determines that the logic is still 1 out of 3. (2) Plausible since a trip unit can remain in trip indefinitely and the candidate might think this time frame also applies to a trip unit in bypass.

Question Information

Topic	SRO-RPS incorrect channel bypassing		
User ID	Q2115841		
System ID	2115841	**Point Value**	1.00
Status	Active	**Time to Complete**	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3

Technical Reference and Revision #	TS 3.3.1, Amend 326 12129-0032, Rev 5
Training Objective	Given a set of plant conditions evaluate the operation of RPS in accordance with the Tech Specs.
Previous NRC Exam Use	None

K/A Links

SF7.012.A2.03	Safety Function 7	Tier 2	Group 1	RO Imp: 3.4	SRO Imp: 3.7
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: (CFR: 41.5 / 43.5 / 45.3 / 45.5) Incorrect channel bypassing					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.02, CFR: 43.2	Facility operating limitations in the technical specifications and their bases.

Question 91	ID: 2121787	Points: 1.00

SRO ONLY

Using Provided References:

Given the following conditions on Unit-1:

- Unit-1 is in Mode 4
- A Loss of Power caused both ESFAS Channel B Logic Cabinet and AFAS Channel B Logic Cabinet to de-energize

Which one of the following describes:

(1) What Tech Spec LCO(s) will the Unit Supervisor enter?
And,
(2) What Tech Spec required component(s) was/were made inoperable due to the Loss of Power?

A. (1) **BOTH** 3.3.5.A and 3.3.5.C
(2) One required channel of manual actuation is inoperable

B. (1) **ONLY** 3.3.5.C
(2) One required channel of manual actuation is inoperable

C. (1) **ONLY** 3.3.5.C
(2) Two required channels of manual actuation are inoperable

D. (1) **BOTH** 3.3.5.A and 3.3.5.C
(2) Two required channels of manual actuation are inoperable

Answer	B

Answer Explanation

A. Incorrect. (1) Incorrect but plausible as stated below. (2) Correct as stated below.

B. Correct. (1) Correct since only ESFAS is required in Mode 4 as shown in Table 3.3.5-

1 and LCO 3.3.5.C is entered for ESFAS inoperability. (2) Tech Spec Basis 3.3.5 describes the two independent channels of actuation as Channel A and Channel B Logic Cabinets and their circuitry. The inoperability of Channel B Logic Cabinet impacts only one of the manual actuation pushbuttons in the Control Room.

C. Incorrect. (1) Correct as stated above. (2) Incorrect but plausible as stated below.

D. Incorrect. (1) Incorrect but plausible since the operator may misinterpret the Mode applicability in Table 3.3.5-1 and conclude that AFAS Logic Cabinet is also required in Mode 4. Then, the operator will determine that both Tech Spec 3.3.5.A and 3.3.5.C will be entered. (2) Incorrect but plausible since the operator may misinterpret the inoperability of the Control Room Channel B actuation pushbutton and the ESFAS Channel B Logic Cabinet manual actuation pushbutton to be two required channels made inoperable. Also plausible if the operator incorrectly concludes that the two required channels made inoperable are ESFAS manual actuation and AFAS manual actuation.

Question Information

Topic	SRO-References: ESFAS Loss of Power Tech Spec Actions		
User ID	Q2121787		
System ID	2121787	**Point Value**	1.00
Status	Active	**Time to Complete**	4
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	Tech Spec 3.3.5, Pages 3.3.5-1 to 3.3.5-5 and 3.3.4
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	
Technical Reference and Revision #	Tech Spec Basis 3.3.5 Rev 66 Tech Spec 3.3.5 Rev 326/304		
Training Objective	Assess the impact on ESFAS system or equipment operability when either actuation logic cabinet is de-energized.		
Previous NRC Exam Use	None		

K/A Links

SF7.016.A2.02	Safety Function 7	Tier 2	Group 2	RO Imp: 2.9*	SRO Imp: 3.2*
Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5) Loss of power supply					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links	
10CFR55.43.01, CFR: 43.1	Conditions and limitations in the facility license.

Question 92	ID: 2097004	Points: 1.00

SRO ONLY

A reactor startup is in progress on Unit-1 with the following conditions:

- Reactor power is 3% and steady
- TBV controller, 1-PIC-4056, is in automatic maintaining T_{COLD} at 532° F
- Condenser vacuum begins to lower
- The reactor startup is stopped
- Vacuum stabilizes at 22 inches Hg

Which one of the following actions are required?

A. Raise power to 10% IAW OP-2.

B. Maintain power at 3% IAW OP-2.

C. Manually trip the reactor IAW AOP-7G.

D. Reduce power to less than or equal to 1% IAW AOP-7G.

Answer	D

Answer Explanation

A.Incorrect - Plausible since Unit-2 has a vacuum region that you raise power quickly through to avoid time in the avoid operation region at low vacuum startup conditions.

B.Incorrect - Plausible if the operator misinterprets that since AFW is able to maintain S/G levels up to approximately 5% power, the current power level is acceptable. It is not desirable to maintain power and perform these actions. Power is reduced below 1% per AOP. OP-2 is the reactor startup procedure but does not give these actions.

C.Incorrect - No trip criteria is being met, but remaining actions are directed per the AOP-7G. Plausible since 22" is trip criteria at higher power levels.

D.Correct - Per AOP-7G, since power is below 5%, it directs lowering power to below 1%.

Question Information

Topic	SRO-Actions during rx startup when Unit-1 vacuum at 22" Hg and reactor power at 3%		
User ID	Q2097004		
System ID	2097004	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-7G-1, 00400		
Training Objective	Identify when a loss of condenser vacuum has occurred and the corrective actions to restore condenser vacuum.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Links

GS.3.0.SF4.SEC.045	Safety Function 4	Tier 2	Group 2	RO Imp:	SRO Imp:
Main Turbine Generator (MT/G) System					
P2.1.20	Safety Function 4	Tier 3	Group	RO Imp: 4.6	SRO Imp: 4.6
Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)					

Associated Objective(s)

Identify when a loss of condenser vacuum has occurred and the actions required to stabilize the plant. **User (Sys) ID** N/A (1203843)
SRO NRC Test **User (Sys) ID** N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 93	ID: 2115996	Points: 1.00

SRO ONLY

Unit-1 is operating at 100% power:

- 11 and 15 Circ Water Pumps are secured due to an intake issue
- AOP-7L, Circulating Water/Intake Malfunction is implemented

Which of the following describes:

(1) Per AOP-7L and OI-14A, Circulating Water System, The Unit Supervisor will direct a downpower to an initial target of ______ MWe.

And,

(2) What is the basis for this value?

A. (1) 300 MWe
(2) supports multiple Circ. Water pumps removed from service without affecting condenser vacuum.

B. (1) 300 MWe
(2) prevents operating in the "Avoid operation in this region" of the MWE/Backpressure curve.

C. (1) 465 MWe
(2) prevents operating in the "Avoid operation in this region" of the MWE/Backpressure curve.

D. (1) 465 MWe
(2) supports multiple Circ. Water pumps removed from service without affecting condenser vacuum.

Answer	A

Answer Explanation

A. Correct. (1) Per OI-14A, the initial target load is 300 MWE. (2) Per AOP-7L-TB, the basis for this value is to support multiple Circ. Water pumps removed from service without affecting condenser vacuum.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) Plausible if the candidate misinterprets the value of the start of the do not operate region from Unit 2. (2) Plausible since this is the basis for the 465 MWe number just for the reason of lowering vacuum not for a loss of CWPs.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	SRO-Given an action in AOP-7L, Identify the basis for the action.		
User ID	Q2115996		
System ID	2115996	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

NRC Exams Only			
Question Type	Bank	**Difficulty**	3
Technical Reference and Revision #	AOP-7L-1, Rev 01201 AOP-7L-TB, Rev 01100 OI-14A-1, Rev 02601		
Training Objective	From memory, demonstrate an understanding of the strategy and basis of AOP-7L without error.		
Previous NRC Exam Use	None		

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

K/A Links

SF8.075.A2.02	Safety Function 8	Tier 2	Group 2	RO Imp: 2.5	SRO Imp: 2.7
Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13) Loss of circulating water pumps					

Associated Objective(s)

SRO NRC Test **User (Sys) ID** N/A (1527462)

Cross Reference Links

Table: **NRC-10 CFR 55.41, 43, and 45 Links**
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 94	ID: 2116011	Points: 1.00

SRO ONLY

Plant operators running the 2B Emergency Diesel Generator have determined that a PPE exemption is required since wearing the prescribed PPE would present an additional heat stress hazard.

Which of the following describes:

(1) Who is required to approve the exemption?

And,

(2) Where is the exemption required to be maintained?

A. (1) Site Duty Manager
(2) At the work execution center

B. (1) Site Duty Manager
(2) At the work location

C. (1) Site Safety Professional
(2) At the work execution center

D. (1) Site Safety Professional
(2) At the work location

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible since many approvals during a normal work week require the site duty manager approval. (2) Plausible if thought that the WEC needs to be aware of all PPE exemptions so they could inform the fire and safety watch if required.

B. Incorrect. (1) See above. (2) Correct.

C. Incorrect. (1) Correct. (2) See above.

D. Correct. (1) Per SA-AA-116, the Site Safety Professional is required to approval any PPE exemptions. (2) Per SA-AA-116, the exemption should be maintained at the work location or with the work package.

Question Information

Topic	SRO-PPE exemptions		
User ID	Q2116011		
System ID	2116011	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	SA-AA-116, Rev 021		
Training Objective	Apply the requirements of SA-AA-116, Personal Protective Equipment.		

Previous NRC Exam Use	None

K/A Links

P2.1.26	Safety Function 8	Tier 3	Group	RO Imp: 3.4	SRO Imp: 3.6
Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen). (CFR: 41.10 / 45.12)					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 95	ID: 2116027	Points: 1.00

SRO ONLY

Which of the following describes:

(1) Per OP-AA-108-101, Control of Equipment and System Status, when are audits of the Temporary Configuration Changes (TCC) Log performed?

And,

(2) A 50.59 Review is required for a TCC to remain installed in the plant at power past _______ days?

A. (1) Semi-Annually
(2) 30 Days

B. (1) Semi-Annually
(2) 90 Days

C. (1) Quarterly
(2) 30 Days

D. (1) Quarterly
(2) 90 Days

Answer	D

Answer Explanation

A. Incorrect. (1) Plausible since many procedurally required audits are semi-annually, such as an audit of the Reactivity Management program. (2) Plausible since the operator will recall that the requirement to perform an audit of active Component Manipulations and an audit of active Locked Valve Deviations per PE 0-102-58-O-M is monthly (30 days). The operator may then conclude that the TCC audit is performed on the same frequency.

B. Incorrect. (1) See above. (2) Correct.

C. Incorrect. (1) Correct. (2) See above.

D. Correct. (1) Per OP-AA-108-101, a quarterly audit is performed. (2) Per CC-AA-112, a 50.59 review is required to go beyond 90 days at power.

Question Information

Topic	SRO-Temporary Configuration Changes (TCC) Audits		
User ID	Q2116027		
System ID	2116027	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	CC-AA-112, Rev 27 OP-AA-108-101, Rev 14		
Training Objective	Apply the requirements of OP-AA-108-101, Control of Equipment and System Status.		

Previous NRC Exam Use	None

K/A Links

P2.2.11	Safety Function 8	Tier 3	Group	RO Imp: 2.3	SRO Imp: 3.3
Knowledge of the process for controlling temporary design changes. (CFR: 41.10 / 43.3 / 45.13)					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 96	ID: 2116030	Points: 1.00

SRO ONLY

Which of the following completes the following statements:

Adherence to Technical Specification Cooldown Rates provides margin against ___(1)___.

And,

Unit ___(2)___ has the more restrictive cooldown rate from Mode 3 to Mode 5.

A. (1) Brittle Fracture
(2) 1

B. (1) Brittle Fracture
(2) 2

C. (1) Pressurized Thermal Shock
(2) 1

D. (1) Pressurized Thermal Shock
(2) 2

Answer	A

Answer Explanation

A. Correct. (1) Per TS 3.4.3 TB, operating limits for cooldown provide margin to brittle failure. (2) Per TS 3.4.3, Unit 1 has the more restrictive CD rates. (256-106° C/D rate is 40°F vice 100°F on Unit 2).

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since this is reason why the pressurizer has certain limits in place. (2) Plausible if the candidate misinterprets when the C/D rates change based on temperature and mode.

Question Information

Topic	SRO-Cooldown basis and differences between units		
User ID	Q2116030		
System ID	2116030	**Point Value**	1.00
Status	Active	**Time to Complete**	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-HIGH		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	TS 3.4.3, Amend 243 and 217 TS 3.4.3 TB, Revision 2		
Training Objective	Given a set of plant or system conditions, evaluate the conditions and apply the appropriate actions in accordance with the Tech Specs.		
Previous NRC Exam Use	None		

K/A Links

P2.2.3	Safety Function 8	Tier 3	Group	RO Imp: 3.8	SRO Imp: 3.9
(multi-unit license) Knowledge of the design, procedural, and operational differences between units. (CFR: 41.5 / 41.6 / 41.7 / 41.10 / 45.12)					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: **NRC-10 CFR 55.41, 43, and 45 Links**
10CFR55.43.02, CFR: 43.2 Facility operating limitations in the technical specifications and their bases.

Question 97	ID: 2116045	Points: 1.00

SRO ONLY

A LOCA has occurred on Unit-1:

- A Site Area Emergency has been declared
- An Equipment Operator has been selected to enter the Auxiliary Building to protect valuable vital equipment
- All required approvals have been obtained

What is the maximum total exposure the Equipment Operator may receive while performing these actions?

A. 5 REM

B. 10 REM

C. 25 REM

D. greater than 25 REM

Answer	B

Answer Explanation

A. Incorrect. Plausible since EP-CE-113 describes 5 REM as the emergency exposure limit for all activities.

B. Correct. Per EP-CE-113, the emergency exposure limit for protecting valuable property is 10 REM.

C. Incorrect. Plausible since EP-CE-113 describes 25 REM is the emergency exposure limit for lifesaving actions.

D. Incorrect. Plausible since EP-CE-113 describes a limit of >25 REM when lifesaving actions are voluntary.

Question Information

Topic	SRO-Radiation Exposure limits for protecting plant equipment		
User ID	Q2116045		
System ID	2116045	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	Difficulty	3
Technical Reference and Revision #	EP-CE-113, Rev 001		
Training Objective	From memory, recall dose limits for entering radiation areas in an emergency allowed by the Emergency Plan.		
Previous NRC Exam Use	None		

K/A Links

P2.3.4	Safety Function 8	Tier 3	Group	RO Imp: 3.2	SRO Imp: 3.7
Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.04, CFR: 43.4 Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Question 98	ID: 2116130	Points: 1.00

SRO ONLY

Which of the following describes:

(1) Per OI-41B, Hydrogen Purge System Operation, what type of pressure vent of containment requires a Gaseous Waste Release Permit?

And,

(2) On the permit what does the Unit Supervisor initial for?

A. (1) Positive
(2) Ensures termination criteria is entered into the Plant Computer Alarm Setpoints as indicated on the Release Permit.

B. (1) Positive
(2) Ensures radiation monitor alarm setpoints on 1C22 have been adjusted to the appropriate values.

C. (1) Negative
(2) Ensures termination criteria is entered into the Plant Computer Alarm Setpoints as indicated on the Release Permit.

D. (1) Negative
(2) Ensures radiation monitor alarm setpoints on 1C22 have been adjusted to the appropriate values.

Answer	A

Answer Explanation

A. Correct. (1) Per OI-41B, a waste permit is required for a positive pressure vent. (2) Per CY-CA-170-604, the US/SM initials for ensuring termination criteria is entered into the plant computer.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible if the candidate misinterprets the flow path of a negative pressure vent and determines a permit is required. (2) Plausible since adjusting RMS setpoints is plausible since the RMS Warning/Alert/Critical alarm values from the permit are inputted to the Plant Computer for termination criteria. However, RMS alarm setpoints are not adjusted.

Question Information

Topic	SRO-Containment pressure venting		
User ID	Q2116130		
System ID	2116130	Point Value	1.00
Status	Active	Time to Complete	3
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	3
Technical Reference and Revision #	OI-41B, Rev 01400 CY-CA-170-604, Rev 000		

Training Objective	Discuss how CCNPP complies with regulatory requirements for the control of wastes.
Previous NRC Exam Use	None

K/A Links

P2.3.6	Safety Function 8	Tier 3	Group	RO Imp: 2.0	SRO Imp: 3.8
Ability to approve release permits. (CFR: 41.13 / 43.4 / 45.10)					

Associated Objective(s)

SRO NRC Test **User (Sys) ID** N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.04, CFR: 43.4 Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Question 99	ID: 2116155	Points: 1.00

SRO ONLY

Which of the following completes the statement below in regard to EOP-2, Loss of Offsite Power:

Safety Function Status Checks are required to be performed every ___(1)___ minutes.

And,

The Unit Supervisor should be notified when a safety function intermediate acceptance criteria is not being met ___(2)___.

A. (1) 15 minutes
(2) upon full completion of all SFSCs.

B. (1) 15 minutes
(2) upon completion of that individual Safety Function.

C. (1) 30 minutes
(2) upon full completion of all SFSCs.

D. (1) 30 minutes
(2) upon completion of that individual Safety Function.

Answer	B

Answer Explanation

A. Incorrect. (1) Correct. (2) See below.

B. Correct. (1) Per EOP-2-TB, the operator is required to verify the SFSC acceptance criteria every 15 minutes. (2) Per OP-CA-103-102-1001, promptly inform the US if any safety function is not being met after all parameters within that safety function are evaluated.

C. Incorrect. (1) Plausible since a parameter can be not met for up to 30 minutes prior to needing a procedure deviation and the candidate misinterprets this to mean the checks are performed every 30 minutes. (2) Plausible since this is the reporting requirements for EOP-0.

D. Incorrect. (1) See above. (2) Correct.

Question Information

Topic	SRO-EOP SFSC times and actions		
User ID	Q2116155		
System ID	2116155	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	EOP-02-TB, Rev 01800 OP-CA-103-102-1001, Rev 005		
Training Objective	Apply the requirements of OP-CA-103-102-1001, Strategies for Successful Transient Mitigation.		

Previous NRC Exam Use	None

K/A Links

P2.4.14	Safety Function 8	Tier 3	Group	RO Imp: 3.8	SRO Imp: 4.5
Knowledge of general guidelines for EOP usage. (CFR: 41.10 / 45.13)					

Associated Objective(s)

SRO NRC Test **User (Sys) ID** N/A (1527462)

Cross Reference Links

Table: NRC-10 CFR 55.41, 43, and 45 Links
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question 100	ID: 2116156	Points: 1.00

SRO ONLY

According to OP-CA-103-102-1001, Strategies for Successful Transient Mitigation:

The Unit Supervisor will ensure that any AOP actions that are directed to be completed immediately after tripping the reactor should be evaluated upon completion of the ___(1)___ safety function of EOP-0.
AND
Then, the Unit Supervisor will direct the AOP follow up actions be completed at a ___(2)___ priority than the safety functions described by EOP-0.

A. (1) Reactivity Control
(2) lower

B. (1) Reactivity Control
(2) higher

C. (1) Pressure and Inventory Control
(2) lower

D. (1) Pressure and Inventory Control
(2) higher

Answer	A

Answer Explanation

A. Correct. (1) Per OP-CA-103-102-1001, AOP actions should be performed upon completion of reactivity control. (2) Per OP-CA-103-102-1001, the AOP shall be set aside and EOP-0 worked.

B. Incorrect. (1) Correct. (2) See below.

C. Incorrect. (1) See below. (2) Correct.

D. Incorrect. (1) Plausible since some actions during reactivity control might entail starting charging pumps and the candidate misinterprets this to mean that PIC needs to be completed. (2) Plausible since the operator will recall that many AOPs direct a reactor trip and then continue with mitigating actions to address the transient condition. Also plausible since AOPs are used concurrently in EOP-1 through EOP-8.

Question Information

Topic	SRO-Use of AOPs in EOP-0		
User ID	Q2116156		
System ID	2116156	Point Value	1.00
Status	Active	Time to Complete	2
Open or Closed Reference	CLOSED		
Site	CC		
Operator Type-Cognitive Level	SRO-MEMORY		
Operator Discipline	LO-I		

Comments

References Provided	None
K/A Justification	No additional information
SRO-Only Justification	No additional information
Additional Information	No additional information

NRC Exams Only			
Question Type	New	**Difficulty**	2
Technical Reference and Revision #	OP-CA-103-102-1001, Rev 005		
Training Objective	Apply the requirements of OP-CA-103-102-1001, Strategies for Successful Transient Mitigation.		

Previous NRC Exam Use	None

K/A Links

P2.4.8	Safety Function 8	Tier 3	Group	RO Imp: 3.8	SRO Imp: 4.5
Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)					

Associated Objective(s)

SRO NRC Test
User (Sys) ID N/A (1527462)

Cross Reference Links

Table: **NRC-10 CFR 55.41, 43, and 45 Links**
10CFR55.43.05, CFR: 43.5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.