Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 1	Group #	1	
	K/A #	G 2.1.20 Ability to execute procedure steps.	
	Importance Rating	4.3	4.2

Proposed Question:
During implementation of E-0, REACTOR TRIP OR SAFETY INJECTION, the main turbine fails to trip automatically or manually.

Which one of the following is the next required action?

A. Dispatch an NSO to locally trip the turbine.

B. Manually run back the turbine, when generator output is ZERO Mwe, then open the generator breaker.

C. Manually open the generator breaker.

D. Close the MSIV's. When generator output is ZERO Mwe, then open the generator breaker.

Proposed Answer: D

A is incorrect. Dispatching an NSO to locally trip the turbine is an action utilized in FR-S.1, step 6, when checking if a turbine trip had occurred.

B is incorrect. Manually running back the turbine is plausible, however, it is an action utilized in FR-S.1, Step 2a RNO.

C is incorrect. Opening the generator breaker is plausible as it is part of the procedure step, however, the generator breaker is not opened until the turbine is either tripped or the MSIV's are closed and electrical load is checked to be at 0 Mwe.

D is correct. E-0, REACTOR TRIP OR SAFETY INJECTION, step 2a, RNO directs the operator to manually trip the turbine. If the turbine will not manually trip the step then directs the operator to "close the MSIV's. When generator output is ZERO Mwe, then open the generator breaker"

Technical Reference(s): E-0, REACTOR TRIP OR SAFETY INJECTION, step 2a, RNO

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.1.20 Ability to execute procedure steps.

Question Source:	Modified from bank	Original question attached to reference.
Question Cognitive Level:	Memory	
10 CFR Part 55 Content:	41.10 / 43.5 / 45.12	
Learning Objective:	Lesson Plan L1202I, Objective L1202I04	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 2	Group #	1	
	K/A #	008 AK3.03	
	Importance Rating	4.1	4.6

Proposed Question:

An indavertant Safety Injection occurred and the operating crew is currently in ES-1.1, 'SI TERMINATION' at step 7, 'Check if SI Pumps Should Be Stopped'. Subsequently, one PORV opens and fails to reclose. Attempts to close the PORV's associated block valve fail. Containment pressure is at 5 psig and slowly increasing.

Which of the following conditions will require the operator to manually start the non-operating charging pump?

A. Pressurizer level drops to less than 30%.

B. RCS subcooling drops to less than 40° F.

C. RCS pressure drops to less than 1700 psig.

D. Total EFW flow to intact Steam Generators is less than 500 gpm.

Proposed Answer: B

B is correct. Per ES-1.1, OAS page, Following SI termination, manually start ECCS pumps as required and go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT Step 1 if either condition below occurs:

RCS subcooling-Less than 40° F.

-or-

Pressurizer level-Cannot be maintained greater than 7% (28% for adverse containment)

A is incorrect but plausible. Pressurizer level is one of the ECCS pump restart criteria, however the level requirement for adverse containment is <28%. This question tests the students ability to analyze the data given and realize that containment is adverse and that the existing pressurizer level data does not warrant a pump restart.

C is incorrect but plausible. RCS pressure "stable or increasing" is one of the SI termination criteria, however, decreasing RCS pressure is no one of the ECCS reinitiation criteria.

D is incorrect but plausible. Adequate heat sink is one of the SI termination criteria in E-1.

Technical Reference(s): ES-1.1, SI TERMINATION

Proposed references to be provided to applicants during examination: None

K/A Topic: 008 AK3.03, Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident:
Actions contained in EOP for Pressurizer Vapor Space Accident/LOCA

Question Source: Direct from bank

Question Cognitive Level: Analysis

10 CFR Part 55 Content: 41.5,41.10,45.6,45.13

Learning Objective: Lesson L1413I, Objective L1413I07
Lesson L1226I, Objective L1226I03

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 3	Group #	1	
	K/A #	009EA2.15	
	Importance Rating	3.3	3.4

Proposed Question:
A Small Break LOCA has occurred. The crew is performing the actions of ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

ECCS pumps have been stopped. Normal Charging is aligned. The crew is depressurizing the RCS to minimize subcooling. Several minutes after the depressurization is stopped, the following conditions exist:

- RCS Subcooling is 37°F and DECREASING
- Pressurizer Level is 18% and DECREASING

Based on these indications, what actions should be taken?

A. ISOLATE Letdown. Check to ensure Pressurizer Level stabilizes above 7%.

B. Manually START ECCS pumps as necessary to regain subcooling.

C. REINITIATE Safety Injection and verify all safeguards equipment has actuated.

D. Energize pressurizer heaters to saturate the pressurizer.

Proposed Answer: B

B is correct. The Operator Action Summary page for ES-1.2 states the required action as follows:
Following SI termination or ECCS flow reduction manually start ECCS pumps as required if EITHER condition listed below occurs:

- RCS Subcooling-Less than 40°F

  -OR-

- Pressurizer level-CANNOT BE MAINTAINED GREATER THAN 7% (28% FOR ADVERSE CONTAINMENT)

A is incorrect. Letdown is already out of service in this event. Additionally, Pressurizer level is above the ECCS level criteria.
C is incorrect. Reinitiation of SI may result in a higher pressure than necessary for the plant conditions, and RHR will be running again at shutoff head. The procedure does not direct reinitiation of SI.
D is incorrect. Pressurizer heaters are energized per the TSC, to saturate the pressurizer, however for the conditions listed (i.e.LOCA) turning on the heaters would not improve subcooling.

Technical Reference(s): ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

Proposed references to be provided to applicants during examination: None

K/A Topic: 009EA2.15 Ability to determine or interpret the following as they apply to a small break LOCA: RCS parameters

Question Source: Direct from bank.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: 43.5/ 45.13

Learning Objective: Lesson Plan L1204I, Objective L1204I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 4	Group #	1	
	K/A #	011EK2.02	
	Importance Rating	2.6	2.7

Proposed Question:
A Large Break LOCA has occurred. The crew has entered E-0, REACTOR TRIP OR SAFETY INJECTION. Containment pressure is 20 psig and slowly decreasing. Which of the following explains the basis for stopping the RCP's as directed by the Operator Action Summary Page.

A. Prevents excessive depletion of RCS inventory through the RCS break.

B. Protects the RCP's. from damage due to a loss of component cooling water.

C. If the RCP's stay running phase separation may occur with resulting partial core uncovery.

D. Preempt the RCP's from tripping due to cavitation.

Proposed Answer: B

A is incorrect. Excessive depletion of RCS inventory from RCP operation is the bases for securing RCP's in the event of a small break LOCA. This bases is explained on WOG Emergency Response Guidelines, Executive Volume, Generic Issue, RCP Trip/Restart.

B is correct. RCP's should be secured within 10 minutes of loss of cooling flow to preclude pump damage. This info is contained in UFSAR, Section 9.2.2.3d, Auxiliary Systems, Water Systems, item d., Failure Analysis. Additional supporting info is included in Seabrook Station Engineering Evaluation EE-04-024, Page 12, item 14, Response to Loss of PCCW Cooling to RCP's.

C is incorrect. Phase separation and partial core uncovery is part of the small break LOCA bases for securing RCP's.

D is incorrect. There are no Operator Action Summary page items that address preemptive actions for RCP's tripping.

Technical Reference(s):	E-0, REACTOR TRIP OR SAFETY INJECTION, and associated WOG basis document.	WOG basis document, Executive Volume, RCP Trip Criteria.

Proposed references to be provided to applicants during examination: None

K/A Topic: 011EK2.02 Knowledge of the interrelationship between the following and a large break LOCA: Pumps

Question Source:	New
Question Cognitive Level:	Analysis
10 CFR Part 55 Content:	43.5/ 45.13
Learning Objective:	Lesson Plan L1202I, Objective L1202I03RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 5	Group #	1	
	K/A #	015AA1.07	
	Importance Rating	3.5	3.4

Proposed Question:
Which one of the following would cause RCP #1 seal leakoff flow to indicate LOW and #2 seal leakoff flow to indicate HIGH?

A. Failure of the #1 seal.

B. Failure of the #1 seal followed by failure of the #2 seal

C. Failure of the #2 seal.

D. Failure of the #3 seal.

Proposed Answer: C

A is incorrect. #1 seal failure would result in high #1 seal leakoff.

B is incorrect. #1 seal failure would result in high #1 seal leakoff.

C is correct. If the #2 seal fails, the #1 seal leakoff will be directed through the #2 seal. In this case the #1 seal leakoff would decrease and the #2 seal leakoff would increase.

D is incorrect. Failure of the #3 seal would not affect operation of either the #1 or the #2 seals.

Technical Reference(s): Westinghouse RCP Manual

Proposed references to be provided to applicants during examination: None

K/A Topic: 015AA1.07 Ability to operate and/or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of Flow):
RCP Seal Water Injection Subsystem

Question Source: Direct from bank. Last used Seabrook 1998 Company Exam

Question Cognitive Level: Analysis

10 CFR Part 55 Content: 41.7/45.5/45.6

Learning Objective: Lesson Plan L8021I, Objective L8021I27
Lesson Plan L1181I, Objective L1181I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 6	Group #	1	
	K/A #	022AK3.02	
	Importance Rating	3.5	3.8

Proposed Question:
Why does the precaution in OS1008.01, CHEMICAL AND VOLUME CONTROL SYSTEM MAKEUP OPERATIONS warn the operator to closely monitor VCT level if CS-LT-112 fails high?

A. Automatic makeup is defeated.

B. Divert on high VCT level is defeated.

C. Makeup will not terminate automatically.

D. Swapover to RWST will not occur on SI actuation.

Proposed Answer: A

A is correct. VCT level will decrease due to full divert. Automatic makeup will fail to initiate.

B is incorrect. Full divert will take place.

C is incorrect. Makeup would fail to initiate.

D is incorrect. Swapover to the RWST would still occur based on SI signal.

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

K/A Topic: 022AK3.02 Knowledge of the interrelations between the Loss of Reactor Coolant Makeup and the following:
Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging.

Question Source:	Direct from bank. Last used Seabrook 1998 Company Exam
Question Cognitive Level:	Comprehension
10 CFR Part 55 Content:	41.5/41.10/45.6/45.13
Learning Objective:	Lesson Plan L8021I, Objective L8021I27RO Lesson Plan L1181I, Objective L1181I02RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 7	Group #	1	
	K/A #	025, G 2.4.11, Knowledge of abnormal condition procedures.	
	Importance Rating	3.4	3.6

Proposed Question:
The following plant conditions exist:

- Reactor vessel level is at –74 inches in preparation to install Steam Generator nozzle dams.
- The Train 'A' RHR pump is in operation with RHR loop flow at 3500 gpm.
- The 'B' RHR pump is in standby.
- The procedure in effect is OS1000.12, 'Operation With RCS At Reduced Inventory/Midloop Conditions'.
- A spurious loss of power to Bus E5 results in a loss of the 'A' RHR pump.

In accordance with OS1213.02, 'Loss Of RHR While Operating At Reduced Inventory OR Midloop Conditions', which action should be taken with respect to the 'B' RHR pump?

A. The 'B' RHR pump should not be started because air entrainment could result in a loss of both RHR pumps.

B. The 'B' RHR pump should be started as vessel level is above the start criteria of –83.5 inches.

C. The 'B' RHR pump should not be started until an RWST gravity fill has been implemented to restore vessel level above –36 inches.

D. The 'B' RHR pump should be started when vessel level is raised above –71 inches by increasing charging flow and reducing letdown flow.

Proposed Answer: B

A is incorrect but plausible. It is generally prudent not to start a standby pump at midloop conditions, however the vessel level of the question stem is above midloop conditions.

C is incorrect but plausible. Increasing vessel level to greater than –36 inches brings level above reduced inventory conditions but this is not required for starting the standby RHR pump. RWST gravity fill is only utilized if no emergency bus has power.

D is incorrect but plausible. Charging and letdown flow adjustments are used to adjust vessel level when level is above –83.5 inches in accordance with OS1213.02. Vessel level does not have to be raised above the top of the hot leg nozzle before the RHR pump can be started.

Technical Reference(s):	OS1213.02, 'Loss Of RHR While Operating At Reduced Inventory OR Midloop Conditions'	OS1213.02, LOSS OF RHR WHILE OPERATING AT REDUCED INVENTORY OR MID-LOOP.

Proposed references to be provided to applicants during examination: None

K/A Topic:	025, G 2.4.11, Knowledge of abnormal condition procedures.
Question Source:	New
Question Cognitive Level:	Comprehension/Application
10 CFR Part 55 Content:	41.10 / 43.5 / 45.13
Learning Objective:	Lesson Plan L1705I, Objective L1705I05RO and L1705I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 8	Group #	1	
	K/A #	027AK1.02	
	Importance Rating	2.8	3.1

Proposed Question:
The plant is operating at 100% power. The following conditions exist:
- RCS temperature is 589 degrees F
- Pressurizer pressure is 2235 psig.
- Pressurizer pressure channel I (PT-455) is selected as the controlling channel.

Which one of the following correctly describes Pressurizer pressure response if PT-455 fails LOW and no operator action is taken?

A. Actual RCS pressure will increase until the pressurizer spray valves open.

B. Actual RCS pressure will increase until one pressurizer code safety valve opens.

C. Actual RCS pressure will increase until the 'A' PORV opens.

D. Actual RCS pressure will increase until the 'B' PORV opens.

Proposed Answer: D

A. Incorrect but plausible. Both pressurizer spray valves receive input signals from the Pressurizer Master Pressure Controller. PT-455 is the selected input to the master pressure controller. Failure low of PT-455 will cause the master pressure controller output signal to decrease, which would inhibit spray operation.
B. Incorrect but plausible. The B PORV will still function based on valid input from the backup pressure control channel (PT-456) and a valid arming signal from PT-457. Operation of the B PORV would prevent pressure from reaching the code safety valve setpoint.
C. Incorrect but plausible. Failure of PT-455 will inhibit operation of the A PORV as this PORV gets its signal from the invalid low master pressure controller output signal. The candidate must know the control circuitry for both PORV channels.
D. Correct. Failure low of PT-455 would cause the master pressure controller output signal to decrease to the low value. This causes the pressurizer control and backup heaters to energize. Energization of the heaters will increase the temperature of the pressurizer fluid causing further fluid/steam expansion at saturation temperature. This would cause the pressurizer steam bubble to rise in pressure to the point where the B PORV would open.

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

K/A Topic: 027AK1.02, Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions:
Expansion of liquids as temperature increases.

Question Source:	New
Question Cognitive Level:	Comprehension/Analysis
10 CFR Part 55 Content:	CFR 41.8/41.10/45.3
Learning Objective:	Lesson Plan L1406I, Objective L1406I03

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 9	Group #	1	
	K/A #	029EA1.01	
	Importance Rating	3.4	3.1

Proposed Question:
The crew is responding to an ATWS event and is implementing FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS. Control rods are inserting in automatic, the main turbine has been verified tripped, and the EFW pumps are running. The crew is ready to initiate Emergency Boration of the RCS.

Which of the following describes the alternate method of borating the RCS if a centrifugal charging pump cannot be started?

A. 1) Align boration path
   2) Open CS-LCV-112D/E
   3) Close CS-LCV-112B/C
   4) Start one Safety Injection pump and open one PZR PORV to establish injection flow
   5) Check pressurizer pressure less than 2385 psig.

B. 1) Start at least one boric acid pump
   2) Open CS-V426, Emergency Boration Valve
   3) Start one Safety Injection pump and open both PZR PORVs to establish injection flow
   4) Check pressurizer pressure less than 2385 psig.

C. 1) Start the positive displacement charging pump
   2) Start at least one boric acid pump
   3) Open CS-V426, Emergency Boration Valve
   4) Align charging flow path
   5) Check pressurizer pressure less than 2385 psig

D. 1) Start the positive displacement charging pump
   2) Align gravity boration flowpath
   3) Align charging flowpath
   4) Check pressurizer pressure less than 2385 psig

Proposed Answer: C

A. Incorrect but plausible. FR-S.1 does include direction for opening PORV's and block valves as necessary, however this is done to reduce RCS pressure to > 2185 psig to ensure boron injection via an operating centrifugal charging pump. Also, the WOG ERG for FR-S.1 states that SI initiation is a possible means of injecting boron, however this method is not utilized at Seabrook Station. Students are required to be familiar with the WOG ERG's.
B. Incorrect but plausible. FR-S.1 does include direction for opening PORV's and block valves as necessary, however this is done to reduce RCS pressure to > 2185 psig to ensure boron injection via an operating centrifugal charging pump. Also, the WOG ERG for FR-S.1 states that SI initiation is a possible means of injecting boron, however this method is not utilized at Seabrook Station. Students are required to be familiar with the WOG ERG's.
C. Correct. This answer describes the FR-S.1, step 4 flowpath in the event that a CCP cannot be started.
D. Incorrect but plausible because it is a physically possible flowpath. FR-S.1 does not dictate use of a gravity boration flowpath to the suction of the charging pumps.

Technical Reference(s): FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.

Proposed references to be provided to applicants during examination: None

K/A Topic: 029EA1.01, Ability to operate and monitor the following as they apply to a ATWS: Charging Pumps.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: 41.7/45.5/45.6

Learning Objective: Lesson Plan L1200I, Objective L1200I02 and L1200I13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 10	Group #	1	
	K/A #	040AK2.02	
	Importance Rating	2.6	2.6

Proposed Question:
The following plant conditions exist:

- A Reactor Trip, Safety Injection, and Main Steamline Isolation have occurred.
- RCS pressure is 1820 psig and decreasing rapidly.
- RCS temperature is 530 degrees F and decreasing rapidly.
- Containment humidity is increasing.
- Main Steamline, SG Blowdown, and Condenser off-gas radiation is normal.
- Containment pressure is 2.4 psig and increasing.
- Containment radiation is normal.

These conditions are indicative of:

A. A faulted steam generator

B. A small break LOCA

C. A steam generator tube rupture.

D. A large break LOCA

Proposed Answer: A

A is correct. A faulted steam generator would be detected/sensed by the main steam isolation and safety injection circuitry. A fault in containment would be sensed by containment humidity and pressure detectors. A fault in containment would not be sensed by containment radiation detectors or any secondary side (steamline/blowdown/condenser) radiation detectors.

B is incorrect. A small break LOCA would cause elevated containment radiation conditions.

C is incorrect. A steam generator tube rupture would be detected by secondary radiation detectors.

D is incorrect. A large break LOCA would cause elevated containment radiation readings as well as a more pronounced containment pressure transient.

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

K/A Topic: 040AK2.02, Knowledge of the interrelationship between the Steam Line Rupture and the following: Sensors and detectors

Question Source:	Direct from bank. DC Cook 2001
Question Cognitive Level:	Analysis
10 CFR Part 55 Content:	41.7/45.7
Learning Objective:	Lesson L1207I, Objective L1207I01

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 11	Group #	1	
	K/A #	055EK3.02	
	Importance Rating	4.3	4.6

Proposed Question:
The crew is performing the actions of ECA-0.0, LOSS OF ALL AC POWER.
The reactor operator places the control switches for the charging pumps in Pull-To-Lock.

This is done to prevent which of the following from occurring when power is restored?

A. Injecting cold seal injection water into the RCP seal packages and potentially damaging the seal package or bowing the RCP pump shaft.

B. Thermally shocking the reactor vessel downcomer with cold, high pressure Safety Injection water after the steam generators are depressurized.

C. Injecting cold seal injection water through the RCP seal packages and potentially thermal shocking the RCP thermal barrier heat exchangers.

D. Pressurizer overfill situation, since CS-FCV-121 will be full open and the pressurizer will be empty due to the cooldown when the steam generators are depressurized.

Proposed Answer: A

A is correct. Defeating automatic loading of Charging/SI pumps functions to protect the RCP's from damage when AC power is restored. This action prevents the automatic delivery of relatively cold seal injection flow into the RCP number 1 seal chamber and shaft area. Injection of cold seal injection has the potential to thermally shock and subsequently damage the RCP seals and shaft.

B is incorrect but plausible. The reason for placing the control switches to pull to lock is to prevent thermal stress conditions, but not to the vessel downcomer region.

C is incorrect but plausible. The reason for placing the control switches to pull to lock is to prevent thermal stress conditions, but not to the thermal barrier heat exchangers. Most of the cold seal injection flow travels from the #1 seal to the seal return header or to the #2 seal. The thermal impact on the thermal barrier heat exchanger would be minimal.

D is incorrect but plausible. If the charging pumps were to auto start they inject into the RCS adding inventory.

Technical Reference(s): ECA-0.0, LOSS OF ALL AC POWER, Background Document, step 6.

Proposed references to be provided to applicants during examination: None

K/A Topic: 055EK3.02 Knowledge of the reasons for the following responses as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power

Question Source: Direct from bank. Last used Seabrook 1998 Company Exam

Question Cognitive Level: Comprehension

10 CFR Part 55 Content: 41.5/41.10/45.6/45.13

Learning Objective: Lesson Plan L8067, Objective L8067I11

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 12	Group #	1	
	K/A #	056AA1.18	
	Importance Rating	3.2	3.2

Proposed Question:
Given the following conditions:

- The plant is in MODE 1 at 100% power.
- A loss of power occurs on Bus E5.
- The "A" Emergency Diesel Generator functions properly and reenergizes Bus E5.

Which of the following correctly describes the affect on Control Building Air system alignment?

A. The emergency makeup filter damper, CBA-DP-27A fails closed. The emergency makeup fan, CBA-FN-16A starts as soon as it's bus is energized. Once CBA-FN16A starts it generates a filter recirc. signal which opens emergency makeup filter damper, CBA-DP-27A.

B. The control room makeup air fan, CBA-FN-27A starts as soon as the bus is energized. The control room makeup air damper,CBA-DP-53A remains open.

C. The emergency makeup filter damper, CBA-DP-27A fails open upon loss of power. The emergency makeup fan, CBA-FN-16A immediately starts when the bus is energized.

D. The emergency makeup filter damper, CBA-DP-27A fails open upon loss of power. The control room makeup air fan, CBA-FN-27A starts at step 1 of the sequencer.

Proposed Answer: C

A. Incorrect. Fan 16A does immediately starts when the bus is energized. This is due to the fact that emergency damper 27A failed open upon loss of power. The start of emergency fan 16A creates a filter recirc signal which inhibits supply fan 27A from starting.
B. Incorrect. A control room filter recirc signal is generated by fan 16A starting. This inhibits fan 27A from starting.
C. Correct. The emergency damper 27A fails open on a LOP. Fan 16A will start as soon as MCC 521 is energized. Damper 27A open is a start permissive for fan 16A.
D. Incorrect. The emergency damper 27A does fail open, however fan 16A would start. This produces a filter recirc signal which would inhibit supply fan 27A from starting.

Technical Reference(s): 1-NHY-503232, 310926 sheet D87a

Proposed references to be provided to applicants during examination: None

K/A Topic: 056AA1.18, Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power: Control room normal ventilation supply fan.

Question Source: New
Question Cognitive Level: Analysis
10 CFR Part 55 Content: 41.7/45.5/45.6
Learning Objective: Lesson L8039I, Objectives L8039I04 and L8039I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 13	Group #	1	
	K/A #	057AA2.15	
	Importance Rating	3.8	4.1

Proposed Question:
The following conditions have occurred as the result of loss of power to an instrument power panel or bus:
- Auto rod withdrawal and turbine loading is blocked
- 2 of 4 feedwater regulating valves have positioned to full open
- The B PORV has armed but is not open
- The Train A safeguard actuation function is lost

Which one of the following instrument panels/bus has been lost?

A. Vital Instrument Power Panel PP-1E

B. Vital Instrument Power Panel PP-1F

C. Vital Instrument Power Panel PP-1D

D. Vital Instrument Power Panel PP-1A

Proposed Answer: D

A. Incorrect. Loss of PP-1E will result in loss of charging and seal injection flow control, loss of letdown. The loss of PP-1E will result in loss of a multitude of primary system functions. Loss of PP-1E has no impact on safeguard actuation function.
B. Incorrect. Loss of PP-1F will result in such conditions as loss of PCCW temp. control, loss of letdown etc. but will not arm the B PORV, cause feedwater regulating valves to fail open or inhibit safeguards actuation.
C. Incorrect. Loss of PP-1D can inhibit auto operation of both PORV's, but will not arm the B PORV nor will it inhibit actuation of Train A safeguards
D. Correct. Loss of PP-1A will inhibit Train A safeguards actuation as PP-1A is the primary power source for Train A ESFAS and RPS equipment. Additionally, per Dwg 310105, System Failure Analysis, EDE-PP-1A, a loss of PP-1A will cause those feedwater regulating valves selected for "Channel 1" level input to fail open, will block auto rod withdrawal and turbine loading, and will arm the B PORV.

Technical Reference(s): OS1247.01, LOSS OF 120VAC INSTRUMENT PANEL 1A, 1B, 1C, OR 1D. OS1247.02, LOSS OF 120VAC INSTRUMENT BUS PP-1E OR PP-1F.

Proposed references to be provided to applicants during examination: None

K/A Topic: 057AA2.15, Ability to determine and interpret the following as the apply to Loss Of Vital AC Instrument Bus:
That a loss of ac has occurred:

Question Source: New
Question Cognitive Level: Analysis
10 CFR Part 55 Content: CRF 43.5/45.13
Learning Objective: Lesson L1186I, Objectives L1186I09 and L1186I12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 14	Group #	1	
	K/A #	G 2.4.11, Knowledge of abnormal condition procedures.	
	Importance Rating	3.4	3.6

Proposed Question:
The plant is at 100% power. The BOP operator announces alarm D6094, DC Bus 11A Volt Lo-Lo and the crew enters OS1248.01, LOSS OF A VITAL 125 VDC BUS.

Under which of the following conditions does OS1248.01 direct a reactor trip?

A. If charging system flow has been lost and cannot be reestablished.

B. If the steam dumps ramp open.

C. If PCCW temperature to the RCP's on the affected train is decreasing.

D. If feedwater control is not available.

Proposed Answer: D

A. Incorrect. Step 3 of the procedure addresses the charging system. If charging is not in service then the step states "Take manual control and align systems as necessary".
B. Incorrect. The condenser steam dump system is not addressed in the procedure.
C. Incorrect. Step 3 of the procedure checks PCCW flow and loop temperatures, however directs the operator to "Take manual control and align systems as necessary". The procedure does not address PCCW temperature to the RCP's.
D. Correct. Step 1 of the procedure checks feedwater status, The step has the operator take manual control if necessary. If feedwater control is not available then the step directs the crew to trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION.

Technical Reference(s): OS1248.01, LOSS OF A VITAL 125 VDC BUS.

Proposed references to be provided to applicants during examination: None

K/A G 2.4.11, Knowledge of abnormal condition procedures.
Topic:
Question Source: New
Question Cognitive Level: Application
10 CFR Part 55 Content: 41.10 / 43.5 / 45.13
Learning Objective: Lesson L1189, Objective L1189I03

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 15	Group #	1	
	K/A #	062AK3.02	
	Importance Rating	3.6	3.9

Proposed Question:
The plant has sustained a LOCA. Safety Injection has actuated. All safeguards systems are functioning as designed.

Which of the following describes the expected Service Water system alignment?

A. PCCW heat exchangers and DG cooling is supplied from the Cooling Tower. SCCW heat exchangers are supplied from the Service Water pumps.

B. PCCW heat exchangers and DG cooling is supplied from the Service Water pumps. SCCW heat exchangers are isolated.

C. PCCW heat exchangers and DG cooling is supplied from the Cooling Tower. SCCW heat exchangers are isolated.

D. PCCW heat exchangers, DG cooling, and SCCW heat exchangers are supplied from the Service Water pumps.

Proposed Answer: B

B is correct. SCCW isolates on either a TA signal or S signal, making A and D incorrect. C is also incorrect because a TA signal would not be generated under the conditions proposed in the stem (Low SW pressure or LOP w/CT running).

Technical Reference(s): SW Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: 062AK3.02, Knowledge of the reasons for the following responses as they apply to the Loss Of Nuclear Service Water:
The automatic actions (alignments) within the nuclear service water resulting from the actuation of ESFAS.

Question Source: Direct from bank. Last used Seabrook 2000.

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR 41.4/41.8/45.7

Learning Objective: Lesson Plan L1119I, Objective L1119I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 16	Group #	1	
	K/A #	065AA2.05	
	Importance Rating	3.4	4.1

Proposed Question:
Procedure ON1242.01, "Loss Of Instrument Air" contains steps that direct the crew to trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION, under which of the following conditions?

A. If feedwater flow to any steam generator is isolated. The main feedwater regulating valves and bypass valves will FAIL CLOSED on loss of control air resulting in loss of feed to the steam generators.

B. If feedwater control to any steam generator is lost. The main feedwater regulating valves and bypass valves will FAIL OPEN on loss of control air resulting in excess feed to the steam generators.

C. If seal injection flow is isolated. CS-HCV-182 will FAIL CLOSED on loss of air resulting in a loss of cooling to the RCP seals.

D. If service air pressure is less than 90 psig. At 90 psig the PCCW containment isolation valves will fail closed isolating cooling water flow to the RCP's.

Proposed Answer: A

A is correct. If feed to the steam generators is lost ON1242.01, Loss Of Instrument Air directs the operators to trip the reactor.
B is incorrect. The feedwater regulating valves and bypass valves fail closed on loss of air.
C is incorrect. The procedure does not give any direction based on charging/seal injection system conditions. CS-HCV-182 fails open on loss of air. CS-HCV-182 is designed to provide flow to the seals via back pressure of the charging header. If CS-HCV-182 were to close then seal injection flow would increase, not isolate.
D is incorrect. If service air pressure is less than 90 psig the procedure directs closure of the service air isolation valves. The reactor is not directed to be tripped unless the PCCW containment isolation valves are failed closed. The OAS page discusses tripping the reactor within 10 minutes of losing PCCW flow to containment.

Technical Reference(s): ON1242.01, Loss Of Instrument Air

Proposed references to be provided to applicants during examination: None

K/A Topic: 065AA2.05, Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR 43.5/45.13

Learning Objective: Lesson Plan L1194I, Objectives L1194I03 and L1194I04

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 17	Group #	1	
	K/A #	E04EK1.3	
	Importance Rating	3.5	3.9

Proposed Question:
The crew has entered ECA-1.2, 'LOCA OUTSIDE CONTAINMENT' from E-0 based upon RDMS indication of high radiation levels in the RHR vaults. After closing RH-V14, 'RHR Train A discharge to the RCS' and RH-V22, 'RHR Train A cross-connect' and placing the 'A' train RHR and CBS pumps in pull-to-lock, the following conditions exist:

- ECCS flow is decreasing
- RCS pressure is 1100 psig and slowly increasing

Which of the following indicates the status of the LOCA and the subsequent procedural action that should be taken?

A. The LOCA is isolated. The crew should transition to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, step 1.

B. The LOCA is not isolated. The crew should transition to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, step 1.

C. The LOCA is isolated. The crew should transition to ES-1.1, SI TERMINATION, step 1.

D. The LOCA is not isolated. The crew should continue with actions in ECA-1.2, LOCA OUTSIDE CONTAINMENT.

Proposed Answer: A

A is correct. Per ECA-1.2, step #4 if RCS pressure is increasing due to successful leak isolation the crew should transition to E-1.
B is incorrect but plausible. The crew would transition to ECA-1.1 at step #4 of ECA-1.2 if RCS pressure is increasing due to leak isolation.
C is incorrect but plausible. A transition to ES-1.1 may ultimately be made to terminate SI based upon isolation of the LOCA , however this transition will most likely be made from E-1 not directly from ECA-1.2.
D is incorrect but plausible. If the initial actions in ECA-1.2 are not successful in isolating the LOCA additional actions may be taken to isolate valves in the other train.

Technical Reference(s): ECA-1.2, LOCA OUTSIDE CONTAINMENT.

Proposed references to be provided to applicants during examination: None

K/A Topic: E04EK1.3 Knowledge of the operational implications of the following concepts as they apply to the LOCA Outside Containment:
Annunciators and conditions indicating signals, and remedial actions associated with the LOCA Outside Containment.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR 41.8/41.10/45.3

Learning Objective: Lesson L1209I, Objective L1209I04RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 18	Group #	1	
	K/A #	E11EA2.2	
	Importance Rating	3.4	4.2

Proposed Question:
The crew is performing actions in ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

What condition would require that all intact steam generators be depressurized to atmospheric pressure?

A. RWST level drops below 80,000 gallons.

B. Containment building level is off scale low.

C. Reactor Vessel Full Range Level drops below 62%.

D. RCS subcooling is less than the 90° F SI termination criteria.

Proposed Answer: A

A is correct. Per ECA-1.1, if RWST level is less than 80,000 gallons then actions are taken to secure running RHR, SI and CBS pumps (and charging pumps if <40,000 gallons), and then attempts are made to add makeup to the RCS concurrent with the subsequent procedural flow path to dump steam by depressurizing the steam generators.

B is incorrect but plausible. Student may choose this distracters containment building level below indicating range could be perceived as a criteria for securing ECCS pumps. This criteria is actually used in step 13 to determine whether CBS pumps can be aligned to the containment building sump.

C is incorrect but plausible. Student could choose this distracter as full range level dropping could be an indicator that ECCS is not adequately cooling the core. This criteria is used in step 29 to determine if RCS makeup flow should be increased and is not part of the decision making criteria for depressurizing the steam generators.

D is incorrect but plausible. The 90° F SI termination criteria is used to determine whether minimum SI flow should be used to remove decay heat, at step 22. This step occurs procedurally if RWST level is >80,000 gallons and the crew was able to cool down the RCS and attemp SI termination. The student may choose this distracter as subcooling is a parameter used to evaluate RCS cooling.

Technical Reference(s): E-1, LOSS OF REACTOR OR SECONDARY COOLANT

Proposed references to be provided to applicants during examination: None

K/A Topic: E11EA2.2 Ability to determine or interpret the following as they apply to Loss Of Emergency Coolant Recrculation:
Adherence to appropriate procedures and operating within the limits of the facilities license/amendments.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: 43.5/45.13

Learning Objective: Lesson Plan L1201I, Objective L1209I01

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 19	Group #	2	
	K/A #	003AK3.04	
	Importance Rating	3.8	4.1

Proposed Question:
The following plant conditions exist:

- Reactor is at 100% power.
- One group D rod is 24 steps below group D bank height.
- The Shift Manager informs you that the Shutdown Margin is currently less than that specified as minimum in the Core Operating Limits Report.
- The misaligned rod has not been declared inoperable.

Which of the following describes the correct response?

A. Immediately commence rapid boration.

B. Immediately step Bank "D" rods inward.

C. Commence normal reactor shutdown to HOT STANDBY within 6 hours.

D. Manually trip the reactor and perform E-0 "Reactor Trip or Safety Injection."

Proposed Answer: A

A is correct. If at any time the MODE 1 Shutdown Margin is inadequate as defined in the COLR, associated Tech. Spec. 3.8.1.1. action requires immediate Rapid Boration.

B is incorrect but plausible. The major strategy for a misaligned rod is to realign the rod with the bank to prevent possible fuel damage, however, inadequate shutdown margin response takes precedence.

C is incorrect but plausible. Tech. Spec. actions associated with a "Dropped Rod" may require a power reduction, however in this case the rod is misaligned and the additional loss of shutdown margin action takes priority

D is incorrect but plausible. Multiple misaligned rods (>48 steps) would require a reactor trip. Students may choose this answer as a 24 step misalignment is significant, but does not warrant a reactor trip.

Technical Reference(s): OS1210.05, DROPPED ROD; OS1210.06, MISALIGNED CONTROL ROD.

Proposed references to be provided to applicants during examination: None

K/A Topic: 003AK3.04, Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod:
Actions contained in the EOP for dropped control rod.

Question Source: Direct from bank. Seabrook 2003.

Question Cognitive Level: Application

10 CFR Part 55 Content: 41.5/41.10/45.6/45.13

Learning Objective: Lesson L1185I, Objective L1185I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 20	Group #	2	
	K/A #	028AA2.03	
	Importance Rating	2.8	3.3

Proposed Question:
The plant is at 100% power with all Control Systems operating in automatic.

The backup pressurizer level control channel fails low causing letdown to isolate. The primary board operator responds by placing CS-FK-121, Charging Flow Controller in MANUAL to reduce charging flow. The Pressurizer Master Level Controller, RC-LK-459, remains in AUTOMATIC.

Letdown is re-established and pressurizer level has been returned to program.

What actions are required to place the Pressurizer Level Control system back in AUTOMATIC in accordance with OS1002.08, 'Pressurizer Level Control System Operation'?

A. Place RC-LK-459 in MANUAL. Adjust the output of CS-FK-121 to match the output of RC-LK-459 and place RC-LK-459 in AUTO. Then place CS-FK-121 in AUTO.

B. Place RC-LK-459 in MANUAL and adjust it's output to match the input and setpoint signals on CS-FK-121. Place CS-FK-121 in AUTO. Then place RC-LK-459 in AUTO.

C. Leave RC-LK-459 in AUTO. Adjust the output of CS-FK-121 to match the input of CS-FK-121. Then place CS-FK-121 in AUTO.

D. Leave RC-LK-459 in AUTO. Adjust the output of CS-FK-121 to match the input of RC-LK-459. Then place CS-FK-121 in AUTO.

Proposed Answer: B

B is correct. The operator had previously returned pressurizer level to programmed value. Taking RC-LK-459 to manual and matching the input and setpoint signals of the charging flow control valve will allow for the flow control valve to have a "bumpless" transfer to AUTO and then ultimately respond to the programmed RC-LK-459 output signal in a controlled manner. This technique is dictated procedurally in OS1002.08, Pressurizer Level Control System Operation. Selection of the correct answer demonstrates "synthesis" of pressurizer control system knowledge into proper application/manipulation of the systems controllers and operating procedure.

A is incorrect but is a plausible distractor. Adjustment if CS-FK-121 to match the output of RC-LK-459 could be done, but would result in a more pronounced initial charging flow transient. This could cause system instability with resultant pressurizer level oscillation, and could result in transient/inadequate cooling flow to the regenerative heat exchanger and RCP seal injection supply.

C is incorrect but plausible. Adjustment if CS-FK-121 output to match it's input could be done, but would result in an immediate change in charging flow. This could cause system instability with resultant pressurizer level oscillation, and could result in transient/inadequate cooling flow to the regenerative heat exchanger and RCP seal injection supply.

D is incorrect but plausible. The operator could adjust the CS-FK-121 output to match the input to RC-LK-459, however these signals are not synchronous and would cause a charging system flow transient. The input signal to RC-LK-459 is actual pressurizer level. The output signal from CS-FK-121 is a charging flow demand signal based on a pressurizer level vs. pressurizer level setpoint error signal generated from RC-FK-459. Use of this method would create a charging system flow transient.

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

K/A Topic:	028AA2.03, Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunction: Charging subsystem flow indicator and controller.
Question Source:	Direct from bank. Seabrook 1998 NRC.
Question Cognitive Level:	Analysis
10 CFR Part 55 Content:	43.5/45.13
Learning Objective:	Lesson L1406I, Objective L1406I03, Lesson L1182I, Objective L1182I01

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 21	Group #	2	
	K/A #	051AA2.02	
	Importance Rating	3.9	4.1

Proposed Question:
The plant was at 100% power. The following plant conditions currently exist:

- Condenser Vacuum is 22.5 in. hg and slowly decreasing.
- Turbine load reduction is in progress.
- Turbine load is 360 MWE.

Which of the following actions should be taken by the crew?

A. Immediately trip the turbine and verify all stop valves close and the generator breaker opens.

B. Continue the load reduction to increase condenser vacuum to > 25 in. hg.

C. Immediately trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION.

D. Continue the load reduction and if vacuum remains > 22.4 in. hg. remove the turbine generator from service per OS1000.06, POWER DECREASE.

Proposed Answer: C

A is incorrect. The Loss of Condenser Vacuum abnormal procedure calls for a manual reactor trip, not a manual turbine trip. Answer is plausible as the megawatt load in the stem is just above the P-9 setpoint, when the turbine could be tripped without a reactor trip.

B is incorrect. The conditions in the question stem call for a reactor trip.

C is correct. Load is less than 360 MWE and condenser vacuum is approaching the turbine trip setpoint.

D is incorrect. A load decrease below 360 MWE should not be conducted. A reactor trip is required

Technical Reference(s):

Proposed references to be provided to applicants during examination: None

K/A Topic: 051AA2.02, Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum:
Conditions requiring reactor and/or turbine trip.

Question Source: Direct from bank. Last used Seabrook 1996

Question Cognitive Level: Application

10 CFR Part 55 Content: 43.5/45.13

Learning Objective: Lesson L1188I, Objective L1188I08

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 22	Group #	2	
	K/A #	E06EA1.3	
	Importance Rating	3.7	4.0

Proposed Question:
The crew has entered FR-C.1, RESPONSE TO INADEQUATE CORE COOLING due to a valid RED path on the Core Cooling CSF.

- CS-P-2A is Danger tagged out of service.
- CS-P-2B cannot be started due to a loss of power to bus E-6.
- RCS pressure is 1900 psig and slowly increasing.
- CETCs are 750 °F and slowly increasing.
- All other ECCS equipment is functioning as designed.
- ECCS flow cannot be verified in either train.

What action should the crew initially take to establish some form of injection flow?

A. Start the Positive Displacement Charging Pump and establish flow through CS-FCV-121.

B. Start one RCP to collapse any voids in the RCS that restrict ECCS flow.

C. Open one PORV to depressurize the RCS to allow accumulator injection.

D. Depressurize all intact SGs to 125 psig to allow accumulator injection.

Proposed Answer: A

A is correct per FR-C.1, step 2 RNO. PDP is alternate high-pressure flow in FR-C.1.
B is incorrect but plausible because starting RCPs will help heat transfer but is not called for in FR-C.1 until CETCs are greater than 1100 °F.
C is incorrect but plausible. Opening the pressurizer PORV's is a functional restoration procedure strategy used to cool the core by injecting the accumulators or establishing some sort of lower pressure injection flow, however it is used in FR-H.1, Loss of Secondary Heat Sink, vice FR-C.1. Opening the PORV will decrease RCS pressure but is not called for by the procedure because it also would increase mass loss.
D is incorrect because this action takes time and injection flow will not be established in the short term.

Technical Reference(s): FR-C.1, RESPONSE TO INADEQUATE CORE COOLING

Proposed references to be provided to applicants during examination: None

K/A Topic: E06EA1.3, Ability to operate and/or monitor the following as they apply to Degraded Core Cooling: Desired operating results during abnormal and emergency situations.

Question Source: Direct from bank. Last used Seabrook 2000.

Question Cognitive Level: Application

10 CFR Part 55 Content: 41.7/45.5/45.6

Learning Objective: Lesson Plan L1227I, Objective L1227I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 23	Group #	2	
	K/A #	2.4.11 Knowledge of abnormal condition procedures.	
	Importance Rating	3.4	3.6

Proposed Question:
The plant is at 100% power. The Duty Chemist calls the Control Room and notifies the crew that the reactor coolant specific activity levels are elevated and approaching Tech. Spec. limits.

Per OS1202.05, Reactor Coolant System High Activity, what action should the crew take next?

A. Reduce plant power to less than 50% within 1 hour.

B. Direct and NSO to place the Letdown Degassifier in service.

C. Reduce letdown flow to minimize the possibility of demineralizer channeling

D. Direct the Duty Chemist to verify the initial sample results.

Proposed Answer: D

D is correct. OS1202.05, Reactor Coolant System High Activity directs the Duty Chemist to verify the initial sample results by resampling or other confirmatory means.

A is incorrect but plausible. Secondary chemistry conditions could result in the need to reduce plant power, however there is no direction in OS1202.05 to reduce plant power.

B is incorrect but plausible. The degassifier removes radioactive non-condensable gasses. RCS high activity is addressed with demineralizers and filters as the source of high activity is typically ionic or crud. The procedure does not address operation of the letdown degassifier.

C is incorrect but plausible. The procedure does address adjustment of letdown, however, letdown flow is raised to maximum flow to support cleanup efforts.

Technical Reference(s): OS1202.05, REACTOR COOLANT SYSTEM HIGH ACTIVITY

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.4.11 Knowledge of abnormal condition procedures.

Question Source: New

Question Cognitive Level: Memory

10 CFR Part 55 Content: 41.10 / 43.5 / 45.13

Learning Objective: Lesson Plan L1181I, Objective L1181I09

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 24	Group #	2	
	K/A #	EPE W/E16 EA 1.1	
	Importance Rating	3.1	3.2

Proposed Question:
Which of the following radiation monitors would cause the Containment Integrity status tree (Z) to indicate a YELLOW path due to high radiation?

A. Manipulator Crane monitors, RM-6535A and B

B. Containment Post LOCA monitors, RM-6576A and B

C. Containment Seal Table Monitor, RM-6534-1

D. COP Train A and B monitors, RM-6527A and B

Proposed Answer: B

B is correct. Computer logic states that either Containment Post LOCA monitor in HIGH alarm, (auctioneered high) causes the status tree to indicate a YELLOW path. These monitors indicate on AF and CF.
A is incorrect but plausible as the manipulator crane rad monitors are in active in containment during outages.
C is incorrect but plausible as the seal table monitor is in containment.
D is incorrect but plausible as the Containment Online Purge rad monitors will indicate containment radiation levels.

Technical Reference(s): Main Plant Computer Logic for Containment Post LOCA Monitor

Proposed references to be provided to applicants during examination: None

K/A Topic: EA 1.1 Ability to operate and/or monitor the following as they apply to High Containment Radiation: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Question Source: Modified from Bank. Seabrook 1998 Company Exam. Original question attached for reference.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: 41.7/45.5/45.6

Learning Objective: Lesson L1212I, Objective L1212I11

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 25	Group #	2	
	K/A #	EPE W/E03 EK3.2	
	Importance Rating	3.4	3.9

Proposed Question:
The following plant conditions exist:

- Due to a Small Break LOCA a Plant Trip and Safety Injection has occurred.
- The crew has entered E-0, "Reactor Trip or Safety Injection" and is currently processing step 11, "Check If RCS Is Intact".
- All automatic equipment has responded as expected.

The following indications are noted:

- Containment pressure is 2.5 psig and stable.
- Containment radiation is 2 R/HR.
- RCS pressure is 1540 psig and stable.
- Core exit thermocouple temperature is 500°F.
- Pzr level is 12% and slowly increasing.

Assuming conditions do not change, in which of the following procedures will the crew be directed to stop one charging pump?

A. E-0 "Reactor Trip or Safety Injection".

B. ES-1.2, "Post-LOCA Cooldown and Depressurization".

C. ES-1.1, "SI Termination".

D. E-1 "Loss of Reactor or Secondary Coolant"

Proposed Answer: C

A is incorrect. Step 14 of E-0 RNO directs you to E-1.
B is incorrect. A transition from E-1 to ES-1.2 will only be made if the Operator reaches Step 12 of E-1. If the Operator properly executes E-1 he will transition out of E-1 at Step 6 .
C is correct. A charging pump will be stopped IAW Step 2 of ES-1.1.
D is incorrect. E-1 step 6 will allow transition to ES-1.1 if four ECCS termination criteria are satisfied. Sub-cooling is ~ 100°F, 40°F is required. Heat Sink is available (AFW auto started, and operators control feed flow based on EOP steps). RCS pressure is stable. PZR level is > 5%. Therefore a transition to ES-1.1 will be made .

Technical Reference(s): E-0, "Reactor Trip or Safety Injection", step 14, "Check If RCS Is Intact". ES-1.1, "SI Termination

E-1 "Loss of Reactor or Secondary Coolant"

Proposed references to be provided to applicants during examination: None

K/A Topic: EK3.2 Knowledge of the reasons for the following responses as they apply to the LOCA Cooldown and Depressurization:

Normal, abnormal, and emergency operating procedures associated with LOCA Cooldown and Depressurization.

Question Source:	Direct from bank. Last used Seabrook 2003 Company Exam
Question Cognitive Level:	Analysis
10 CFR Part 55 Content:	41.5/41.10/45.6/45.13
Learning Objective:	Lesson L1226I, Objective L1226I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 26	Group #	2	
	K/A #	EPE W/E09 EK1.2	
	Importance Rating	3.3	3.7

Proposed Question:
The crew has entered ES-0.2, NATURAL CIRCULATION COOLDOWN. After entering the procedure, conditions are established to start a RCP.

According to the procedure, which of the following may occur when the RCP is started?

A. Pressurizer PORV opens.

B. Letdown isolation.

C. Steam Generator safety valve opens.

D. Decrease in RVLIS Full Range Level.

Proposed Answer: C

C is correct. The note prior to step 1 of the procedure states, "Starting an RCP while on natural circulation may cause SG safety valve actuation due to a rapid rise in SG level and pressure".

A is incorrect but plausible. The procedure specifically states "Starting an RCP while on natural circulation may cause SG safety valve actuation due to a rapid rise in SG level and pressure", which is due to a transfer of thermal energy from the RCS to the SG's. This answer is a plausible distracter as the RCP is transporting thermal energy through the reactor coolant system.

B is incorrect but plausible. Starting and RCP will have temperature reduction effects on the RCS but not to the point where pressurizer level would drop to the letdown isolation value.

D is incorrect but plausible. Starting a reactor coolant pump will have temperature reduction effects on the loop with forced cooling but not to the point where RVLIS would indicate a head bubble.

Technical Reference(s): ES-0.2, NATURAL CIRCULATION COOLDOWN.

Proposed references to be provided to applicants during examination: None

K/A Topic: EPE W/E09 EK1.2 Knowledge of the operational implications of the following concepts as they apply to the Natural Circulation Cooldown:
Normal, abnormal, and emergency operating procedures associated with Natural Circulation Cooldown.

Question Source: Direct from bank. Last used Seabrook 1998 Company Exam.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: 41.8/41.10/45.3

Learning Objective: Lesson L1225I, Objective L1225I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
Question # 27	Group #	2	
	K/A #	EPE W/E08EK2.1	
	Importance Rating	3.4	3.7

Proposed Question:
FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS, step 15 directs the operators to isolate all SI Accumulators.

What is the EOP basis for isolating the SI Accumulators?

A. To prevent injecting gasses into the RCS and creating a gas bubble in the vessel head region.

B. To prevent injecting gasses into the RCS that could potentially gas bind the steam generator u-tubes.

C. To prevent the additional stress that would be created by injection of cold SI accumulator water.

D. To preserve a source of highly borated water to prevent recriticality during cooldown.

Proposed Answer: C

C is correct. Per the Westinghouse FR-P.1 Background Document, "The injection of the cold SI accumulator water into the RCS should be avoided due to the additional thermal stresses it could cause. Since the SI termination criteria of RCS subcooling and RVLIS are satisfied at this time, the accumulators are no longer required and can be isolated."
A is incorrect. Although gas intrusion from the accumulators is a concern elsewhere in the EOP's, the specific concern in this situation is prevention of additional thermal stress that would be created by injection of cold SI accumulator water.
B is incorrect. Gas binding of SG tubes is not a concern addressed in FR-P.1.
D is incorrect. Recriticality from cooldown is not a concern at this time. Thermal shock conditions are a result of cooldown combined with a relative high pressure. Further cooldown is undesirable.

Technical Reference(s): FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS

Proposed references to be provided to applicants during examination: None

K/A Topic: EPE W/E08EK2.1, Knowledge of the interrelationships between the Pressurized Thermal Shock and the following:
Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question Source: Direct. McGuire 2000

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: 41.7/45.7

Learning Objective: Lesson L1208I, Objective L1208I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 28	Group #	1	
	K/A #	003K1.04	
	Importance Rating	2.6	2.9

Proposed Question:
The following plant conditions exist:

- The plant is at 100% power.
- The reactor operator is performing a VCT divert.
- Volume Control Tank pressure drops from 23 psig to 19 psig.

Which of the following describes the effect on RCP seal flows?

A. RCP #1 seal leakoff flow will decrease due to the decreased back pressure.

B. RCP #1 seal leakoff flow will increase due to the decrease in #2 seal leakoff back pressure.

C. RCP #2 seal leakoff flow will decrease due to the decrease in #1 seal leakoff back pressure.

D. RCP #2 seal leakoff flow will increase due to the decreased back pressure.

Proposed Answer: C

C is correct. #2 seal flow will decrease because #1 seal leakoff flow will experience less VCT back pressure. With less VCT backpressure more of the #1 seal leakoff flow will exit via the seal return line and less will be directed towards the #2 seal.

This question tests the students knowledge of the physical arrangement of the integrated RCP seal package. Common misconceptions about the configuration and operation of the seal package make all distractors plausible.

A is incorrect but plausible. #1 seal leakoff flow will change due to the decreased backpressure, however the flow will increase.
B is incorrect but plausible. #1 seal leakoff flow will increase, but this is due to a decrease in #1 seal leakoff backpressure.

D is incorrect but plausible. There is less backpressure, however this effect is on the supply side of the #2 seal. There would be less flow to the #2 seal, so #2 seal leakoff would tend to decrease.

Technical Reference(s): OS1001.05, Reactor Coolant Pump Operation

Proposed references to be provided to applicants during examination: None

K/A Topic: 003K1.04 Knowledge of the physical connections and/or cause effect relationships between the RCPS and the following systems:
CVCS

Question Source: New

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.2 to 41.9/45.7 to 45.8

Learning Objective: Lesson L1167I, Objective L1167I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 29	Group #	1	
	K/A #	003K6.04	
	Importance Rating	2.8	3.1

Proposed Question:
An inadvertent Phase 'A' Containment Isolation signal has occurred. What is the status of Reactor Coolant Pump No. 1 seal leak off flow under this condition?

A. No. 1 seal leak off flow is directed to the Volume Control Tank.

B. No. 1 seal leak off flow is directed to the Containment Structure Sump.

C. No. 1 seal leak off flow is directed to the Reactor Coolant Drain Tank.

D. No. 1 seal leak off flow is directed to the Pressurizer Relief Tank.

Proposed Answer: D

D is correct. An inadvertent Phase A Containment Isolation signal will close either CS-V-167 or CS-V-168 depending on which train had the inadvertent signal. These two valves are in series. Closure of either valve will route RCP #1 seal leakoff flow to the Pressurizer Relief Tank through relief valve CS-V-794.
A is incorrect. An inadvertent Phase A Containment Isolation signal will close either CS-V-167 or CS-V-168 depending on which train had the inadvertent signal. The flowpath out of containment to the VCT is isolated.
B is incorrect. #1 seal leakoff flow will remain contained and routed to the PRT.
C is incorrect. CS-V-794 discharges to the PRT, not the RCDT.

Technical Reference(s): PID 1-CS-B20726 — OS1205.01, INADVERTANT PHASE A CONTAINMENT ISOLATION

Proposed references to be provided to applicants during examination: None

K/A Topic: 003K6.04 Knowledge of the effect of a loss or malfunction one of the following will have on the RCPS: Containment isolation valves affecting RCP operation.

Question Source: New

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: 41.7/45.5

Learning Objective: Lesson L1181I, Objective L1181I05
Lesson L8021I, Objective L8021I27

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 30	Group #	1	
	K/A #	004K2.03	
	Importance Rating	3.3	3.5

Proposed Question:
Given the following sequence of events:

- The plant is initially at 100% power.
- The 'B' EDG is running fully loaded in parallel with the BUS E-6 UAT supply for monthly surveillance testing.
- The 'B' Centrifugal Charging Pump (CS-P-2B) is running.
- A major RCS Loss of Coolant Accident (LOCA) occurs resulting in Safety Injection (SI) actuation.
- Offsite power remains available.

Which of the following describes the effect on CS-P-2B?

A. It is stripped from BUS E-6, then immediately reloaded on the B EDG by the B EPS.

B. It is stripped from BUS E-6, then immediately reloaded on the UAT supply to BUS E-6.

C. It will continue to run without interruption, powered by the UAT supply to BUS E-6.

D. It will continue to run without interruption, powered by the B EDG.

Proposed Answer: C

C is correct. The B EDG will keep running due to the SI signal, however it's output breaker will trip open. Offsite power is still available via the UAT so CS-P-2B remains energized.

A is incorrect but plausible. The charging pump would be energized if EPS had actuated, however there is no EPS sequence with just an SI signal if offsite power is still available.

B is incorrect but plausible. The charging pump will be energized by the UAT, however there would be no stripping sequence.

D is incorrect but plausible. The charging pump will continue to run, however it will be powered from the UAT.

Technical Reference(s): M Print 310891, sheet A62

Proposed references to be provided to applicants during examination: None

K/A 004K2.03 Knowledge of bus power supplies to the following:
Topic: Charging Pumps
Question Source: Direct from bank. TEB 20257
Question Cognitive Level: Analysis
10 CFR Part 55 Content: 41.7
Learning Objective: Lesson L8024I, Objective L8024I10

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 31	Group #	1	
	K/A #	004K5.19	
	Importance Rating	3.5	3.9

Proposed Question:
The plant is at 50% power.

Due to a malfunction, the temperature of the letdown flow entering the mixed bed demineralizers increases by 10°F.

What effect, if any, does higher temperature water entering the mixed beds have on Shutdown Margin?

A. No effect, if rods are in Manual.

B. No effect, if rods are in Auto.

C. INCREASES with rods in Auto or Manual control.

D. DECREASES with rods in Auto or Manual control.

Proposed Answer: C

Increasing temperature of the water entering the mixed beds causes boron to be released by the beds. This increases boron concentration in the RCS. As boron leaks into the core, negative reactivity is inserted. This will have to be offset by positive reactivity if the plant is to remain at the initial power level. If rods are in Manual, Tave will fall, on a plant trip, the amount of reactivity inserted from Tave going to No-load, is less, therefore, SDM is increased, (A incorrect).

If rods are in Auto, rods will move out to maintain Tave and reactor power. On a trip, the rods will insert more reactivity, therefore SDM is increased, (B incorrect).

In both cases, SDM increased, (C correct, D incorrect).

Technical Reference(s): Tech Spec definition of Shutdown Margin; RE Reactivity Curves

Proposed references to be provided to applicants during examination: None

K/A Topic: 004K5.19 Knowledge of the operational implications of the following concepts as they apply to the CVCS: Concept of SDM

Question Source: Direct from Bank. Last used Seabrook 1998 Company Exam

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.5/45.7

Learning Objective: Lesson L1404I, Objective L1404I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 32	Group #	1	
	K/A #	005K3.01	
	Importance Rating	3.9	4.0

Proposed Question:
The following plant conditions exist:

- The plant is in MODE 5.
- Train "B" RHR is in service in COOLDOWN mode.
- Core Exit Thermocouple Temperature is 182°F and STABLE
- RHR HEAT EXCHANGER OUTLET VALVE, RH-HCV-607 is 10% OPEN
- RHR HEAT EXCHANGER BYPASS FLOW CONTROL VALVE, RH-FCV-619, is maintaining total RHR flow at 3500 gpm
- A loss of Instrument Air pressure occurs.

Which of the following describes the effect on the RHR system and on RCS temperature?

	RH-HCV-607	RH-FCV-619	RCS Temperature
A.	FAILS AS IS	FAILS AS IS	INCREASES
B.	FAILS AS IS	FAILS CLOSED	INCREASES
C.	FAILS OPEN	FAILS CLOSED	DECREASES
D.	FAILS OPEN	FAILS AS IS	DECREASES

Proposed Answer: C

A and B are incorrect. RH-HCV-607 is a fail-open valve.
D is incorrect. RH- FCV-619 is a fail closed valve.
C is correct. Each valve will fail in the safe position, directing full flow through the RH heat exchanger. The increased flow through the heat exchanger will result in an RCS cooldown.

Technical Reference(s): Seabrook print PID-1-RH-B20663

Proposed references to be provided to applicants during examination: None

K/A Topic: 005K3.01, Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: RCS

Question Source: New
Question Cognitive Level: Analysis
10 CFR Part 55 Content: CFR 41.7/45.6
Learning Objective: Lesson L8033I, Objective L8033I07

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 33	Group #	1	
	K/A #	G 2.1.28	
	Importance Rating	3.2	3.3

Proposed Question:

Which of the following is a potential consequence if no RHR pumps are available during a large break LOCA?

A. DBNR decreases to less than 1.3.

B. No long term cooling for decay heat removal.

C. No immediate injection flow is available to begin refilling the reactor vessel.

D. Insufficient injection flow is available to remove decay upon break initiation.

Proposed Answer: B

B is correct. The RHR pumps provide flow when RCS pressure decreases below their shutoff head. At least one train of RHR is assumed to be available to provide long term decay heat removal and are used when the plant switches over to recirculation.

A is incorrect. The consequence of loss of RHR pumps is particular to the recirc phase when the pumps provide suction for the SI pumps and heat removal via the hx's.

C is incorrect but plausible. It is plausible that RCS pressure could drop below the shutoff head of the RHR pumps, however it is the function of the accumulators to refill the vessel.

D is incorrect. Heat removal would be accomplished out of the break with ECCS injection via the SI pumps and CCP's.

Technical Reference(s): FSAR section 6.3

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.1.28 Knowledge of the purpose and function of major system components and controls.

Question Source: Direct from bank. Last used Seabrook 1998 Company exam

Question Cognitive Level: Memory

10 CFR Part 55 Content: CFR: 41.7

Learning Objective: Lesson L8034I, Objective L8034I06
Lesson L8033I, Objective L8033I01

Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
Question # 34	Group #	1	
	K/A #	006K4.07	
	Importance Rating	3.4	3.8

Proposed Question:
A Large Break LOCA has occurred. All safeguards equipment functioned as designed. NO safeguards actuation signals have been RESET.

The RWST LO-LO level alarm has actuated.

Which of the following describes how swapover to Cold Leg recirculation will be accomplished?

A. Containment recirculation sump valves, CBS-V8 and CBS-V14, will automatically open. RWST suction valves, CBS-V2 and CBS-V5, will automatically close when the containment recirculation suction valves are fully open.

B. Containment recirculation sump valves, CBS-V8 and CBS-V14, will automatically open. RWST suction valves, CBS-V2 and CBS-V5, must be manually closed when the containment recirculation valves are open.

C. Containment recirculation sump valves, CBS-V8 and CBS-V14, must be manually opened. RWST suction valves, CBS-V2 and CBS-V5, must be manually closed.

D. Containment recirculation sump valves, CBS-V8 and CBS-V14, must be manually opened. RWST suction valves, CBS-V2 and CBS-V5, automatically close when the containment recirculation valves are open.

Proposed Answer: B

B is correct. As long as an S signal is present, the containment valves, CBS-V8 and CBS-V14, will auto open. After they are open, S can be reset, and the RWST suctions, CBS-V2 and CBS-V5, may be manually closed from the control room.

A is incorrect. CBS-V-2 and 5 do not auto close.

C is incorrect. CBS-V-8 and 14 will auto open.

D is incorrect. CBS-V-8 and 14 will auto open. CBS-V-2 and 5 must be manually closed.

Technical Reference(s): ES-1.3, TRANSFER TO COLD LEG RECIRCULATION

Proposed references to be provided to applicants during examination: None

K/A Topic: 006K4.07, Knowledge of ECCS design features and/or interlocks which provide the following: Normal water supply for SIS

Question Source: Direct from Bank. Last used Seabrook 2000 NRC.

Question Cognitive Level: Memory

10 CFR Part 55 Content: CFR 41.7

Learning Objective: Lesson L8035I, Objective L8035I13
Lesson L1203I, Objective L1203I08

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 35	Group #	1	
	K/A #	007K4.01	
	Importance Rating	2.6	2.9

Proposed Question:
The following plant conditions exist:

The plant is in Mode 1 at 100% power.
One pressurizer PORV is leaking.
Pressurizer Relief Tank (PRT) level and temperature are slowly increasing.
Assume a normal system lineup with no other abnormalities.

Assuming no operator action, which of the following best describes the expected system response?

A. At 120° F PRT water will automatically be transferred to the RCDT.

B. At 92% level PRT water will automatically be transferred to the RCDT.

C. At 120° F PRT water will automatically begin circulating through the PRT Heat Exchanger.

D. At 92% level PRT water will automatically begin circulating through the PRT heat exchanger.

Proposed Answer: C

A is incorrect. The PRT cooling system is normally placed in the recirculation mode.
B is incorrect. The PRT cooling system is normally placed in the recirculation mode.
C is correct. At 120° F, PRT water will automatically begin circulating through the PRT Heat Exchanger.
D is incorrect. Circulation through the PRT heat exchanger is based on upon temperature and not level.

Technical Reference(s): OS1001.08, Pressurizer Relief Tank Operations

Proposed references to be provided to applicants during examination: None

K/A Topic: 007K4.01, Knowledge of the PRTS design features and/or interlocks which provide for the following: Quench tank cooling

Question Source: New

Question Cognitive Level: Analysis/Comprehension

10 CFR Part 55 Content: CFR 41.7

Learning Objective: Lesson L8022I, Objective L8022I09

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 36	Group #	1	
	K/A #	008A1.04	
	Importance Rating	3.1	3.2

Proposed Question:
Given the following:

- The plant is at 80% power.
- The level in the "A" Primary Component Cooling Water Head Tank is increasing.

Which of the following is the potential source of leakage?

A. Letdown Heat Exchanger.

B. "A" PCCW Heat Exchanger.

C. Spent Fuel heat exchanger.

D. Excess Letdown Heat Exchanger.

Proposed Answer: A

A is correct but plausible. CVCS pressure at the letdown heat exchanger is approx. 350 psig which is higher than Train "A" PCCW pressure which is approx. 105 psig. The letdown heat exchanger is cooled by Train "A" PCCW.
B is incorrect but plausible. SW pressure through the Train "A" PCCW heat exchanger is approx. 49.5 psig. Train "A" PCCW system pressure is approx 105 psig.
C is incorrect but plausible. Train "A" PCCW does supply cooling to the "A" Spent Fuel Cooling heat exchanger, however, SF Cooling pressure is approx. 20 psig. Train "A" PCCW pressure is approx 105 psig.
D is incorrect but plausible. The excess letdown heat exchanger, when in service, would be at approx. 125 psig. This pressure is slightly higher than PCCW system pressure and could lead to in leakage, however, the heat exchanger is cooled from train 'B' PCCW.

Technical Reference(s): OS1212.01, PCCW SYSTEM MALFUNCTION

Proposed references to be provided to applicants during examination: None

K/A Topic: A1.04, Ability to predict and/or monitor changes in parameters associated with operating the CCWS controls including:
Surge Tank Level

Question Source: Direct from bank.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.5/45.5

Learning Objective: Lesson L8036I, Objective L8036I12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 37	Group #	1	
	K/A #	010K6.01	
	Importance Rating	2.7	3.1

Proposed Question:
A plant cooldown is in progress. The plant has just entered MODE 4. Wide range Thot instrumentation is reading 349°F. Wide range Tcold instrumentation is reading 340°F.

Wide range pressure transmitter, PT-403 fails HIGH.

Which of the following describes expected plant response?

A. Both PORV's remain closed.

B. PORV 456A opens

C. PORV 456B opens

D. Both PORVs open

Proposed Answer: A

A is correct. To open, the pressure must be high and the temperature low, additionally, an arming signal must exist from the other train. Above 342 degrees, the pressure setpoint is 2385 psig. Neither PORV will open.
B is incorrect. Although "A" train of LTOP is armed, it utilizes PT-405 to sense actual RCS pressure.
C is incorrect. The B PORV will not open.
D is incorrect, although "A" train of LTOP is armed, it utilizes PT-405 to sense actual RCS pressure.

Technical Reference(s): Westinghouse Process Control Block Diagram: 1-NHY-509038, RCS Cold Overpressurization Control

Proposed references to be provided to applicants during examination: None

K/A Topic: 010K6.01, Knowledge of the effect of a loss or malfunction that the following will have on the PZR PCS: Pressure Detection Systems

Question Source: Direct from bank. Last used Seabrook 1998 Company Exam

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7/45.7

Learning Objective: L8027I, Objective L8027I07

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 38	Group #	1	
	K/A #	010A1.06	
	Importance Rating	3.1	3.2

Proposed Question:
A rapid load reduction is being performed from 100% power. Shortly after the downpower is commenced, the pressurizer backup heaters energize even though spray valves are open.

Which of the following describes why the heaters are energized?

A. The pressurizer pressure controller is responding to the rate/lag compensated pressure channel inputs.

B. The controlling pressure channel has failed low.

C. The pressurizer level controller is responding to a greater than 5% outsurge from the downpower.

D. The pressurizer level controller is responding to a greater than 5% insurge from the downpower.

Proposed Answer: D

D is correct. The downpower will cause RCS temperature to increase due to a decrease in heat removal. This will cause RCS water to expand, resulting in an insurge to the pressurizer, so both PZR pressure and level will increase. The increase in pressure causes spray valves to open, and when pressurizer level increases by 5%, the heaters will energize. The reason for this is that the temperature of the insurging water is not as hot as the pressurizer water, and if an outsurge follows with the pressurizer water at less than saturation temperature, RCS pressure could rapidly drop.

A and B are incorrect. Backup heaters cycle below 2235 psig, and spray valves cycle above 2235 psig, and should not be on concurrently.

C is incorrect. The backup heaters do not energized with a –5% level deviation.

Technical Reference(s): Westinghouse Process Control Block Diagram: 1-NHY-509027, Pressurizer Level Control

Proposed references to be provided to applicants during examination: None

K/A Topic: 010A1.06, Ability to predict and/or monitor changes in parameters associated with operating the PZR PCS controls including:
RCS heatup and cooldown effect on pressure.

Question Source: Direct from bank. Last used Millstone 2000.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.5/45.5

Learning Objective: Lesson L8027I, Objectives L8027I06 and L8027I08

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 39	Group #	1	
	K/A #	012K1.03	
	Importance Rating	3.7	3.8

Proposed Question:
Which trip devices (UV or Shunt) will actuate for Reactor Trip Breaker "B" (RTB) and Bypass Breaker "A" (BYA) if SSPS Train "B" initiates an automatic reactor trip?

A. RTB UV only, BYA UV only.

B. RTB UV and Shunt, BYA UV only.

C. RTB UV only, BYA UV and Shunt.

D. RTB UV and Shunt, BYA UV and Shunt

Proposed Answer: B

B is correct. An automatic reactor trip signal from Train B of SSPS will send both a UV and Shunt Trip signal to the B Reactor Trip Breaker. The A Bypass Breaker will only receive a UV signal.
A is incorrect. The RTB will receive a shunt trip signal.
C is incorrect. RTB will receive a shunt trip. BYA will not receive a shunt trip.
D is incorrect. BYA will not receive a shunt trip.

Technical Reference(s): Westinghouse print 1-NHY-50902, Reactor Trip Signals

Proposed references to be provided to applicants during examination: None

K/A Topic: K1.03, Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems:
CRDS

Question Source: Direct from Seabrook exam bank.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.2 to 41.9/45.7 to 45.8

Learning Objective: Lesson L8056I, Objective L8056I10

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 40	Group #	1	
	K/A #	012A2.05	
	Importance Rating	3.1	3.2

Proposed Question:
The plant is at 100% power and the Loop 1 narrow range T_{hot} instrument has failed low. The operating crew has completed all the necessary actions of the applicable abnormal operating procedure for this failure, tripped applicable bistables, and all controls are back in automatic.

Subsequently, pressurizer pressure instrument PT-456 fails LOW.

Which of the following describes the expected plant response?

A. Reactor trip will occur on Low Pressurizer Pressure.

B. Reactor trip will occur due to OTΔT trip coincidence being met.

C. Reactor trip will occur due to OPΔT trip coincidence being met.

D. "S" signal on Pressurizer Low Pressure.

Proposed Answer: B

Per OS1201.08, TAVG/DELTA T INSTRUMENT FAILURE, both the loop 1 OTΔT and OPΔT trip bistables are taken to the tripped position. A subsequent PT-456 instrument failure will create the required 2 of 4 coincident for a reactor trip on OTΔT.

A is incorrect. The reactor will trip, but not on low pressurizer pressure. The pressurizer low pressure reactor trip requires a 2/4 logic @1945 psig.

C is incorrect. Pressure channels do not input to the OPΔT reactor protection circuitry.

D is incorrect. Low Pressure SI would require a 2/4 pressure low logic.

Technical Reference(s): OS1201.08, TAVG/DELTA T INSTRUMENT FAILURE

Proposed references to be provided to applicants during examination: None

K/A Topic: 012A2.05 Ability to predict the impacts of the following malfunctions or operations on the RPS, and based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Faulty or erratic operation of detectors and function generators.

Question Source: Direct from bank. Last used Seabrook 1996.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: 41.5/43.5/45.3/45.5

Learning Objective: Lesson L1182I, Objective L1182I13
Lesson L8026I, Objective L8026I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 41	Group #	1	
	K/A #	013A3.01	
	Importance Rating	3.7	3.9

Proposed Question:
The following plant conditions exist:

- The plant is at 100% power.
- Pressurizer pressure channel PT-455 has failed high.
- The operating crew has carried out the actions of OS1201.06, PZR PRESSURE INSTRUMENT/COMPONENT FAILURE.
- Pressurizer pressure channel PT-457 is now the controlling channel.
- All systems have been returned to automatic control.

A loss of 120 VAC vital instrument panel PP-1C has just occurred. Which of the following describes the impact on the plant?

A. The plant will remain at 100% power. RCS pressure control will be in manual and automatic actuation of the PORV's has been lost.

B. Safety injection will actuate due to the low pressurizer pressure logic coincidence being met.

C. The PORV's will open due to a high pressure signal and this will eventually lead to a safety injection on low RCS pressure.

D. The master pressure controller will cause the pressurizer control heaters to go to minimum output and close the spray valves.

Proposed Answer: B

B is correct. The operating crew will have tripped the bistable associated with the failure high of PT-455. This includes the low pressure SI bistable. When the vital AC instrument panel is lost a second low pressure bistable will be received and the 2 of 4 low pressure SI signal coincidence will be met.
A is incorrect. A Safety Injection signal will be generated. The plant will not remain at power.
C is incorrect. There would be no high pressure signal to the PORV's.
D is incorrect. A low signal to the master controller would cause control group heaters to go to maximum output.

Technical Reference(s): OS1201.06, PZR PRESSURE INSTRUMENT/COMPONENT FAILURE.

Proposed references to be provided to applicants during examination: None

K/A Topic: 013A3.01 Ability to monitor automatic operation of the ESFAS including: Input channels and logic.

Question Source: Direct from bank. Last used Seabrook 1996

Question Cognitive Level: Analysis

10 CFR Part 55 Content: 41.5/43.5/45.3/45.13

Learning Objective: Lesson L8056I, Objective L8056I19
Lesson L1182I, Objective L1182I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 42	Group #	1	
	K/A #	022A4.05	
	Importance Rating	3.8	3.8

Proposed Question:
Which of the following would prevent the operator from starting CAP-FN-35, Containment Air Purge Exhaust Fan?

A. Containment Rad Monitor RM-6535A in ALERT

B. Supply or Exhaust Damper CAP-DP-8A or 8B OPEN

C. Containment Ventilation Isolation Signal ACTUATED

D. CAP-FN-9, Pre-Entry Purge Fan, or CAP-FN-34, Refueling Purge Supply Fan NOT RUNNING

Proposed Answer: A

A is correct. One of the starting interlocks for CAP-FN-35 is that RM-6535A cannot be in alert.

B is incorrect but plausible. CAP-DP-8A or 8B are interlocks for the fan start circuit, however the dampers will inhibit a fan start if the are Full Closed.

C is incorrect but plausible. A CVI signal does inhibit a start of CAP system fan CAP-FN-9, however, it is not associated with CAP-FN-35.

D is incorrect but plausible. It is conceivable that an exhaust fan would have a start permissive that associated with supply fans, however, CAP-FN-35's start permissive is associated with having its associated dampers 8A and 8B not full closed.

Technical Reference(s): Print 1-NHY-503223, Containment Air Purge Exhaust Fan FN-35 Logic Diagram

Proposed references to be provided to applicants during examination: None

K/A Topic: 022A4.05, Ability to manually operate and/or monitor in the control room: Containment readings of temperature, pressure, and humidity.

Question Source: Direct from bank.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.7/45.5 to 45.8

Learning Objective: Lesson L8038I, Objective L8038I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 43	Group #	1	
	K/A #	G 2.4.2, Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	
	Importance Rating	3.9	4.1

Proposed Question:

The following plant conditions exist:

- The plant has sustained a Steam Line Break.
- The reactor has tripped.

The following indications are noted:

- SG A pressure - 700 psig and slowly DECREASING
- SG B pressure - 600 psig and steadily DECREASING
- SG C pressure - 850 psig and STABLE
- SG D pressure - 850 psig and STABLE
- RCS pressure - 1880 psig and DECREASING
- Containment pressure - 6 psig and INCREASING

NO additional actions have been taken. All safeguards systems have functioned as designed.

Which of the following ESF Actuations have occurred?

A. SI, CONTAINMENT ISOLATION phase A ONLY.

B. SI, CONTAINMENT ISOLATION phase A, and MAIN STEAMLINE ISOLATION ONLY.

C. SI, CONTAINMENT ISOLATION phase A, MAIN STEAMLINE ISOLATION, and EFW ACTUATION ONLY.

D. SI, CONTAINMENT ISOLATION phase A, MAIN STEAMLINE ISOLATION, EFW actuation, and CONTAINMENT SPRAY ACTUATION/ phase B.

Proposed Answer: C

Plant conditions require actuation of containment HI-1 (SI, CIS-A) and HI-2 (MSLIS) (4.3 psig). Steam pressure has not dropped to the Steam Line pressure MSLIS setpoint, but containment pressure has increased above HI-2. An EFW actuation occurs on an SI signal.
A is incorrect. A Main Steam Isolation and EFW Actuation also occurs.
B is incorrect. An EFW actuation also occurs.
D is incorrect. A CBS actuation and Phase B actuation do not occur because the setpoint is 18 psig containment pressure.

Technical Reference(s): PLS, Pages 10 and 11

Westinghouse Functional diagram 509048.

Proposed references to be provided to applicants during examination: None

K/A Topic:	G 2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. Note: The issue of setpoints and automatic safety features is not specifically covered in the systems sections).

Question Source:	Modified from bank. Seabrook 2005	Original question attached to reference.
Question Cognitive Level:	Analysis	
10 CFR Part 55 Content:	CFR: 41.7 / 45.7 / 45.8	
Learning Objective:	Lesson L8035I, Objective L8035I08 Lesson L8056I, Objective L8056I19	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 44	Group #	1	
	K/A #	039A4.07	
	Importance Rating	2.8	2.9

Proposed Question:
The following plant conditions exist:

- The reactor has just tripped from 100% power.
- 'B' Reactor Trip Breaker (RTB) is CLOSED.
- 'A' Reactor Trip Breaker (RTA) is OPEN.
- The steam dumps are in the Tavg mode.

Which of the following describes the automatic operation of the Steam Dumps as a result of this transient?

A. The Steam Dumps will OPEN on the Plant Trip Controller.

B. The Steam Dumps will OPEN on the Load Rejection Controller.

C. The Steam Dumps will remain CLOSED until the Steam Pressure Mode is selected.

D. The Steam Dumps will remain CLOSED because demand is less than required for operation.

Proposed Answer: B

B is correct. The steam dumps will arm based on a Train 'A' P-4 signal, however the steam dump controller will not swap to the Plant Trip Controller as this requires a Train 'B' P-4 signal. There is no Train 'B' P-4 signal due to failure of the 'B' Reactor trip breaker to open.

A is incorrect but plausible. The steam dumps will open, however they would not be operating with the Plant Trip Controller because the Train 'B' P-4 signal is not present.

C is incorrect but plausible. The Train 'A' and Train 'B' Reactor Trip Breakers serve to arm the steam dumps and swap controllers. The student must know that Train 'A' P4 arms the Steam Dumps. If the student decides that Train 'B' P-4 arms the steam dumps then they may believe that the controller would have to be swapped to the Steam Pressure Mode.

D is incorrect but plausible. The student may think that the steam dumps are on the load rejection controller and there is not enough of a demand due to the reactor tripping.

Technical Reference(s): Westinghouse Process Print-Steam Dumps

Proposed references to be provided to applicants during examination: None

K/A Topic: 039A4.07, Ability to manually operate or monitor in the control room: Steam dump valves

Question Source: Direct from bank.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7/45.5 to 45.8

Learning Objective: Lesson L8047I, Objective L8047I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 45	Group #	1	
	K/A #	059K3.04	
	Importance Rating	3.6	3.8

Proposed Question:
With the plant initially at 100% power, a significant feed water heater level transient results in HI-HI levels in the 25A and 26A feedwater heaters. Extraction steam isolates to the heaters.

Assuming the reactor does not trip, what will be the initial effect on indicated narrow range SG levels, RCS Tcold and reactor power?

A. SG NR levels will increase, RCS Tcold will increase, and actual reactor power will increase.

B. SG NR levels will increase, RCS Tcold will increase, and actual reactor power will decrease.

C. SG NR levels will decrease, RCS Tcold will decrease, and actual reactor power will increase.

D. SG NR levels will decrease, RCS Tcold will decrease, and actual reactor power will decrease.

Proposed Answer: C

C is correct. Loss of feed preheating will add colder water to the SGs, reducing boiling, resulting in "shrink". The drop in Tcold adds positive reactivity, causing reactor power to increase

A is incorrect. SG levels would decrease. Tcold would decrease.

B is incorrect. SG levels would decrease. Tcold would decrease. Reactor power would increase.

D is incorrect. Reactor power would increase.

Technical Reference(s): OS1290.02, RESPONSE TO SECONDARY SYSTEM TRANSIENT

Proposed references to be provided to applicants during examination: None

K/A Topic: 059K3.04, Knowledge of the effect that a loss or malfunction of the MFW will have on the following: RCS

Question Source: Modified from bank. Millstone 2001. Original question attached to reference.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7/45.6

Learning Objective: Lesson L1191I, Objective L1191I08

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 46	Group #	1	
	K/A #	059A3.02	
	Importance Rating	2.9	3.1

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- All plant systems are in normal alignment.
- The Steam Generator 'A' controlling feed flow channel fails high.

Assuming no operator action, which of the following describes the response of the Feedwater Regulating Valve to this condition?

A. The valve controller is comparing the failed high feed flow channel to programmed level and creating an error signal. The Feedwater Regulating Valve will throttle closed and eventually throttle back open. Steam Generator level returns to 50%.

B. The valve controller is comparing the failed high feed flow channel to steam flow and creating a flow error signal. The Feedwater Regulating Valve will throttle closed and eventually throttles back open. The reactor eventually trips at the P-14 setpoint.

C. The valve controller is comparing the failed high feed flow channel to programmed level and creating an error signal. The Feedwater Regulating Valve will throttle closed and eventually throttle back open. The reactor eventually trips at the P-14 setpoint.

D. The valve controller is comparing the failed high feed flow channel to steam flow and creating a flow error signal. The Feedwater Regulating Valve will throttle closed and eventually throttles back open. Steam Generator level returns to 50%.

Proposed Answer: D

D is correct. A flow error signal is created. The Feedwater reg. valve controller is "level dominant" with the level error signal having an approximate 3:1 gain over the flow error signal. The flow error signal will initially cause the reg. valve to throttle close. As level drops below 50% a level erro signal will develop and cause the reg. valve to throttle open and return level towards the 50% level setpoint value.

A is incorrect but plausible. It is true that the level will go below 50% and then return towards 50%, however the feed flow is not compared to programmed level to create an error signal.

B is incorrect but plausible. A flow error signal is created. The valve does throttle in the closed direction and then back open, however, level will never reach the P-14 setpoint.

C is incorrect but plausible. It is true that the reg valve will throttle closed and then back open however the feed flow is not compared to programmed level to create an error signal. Additionally, level will never reach the P-14 setpoint.

Technical Reference(s): OS1235.03, SG LEVEL INSTRUMENT FAILURE

Westinghouse print. 1-NHY-509033, Steam Generator Level Control. Process Control Block Diagram

Proposed references to be provided to applicants during examination: None

K/A Topic: 059A3.02, Ability to monitor automatic operation of the MFW, including: Programmed levels of the S/G.

Question Source: New

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.7/45.5

Learning Objective: Lesson L8046I, Objective L8046I07

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 47	Group #	1	
	K/A #	061K4.04	
	Importance Rating	3.1	3.4

Proposed Question:
The reactor has tripped. The following conditions exist:

- RCS Tave is 557°F and STABLE
- EFW flow to SG A, B, and D is 220 gpm each, and STABLE
- EFW flow to SG C is 480 gpm and INCREASING

Assuming the current trends continue, with NO operator action, which of the following describes the expected plant response?

A. EFW flow to SG C will be limited to 525 gpm by DP across a venturi in the EFW piping.

B. SG C EFW flow control MOVs will close when flow reaches 525 gpm.

C. EFW flow to SG C will be limited to 750 gpm by DP across a flow orifice in the EFW piping.

D. EFW flow to SG C will be limited to 750 gpm by the size of the EFW piping.

Proposed Answer: B

B is correct. SG C EFW flow control MOV's will close when flow reaches 525 gpm.
A is incorrect because the venturi will limit flow to 750 gpm if there is a pipe rupture and the MOVs fail to close.
C is incorrect because the flow orifice has transmitters that send signals to the MOVs to close at 525 gpm.
D is incorrect because the piping would allow more flow if there was no venturi in the lines.

Technical Reference(s): P&ID print 1-FW-B20688, Emergency Feedwater System Details

Proposed references to be provided to applicants during examination: None

K/A Topic: 061K4.04, Knowledge of the AFW design features and/or interlocks which provide for the following: Prevention of AFW runout by limiting AFW flow.

Question Source: Direct from bank. Last used Seabrook 2003 Company exam.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.7

Learning Objective: Lesson L8045I, Objective L8045I07

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 48	Group #	1	
	K/A #	061A2.07	
	Importance Rating	3.4	3.5

Proposed Question:
A steam break in the 'C' steam generator resulted in a reactor trip and Safety Injection. The EFW system functioned normally. The control room crew eventually restored an adequate heat sink with the following alignment for the EFW flow control valves:

Steam Generator A
- CS-4214-A1 Throttled Full Closed
- CS-4214-B1 Auto-Full Open

Steam Generator B
- CS-4224-A1 Auto-Full Open
- CS-4224-B1 Throttled Full Closed

Steam Generator C
- CS-4234-A1 Auto-Full Closed
- CS-4234-B1 Auto-Full Closed

Steam Generator D
- CS-4244-A1 Throttled Full Closed
- CS-4244-B1 Auto-Full Open

Subsequently power to MCC-615 is lost

What is the effect on the control room operator's ability to control steam generator level?

A. No effect since the loss of MCC affects only one of the two flow control valves to each steam generator.

B. The operator will be unable to initiate flow to the 'A' and 'D' steam generators. Local valve operation would need to be coordinated with an NSO.

C. The operator will be unable to initiate flow to the 'B' and 'D' steam generators. Local valve operation would need to be coordinated with an NSO.

D. The operator will be unable to initiate flow to the 'B' steam generator. Local valve operation would need to be coordinated with an NSO.

Proposed Answer: D

D is correct. The 'B' steam generator throttle valve control switch CS-4224-B1 is Throttled Full Closed. This valve is powered from MCC-615. With no power the operator will not be able to move the valve from its Throttled Full Closed position.

A is incorrect. The operator would not be able to control the 'B' steam generator as described above.

B is incorrect. The operator could throttle flow to the 'A' and 'D' steam generators via their Train A throttle valves.

C is incorrect. The operator would be able to throttle flow to the 'D' steam generator with it's Train A throttle valve.

Technical Reference(s): OS1036.01, Aligning The Emergency

Feedwater System For Automatic
Initiation, Attachment A, Emergency
Feedwater System Lineup

Proposed references to be provided to applicants during examination: None

K/A Topic: 061A2.07, Ability to predict the impacts of the following malfunctions or operations on the AFW, and based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Air or MOV failure.

Question Source: Direct from bank. Seabrook 1996 NRC exam.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: 41.5/43.5/45.3/45.13

Learning Objective: Lesson L8045I, Objective L8045I01

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 49	Group #	1	
	K/A #	062A1.01	
	Importance Rating	3.4	3.8

Proposed Question:
An operator is loading the 'A' Emergency Diesel Generator per OS1026.01, Operation of DG 1A. Which of the following would result in exceeding procedural limitations?

A. The operator takes the 'Engine Speed Adjust' switch to 'Raise' and adjusts load to 5985 KW.

B. The operator takes the 'Engine Speed Adjust' switch to 'Raise' and adjusts VAR loading to greater than 2/3 of KW load.

C. The operator takes the 'Auto Voltage Adjust' switch to 'Raise' and adjusts load to 5985 KW.

D. The operator takes the 'Auto Voltage Adjust' switch to 'Raise' and adjusts VAR loading to greater than 3/4 of KW load.

Proposed Answer: D

D is correct. VAR loading is adjusted with the Auto Voltage Adjust Switch. Additionally, the procedural VAR load limit is ¾ of KW load.

A is incorrect but plausible. The Engine Speed Adjust switch is used to raise load, however, 5985 KW is below the continuous load limit of 6000 KW

B is incorrect but plausible. The Engine Speed Adjust switch is used to raise load, however, it is plausible that Load on the machine could be excessive if the Auto Voltage Adjust switch is used to excessively raise VAR loading.

C is incorrect but plausible. Going to Raise on the Auto Voltage adjust switch could cause excessive VAR laoding, However the Engine Speed adjust switch is used to raise load. Additionally, 5985 KW is below the continuous load limit of 6000 KW

Technical Reference(s): OS1026.01, Operation of DG 1A

Proposed references to be provided to applicants during examination: None

K/A Topic: 062A1.01, Ability to predict and/or monitor changes in parameters associated with operating the ac distribution controls, including:
Significance of D/G load limits

Question Source: New

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.5/45.5

Learning Objective: Lesson L8020I, Objective L8020I22

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 50	Group #	1	
	K/A #	063K3.02	
	Importance Rating	3.5	3.7

Proposed Question:
The plant is at 80% power, steady state.

The following sequence of events occur:

- t = 0, Reactor trips and SI actuates
- t = 30 seconds, Loss of offsite power
- t = 1 minute, Loss of 125 VDC bus 11A

What is the expected status of the "A" EDG and bus E5?

A. 'A' EDG running, bus E5 de-energized.

B. 'A' EDG tripped, bus E5 de-energized.

C. 'A' EDG running, bus E5 energized, some EPS associated equipment running

D. 'A' EDG running, bus E5 energized, all EPS associated equipment running

Proposed Answer: C

All control power for the A diesel is from 11A, there will be no control of the diesel available. The EPS will be in the process of completing its sequence, any started equipment should remain running (all protective trips defeated). However, any equipment not yet running will not be started.

Technical Reference(s): Bus Failure Analysis for Vital DC Power

Proposed references to be provided to applicants during examination: None

K/A Topic: 063K3.02, Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following:
Components using DC control power

Question Source: Direct from bank. Seabrook 1998.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7/45.6

Learning Objective: Lesson L1189I, Objective L1189I02
Lesson L8017I, Objective L8017I01

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 51	Group #	1	
	K/A #	064K4.02	
	Importance Rating	3.9	4.2

Proposed Question:
Under which operating condition is the high jacket water temperature trip protection for the EDG bypassed?

A. The diesel generator running unloaded after an emergency manual start from the MCB.

B. The diesel generator carrying the bus after a LOP, the EPS sequencer has completed stepping and RMO has been reset.

C. The diesel generator carrying the bus after an LOP that was followed 10 seconds later by an SI.

D. After a local engine start.

Proposed Answer: C

C is correct. The RA relay bypasses the HI temp trips when SI is actuated.
A is incorrect. The high temperature trips are active with a normal start.
B is incorrect. The high temperature trips are still active as there was no SI.
D is incorrect. The high temperature trips are active with a local start.

Technical Reference(s): OS1026.01, Operation of DG 1A

Proposed references to be provided to applicants during examination: None

K/A Topic: 064K4.02, Knowledge of the ED/G system design features and/or interlocks which provide for the following:
Trips for ED/G while operating (normal or emergency)

Question Source: Direct from Bank. Seabrook 2003 Company exam.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7

Learning Objective: Lesson L8019I, Objective L8019I16

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 52	Group #	1	
	K/A #	073K1.01	
	Importance Rating	3.6	3.9

Proposed Question:
The following sequence of events occurs:

- At Time=0: SB-CV-6519, Steam Generator Blowdown Flash Tank Discharge Control Valve closes.
- At Time=1: D7402, SG BLDWN FLASH TANK LEVEL HIGH goes into alarm.
- At Time=2: The Steam Generator Blowdown Inside Reactor Containment (IRC) Isolation Valves (SB-V-1,3,5, and 7) close.

Which of the following events would have caused the above conditions to occur?

A. Actuation of a 'T' signal.

B. Train 'B' HELB actuation.

C. EFW-P-37B, Motor Driven EFW Pump started.

D. RM6512-1, Steam Generator 'C' Blowdown Radiation Monitor in HIGH alarm.

Proposed Answer: D

A is incorrect. A 'T' signal closes the Outside Reactor Containment (ORC) Isolation Valves. A'T' signal would not cause SB-V-6519 to close.
B is incorrect. A train B HELB would close the ORC isolation valves
C is incorrect. A start of either the steam driven or motor driven EFW pump will close the ORC isolation valves.

Technical Reference(s): OS1252.01, PROCESS OR EFFLUENT HIGH RADIATION

Proposed references to be provided to applicants during examination: None

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

Question Source: Direct from bank. Seabrook 2000.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: 41.2 to 41.9/45.7 to 45.8

Learning Objective: Lesson L1187I, Objective L1187I02
Lesson L8063I, Objectives L8063I03 and L8063I11

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 53	Group #	1	
	K/A #	076K1.16	
	Importance Rating	3.6	3.8

Proposed Question:
The following plant conditions exist:
- Train 'A' Service Water is aligned to the ocean with SW-P-41A running.
- A SI/LOP occurs.

Which of the following will allow manual opening of SW-V-4, SW ISO TO SEC LOAD?

A. SI reset, RMO reset.

B. T signal reset, RMO reset.

C. SI reset, Bus E-5 UAT breaker closed.

D. T signal reset, Bus E-5 UAT breaker closed.

Proposed Answer: C

A is incorrect. The OPEN control circuit for SW-V-4 does not have an RMO reset contact.
B is incorrect. A 'T' signal is not generated by the above described plant conditions.
C is correct. An SI/LOP would cause the K-603A contact and the PR1 contact to open on the SW-V-4 OPEN control Circuit. SI reset will de-energize the K-603A contact, causing it to close. Reenergization of Bus E-5 will de-energize the PR1 contact causing it to close. This will allow for manual opening of SW-V-4.
D is incorrect. A 'T' signal is not generated by the above described plant conditions.

Technical Reference(s): Service Water System Schematic Diagram 301107, sheet DA6a.

Proposed references to be provided to applicants during examination: None

K/A Topic: 076K1.16 Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems:
ESF

Question Source: Direct from bank. TEB 18471

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.2 to 41.9/45.7 to 45.8

Learning Objective: Lesson L8037I, Objective L8037I13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 54	Group #	1	
	K/A #	078A3.01	
	Importance Rating	3.1	3.2

Proposed Question:

The plant is in MODE 1 at 100% power. The following sequence of events has occurred:

- Service Air header pressure has dropped to 84 psig.
- The crew has entered ON1242.01, LOSS OF INSTRUMENT AIR.
- An NSO found a maintenance crew using Service Air to perform sand blasting. The air hose had blown apart. The NSO isolated the leak by closing the local supply valve.
- Air pressure has recovered. The Instrument Air dryer outlet pressure is at 91 psig.

Which of the following describes the expected condition of the Service Air Isolation Valves, SA-V92 and SA-V-93?

A. SA-V92 and SA-V93 remain open. The Service Air and Instrument Air header pressures will recover on their own.

B. SA-V92 and SA-V93 automatically closed when air pressure dropped below 85 psig. The valves will automatically re-open at 90 psig. The Service Air and Instrument Air header pressure will recover on their own when the valves re-open.

C. SA-V92 and SA-V93 remain open. The operators will be directed by procedure to manually CLOSE SA-V92 and SA-V93 to allow air pressure to recover above 95 psig.

D. SA-V92 and SA-V93 automatically closed when air pressure dropped below 90 psig. The automatic closure signal resets at 93 psig. The operators will be directed by procedure to manually cycle OPEN the valves when air dryer outlet pressure is above 95 psig.

Proposed Answer: D

A is incorrect. SA-V-92 and 93 will not remain open. SA-V92 and SA-V93 close <90 psig and reset >93 psig. The reset allows manual opening via control switch.

B is incorrect because SA-V92 and V93 do not auto open when pressure is regained.

C is incorrect. SA-V-92 and 93 will not remain open. SA-V92 and SA-V93 close <90 psig and reset >93 psig. The reset allows manual opening via control switch.

Technical Reference(s): ON1242.01, LOSS OF INSTRUMENT AIR

Proposed references to be provided to applicants during examination: None

K/A Topic: 078A3.01 AAAbiliy to monitor automatic operation of the IAS, including: Air Pressure

Question Source: New

Question Cognitive Level: Analysis/Application

10 CFR Part 55 Content: CFR 41.7/45.5

Learning Objective: Lesson L1194I, Objective L1194I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question #55	Group #	1	
	K/A #	103A3.01	
	Importance Rating	3.9	4.2

Proposed Question:
The following conditions exist:

A large break LOCA has occurred with a subsequent Loss of Offsite Power.
Both vital busses are energized by the Emergency Diesel Generators.
Emergency Core Cooling System is functioning properly.
The PZR is empty and there is a steam void in the reactor vessel head.
Containment pressure is 35 psig.

What is the expected response of the containment structure cooling system to these conditions?

A. The containment structure cooling fans trip when the fan control logic receives a "P" signal.

B. The containment structure cooling fans will trip on a loss of component cooling water after a "P" signal is actuated.

C. The containment structure cooling fans are powered from a non-vital bus and are not available after a loss of offsite power.

D. All containment structure cooling fans are running as required during RCS blowdown to maintain containment pressure below design pressure.

Proposed Answer: B

B is correct. P signal is generated at 18 psig in containment. A P signal isolates both the Train "A" and Train "B" PCCW containment isolation valves. The containment structure cooling units will trip on low PCCW flow.
A is incorrect. The cooling unit controls do not receive a trip signal directly from the P signal.
C is incorrect. The structure cooling units are supplied from vital Unit Subs E53 and E63.
D is incorrect. The units trip due to loss of PCCW from the P signal.

Technical Reference(s): OS1252.03, AREA HIGH RADIATION

Proposed references to be provided to applicants during examination: None

K/A Topic: 103A3.01 Ability to monitor automatic operation of the containment system, including: Containment isolation.

Question Source: Direct from bank. Seabrook 2003

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7/45.5

Learning Objective: Lesson L8038I, Objectives L8038I04

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 56	Group #	2	
	K/A #	001K1.05	
	Importance Rating	4.5	4.4

Proposed Question:
Which of the following conditions will prevent outward control rod motion in both automatic and manual?

A. Turbine impulse channel PT-505 is reading 12% equivalent power.

B. One Intermediate Range NI channel is reading 11% equivalent current.

C. Control Bank D rods are positioned at 224 steps.

D. One Power Range NI channel is reading 104% power.

Proposed Answer: D

A is incorrect. Control permissive signal C-5 will block automatic control rod withdrawal below 15% equivalent power, however, manual rod withdrawal is still available.

B is incorrect. Control permissive signal C-1 blocks automatic and manual rod withdrawal when 1 of 2 IR channels are greater than 20% IR equivalent current (if not previously blocked by P-10).

C is incorrect. Control permissive signal C-11 will block automatic rod withdrawal when control bank D withdraws to 223 steps, however manual rod withdrawal is still available above 223 steps

D is correct. Control permissive signal C-2 will block both automatic and manual rod withdrawal when 1 of 4 Power Range NI instruments read greater than 103% power.

Technical Reference(s): PLS Book

Proposed references to be provided to applicants during examination: None

K/A Topic: 001K1.05 Knowledge of the physical connections and/or cause-effect relationships between the CRDS and the following:
NIS and RPS

Question Source: New

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR: 41.2 to 41.9 / 45.7 to 45.8

Learning Objective: Lesson Plan L8031I, Objective L8031I15

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 57	Group #	2	
	K/A #	002K1.09	
	Importance Rating	4.1	4.1

Proposed Question:
The plant has sustained a Small Break LOCA. The following conditions exist:

- PORV 456B is stuck OPEN, and has NOT been isolated
- RCS pressure is 1050 psig
- Core Exit Thermocouples are approximately 550°F
- All RCPs are TRIPPED.

Which of the following instruments will provide the most reliable indication of actual RCS inventory?

A. Pressurizer Hot-calibrated level instrument LT-459

B. Pressurizer Cold-calibrated level instrument LT-462

C. Reactor Vessel Dynamic Range DP

D. Reactor Vessel Full Range DP

Proposed Answer: D

A and B are incorrect because pressurizer level is not an accurate indication of inventory with a steam space leak in the pressurizer.
C is incorrect because it is most reliable when RCPs are running.
D is correct because it measures DP across the vessel under static conditions.

Technical Reference(s): E-1 Westinghouse background document

Proposed references to be provided to applicants during examination: None

K/A Topic: 002K1.09, Knowledge of the physical conection and/or cause and effect relationships between the RCS and the following:
PZR

Question Source: Direct from Bank. Seabrook 2000

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.2 to 41.9/45.7 to 45.8

Learning Objective: Lesson L1413I, Objective L1413I07

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 58	Group #	2	
	K/A #	011K2.02	
	Importance Rating	3.1	3.2

Proposed Question:
The plant has sustained a Loss of Off-Site power. All safeguards equipment is functioning as designed.

Assuming no operator action, which of the following equipment is running / energized?

A. Charging pump A, Emergency Feedwater pump B, Service Water pump A.

B. Charging pump B, Emergency Feedwater pump A, Pressurizer Heater Backup group B.

C. Service Water pump A, PCCW pump A, Pressurizer Control Group Heaters.

D. Charging pump A, Service Water pump A, Pressurizer Heater Backup group A.

Proposed Answer: A

A is correct because all listed loads are powered from bus E5 or E6 and would be started on LOP.
B is incorrect because PZR HTR Backup Group B is not energized until RMO is RESET.
C is incorrect because the Pressurizer Control Group Heaters are not powered from an emergency bus.
D is incorrect because the PZR HTR Backup Group A is not energized until RMO is RESET.

Technical Reference(s):	OS1246.02, DEGRADED VITAL AC POWER	Print 310102, 4160V Dist. Schematic
Print 310891, CVCS Schematic	Print 310108 Emergency Power Sequencer	Print 310882, Reactor Coolant System Schematic

Proposed references to be provided to applicants during examination: None

K/A Topic: 011K2.02, Knowledge of bus power supplies to the following: PZR Heaters

Question Source: Direct from bank. Seabrook 2000

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7

Learning Objective: L8013I, Objective L8013I15

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 59	Group #	2	
	K/A #	015K3.01	
	Importance Rating	3.9	4.3

Proposed Question:
Given the following plant conditions:

- A Plant Shutdown is in progress.
- Reactor power is 6% and decreasing.
- Intermediate range channel N-36 fails HIGH.

Which of the following statements describes how this failure affects the reactor shutdown and subsequent operation of the Nuclear Instrumentation (NI) System?

A. The reactor will not trip, and source range NIs will have to be manually re-energized.

B. The reactor will trip on high IR flux, and source range NIs will have to be manually re-energized.

C. The reactor will not trip, and source range NIs will re-energize when N-35 decreases to the proper setpoint.

D. The reactor will trip on high IR flux, and source range NIs will re-energize when N-35 decreases to the proper setpoint.

Proposed Answer: B

B is correct. Permissive signal P-10 is reset as reactor power has gone below 8%. The IR High Flux Trip is active. The IR High Flux Trip logic is 1 of 2. Additionally, the Source Range channels are auto re-energized by the rest of protective signal P-6. The reset logic is 2 of 2 IR channels < $5X10^{-11}$amps. With a failed high IR channel, the P-6 signal would not reset.
A is incorrect. The reactor will trip.
C is incorrect.The reactor will trip. The Source Range chanels will not auto re-energize.
D is incorrect. The Source Range channels wil not auto re-energize.

Technical Reference(s): PLS Document

Proposed references to be provided to applicants during examination: None

K/A Topic: 015K3.01, Knowledge of the effect that a loss or malfunction of the NIS will have on the following: RPS

Question Source: Direct from bank. Seabrook 2003 Company exam.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.7/45.6

Learning Objective: L8030I, Objective L8030I08

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 60	Group #	2	
	K/A #	017K4.02	
	Importance Rating	3.1	3.6

Proposed Question:
Which one of the following correctly lists the function of signals generated by the Core Exit Thermocouples?

A. Bulk average core temperature provides input to LTOP, Hot channel core exit temperature provides input to the Core Cooling Critical Safety Function status, Highest quadrant average temperature provides input to the subcooling margin calculation.

B. Hot channel core exit temperature provides input to the Core Cooling Critical Safety Function status, Bulk average core temperature provides density compensation input for the reactor vessel level calculation, Highest quadrant average temperature provides input to the subcooling margin calculation.

C. Bulk average core temperature provides input to LTOP, Bulk average core temperature provides input to the subcooling margin calculation, Highest quadrant average temperature provides input to the Core Cooling Critical Safety Function status.

D. Highest quadrant average temperature provides input to the subcooling margin calculation, Hot channel core exit temperature provides density compensation input for the reactor vessel level calculation, Bulk average core temperature provides input to the Core Cooling Critical Safety Function status.

Proposed Answer: B

B is correct. The Core Exit Thermocouples output provides the following functions:
- The hot channel core exit temperature provides input to the Core Cooling Critical Safety Function status.
- The bulk average core temperature provides density compensation for the reactor vessel level indication.
- The highest quadrant average temperature temperature provides input to the subcooling margin calculation.
- In-core temperature instruments do not input into LTOP. LTOP uses loop T-hot and T-cold temperatures.

A is incorrect. Bulk average temperature does not input to LTOP.
C is incorrect. Bulk average temperature does not input to LTOP. Bulk average core temperature does not provide input to the subcooling margin calculation. Highest quadrant average temperature provides does not provide input to the Core Cooling Critical Safety Function status.
D is incorrect. Hot channel core exit temperature does not provide density compensation input for the reactor vessel level calculation. Bulk average core temperature does not provide input to the Core Cooling Critical Safety Function status.

Technical Reference(s): Main Plant Computer Core Cooling Safety Function Status display. RVLIS display

Westinghouse print 1-NHY-509038, RCS Cold Overpressrization Control

Proposed references to be provided to applicants during examination: None

K/A Topic: 017K4.02, Knowledge of ITM system design features and/or interlocks which provide for the following: Sensing and determination of location core hot spots.

Question Source: New

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 41.7

Learning Objective: L8029I, Objective L8029I03

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 61	Group #	2	
	K/A #	035K5.03	
	Importance Rating	2.8	3.1

Proposed Question:
Given the following conditions:

- The main generator has just been synched to the grid and the plant is at 17% power.
- The 'B' Reactor Coolant Pump trips.

Assuming no operator actions, 'B' Steam Generator steam flow will ________ and 'B' Steam Generator level will ________. (Consider only the immediate effects)

A. Increase, increase

B. Increase, decrease

C. Decrease, increase

D. Decrease, decrease

Proposed Answer: D

Steam flow will decrease as pressure drops and level will decrease, due to colder RCS when flow reverses.
When the RCP stops the loop coolant flow drops and reverses.
The idle loop temp. approaches T-cold of the other 3 loops.
Idle loop Steam Generator steam flow will decrease and level will decrease.

Technical Reference(s): OS1201.01, RCP MALFUNCTION

Proposed references to be provided to applicants during examination: None

K/A Topic: 035K5.03, Knowledge of operational implications of the following concepts as they apply to the S/G's: Shrink and swell concepts

Question Source: Direct from bank. Millstone 1997.

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR 41.5/45.7

Learning Objective: Lesson L1181I, Objective L1181I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 62	Group #	2	
	K/A #	041K6.03	
	Importance Rating	2.7	2.9

Proposed Question:
Given the following conditions:

- The plant is in Mode 3 preparing for a plant cooldown.
- The Steam Dumps are in the Steam Pressure Mode controlling Tavg at 545°F in automatic.
- Steam Header Pressure Transmitter PT-507 has failed HIGH.

Which of the following describes the manual actions that the operator must take to regain temperature control in accordance with ON1230.01, 'STEAM HEADER PRESSURE PT-507 INSTRUMENT FAILURE'?

A. Manually close the steam dumps by switching the steam dump controller to the Tavg mode. Place the steam dump controller, MS-PK-507, to manual/minimum output. Place the Steam Dump Interlock control switches to the NA RESET NA BYPASS INTERLOCK position. Control temperature manually in the Tavg mode.

B. Manually close the steam dumps by placing either Steam Dump Interlock control switch to the OFF position. Place MS-PK-507, to manual/minimum output. Place the Steam Dump Interlock control switches to the NA RESET NA BYPASS INTERLOCK position. Manually operate the steam dumps to control temperature.

C. Manually re-open the steam dumps by placing the steam dump controller, MS-PK-507, in manual and raising the output. Continue to operate the steam dumps manually to control temperature.

D. Manually close the steam dumps by switching the steam dump controller to the Tavg mode. The steam dumps are inoperable. Manually control temperature with the Atmospheric Steam Dumps (ASDV's).

Proposed Answer: B

B is correct. This is the correct process as described in ON1230.01, STEAM HEADER PRESSURE PT-507 INSTRUMENT FAILURE when PT-507 has failed high and steam dumps are in the steam pressure mode.

A is incorrect. There is no procedural direction to swap from the Steam Pressure Mode to the Tavg Mode. Tavg was below the 550 degree bypass setpoint, so there would be no reason to have to go to bypass.

C is incorrect. The steam dumps would have gone open on a PT507 high failure. "Re-opening" the valves is incorrect.

D is incorrect. The steam dumps would still be operable with MS-PK-507 in manual.

****** This question tests the candidate's knowledge of the effect that a malfunctioning controller will have on the steam dumps. The steam dump control scheme is complex and this question tests the operators knowledge of the control scheme. The question is directly tied to the K/A .******

Technical Reference(s): ON1230.01, STEAM HEADER PRESSURE PT-507 INSTRUMENT FAILURE

Proposed references to be provided to applicants during examination: None

K/A Topic:	041K6.03, Knowledge of the effect of a loss or malfunction that the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS	
Question Source:	Modified from bank. Beaver Valley 1997.	Original question attached to reference.
Question Cognitive Level:	Analysis	
10 CFR Part 55 Content:	CFR 41.7/45.7	
Learning Objective:	L1193I, Objective L1193I08	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 63	Group #	2	
	K/A #	045A1.05	
	Importance Rating	3.8	4.1

Proposed Question:
The plant was operating at 100% power. A major electrical grid disturbance causes an automatic trip of the main turbine due to electrical fault protection. A loss of offsite power occurs. All other plant systems respond to the event as designed.

Which one of the following correctly describes how the reactor coolant temperature will respond?

A. Reactor coolant temperature will stabilize at 557° F due to response of the Main Steam Dump plant trip controller.

B. Reactor coolant temperature will stabilize at 557° F due to operation of the Atmospheric Steam Dumps.

C. Reactor coolant temperature will stabilize at 561° F due to response of the Main Steam Dump plant trip controller.

D. Reactor coolant temperature will stabilize at 561° F due to operation of the Atmospheric Steam Dumps.

Proposed Answer: D

D is correct. The Main Steam Dumps are not available due to loss of offsite power. RCS temperature will stabilize at 561° F as the ASDV's open at their 1125 psig setpoint. 561° F is associated with the saturation conditions @ 1125 psig.

A is incorrect. The Main Steam Dumps are not available due to loss of offsite power. There would be no Circ Water Pumps running.

B is incorrect. The ASDV setpoints are 1125 psig, which corresponds to a saturation temperature of 561° F.

C is incorrect. This description implies that the Main Steam Dumps are available. If they were available they would not function as described.

Technical Reference(s): ES-0.1, REACTOR TRIP RESPONSE, Step 1. OS1000.01, Heatup From Cold Shutdown To Hot Standby, Step 4.1.30.

Proposed references to be provided to applicants during examination: None

K/A Topic: 045A1.05, Ability to predict and/or monitor changes in parameters associated with operating the MT/G system controls including:
Expected response of primary system parameters following T/G trip.

Question Source: New
Question Cognitive Level: Analysis
10 CFR Part 55 Content: CFR 41.5/45.5
Learning Objective: Lesson L8041I, Objective L8041I03
Lesson L1202I, Objective L1202I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 64	Group #	2	
	K/A #	056A2.04	
	Importance Rating	2.6	2.8

Proposed Question:
Given the following conditions:

- The plant is operating at 60% power
- 'A' & 'B' Condensate Pumps are running with 'C' Condensate Pump tagged out for maintenance.
- 'B' Heater Drain Pump is running with 'A' Heater Drain Pump in Standby
- One of the running Condensate pumps trips.
- The standby Condensate pump receives an auto start signal but its breaker immediately trips.

Which of the following describes how the secondary plant will respond?

	Turbine Load	'A' Main Feed Pump	'B' Main Feed Pump
A.	A turbine runback signal will reduce turbine load to 55%.	will trip if suction pressure drops below 250 psig for greater than 6 seconds.	will trip if suction pressure drops below 250 psig for greater than 12 seconds.
B.	A turbine runback signal will reduce turbine load to 55%.	will trip if suction pressure drops below 220 psig for greater than 6 seconds	will trip if suction pressure drops below 220 psig for greater than 12 seconds
C.	A turbine setback signal will reduce turbine load to 55%.	will trip if suction pressure drops below 250 psig for greater than 6 seconds.	will trip if suction pressure drops below 250 psig for greater than 12 seconds.
D.	A turbine setback signal will reduce turbine load to 55%.	will trip if suction pressure drops below 220 psig for greater than 6 seconds	will trip if suction pressure drops below 220 psig for greater than 12 seconds

Proposed Answer: D

Overall plausibility statement: there are a number of historically challenging items being tested here. The difference between what signals initiate a 'runback' and those that initiate a 'setback' has been a source of confusion for candidates in the past. Similarly the response of the Main Feedwater pumps and their specific and unique suction pressure trips and automatic start setpoints for the CO pumps has also been an area of candidate weakness. This is also woven in with needing to understand the power supply distribution for the condensate pumps and the resultant complications for the situation given in the stem.

A is incorrect. The 2/3 Condensate Pump Trip logic actuates a turbine Setback vice a runback.
B is incorrect. The 2/3 Condensate Pump Trip logic actuates a turbine Setback vice a runback. The feedwater pump low suction trip setpoint is 220 psig. 250 psig is the standby auto start setpoint for the condensate pumps.
C is incorrect. The feedwater pump low suction trip setpoint is 220 psig. 250 psig is the standby auto start setpoint for the condensate pumps.

Technical Reference(s): OS1231.03, TURBINE RUNBACK/SETBACK

OS1290.02, RESPONSE TO SECONDARY SYSTEM TRANSIENT

ON1035.10, Main Feed Pump Standby And Sartup Operation

Proposed references to be provided to applicants during examination: None

K/A Topic:	056A2.04, Ability to predict the impacts of the following malfunctions or operations on the Condensate System: Loss of Condensate pumps
Question Source:	New
Question Cognitive Level:	Analysis
10 CFR Part 55 Content:	CFR 41.5/ 43.5/ 45.3/ 45.13
Learning Objective:	Lesson L1191I, Objective L1191I08

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
Question # 65	Group #	2	
	K/A #	G 2.1.32 Ability to explain and apply all system limits and precautions.	
	Importance Rating	3.4	3.8

Proposed Question:
The plant is operating at 100% power with all three Circulating Water Pumps in service. Due to bearing problems CW-P-39A must be removed from service.

Which of the following describes a precaution that applies to two pump operation at power?

A. Environmental Compliance should be notified upon entry into or exit from two CW pump operation at power. Environmental Compliance tracks CWΔT during two pump operation to ensure the 15 day NPDES administrative limit is not exceeded.

B. Environmental Compliance should be notified if two pump operation results in the daily CWΔT value exceeding 47°F. Environmental Compliance is responsible for tracking CWΔT during two pump operation if the daily CWΔT value exceeds 47°F.

C. Operation of two pumps with all three water boxes in service should be minimized to prevent pump damage due to pump runout. Two pump operation may require throttling of one or more water box discharge valves.

D. Prior to going from three circulating water pumps to two circulating water pumps, reduce turbine load to <75%. Two pump operation at greater than 75% power may result in condenser tubes overheating.

Proposed Answer: A

A is correct. This precaution is listed as PRECAUTION 3.16 of procedure ON1038.02, Circulating Water System Pump Shutdown. Per the NAEC, Environmental Compliance Manual, an average monthly ΔT limit of 45°F and a maximum daily ΔT limit of 47°F are allowed up to a maximum of 15 days per year and only when one circulating water pump has been taken out of service for corrective or preventative maintenance.

B is incorrect. Environmental Compliance should be notified prior to securing the circulating water pump. The 47°F value is the 15 day administrative limit value for operating with one pump removed from service for maintenance.

C is incorrect. Pump runout concern applies to one pump operation against 3 waterboxes. Two pump operation at power is allowed and proceduralized with no concern for pump runout.

D is incorrect. Two pump operation supports turbine operation.

Technical Reference(s):	PRECAUTION 3.16 of procedure ON1038.02, Circulating Water System Pump Shutdown.	NAEC, Environmental Compliance Manual.

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.32 Ability to explain and apply all system limits and precautions.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 41.10 / 43.2 / 45.12

Learning Objective: Lesson L8053I, Objective L8053I03 and L8053I12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 66	Group #	2.1	
	K/A #	2.1.1 Knowledge of conduct of operations requirements.	
	Importance Rating	3.7	3.8

Proposed Question:
In accordance with OP 9.2, Emergency Operating Procedure User's Guide, which one of the following is listed as a Skill of the Operator Task?

A. Performing a rapid boration.

B. Manual ESF actuation when automatic actuation setpoint is or will be exceeded.

C. Re-opening letdown isolation valves in response to an instrument failure.

D. Filling an RCP standpipe.

Proposed Answer: B

Answer B is the only choice specifically listed as "skill of the operator" in OP 9.2, Emergency Operating Procedure User's Guide.

A is incorrect. Rapid boration is performed per procedural guidance and is not listed as a Skill of the Operator Task in OP9.2
C is incorrect. Re-opening letdown isolation valves is done per procedural guidance in the applicable instrument failure abnormal procedure.
D is incorrect. Filling RCP standpipes is performed per procedural guidance and is not listed as a Skill of the Operator Task in OP9.2

NOTE: Skill of the operator tasks are specifically listed in OP 9.2, Emergency Operating Procedure User's Guide. These tasks are not exclusive to Emergency Operating Procedure's. The following information is from OP 9.2, section 4.7.2:

Skill of the operator task is permitted when all of the following conditions are met:

- **A system or component is deviating from it's design function state or it is anticipated that the system or component will deviate from it's design function state.**
- **The task is simple and is considered a routine activity based on operational experience or training.**
- **Written instruction does not exist or written instruction is not immediately available.**
- **The operator obtains concurrence from his/her supervisor prior to performing the task.**
- **As time allows, any written instruction should be used as follow up to verify task completion.**

The following is a complete list of Skill of The Operator Tasks:

- **Manual ESF actuation when automatic actuation setpoint is or will be exceeded.**
- **Manual reactor trip actuation when automatic actuation setpoint is or will be exceeded.**
- **Manual control of any component using manual/auto controller station.**
- **Manual closure of failed open PZR PORV, or manua closure of the block valve to isolate a failed open PORV.**
- **Manual closure of a failed open PZR spray valve.**
- **Placing rod control in manual or automatic.**
- **Adjusting charging and letdown flows.**
- **Closing CS-V-145 to isolate letdown flow path.**
- **Manual control of steam dump valves, including use of P-12 switches.**
- **Resetting remote manual override (RMO).**

- **Stopping a pump that is not pumping.**
- **Starting a pump that has failed to start automatically.**
- **Placing a diesel generator in maintenance to prevent restart when engine support conditions are degraded.**

Technical Reference(s): OP9.2, Emergency Operating Procedure User's Guide

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.1 Knowledge of conduct of operations requirements.

Question Source: Modified from bank. Original question attached to reference.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR: 41.10 / 45.13

Learning Objective: Lesson L1505I, Objective L1505I23

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 67	Group #	2.1	
	K/A #	2.1.11 Knowledge of less than one hour technical specification action statements for systems.	
	Importance Rating	3.0	3.8

Proposed Question:
Given the following conditions:
- The plant is in Mode 1 at 100% power.
- The Shift Manager declares the 'B' Emergency Diesel Generator INOPERABLE.

Per Technical Specification 3.8.1.1, AC Sources, which of the following describes an action that must take place within 1 hour?

A. Demonstrate OPERABILITY of the 'A' Emergency Diesel Generator.

B. Verify that the steam driven EFW pump is operable.

C. Demonstrate OPERABILITY of the remaining A.C. sources.

D. Restore OPERABILITY of the 'B' Emergency Diesel Generator.

Proposed Answer: C
C is correct per Tech. Spec. 3.8.1.1b.1.
A is incorrect but plausible. The tech spec action requires demonstrating operability of the remaining diesel, however it must be done within 24 hours.
B is incorrect but plausible. The tech spec requires demonstration of operability of the steam driven EFW pump, however it is required within 4 hours.
D is incorrect but plausible. The Tech. Spec. requires restoration of both diesel generators within 72 hours. Additionally, an operable SEPS diesel allows this time to be further extended.
Technical Reference(s): Tech. Spec. 3.8.1.1b.1.

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.11 Knowledge of less than one hour technical specification action statements for systems.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 43.2 / 45.13

Learning Objective: Lesson L8020I, Objective L8020I27

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 68	Group #	2.1	
	K/A #	2.1.28 Knowledge of the purpose and function of major system components and controls.	
	Importance Rating	3.2	3.3

Proposed Question:
The following conditions exist:

- The plant is at 2% power with a plant startup underway.
- The controlling PZR Pressure Instrument PT-455 fails high.

Assuming no operator actions, which of the following will occur?

A. OTΔT Reactor Trip

B. OPΔT Reactor Trip

C. Low PZR Pressure Reactor Trip

D. Low Pressure SI

Proposed Answer: D

D is correct. The failed high pressure channel will actuate pressurizer sprays, inhibit heaters, and cause pressurizer pressure to drop to the low pressurizer SI setpoint.

A is incorrect but plausible. Pressure does input to the OTΔT Reactor Trip, but is only a significant trip concern at higher power levels.

B is incorrect but plausible. It is a common student misconception that pressure inputs to the OPΔT Reactor Trip.

C is incorrect but plausible. PT-455 does feed into the Low PZR Pressure Reactor Trip, however this trip is blocked below the P-7 interlock point (10% power).

****** The question tests the candidate's knowledge of the function of the pressurizer pressure control system. The student must understand that the failed high channel will cause pressurizer sprays to actuate and drive pressurizer pressure down. The question also tests the candidate's knowledge of the reactor protection systems components and interlocks as they must know that the OTΔT trip would be challenged at high power levels, the OPΔT trip is not affected by pressure, and that the Low PZR Reactor Trip is not active below the P-7 interlock.******

Technical Reference(s): Westinghouse Functional Prints- Reactor Protection System

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.28 Knowledge of the purpose and function of major system components and controls.

Question Source: New

Question Cognitive Level: Comprehension

10 CFR Part 55 Content: CFR: 41.7

Learning Objective: Lesson L8021I, Objective 8021I29

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 69	Group #	2.1	
	K/A #	2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.	
	Importance Rating	3.4	4.0

Proposed Question:
According to Seabrook Technical Specifications which of the following conditions is considered a Loss of Containment Integrity?

A. During movement of fuel within containment, the equipment hatch door is closed and held in place by six bolts.

B. Both containment airlock doors are closed but do not meet containment leakage rate testing program requirements in Mode 1.

C. A locked closed containment isolation valve is opened on an intermittent basis under administrative control in Mode 1.

D. An eight inch containment purge supply and exhaust isolation valve is open to facilitate personnel entry into containment in Mode 1.

Proposed Answer: B

B is correct. Per Tech. Spec. 3.6.1.1, Containment Integrity, the airlock doors must be in compliance with Tech. Spec. 3.6.1.3, Containment Air Locks. 3.6.1.3 states that each containment air lock shall be demonstrated OPERABLE, with the leakage rate in accordance with the Containment Leakage Rate Testing Program.
A is incorrect. During fuel movement the equipment hatch must be held in place by at least 4 bolts.
C is incorrect. The containment tech. Spec allows locked closed containment isolation valves to be opened on an interim basis under administrative control.

D is incorrect. Tech. Spec. 3.6.1.7, Containment Ventilation System, allows for the opening of the purge valves for control of pressure, ALARA, respirable, and air quality considerations.

Technical Reference(s): Tech. Spec. 3.6.1.1, Containment Integrity; Tech. Spec. 3.6.1.3, Containment Air Locks

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.

Question Source: Direct from bank. Seabrook 2003 company exam.

Question Cognitive Level: Memory

10 CFR Part 55 Content: CFR: 43.2 / 43.3 / 45.3

Learning Objective: Lesson L8026I, Objective L8026I12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 70	Group #	2.2	
	K/A #	2.2.13 Knowledge of tagging and clearance procedures.	
	Importance Rating	3.6	3.8

Proposed Question:
Which of the following lists the MINIMUM temperature and pressure for fluid or gas systems that require double valve isolation when preparing a tagging clearance?

A. 150 ° F, 200 psig.

B. 200 ° F, 500 psig.

C. 150 ° F, 500 psig.

D. 200 ° F, 200 psig.

Proposed Answer: B

B is correct. Per MA 4.2, Equipment Tagging and Isolation, Section 4.3, Clearance Section Preparation and Review, 4.3.1, Specific Requirements, item 6, "When breaching a fluid or gas system that operates with temperatures greater than 200 ° F or pressures greater than 500 psig, the worker shall have :
- Two closed valves in series, and
- An open and tagged telltale vent or drain, between the two valves.

A is incorrect. The requirement is 200 ° F and 500 psig.
C is incorrect. The requirement is 200 ° F.
D is incorrect. The requirement is 500 psig.

Technical Reference(s): MA 4.2, Equipment Tagging and Isolation, Section 4.3, Clearance Section Preparation and Review, 4.3.1, Specific Requirements

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.2.13 Knowledge of tagging and clearance procedures.

Question Source: New
Question Cognitive Level: Memory
10 CFR Part 55 Content: CFR: 41.10 / 45.13
Learning Objective: L1514I

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 71	Group #	2.2	
	K/A #	2.2.22 Knowledge of limiting conditions for operations and safety limits.	
	Importance Rating	3.4	4.1

Proposed Question:
The following plant conditions exist:

- The plant is in MODE 3 following a reactor trip.
- A cooldown of the RCS is in progress.
- RCS temperature is at 380° F.

Which of the following RCS overpressure systems is required to be OPERABLE to satisfy Technical Specifications?

A. Both pressurizer PORV's.

B. Both RHR suction reliefs.

C. One pressurizer PORV and the RCS heat vent.

D. One pressurizer PORV and one RHR suction relief.

Proposed Answer: A

A is correct. Per Tech. Spec. 3.4.4, RCS Relief Valves, both PORV's are required to be operable.

B is incorrect but plausible. Both RHR suction reliefs is a condition for satisfying Tech. Specs., however it is applicable to MODE 4 with RCS temp < 290°F.

C is incorrect but plausible. The PORV's do meet the Tech. Spec. requirement, however both PORV's must be operable. Additionally, the RCS heat vent is used for overpressure protection, however it is only Tech. Spec. applicable in MODE's 5 and 6.

D is incorrect but plausible. The PORV's do meet the Tech. Spec. requirement, however both PORV's must be operable. Additionally, the combination of 1 PORV and 1 RHR suction relief is a condition for meeting Tech. Specs., however it applies in MODE 4.

Technical Reference(s): Tech. Spec. 3.4.4, RCS Relief Valves; Tech.Spec 3.4.9.3, Overpressure Protection Systems.

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.2.22 Knowledge of limiting conditions for operations and safety limits.

Question Source: Direct from bank.

Question Cognitive Level: Applicability

10 CFR Part 55 Content: CFR: 43.2 / 45.2

Learning Objective: Lesson L8022I, Objective L802213

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 72	Group #	2.3	
	K/A #	2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements.	
	Importance Rating	2.6	3.0

Proposed Question:
In accordance with the Seabrook Radiation Protection Manual, which of the following would require a Specific Radiation Work Permit, and how long would it remain in effect?

A. Entry under the reactor vessel, for the duration of the job.

B. Containment Entry, for 1 year.

C. Primary chemistry sampling, for the duration of the job.

D. Operations rounds, 1 year.

Proposed Answer: A

A is correct. In accordance with the Seabrook Radiation Protection Manual, Entry under the vessel is listed as one of the items requiring a Specific Radiation Work Permit. Specific Radiation Work Permits remain in effect for the duration of the job.

B is incorrect but plausible. Containment entry could be construed as requiring a specific RWP as the task is somewhat non-routine, however it is not specifically listed as requiring a Specific Radiation Work Permit.

C is incorrect but plausible. Primary chemistry sampling could be construed as requiring a Specific Radiation Work Permit as primary fluid and gas systems could be breached, however it is not specifically listed as requiring a Specific Radiation Work Permit.

D is incorrect but plausible. The listed time that the Operator Rounds RWP is in effect is correct (1 year), however this is considered a Routine RWP.

Technical Reference(s): SSRP-Radiation Protection Manual

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements.

Question Source: Modified from bank — Original question attached to reference.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR: 41.12 / 43.4.
45.9 / 45.10

Learning Objective: Lesson L1525I, Objective L1525I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 73	Group #	2.3	
	K/A #	2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	
	Importance Rating	2.5	3.1

Proposed Question:
The following radiological conditions exist in the 'A' Charging Pump room:

- General dose rates range from 25-45 mrem/hr.
- Measurements taken on pipes and valves include:

  * Point 1 is 100 mrem/hr at 30 cm.
  * Point 2 is 500 mrem/hr at 30 cm.
  * Point 3 is 1100 mrem/hr at 30 cm.

- The room is accessible to plant personnel.
- A worker has obtained the required authorization to enter the room, but needs an exposure limit upgrade to 2000 mrem.

Based on these conditions what is the radiological posting required for this room and who can authorize an exposure limit upgrade?

A. High Radiation Area, HP Department Manager

B. Technical Specification Locked High Radiation Area, HP Department Manager

C. High Radiation Area, Health Physics Supervisor

D. Technical Specification Locked High Radiation Area, Health Physics Supervisor

Proposed Answer: D

D is correct. A Technical Specification Locked High Radiation Area is any area accessible to individuals in which radiation levels could result in an individual receiving a dose equivalent >1000 mrem DDE in one hour at 30 centimenters from the radiation source. Additionally, dose extention from 1000-3000 mrem requires authorization of the HP Supervisor.

A is incorrect but plausible. The student must know that the Point 3 dose rate meets the criteria for defining this area as a Technical Specification Locked High Radiation Area. It is plausible for the student to choose the area as a High Radiation Area as that area is defined as having rad levels such that an individual could receive a dose in excess of 100 mrem in one hour. The student may identify that the three specific rad point readings qualify as a High Rad area. The student must understand the threshold for a Technical Specification Locked High Radiation Area. Additionally, it is plausible that an extension of an individuals dose limit could require the authorization of the HP Department Manager, however, this would only be the case if the individual was being extended above 3000 mr and up to the federal limit of 5000 mrem.

B is incorrect but plausible. The area is a Technical Specification Locked High Radiation Area, however, an extension of an individuals dose limit up to 3000 mrem requires the authorization of the HP Department Superisor. It is plausible that an extension of an individuals dose limit could require the authorization of the HP Department Manager, however, this would only be the case if the individual was being extended above 3000 mr and up to the federal limit of 5000 mrem.

C is incorrect but plausible. Dose extention from 1000-3000 mrem do require authorization of the HP Supervisor, however, the area should be defined as a Technical Specification Locked High Radiation Area..The student must know that the Point 3 dose rate meets the criteria for defining this area as a Technical Specification Locked High Radiation Area. It is plausible for the student to choose the area as a High Radiation Area as that area is defined as having rad levels such that an individual could receive a dose in excess of 100 mrem in one hour. The student may identify that the three specific rad point readings qualify as a High Rad area.

Technical Reference(s): SSRP-Chapter 1, Section 3.4.6.7. SSRP, Procedure RP-5.1, Figure 5.3.

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 43.4 / 45.10

Learning Objective: L1525I, Objective L1525I13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 74	Group #	2.4	
	K/A #	2.4.27 Knowledge of fire in the plant procedure.	
	Importance Rating	3.0	

Proposed Question:
As a result of a fire in the Train A Essential Switchgear Room, the crew is required to place the DISABLE switch of the "A" PORV, RC-PCV-456A, in DISABLE.

How, if at all, can the PORV be operated?

A. The PORV cannot be operated, all operation is defeated.

B. The PORV automatic operation is defeated, the valve can be opened by the operator.

C. The PORV will open automatically if RCS pressure reaches the PORV setpoint and the LOCAL/REMOTE selector switch is placed in LOCAL.

D. The PORV can be opened by the operator from the RSS panel if the LOCAL/REMOTE selector switch is placed in LOCAL.

Proposed Answer: A

For the PORV to open at all, the DISABLE switch must be in NORMAL. If in DISABLE no automatic or manual operation is possible.

Technical Reference(s): Logic print 503746 OS1200.00A — OS1200.00A, Response To Fire Or Fire Alarm Actuation, Step 5.

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.4.27 Knowledge of fire in the plant procedure.

Question Source: Direct from bank. Last used Seabrook 1998 Company exam.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR: 41.10 / 43.5 / 45.13

Learning Objective: L8210I, Objective L8210I03

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
Question # 75	Group #	2.4	
	K/A #	2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	
	Importance Rating	4.0	

Proposed Question:
Given the following conditions:

- The plant is being shut down for a refueling outage.
- Reactor power is at 20% and decreasing.
- The main turbine trips.
- All plant systems respond as expected.
- No operator actions have been taken.

Based on the above conditions what action should the crew perform next:

A. Enter E-0, REACTOR TRIP OR SAFETY INJECTION and verify that the reactor has tripped.

B. Enter ON1231.02, TURBINE TRIP BELOW P-9.

C. Shutdown the reactor per Main Plant Evolution procedural guidance.

D. Verify control rods are inserting in AUTO at maximum rate per FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.

Proposed Answer: A

A is correct. The plant is above the P-9, Turbine Trip-Reactor Trip permissive (setpoint 20% NI power). The reactor should automatically trip. The crew should be entering procedure E-0. The first step of E-0 is to verify that the reactor has tripped.

B is incorrect. Reactor power is at 20% and decreasing. The reset for P-9 is 18%. There should still be a valid reactor trip signal from the turbine trip.

C is incorrect. Reactor power is at 20% and decreasing. The reset for P-9 is 18%.

D is incorrect. This is an FR-S.1 action. FR-S.1 is not entered until the RNO action for reactor trip verification (manually trip the reactor) is not successful in E-0, REACTOR TRIP OR SAFETY INJECTION.

Technical Reference(s): OP 9.2, Emergency Operator Procedure User's Guide; E-0, REACTOR TRIP OR SAFETY INJECTION; FR-S.1, ATWS

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Question Source: New

Question Cognitive Level: Analysis/Application

10 CFR Part 55 Content: CFR: 41.10 / 43.2 / 45.6

Learning Objective: Lesson L1505I, Objectives L1505I07 and L1505I23

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 76	Group #		1
	K/A #	007EA2.04	
	Importance Rating		4.6

Proposed Question:
Given the following conditions:

- The plant is initially at 100% power.
- A valid reactor trip signal occurred and the reactor did not trip.
- Attempts to trip the reactor from the control room were unsuccessful.

Which of the following describes the actions that the crew should take next?

A. Perform the immediate actions of FR-S.1, 'RESPONSE TO NUCLEAR POWER GENERATION/ATWS', Dispatch and NSO to locally trip the reactor. When the NSO reports that the reactor has been locally tripped then transition to procedure and step in effect.

B. Perform the immediate actions of FR-S.1, 'RESPONSE TO NUCLEAR POWER GENERATION/ATWS', Dispatch an NSO to locally trip the reactor. When the NSO reports that the reactor has been locally tripped continue in FR-S.1 until the reactor is verified subcritical, then transition to procedure and step in effect.

C. Perform the immediate actions of FR-S.1, 'RESPONSE TO NUCLEAR POWER GENERATION/ATWS', Dispatch an NSO to locally trip the reactor. When the NSO reports that the reactor has been locally tripped remain in FR-S.1 until the procedure is completed.

D. Perform the immediate actions of FR-S.1, 'RESPONSE TO NUCLEAR POWER GENERATION/ATWS', Dispatch and NSO to locally trip the reactor. When the NSO reports that the reactor has been locally tripped remain in FR-S.1 until the procedure is completed, unless a higher priority Critical Safety Function Red Path condition occurs.

Proposed Answer: B

B is correct. FR-S.1, Step 7 verifies that the reactor is subcritical. If the reactor is subcritical then step 7 dictates returning to procedure and step in effect. Step 7 is a continuous action step which applies throughout the remaineder of FR-S.1.

A is incorrect but plausible. If the reactor is successfully tripped locally the procedural direction will be to transition to procedure and step in effect, however this is done by executing FR-S.1, Step 7 to verify the reactor subcritical.

C is incorrect but plausible. FR-S.1 is a Red Path high priority Functional Restoration Procedure. The student may assume that the procedure should be completed if they are not aware of the contents and "continuous action step" purpose of FR-S.1, Step 7.

D is incorrect but plausible. FR-S.1 is a Red Path high priority Functional Restoration Procedure. The student may assume that the procedure should be completed if they are not aware of the contents and "continuous action step" purpose of FR-S.1, Step 7. The student should be aware that subcriticality is the highest priority critical safety function.

Technical Reference(s): FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS

Proposed references to be provided to applicants during examination: None

K/A Topic:	07EA2.04, Ability to determine or interpret the following as they apply to a reactor trip: If reactor should be tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP.
Question Source:	New
Question Cognitive Level:	Application
10 CFR Part 55 Content:	CFR 41.7/45.5/45.6
Learning Objective:	Lesson L1202I, Objective L1202I04

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 77	Group #		1
	K/A #	011EA2.10	
	Importance Rating		4.7

Proposed Question:
Given the following conditions:

- A large-break LOCA has occurred.
- The crew responded in accordance with E-0, REACTOR TRIP OR SAFETY INJECTION.
- No RNO actions were required.
- The crew has just transitioned to E-1, "Loss of Reactor or Secondary Coolant," from E-0, step 11, due to abnormal containment pressure.
- Containment pressure has reached 15 psig.
- RCS average temperature is 420°F.
- RCS pressure is stable at 200 psig.

Which of the following statements describes the primary method of decay heat removal?

A. Heat transfer between the RCS and the S/Gs due to reflux cooling.

B. Heat transfer between the RCS and the S/Gs due to RHR forced circulation flow.

C. The injection of water from the containment sump and the removal of steam/water out the break.

D. The injection of water from the RWST and the removal of steam/water out the break.

Proposed Answer: D

D is correct. Injection of water from the RWST and the removal of steam/water out the break.
A is incorrect but plausible. Reflux cooling is a mechanism for cooling during a large break LOCA but it is not the primary means.
B is incorrect. For smaller break sizes forced circulation is a viable method of heat removal during the initial phase of the transient.
C is incorrect. At the point of entering E-1, ECCS would still be in the injection mode and not yet swapped to sump recirc mode.

Technical Reference(s): Westinghouse background document, E-1.

Proposed references to be provided to applicants during examination: None

K/A Topic: 011EA2.10, Ability to determine or interpret the following as they apply to a Large Break LOCA: Verification of adequate core cooling.

Question Source: Direct from bank. North Anna 1998, question reworded to be Seabrook applicable.

Question Cognitive Level: Comprehension

10 CFR Part 55 Content: CFR 43.5/45.13

Learning Objective: Lesson L1413I, Objective L1413I03

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 78	Group #		1
	K/A #	G 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	
	Importance Rating		4.4

Proposed Question:
The plant is at 52% power. The crew is in the process of raising power at 5% per hour. The following events occur:

- At 1023 alarm D5775, RCP A SHAFT VIBRATION HIGH is received. The RO observes on color graphics that shaft vibration is at 15 mils and increasing.
- The crew enters OS1201.01, RCP MALFUNCTION and determines that the alarm is valid.
- At 1029 the RO observes from color graphics that shaft vibration is at 18 mils and increasing at greater than 10 mils per hour.

Based on these conditions, which one of the following actions should be taken.

A. Trip the reactor, enter E-0, REACTOR TRIP OR SAFETY INJECTION, stop the A RCP after the E-0 immediate actions are complete.

B. Raise the 'A' steam generator level to 60-70%, stop the A RCP, shutdown the plant to Mode 3 within 6 hours.

C. Continue to monitor pump vibration levels. If shaft vibration increases to greater then 20 mils then trip the reactor, enter E-0, REACTOR TRIP OR SAFETY INJECTION, stop the 'A' RCP after the E-0 immediate actions are complete.

D. Continue to monitor pump vibration levels. If shaft vibrations increase to greater than 20 mils then raise the 'A' steam generator level to 60-70%, stop the 'A' RCP, shutdown the plant to Mode 3 within 6 hours.

Proposed Answer: A

A is correct. Per OS1201.01, RCP MALFUNCTION, step 2 RNO, if shaft vibrations are greater than 15 mils and increasing at greater than 1 mil per hour then the reactor coolant pump should be removed from service. The procedure dictates a reactor trip of plant power is greater than P-8 (50% power-reset 48%)

All distractors (selections B, C, & D) are plausible as they require application of knowledge of the procedure requirements **and** the understanding of permissive relay and when they are active. This is a common area of confusion for students.

B is incorrect. Plant power is greater than P-8. The procedure dictates tripping the reactor. This is plausible as it would be the action taken if power were below P-8. This is a common area of misconception for students (P-8 @ 50%, reset @ 48%).

C is incorrect. Shaft vibration level is greater than 15 mils and increasing at greater than 1 mil per hour. The procedure dictates tripping the reactor and removing the RCP from service under these conditions.

D is incorrect. Shaft vibration level is greater than 15 mils and increasing at greater than 1 mil per hour. The procedure dictates tripping the reactor and removing the RCP from service under these conditions.

Technical Reference(s): OS1201.01, RCP MALFUNCTION

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Question Source: New

Question Cognitive Level: Analysis/Application

10 CFR Part 55 Content: CFR: 43.5 / 45.12 / 45.13

Learning Objective: Lesson L1181I, Objectives L1181I03 and L1181I04

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 79	Group #		1
	K/A #	G 2.4.4, Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	
	Importance Rating		4.3

Proposed Question:

The following plant conditions exist:

- The plant is operating at 40% power.
- Pressurizer Spray Valve RC-PCV-455A is stuck OPEN.
- The crew entered OS1201.06, PZR PRESSURE INSTRUMENT/COMPONENT FAILURE.
- All efforts to close the spray valve have failed.
- Pressurizer pressure is at 2000 psig and decreasing steadily.

Which of the following actions should be taken?

A. Trip the 'C' RCP and then trip the Reactor.

B. Feed the 'C' Steam Generator to 60-70% and then trip the 'C' RCP.

C. Trip the Reactor and then trip the 'A' Reactor Coolant Pump.

D. Trip the Reactor and then trip the 'C' Reactor Coolant Pump.

Proposed Answer: D

D is correct. Per OS1201.06, PZR PRESSURE INSTRUMENT/COMPONENT FAILURE step 2 RNO, if a spray valve is failed open and pressurizer pressure cannot be maintained greater than 1945 then the reactor is tripped and the associated spray valve RCP is tripped following E-0 immediate actions. The 'C' RCP feeds RC-PCV-455A.

A is incorrect. The reactor should be tripped prior to tripping the RCP .

B is incorrect. Feeding the steam generators to 60-70% is a method for tripping RCP's below P-8 during an RCP malfunction event.

C is incorrect. The 'B' RCP does not supply either pressurizer normal spray valve.

Technical Reference(s): OS1201.06, PZR PRESSURE INSTRUMENT/COMPONENT FAILURE.

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.4.4, Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Question Source: Modified from bank. Original question attached to reference.

Question Cognitive Level: Analysis/Application

10 CFR Part 55 Content:	CFR 41.10 / 43.2 / 45.6
Learning Objective:	Lesson L1182I, Objective L1182I05 Lesson L1202I, Objective L1202I01

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 80	Group #		1
	K/A #	038EA2.15	
	Importance Rating		4.2

Proposed Question:
A cooldown is being conducted in accordance with ES-3.1, POST-SGTR COOLDOWN USING BACKFILL. The depressurization of the RCS may be stopped when RCS pressure decreases to less than 360 psig.

Which of the following is the reason for stopping depressurization at this pressure?

A. The vessel must be soaked for 1 hour to remove thermal stress from the cooldown.

B. To prevent the SI Accumulators from degrading ECCS cooling capability.

C. To maintain adequate RCP number 1 seal differential pressure.

D. To stop backfill from the ruptured steam generator prior to establishing cooling with RHR.

Proposed Answer: C

C is correct. Per Westinghouse Background Document, ES-3.1, POST-SGTR COOLDOWN USING BACKFILL.
A is incorrect. There are no soak time holds in ES-3.1. There is a 100 degree per hour cooldown limit.
B is incorrect. Degradation of ECCS cooling capability is not addressed by the 360 psig criteria. This criteria is specific to protection of the RCP #1 seal.
D is incorrect. There is no step to stop backfill prior to establishing cooling with RHR. There is direction to maintain pressurizer level and subcooling. The depressurization rate is controlled to maintain pressurizer level. Depressurization is stopped at less than 360 psig and less than 350 degrees F prior to placing RHR in service.

Technical Reference(s): Westinghouse Background Document, ES-3.1, POST-SGTR COOLDOWN USING BACKFILL.

Proposed references to be provided to applicants during examination: None

K/A 038EA2.15, Ability to determine and interpret the following as they apply to a SGTR:
Topic: Pressure at which to maintain RCS during S/G cooldown.

Question Source: Modified from bank. Original question attached to reference.

Question Cognitive Level: Comprehension

10 CFR Part 55 Content: CFR 43.5/45.13

Learning Objective: Lesson L1206I, Objective L1206I17

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 81	Group #		1
	K/A #	055EA2.03	
	Importance Rating		4.7

Proposed Question:
The crew is implementing ECA-0.0, "Loss of All AC Power".

The dispatcher reports he is ready to restore offsite power to the site. The crew is attempting to restore power to bus E-5 in accordance with step 8, "Coordinate Effort to Repower Emergency Busses".

What actions will the crew have to take to restore power to bus E-5 from the UAT?

A. Place E5 Synchronizing Switch in "UAT" position, reset RMO, close the UAT breaker.

B. Place E5 Synchronizing Switch in "UAT" position, place RMO bypass switch in "Bypass", close the UAT breaker.

C. Reset RMO, close the UAT breaker.

D. Place RMO bypass switch in "Bypass", close the UAT breaker.

Proposed Answer: D

D is correct. ECA-0.0, Attachment 'G', step 4 instructs the operator to bypass RMO and close the UAT or RAT breaker.
A and B are incorrect. There is no need for sync check because the bus is dead.
C incorrect, RMO cannot be reset.
*********This is a station priority item as the RMO Bypass switch operation is a common conceptual challenge among Seabrook Station SRO's. Usage of RMO Bypass is not consistently delineated in the procedures. Seabrook Station is a 4 hour coping plant. This is a high priority knowledge task for SRO's.**********

Technical Reference(s): ECA-0.0 instructs the operator to bypass RMO and close the breaker.

Proposed references to be provided to applicants during examination: None

K/A Topic: 055EA2.03, Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

Question Source: Direct from bank. Seabrook 1998.

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR 43.5/45.13

Learning Objective: Lesson L8067I, Objective L8067I03RO
Lesson L8020I, Objective L8020I29RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 82	Group #		2
	K/A #	001AA2.05	
	Importance Rating		4.6

Proposed Question:
A plant startup is in progress. The following conditions exist:

- Reactor power is approximately 15%, with control rods in manual.
- The Main Turbine has been phased on to the grid.
- Steam Dumps are in the Steam Pressure Mode.
- The crew is in the process of transferring steam load off of the steam dumps.

Which of the following conditions would require the crew to trip the reactor?

A. Loss of a 120 VAC vital instrument panel.

B. One control rod drops to the bottom of the core with stable plant conditions.

C. As the BOP operator raises turbine load the control rods begin to slowly withdraw.

D. The turbine trips.

Proposed Answer: C

A is incorrect. Loss of a 120 VAC vital instrument panel should not result in a condition requiring a reactor trip.
B is incorrect. OS1210.05, DROPPED ROD, note prior to step 1 defines a dropped rod as a rod that has dropped to the bottom of the core. A reactor trip would be warranted for more than one dropped rod.
C is correct, a reactor trip is required. Per OS12101.04, CONTINUOUS CONTROL ROD WITHDRAWAL, step 1, if control rods are in manual and rods withdraw, then the RNO requires a reactor trip.
D incorrect, turbine trip below P-9 does not require a reactor trip. The crew would utilize procedure OS1231.02, TURBINE TRIP BELOW P-9.

Technical Reference(s): OS12101.04, CONTINUOUS CONTROL ROD WITHDRAWAL; OS1210.05, DROPPED ROD; OS1231.02, TURBINE TRIP BELOW P-9.

Proposed references to be provided to applicants during examination: None

K/A Topic: 001AA2.05, Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal:
Uncontrolled rod withdrawal, from available indications.

Question Source: New

Question Cognitive Level: Analysis/Application

10 CFR Part 55 Content: CFR 43.5/45.13

Learning Objective: Lesson L1184I, Objective L1184I12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 83	Group #		2
	K/A #	G 2.2.27, Knowledge of the refueling process.	
	Importance Rating		3.5

Proposed Question:
The crew is in the process of responding to a loss of refueling cavity level per OS1215.05, LOSS OF REFUELING CAVITY WATER. The refueling machine is holding a fuel assembly at the transfer canal and is awaiting arrival of the transfer cart from the Fuel Storage Building. The refueling SRO determines that there is not enough time to move the fuel assembly back into the core. Which of the following describes the next action that should be taken?

A. Leave the fuel assembly in the transfer canal with the mast fully extended.

B. Lower the assembly to the transfer canal floor. The mast will cut out on slack cable.

C. Put the assembly into the RCCA change fixture.

D. Continue transferring the fuel assembly to the fuel storage building.

Proposed Answer: B

B is correct. Per OS1215.05, LOSS OF REFUELING CAVITY WATER, step 4, if a fuel assembly in containment is located in the refueling machine mast or the RCCA change fixture and cannot be returned to the core, then it should be moved to the transfer canal and lowered to the floor.

A is incorrect. Per OS1215.05, LOSS OF REFUELING CAVITY WATER, precaution prior to step 4, leaving the fuel assembly in the transfer canal with the refueling machine mast fully extended could result in radiation levels exceeding 1000 Rem/Hr.

C is incorrect. Per OS1215.05, LOSS OF REFUELING CAVITY WATER, precaution prior to step 4, a fuel assembly located in the RCCS change fixture could result in radiation levels exceeding 1000 Rem/Hr.

D is incorrect. Per OS1215.05, LOSS OF REFUELING CAVITY WATER, step 4, the procedure does not prescribe trying to move a fuel assembly out of containment during a loss of refueling cavity water event.

Technical Reference(s): OS1215.05, LOSS OF REFUELING CAVITY WATER

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.2.27, Knowledge of the refueling process.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 43.6 / 45.13

Learning Objective: Lesson L1192I, Objective L1192I03

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 84	Group #		2
	K/A #	G 2.1.20, Ability to execute procedure steps.	
	Importance Rating		4.2

Proposed Question:
The following plant conditions exist:

- The plant is at 100% power.
- The fire brigade leader confirms that there is a fire in the cable spreading room.
- The Unit Supervisor has entered OS1200.02, 'Safe Shutdown and Cooldown From The Remote Safe Shutdown Facilities', as directed by OS1200.02A, 'Fire Hazard Analysis For Affected Area/Zone'.
- While the crew is performing Step 1 actions to establish remote safe shutdown conditions prior to control room evacuation a spurious Train 'B' Safety Injection actuation occurs.

What action should the crew take?

A. Continue with the actions of OS1200.02. E-0,'REACTOR TRIP OR SAFETY INJECTION' should not be implemented.

B. Transition to E-0, 'REACTOR TRIP OR SAFETY INJECTION', perform the immediate actions and then transition back to OS1200.02 to continue actions to evacuate the control room.

C. Continue with actions of OS1200.02. E-0,'REACTOR TRIP OR SAFETY INJECTION' should not be implemented until the Remote Safe Shutdown facilities are manned.

D. Transition to E-0, 'REACTOR TRIP OR SAFETY INJECTION' and consult with the TSC for further guidance.

Proposed Answer: D

D is correct per caution statement prior to Step 1 of OS1200.02.

A is incorrect but plausible. There is a caution prior to step 1 of OS1200.02 that states "E-0 should not be implemented during performance of this procedure".

B is incorrect but plausible. The caution prior to Step 1 of OS1200.02 requires E-0 entry if an SI occurs during performance of OS1200.02, however actions to continue control room evacuation should not be implemented.

C is incorrect but plausible per the caution prior to step 1 of OS1200.02, which states that E-0 should not be implemented during performance of OS1200.02.

Technical Reference(s): OS1200.02, SAFE SHUTDOWN AND COOLDOWN FROM THE REMOTE SHUTDOWN FACILITIES.

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.1.20, Ability to execute procedure steps.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 41.10 / 43.5 / 45.12

Learning Objective: Lesson L8210I, Objective L8210I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 85	Group #		2
	K/A #	E02EA2.2	
	Importance Rating		4.0

Proposed Question:
Which of the following describes the basis for Critical Operator Action Times associated with SI Termination following an inadvertent Safety Injection?

A. Prevent pressurizer water solid conditions which could lead to a LOCA due to PORV/Safety Valve failure to close.

B. Prevent pressurizer water solid conditions which could lead to pressurized thermal shock conditions.

C. Prevent pressurizer water solid conditions which could inhibit adequate ECCS boron injection.

D. Prevent pressurizer water solid conditions which results in loss of pressurizer pressure control.

Proposed Answer: A

******This is a plant specific priority based on Engineering Evaluation EE-04-024 which outlines adherence to procedural guidance for SI termination, including specific time critical actions.******

A is correct. Per EE-04-024, Rev. 03, Operator Action Response Times Assumed in the UFSAR, Pressurizer water solid conditions should be avoided to prevent challenging the PORV's and safety valves from opening and failing to reclose.

B is incorrect but plausible because this condition leads to an elevated pressure but there is no associated significant cooldown. This is not the primary challenge from an inadvertent safety injection.

C is incorrect but plausible. High RCS pressure would inhibit ECCS injection, however this is not a concern during an inadvertent safety injection.

D is incorrect but plausible because you would lose pressure control, however the primary concern is the potential LOCA from PORV/Safety valve failure.

Technical Reference(s): EE-04-024, Rev. 03, Operator Action Response Times Assumed in the UFSAR

Proposed references to be provided to applicants during examination: None

K/A Topic: E02EA2.2, Ability to determine and interpret the following as they apply to the SI Termiation: Adherence to appropriate procedures and operation within the limitations in the facilities license and amendments.

Question Source: New

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR 43.5/45.13

Learning Objective: Lesson L1230I, Objective L1230I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 86	Group #		1
	K/A #	004A2.07	
	Importance Rating		3.7

Proposed Question:
The following plant conditions exist:

- The plant is at 100% power.
- Pressurizer level is at 62% and increasing.
- Letdown flow has been isolated due to CS-V-150, Letdown Heat Exchanger ORC Isolation Valve failing closed.
- The crew has entered OS1202.01, Loss Of Letdown.
- When the reactor operator attempted to re-open CS-V-150 the valve would not open.

Which course of action should be taken as required by OS1202.01, 'LOSS OF LETDOWN'?

A. Minimize charging flow to the RCP seal only and initiate repairs to CS-V-150.

B. Establish excess letdown flow and maintain seal injection flow between 8 and 13 gpm per RCP.

C. Establish charging flow greater than 50 gpm in anticipation of restoring letdown flow.

D. Isolate the normal charging line by closing CS-V-142 and CS-V-143.

Proposed Answer: B

B is correct. Per OS1202.01, Loss Of Letdown, if normal letdown flow cannot be established then excess letdown flow should be established to support charging flow to the RCP seals.

A is incorrect but plausible. The abnormal does direct initiation of valve repair, however the correct procedural course of action is to continue with the procedure and establish excess letdown. Without excess letdown the pressurizer level would continue to increase due to seal injection flow. Repair of CS-V-150 would be done in parallel with the actions to establishing excess letown and regain control or pressurizer level. Additionally, minimization of charging flow is an expected action per the abnormal.

C is incorrect but plausible. Establishment of greater than 50 gpm charging is the correct action if normal letdown were to be established. In this case CS-V-150 does not function and pressurizer level is continuing to increase. Charging flow should be regulated to seal injection flow requirements of 8 to 13 minutes in concert with establishing excess letdown.

D is incorrect but plausible. Closing CS-V-142 and CS-V-143 isolates charging flow, which would minimize a rise in pressurizer level. This action would be directed by the charging system malfunction abnormal but not by the loss of letdown abnormal.

Technical Reference(s): OS1202.01, Loss of Letdown

Proposed references to be provided to applicants during examination: None

K/A Topic: 004A2.07, Ability to predict the impacts of the following malfunctions or operations on the CVCS, and based on these predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.
Isolation of Letdown/Makeup

Question Source: New

Question Cognitive Level: Analysis

10 CFR Part 55 Content:	CFR 41.5/43.5/45.3 /45.5
Learning Objective:	Lesson Plan L8024I, Objective L8024I04 and L8024I10

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 87	Group #		1
	K/A #	G 2.1.12, Ability to apply technical specifications for a system.	
	Importance Rating		4.0

Proposed Question:
The plant is operating at 100% power. The following conditions exist:

- CS-P-2B, Train 'B' Centrifugal Charging Pump has been removed from service and Danger Tagged for pump outboard bearing replacement.
- The crew has entered the appropriate Technical Specification ACTION to support removing CS-P-2B from service.

Subsequently, an NSO reports that RH-P-8A, Train 'A' RHR Pump, has a large puddle of lube oil below its inboard pump bearing. The NSO reports that most of the lube oil has drained out of the pump bearing housing. The Shift Manager declares that RH-P-8A is inoperable due to the leak.

What Technical Specification ACTION should the Shift Manager enter?

A. Restore CS-P-2B, Train 'B' Centrifugal Charging Pump to OPERABLE status as soon as possible. Subsequently, restore RH-P-8A to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

B. Restore both ECCS subsystems to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

C. With one of the RHR loops inoperable immediately initiate corrective action to return the operable RHR loop to OPERABLE status.

D. Apply Tech. Spec. 3.0.3. Within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply.

Proposed Answer: D

A is incorrect. The second sentence of this answer is the action for having 1 train of ECCS inoperable. Tech. Spec. 3.5.2, ECCS Subsystems does not have an action for when both trains are inoperable. Tech. Spec. item 3.0.3 applies.
B is incorrect. This answer is the action for having 1 train of ECCS inoperable.
C is incorrect. This is the required action for an inoperable RHR loop in MODE 5.
D is correct.

Technical Reference(s): Tech. Spec. item 3.0.3. Tech. Spec. 3.5.2, ECCS Subsystems

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.12 Ability to apply technical specifications for a system.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 43.2 / 43.5 / 45.3

Learning Objective: Lesson Plan L8024I, Objective L8024I04 and L8024I10

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 88	Group #		1
	K/A #	G 2.1.20	
	Importance Rating		4.2

Proposed Question:
The following plant conditions exist:

- The crew is responding to a steam generator tube rupture on the 'A' and 'B' steam generators.
- The 'A' Steam Generator indicates 20% narrow range level.
- The 'B' Steam Generator indicates 3% narrow range level.
- The 'C' and 'D' Steam Generators indicate 68% wide range level.
- Pressure in all steam generators is 1000 psig and stable.
- Both EFW pumps are running.

In accordance with E-3, STEAM GENERATOR TUBE RUPTURE, which of the following actions should be taken regarding the steam supply to the turbine-driven EFW Pump?

A. Close both MS-V-393 and MS-V-394.

B. Close MS-V-395.

C. Close either MS-V-393 or MS-V-394.

D. Close MS-V-393. When 'B' Steam Generator narrow range level is greater than 6% close MS-V-394.

Proposed Answer: A

A is correct. The motor driven EFW pump is also running.400 gpm to each steam generator is expected with both pumps running. Both the 'A' and 'B' Steam Generators have tube ruptures, so E-3 directs isolating steam supplies from both generators.

B is incorrect but plausible. MS-V-395 would isolate steam, however it is not directed in E-3, and it is interlocked to open if MS-V-393 or MS-V-394 are open.

C is incorrect but plausible. If only one generator had a tube rupture then either MS-V-393 or 394 would be closed, depending on the generator. It is plausible that one of the two steam supplies would be closed to minimize release, however E-3 directs closure of both.

D is incorrect but plausible. The procedure does direct isolating both valve, however the procedure does not direct waiting for 6% narrow range level, additionally the motor driven EFW pump is running.

Technical Reference(s): E-3, STEAM GENERATOR TUBE RUPTURE

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.20 Ability to execute procedure steps.

Question Source: New
Question Cognitive Level: Knowledge
10 CFR Part 55 Content: CFR: 41.10/43.5/45.12
Learning Objective: Lesson Plan L8056I, Objective L8056I27

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 89	Group #		1
	K/A #	G 2.1.28 Knowledge of the purpose and function of major system components and controls.	
	Importance Rating		3.3

Proposed Question:
Per Technical Specification 3.8.1.1, Electrical Power Sources, AC Systems, which of the following describes the functional basis of the Supplemental Emergency Power System (SEPS):

A. Supply 4.16kv power to either emergency Bus E5 or E6 in the event of a loss of offsite power with both Emergency Diesel Generators failing to start and/or load, provided there is no concurrent seismic event or an event that requires safeguards actuation.

B. Provide backup power to either emergency bus E5 or E6 when both Emergency Diesel Generators are out of service in MODES 1 through 4.

C. Supply 4.16kv power to either emergency Bus E5 or E6 in the event of a loss of offsite power concurrent with a seismic event or an event that requires safeguards actuation.

D. Provide backup power to both emergency busses E5 and E6 when both Emergency Diesel Generators are out of service in MODES 1-4, provided there is no concurrent seismic event or an event that requires safeguards actuation.

Proposed Answer: A

This question addresses a specific Seabrook Station priority. 03DCR002, SEPS System. This priority is specifically described in the Tech. Spec. bases and is SRO only knowledge as required by 10CFR55.43 (2).

A is correct. The SEPS is designed to Supply 4.16kv power to Emergency Bus 5 or Emergency Bus 6 in the event of a loss of offsite power with both Emergency Diesel Generators failing to start and/or load. Provided there is no concurrent seismic event or an event that requires safeguards actuation.

B is incorrect. Per Tech Spec. 3.8.1.1, Electrical Power Systems, AC Sources, with a single Emergency Diesel Generator out of service in Modes 1 through 4, SEPS allows the Tech. Spec. requirement to restore at least two Emergency Diesel Generators to operable status to be extended from 72 hours to 14 days. The basis states "The SEPS is designed to provide backup power to either emergency bus whenever one of the emergency diesel generators is out of service.

C is incorrect. SEPS is designed to provide loss-of-power to bus 5 or 6 in the event of a loss-of-offsite-power alone, not with a concurrent seismic event or an event that requis safeguards actuation.

D is incorrect. SEPS is not designed to provide power both emergency busses. Additionally, it only applies to one emergency diesel generator being out of service in Modes 1-4.

Technical Reference(s):	DCR 03-002, SEPS System	License Amendment Request 03-01, Inclusion of the Supplemental Emergency Power System.
	Tech Spec. 3.8.1.1, Electrical Power Systems, AC Sources	Technical Specification, Bases, B3/4, page 8-2.

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.28 Knowledge of the purpose and function of major system components and controls.

Question Source: New

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR: 41.7

Learning Objective: Lesson L8020I, Objectives L8020I28

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 90	Group #		1
	K/A #	064A2.02	
	Importance Rating		2.9

Proposed Question:
The crew is responding to a loss of all AC power in accordance with ECA-0.0, LOSS OF ALL AC POWER.

The following conditions exist:

- The crew was in the process of performing Step 6, to disable automatic loading of equipment
- The following pumps had been placed in Pull-To-Lock:
  - Charging Pumps
  - Containment Spray Pumps
  - RHR Pumps
  - SI Pumps
  - Motor Driven EFW Pump
  - Startup Feed Pump (Bus 5)
- The Secondary NSO reports the "A" Emergency Diesel Generator has been started locally.

Which of the following describes the correct procedural action(s) to be taken next?

A. Step 6 should be completed prior to proceeding with recovery actions. The PCCW and Service Water Pumps should be placed in Pull-To-Lock to avoid potential overload of the energized AC bus E5.

B. Step 6 should be suspended per procedural direction to proceed with recovery actions. Defeating automatic loading of LOP equipment is done, if time permits, to enhance recovery actions in ECA-0.1, LOSS OF ALL AC RECOVERY WITHOUT SI REQUIRED.

C. Step 6 should be completed prior to proceeding with recovery actions. The PCCW pumps should have been placed in Pull-To-Lock to avoid potential overload of the energized AC bus E5.

D. Step 6 should be suspended per procedural direction to proceed with recovery actions. Defeating automatic Loading of LOP equipment is done, if time permits, to support recovery in ECA-0.2, LOSS OF ALL AC RECOVERY WITH SI REQUIRED.

Proposed Answer: C

****** This question tests the candidate's understanding of a deliberate procedural requirement to take appropriate actions for protecting the Emergency Diesel Generator from a potential malfunction due to overload of the machine.******

C is correct. Per ECA-0.0, Background Document, "Defeating automatic blackout or SI loading of as many loads as practical is intended to avoid potential overload of the energized ac emergency bus." The PCCW pumps are listed in step 6 as needing to be placed in Pull-To-Lock. The caution prior to step 6 states "Once initiated, Step 6 actions must be completed prior to proceeding with recovery actions." The caution also states "A Service Water pump should be kept available to automatically load on it's AC emergency bus to provide diesel generator cooling."

A is incorrect but plausible. The caution prior to Step 6 states "A Service Water pump should be kept available to automatically load on it's AC emergency bus to provide diesel generator cooling." This is the only load that is not placed in Pull to Lock to limit overload concerns.
B is incorrect but plausible. As noted above once initiated Step 6 actions must be completed prior to proceeding with

recovery actions."
D is incorrect but plausible. As noted above once initiated Step 6 actions must be completed prior to proceeding with recovery actions."

Technical Reference(s): Per ECA-0.0, Background Document

Proposed references to be provided to applicants during examination: None

K/A Topic: 064A2.02, Ability to predict the impacts of the following malfunctions or operations on the ED/G system: Load, VARS, pressure on air compressor, speed droop, frequency, voltage, fuel oil level, temperatures.

Question Source: New

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR 41.5/43.5/45.3/ 45.13

Learning Objective: Lesson Plan L8067I, Objective L8067I11

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 91	Group #		2
	K/A #	G 2.2.25, Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	
	Importance Rating		3.7

Proposed Question:
The crew is responding to a failed Pressurizer Pressure Instrument. During the associated transient reactor coolant system pressure dropped to 2150 psig and was subsequently returned to program value.

What is the Technical Specification implication of this event?

A. Reactor Coolant System pressure dropped below the allowed DNB value. Returning pressure above the Tech. Spec. limit ensures that pressure is maintained within the normal steady state envelope of operation assumed in the transient and accident analysis.

B. Reactor Coolant System pressure did not drop below the allowed DNB value. Maintaining pressure above the Tech. Spec. limit ensures that pressure is maintained within the normal steady state envelope of operation assumed in the transient and accident analysis.

C. Reactor Coolant System pressure dropped below the allowed DNB value. Returning pressure above the Tech. Spec. limit ensures that there will be a 95% probability that the hot channel fuel assemblies in the core do not experience DNB.

D. Reactor Coolant System pressure did not drop below the allowed DNB value. Maintaining pressure above the Tech. Spec. limit ensures that there will be a 95% probability that the hot channel fuel assemblies in the core do not experience DNB.

Proposed Answer: A

A is correct. Per Tech. Spec. basis, ¾ .2.5, DNB Parameters, maintaining pressure above the Tech. Spec. limit ensures that pressure is maintained within the normal steady state envelope of operation assumed in the transient and accident analysis.

B is incorrect but plausible. The correct basis is quoted, however the Tech. Spec. limit of 2185 psig was violated.

C is incorrect but plausible. RCS pressure did drop below the DNB value, however the described limit refers to the Tech. Spec. section 2.1, Safety Limits, basis for precluding violation of fuel design criteria.

D is incorrect but plausible. The answer does test the candidates ability to determine if the RCS pressure is below the DNB limit, however, the described limit refers to the Tech. Spec. section 2.1, Safety Limits, basis for precluding violation of fuel design criteria.

Technical Reference(s):	Tech. Spec. , Bases, B3/4.2.5, DNB Parameters	Tech. Spec. , Bases, B3/4.1.1.4, Minimum Temperature For Criticality

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.2.25, Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Question Source: New

Question Cognitive Level:	Knowledge
10 CFR Part 55 Content:	CFR: 43.2
Learning Objective:	Lesson L8021I, Objective L8021117

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 92	Group #		2
	K/A #	034A2.01	
	Importance Rating		4.4

Proposed Question:
The following conditions exist:
- There is a core offload in progress.
- The fuel handlers were moving irradiated fuel from the reactor core to the fuel transfer canal.
- You are notified that the spent fuel bundle was accidentally dropped.
- Manipulator Crane Radiation Monitors RM-6535A-1 and RM-6535B-1 have gone into alarm.

What actions are required?

A. Enter procedure OS1215.06, Fuel Handling Accident. Instruct the Refueling SRO to verify that the fuel bundle is located on the refueling cavity floor. Verify that the containment ventilation systems are isolated.

B. Enter procedure OS1215.06, Fuel Handling Accident. Evacuate non-essential personnel from the containment building. Verify that the containment ventilation systems are isolated.

C. Enter procedure OS1215.03, Area High Radiation. Notify the Shift Manager, HP, and Chemistry. Verify that the containment ventilation systems are isolated.

D. Enter procedure OS1215.03, Area High Radiation. Evacuate non-essential personnel from the containment building. Verify that the Containment Ventilation Systems are isolated.

Proposed Answer: B

B is correct. Notification of a fuel handling accident is an entry condition into OS1215.06. The procedure directs evacuation of non-essential personnel and isolation of containment ventilation.

A is incorrect but plausible. The correct procedure is entered. The procedure does direct isolation of containment building ventilation, however, the procedure directs immediate evacuation of non-essential personnel.

C is incorrect but plausible. The manipulator crane is an area radiation monitor for refueling operations, however the given conditions in the question stem are such that OS1215.06, Fuel Handling Accident is the appropriate procedure. Additionally, immediate evacuation of non-essential personnel is appropriate.

D is incorrect but plausible. The manipulator crane is an area radiation monitor for refueling operations, however the given conditions in the question stem are such that OS1215.06, Fuel Handling Accident is the appropriate procedure. Additionally, isolation of containment ventialation is the appropriate action vice verifying that the fuel bundle is on the refueling cavity floor.

Technical Reference(s): OS1215.06, Fuel Handling Accident.

Proposed references to be provided to applicants during examination: None

K/A Topic: 034A2.02, Ability to predict impacts of the following malfunctions on the FHS and based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Dropped fuel element.

Question Source: New

Question Cognitive Level:	Application
10 CFR Part 55 Content:	55.43
Learning Objective:	Lesson L1192I, Objective L1192I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 93	Group #		2
	K/A #	G 2.3.6, Knowledge of the requirements for reviewing and approving release permits.	
	Importance Rating		3.1

Proposed Question:
The crew is in the process of discharging the "A" Waste Test Tank. The required release permit samples were taken at 0930 on Monday. It is now 1030 on Tuesday. Per the requirements of the LEW Release Permit, what action should the crew take regarding the ongoing discharge?

A. Temporarily suspend the discharge and perform an additional Rad. Monitor 6509 Functional Test. If the functional test is satisfactory the release may be continued.

B. Verify that the discharge dilution flowrate still meets the release permit requirements. If not, then stop the release.

C. Stop the discharge. The LEW Release Permit is only in effect for 24 hours. A new LEW Release Permit must be issued.

D. Have Chemistry perform additional NPDES sampling. If the sample results are not within specification then stop the discharge.

Proposed Answer: C
C is correct. LEW Release Permits are only in affect for 24 hours from the initial sample time.

A is incorrect. LEW Release Permits are only in affect for 24 hours from the initial sample time. A rad monitor functional check would be done to support approval of the new release permit.
B is incorrect. LEW Release Permits are only in affect for 24 hours from the initial sample time. Verification of discharge dilution flowrate would be done to support approval of the new release permit.
D is incorrect. LEW Release Permits are only in affect for 24 hours from the initial sample time. NPDES samples would be taken to support approval of the new release permit.

Technical Reference(s): ON1018.08, WASTE TEST TANK A DISCHARGE TO TRANSITION STRUCTURE; OX0917.01, Form B, LEW Release Permit.

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.3.6, Knowledge of the requirements for reviewing and approving release permits.

Question Source: New — Original question attached to reference.

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 43.4 / 45.10

Learning Objective: Lesson L1517I, Objective L1517I06

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 94	Group #		
	K/A #	G 2.1.4, Knowledge of shift staffing requirements.	
	Importance Rating		3.4

Proposed Question:
The plant is at 100% power. An on-shift fire brigade member becomes ill and must be taken to the hospital. There are 4 hours left until shift change. What action is required?

A. Action must be taken to obtain a replacement fire brigade member within two hours.

B. A replacement fire brigade member must be on-shift within two hours AFTER the scheduled shift turnover.

C. Action must be taken to obtain a replacement fire brigade member within four hours.

D. The vacant fire brigade position can remain unmanned until shift turnover.

Proposed Answer: A

A is correct. Per OPMM, Chapter 2, Shift Composition, Section 1.2, Shift Compliment contains the following NOTE:

The shift crew composition, including a Health Physics Technician and the Fire Brigade may be one less than the minimum requirements of Table 6.2-1 of Technical Specifications for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required position. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crew member being late or absent.

B is incorrect but plausible. There is a time limit for replacement of the brigade member, however it is from the time of incident vice time from turnover.

C is incorrect but plausible. There is a time limit for replacement of the brigade member, however it is 2 hours vice 4 hours.

D is incorrect but plausible. There is a time allowance for replacement, however it is 2 hours vice "until turnover".

Technical Reference(s): OPMM, Chapter 2, Shift Composition, Section 1.2, Shift Compliment; Table 6.2-1 of Technical Specifications

Proposed references to be provided to applicants during examination: None

K/A G 2.1.4, Knowledge of shift staffing requirements.
Topic:
Question Source: New
Question Cognitive Level: Application
10 CFR Part 55 Content: CFR: 41.10 / 43.2
Learning Objective: Lesson L1505I, Objective L1505I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 95	Group #		
	K/A #	G 2.1.10, Knowledge of conditions and limitations in the facility license.	
	Importance Rating		3.9

Proposed Question:
The following plant conditions exist:

- The plant is in MODE 6.
- Core offload is currently underway.
- The protected train is 'A'. All equipment needed for a proper protected train alignment is available.
- DC bus 11B is tagged out and tags are ready to be cleared.
- DC bus 11D deenergizes due to equipment failure.

Which action must occur due to the deenergized bus?

A. Immediately suspend core alterations.

B. Troubleshoot DC bus 11D to determine the cause of the failure.

C. Reenergize DC bus 11B or 11D within 1 hour.

D. Reenergize DC bus 11B and close the crosstie breaker between DC bus 11B and 11D within 2 hours.

Proposed Answer: B

B is correct. Bus 11B and 11D are Train 'B' busses. Tech Spec. 3.8.3.2, Onsite Power Distribution, MODES 5 and 6, requires "Two 125-VDC busses (in the same train) energized from their associated battery banks." The question stem states that Train 'A' is protected and all of the trains equipment is available. This means that DC busses 11A and 11C are operable, and the Tech. Spec. is satisfied.

A is incorrect but plausible. If the student misinterpreted the information in the question stem they could errantly determine that the Tech. Spec. requirement "Two 125-VDC busses (in the same train) energized from their associated battery banks" is not satisfied. In this case the student would apply their knowledge of the 'immediate action' for suspending core alterations, which is the associated required action for this Tech. Spec. item.

C is incorrect but plausible. If the student misinterpreted the information in the question stem they could errantly determine that the Tech. Spec. requirement "Two 125-VDC busses (in the same train) energized from their associated battery banks" is not satisfied. It is a common student challenge to differentiate between the 1 hour action Tech. Spec. items and the "immediate action" Tech. Spec. items. This question tests the students ability to identify the need for immediate action vice a 1 hour action.

D is incorrect but plausible. This action is correct from Tech. Spec. 3.8.3.1, Onsite Power Distribution, MODES 1,2,3, and 4, "When one DC bus is not energized from it's associated battery bank, reenergize the DC bus from it's associated battery bank or close the bus tie to the alternate OPERABLE battery of the same train within 2 hours.

****This is a challenging question for SRO's given that there are no references provided. This is a Tech. Spec. which includes immediate actions, so the SRO's are required to know the Tech. Spec. and applicable actions without the use of references.****

Technical Reference(s): Technical specification 3.8.3.2, Onsite Power Distribution, MODES 5 and 6.

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.1.10, Knowledge of conditions and limitations in the facility license.

Question Source: Direct from bank. Seabrook 2003 NRC Exam

Question Cognitive Level: Analysis

10 CFR Part 55 Content: CFR: 43.1 / 45.13

Learning Objective: Lesson L8017I, Objective L8017I13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 96	Group #		
	K/A #	G 2.2.21, Knowledge of pre- and post-maintenance operability requirements.	
	Importance Rating		3.5

Proposed Question:
Which of the following would be considered an acceptable practice regarding preconditioning of equipment immediately prior to performing a Technical Specification surveillance test?

A. Stroking a valve prior to stroke time testing.

B. Cleaning and lubricating breakers.

C. Venting the RHR pumps.

D. Venting test equipment.

Proposed Answer: D

This is a plant specific priority. A number of years ago the NRC Senior Resident had identified potential "pre-conditioning" of equipment prior to Tech. Spec. surveillances as a potential issue at Seabrook. WM 8.0, Section 4.2, Preconditioning of Equipment exists to guard against this potential occurrence.

Seabrook Station Work Control Supervisors (Senior Licensed Operators) are regularly challenged to guard against "preconditioning" during performance of Tech. Spec. surveillances, including valve stroke testing, pump operability surveillances, breaker operability testing, and control of surveillance test equipment.

NRC Information Notice 97-16 discussed the potential for maintenance activities performed before surveillance testing to adversely affect the validity of the surveillance test results for structures, systems, and components.

The Seabrook Station Operations and Training Departments has continued to address "preconditioning" as a high priority item. The following are examples of "preconditioning" challenges that have occurred since NRC Information Notice 97-16 was issued:

- **1998-NRC identified "preconditioning" concerns regarding EFW pump surveillances. These concerns were included in the February, 12, 1998 NRC Exit Meeting pursuant to Inspection Report 97-08.**
- **Inspection Report 98-01. NRC questioned "preconditioning" with reference to venting via FW-V-477.**
- **Inspection Report 99-01. Non-cited violation NCV-99-01-01 identified a concern with PAH-FN-42A/B testing. This was a violation of Appendix B, Criterion XI, Test Control. Seabrook Station Condition Report 99-474 was written to review "preconditioning".**
- **LER-00-001-00 was written which included the conclusion that several electrical breakers were considered to have been "preconditioned".**

D is correct. Per NAMW, Procedure WM8.0, Work Control Practices, Section 4.2, Preconditioning of Equipment, Part 4.2.4, Acceptable Actions, Venting of attached test transmitters prior to performing surveillance testing is specifically listed as an acceptable practice.

A is incorrect but plausible. It is common practice to perform Technical Specification required surveillance testing activities which involve integrated activities such as pump operability runs, ESFAS testing, and valve stroke timing. The student may assume that it is an acceptable practice to allow the stroking of a particular valve in conjunction with an ESFAS actuation or in support of a pump run and then stroke test an additional time in support of valve stroke time testing. Valve stroke time testing must be completed prior to any further strokes of a particular valve. Surveillance test procedural coordination is such that valve stroke preconditioning is prevented.

B is incorrect but plausible. Cleaning and lubricating breakers may well be performed as part of a particular work package, however it is not allowed immedtiately prior to Technical Specification required surveillance testing. This is specifically described as unacceptable in NAMW, Procedure WM8.0, Work Control Practices, Section 4.2.2, Unacceptable Actions.

C is incorrect but plausible. The Seabrook Operators are aware of the importance of a properly venting the RHR pump seal package during maintenance runs or system filling and venting activities, however, with regard to Technical Specification required surveillance testing, this unacceptable. This is specifically described as unacceptable in NAMW, Procedure WM8.0, Work Control Practices, Section 4.2.2, Unacceptable Actions.

Technical Reference(s): NAWM, WM 8.0, Work Control Practices, Section 4.2, Pre-Conditioning of Equipment, Section 4.2.2, Unacceptable Actions, and 4.2.4, Acceptable Actions.

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.2.21, Knowledge of pre- and post-maintenance operability requirements.

Question Source: Direct from bank.

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR: 43.2

Learning Objective: Lesson L1514I, Objective L1514I02

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 97	Group #		
	K/A #	G 2.2.25, Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	
	Importance Rating		3.7

Proposed Question:
Which of the following describes the primary reason for maintaining Refueling Cavity water level greater than 23 feet over the top of the reactor vessel flange?

A. Sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

B. Sufficient water depth is available to provide time for the operators to recognize the indications of a dilution accident before Keff can exceed 95% Δk/k.

C. Sufficient water volume is available to provide time for the operators to respond to a loss of RHR accident before water temperature in the reactor vessel and refueling cavity exceeds 140° F.

D. Sufficient water is maintained above the top of the fuel assemblies during fuel movement to ensure that the radiation levels at the operating elevation of containment remain below 5 mR/hr.

Proposed Answer: A

A is correct. Per Tech. Spec. Bases 3.9.10 and 3.9.11, the restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

B is incorrect. The required boron concentration has a basis for maintaining subcriticality per Tech. Spec. Bases 3.9.1, Boron Concentration.

C is incorrect. The basis for maintaining ONE RHR loop OPERABLE (T.S. 3.9.8) with the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, is that a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

D is incorrect. Sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity the rupture of an irradiated fuel assembly.

Technical Reference(s): Technical Requirements

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Question Source: Direct from bank. Salem 1998. Modified to make Seabrook Specific.

Question Cognitive Level: Memory

10 CFR Part 55 Content: CFR: 43.2

Learning Objective: Lesson L1192I, Objective L1192I12

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 98	Group #	2.3	2.3
	K/A #	2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	
	Importance Rating		3.1

Proposed Question:
An outage task is going to be performed in the "A" Steam Generator. A Planned Special Exposure is going to be used for an individual employee. The Planned Special Exposure has been justified as it will reduce the collective doses of all the personnel working on the task.

Which of the following correctly lists items that are included in processing the Planned Special Exposure?

A.
- Obtain approval from the Health Physics Department Supervisor.
- Obtain Approval from the Health Physics Department Manager.
- Obtain Approval from the Operations Manager.

B.
- Obtain approval from the Health Physics Department Supervisor.
- Obtain Approval from the Health Physics Department Manager.
- Obtain Approval from the Station Director.

C.
- Obtain approval from the Outage Containment Coordinator.
- Obtain Approval from the Health Physics Department Manager.
- Obtain Approval from the Station Director.

D.
- Obtain verification that the NRC regional office has reviewed the Planned Special Exposure, if time permits.
- Obtain Approval from the Health Physics Department Manager
- Obtain Approval from the Station Director.

Proposed Answer: D
D is correct per RP 5.2, Planned Special Exposures.

A is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed, however it is from the Station Director vice the Operations Manager.

B is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed from the Station Director . There is no approval required from the HP Supervisor. HP Supervisor approval is for lower administrative dose approval.

C is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed from the Station Director . There is no requirement for approval from the Outage Containment Coordinator.

Technical Reference(s): RP5.2, Planned Special Exposures

Proposed references to be provided to applicants during examination:	None
K/A Topic:	2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.
Question Source:	New
Question Cognitive Level:	Memory
10 CFR Part 55 Content:	CFR: 43.4 / 45.10
Learning Objective:	L1525I, Objective L1525I13

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 99	Group #		
	K/A #	G 2.4.21, Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control 2. Core cooling and heat removal 3. Reactor coolant system integrity 4. Containment conditions 5. Radioactivity release control.	
	Importance Rating		4.3

Proposed Question:
The following plant conditions exist:

- The unit has tripped and safety injection has actuated.
- While performing E-1, "Loss of Reactor or Secondary Coolant," an ORANGE path condition was noted for the Core Cooling ~~Heat Sink~~ critical safety function.
- FR-C.2, "Response to Degraded Core Cooling" was entered in response to this condition.
- While performing the steps of this procedure, the crew notes that valid RED path conditions exist for **BOTH** the Heat Sink and Containment critical safety functions.

Per Pertelin Tim Cassidy 7/5/07

Based on these conditions, what course of action should the Unit Supervisor take?

A. Stop performing FR-C.2, and immediately address the Heat Sink RED path.

B. Stop performing FR-C.2, and immediately address the Containment RED path.

C. Complete the actions of FR-C.2, and then address the Heat Sink RED path.

D. Complete the actions of FR-C.2, and then address the Containment RED path.

Proposed Answer: A

A is correct. Per OP 9.2, Emergency Operators Users Guide, Section 4.3, Control Room Usage of Status Trees, the order of priority for critical safety functions is Subcriticality, Core Cooling, Heat Sink, Integrity, Containment, Inventory, Emergency Recirculation, and RDMS. The order of severity priority is Red, Orange, Yellow, and Green. If any Orange terminus is encountered, the operator is expected to monitor all of the remaining trees, if no Red terminus is present, then suspend any ERP or ECA and address the Orange condition. If during the performance of an Orange condition FRP, any Red condition or higher priority Orange arises, then the higher priority condition should be addressed and the original Orange FRP is suspended.

B is incorrect. Containment is of a lesser priority than Heat Sink.

C is incorrect. If a Red priority occurs then the Orange FRP should be suspended.

D is incorrect. If a Red priority occurs then the Orange FRP should be suspended.

Technical Reference(s): OP 9.2, Emergency Operators Users Guide, Section 4.3, Control Room Usage of Status Trees

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.4.21, Knowledge of the parameters and logic used to assess the status of safety functions including:
1. Reactivity control
2. Core cooling and heat removal
3. Reactor coolant system integrity
4. Containment conditions
5. Radioactivity release control.

Question Source: Direct from bank. Seabrook #26546

Question Cognitive Level: Application

10 CFR Part 55 Content: CFR: 43.5 / 45.12

Learning Objective: Lesson L1195I, Objective L1195I05

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 100	Group #		
	K/A #	G 2.4.38, Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator.	
	Importance Rating		4.0

Proposed Question:
What actions are required per ER 1.2, "Emergency Plan Activation", in the event of a radiological release via the turbine driven Emergency Feedwater Pump exhaust?

A. Use the highest reading MAIN STEAM LINE radiation monitor to estimate the UNMONITORED release rate using the Offsite Dose Projection System.

B. Use the UNMONITORED **OR** the MAIN STEAM LINE release pathway of the Offsite Dose Projection System, whichever is more conservative.

C. Dispatch a monitoring team to the downwind site boundary location to obtain a site boundary dose rate, and use the UNMONITORED release pathway of the Offsite Dose Projection System.

D. Dispatch a monitoring team to the downwind site boundary location to obtain a site boundary dose rate, and use the MAIN STEAM LINE release pathway of the Offsite Dose Projection System.

Proposed Answer: C

C is correct per ER1.2, Emergency Plan Activation in the event of a radiological release via the turbine driven Emergency Feedwater Pump exhaust a monitoring team is dispatched to the downwind site boundary location to obtain a site boundary dose rate, and the UNMONITORED release pathway is used on the ODPS.

Technical Reference(s): ER 1.2, Emergency Plan Activation

Proposed references to be provided to applicants during examination: None

K/A Topic: G 2.4.38, Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator.

Question Source: New

Question Cognitive Level: Knowledge

10 CFR Part 55 Content: CFR: 43.5 / 45.11

Learning Objective: Lesson L1509I, Objective L1509I20