Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 1	Group #	1	1
	K/A #	EPE.007.GEN.2.4.21	
	Importance Rating	3.7	4.3

Proposed Question:

The control room crew have just completed immediate actions of E-0, "Reactor Trip or Safety Injection" and completed transition to ES-0.1, "Reactor Trip Response". Shortly after entering this procedure, Safety Injection actuates and the Balance of Plant Operator notes Critical Safety Function Status as follows:

- Subcriticality – GREEN
- Core Cooling – YELLOW
- Heat Sink – GREEN
- Integrity – YELLOW

The Shift Manager checks containment conditions to determine if the containment barrier is intact. The following conditions exist:

- Containment pressure: 42 psig and slowly rising
- Containment sump level: 3.2 feet and slowly rising
- Containment Radiation: 8.0 R/hr and stable
- All Containment Phase A & B penetrations are isolated

Given these conditions, what is the correct crew response?

A. Transition to FR-Z.2, "Response to Containment Flooding".

B. Transition to FR-Z.1, "Response to High Containment Pressure".

C. Remain in ES-0.1, Yellow Path Functional Restoration Procedures are entered on discretion only.

D. Remain in ES-0.1, until first transition at which time Functional Restoration Procedures become applicable.

Proposed Answer: B

Explanation of answer: B is correct because containment conditions generate an ORANGE path containment CSF status tree. Per rules of usage an orange path requires transition to appropriate FRP. In this case FR-Z.1 is the correct procedure.

Explanation of Distractors:
A- incorrect because containment building level is <5ft, which is the trigger point of becoming orange.
C- incorrect because an orange path exists and rules of usage require transition, the second part of distractor is plausible because it is a true statement.
D- incorrect because an orange path exists and rules of usage require transition and FRPs applied when the first transition from E-0 occurred versus ES-0.1.

Technical Reference(s): FR-Z Status Tree, OPMM
E-0, step 10

Proposed references to be provided to applicants during examination: None

K/A Topic:	Knowledge of the parameters and logic used to assess the status of safety functions including, reactivity control, core cooling and heat removal, RCS integrity, containment conditions, and radioactivity release control.
Question Source:	Modified Bank #22583 Original question attached to reference.
Question Cognitive Level:	Higher
10 CFR Part 55 Content:	55.41.10/55.43.5
Learning Objective:	L1212I06RO State the symptoms or entry conditions for FR-Z.1.

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

Proposed Question:

The following plant conditions exist:

- PCV-456A PORV and associated block valve is stuck open.
- The reactor trips and SI is initiated.
- The crew transitions from E-0, "Reactor Trip or Safety Injection" to E-1, "Loss of Reactor or Secondary Coolant".
- All systems are functioning as expected with the exception of the stuck open PORV & block valve.
- Pressure is currently at 1350 psig and continues to decrease.

Which of the following statements provides accurate guidance for operation of RHR pumps as required by E-1 and what is the basis?

A. Do not secure RHR pumps because they may be needed at lower primary pressure.

B. Do not secure RHR pumps because primary pressure and level instrumentation are not reliable during steam space accident.

C. Secure RHR pumps because primary pressure is greater than 260 psig to reduce burden on EDGs during design basis accident.

D. Secure RHR pumps and place in standby to prevent damage to the pump because pressure will not drop below 1250 psig for 1 stuck open PORV.

Proposed Answer: A

Explanation of answer: Step 8 does not allow securing RHR pump if pressure is decreasing
Explanation of distractors:
B. Incorrect because the level instrument is not reliable, however the pressure instrument is OK.
C. Incorrect because this is not the basis for securing RHR
D. Incorrect because cannot secure pump while pressure dropping

Technical Reference(s): E-1, L1413I Loss of Coolant TAA, Westinghouse ERG for E-1

Proposed references to be provided to applicants during examination: None

K/A Topic: PZR Vapor Space Accident/knowledge of the reasons for the following responses as they apply to the PZR vapor space accident: actions contained in EOP for PZR vapor space accident/LOCA.

Question Source: 2003 NRC Exam (#83)/ Bank

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1413I03RO For any LOCA, analyze the cooling mechanism for the core throughout the transient.

L1203I05RO State the basis for the following steps, cautions, or notes found in E-1.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 3	Group #	1	1
	K/A #	EPE.009.EA2.04	
	Importance Rating	3.8	4.0

Proposed Question:

The following sequence of events and plant conditions exist:

- A reactor trip and safety actuation has occurred.
- The "A" Safety Injection pump FAILED to start both automatically and manually.
- Pressurizer level is 17% and INCREASING SLOWLY.
- Pressurizer pressure is 1900 psig and INCREASING SLOWLY.
- Tave is 552 F and STABLE.
- All other conditions and equipment are functioning normally.

If the "B" Safety Injection pump TRIPS on electrical overload, how will this affect pressurizer level and Reactor Coolant System (RCS) subcooling?

	PRESSURIZER LEVEL	RCS SUBCOOLING
A.	DECREASING	DECREASING
B.	DECREASING	STABLE
C.	STABLE	DECREASING
D.	INCREASING	INCREASING

Proposed Answer: D

Explanation of answer: 1900 is above the shutoff head of the SI pumps and pressurizer level and RCS pressure are increasing due to high head injection. Tavg is stable, indicating that the pressure and level increase are not due to temperature.

Explanation of distractors:

A- Incorrect because it represents conditions where the SI pumps would be necessary to maintain RCS inventory (RCS pressure below SI shutoff head)

B- Incorrect because pressurizer level will not decrease with high head pumps running and SI pumps at shutoff head.

C- Incorrect because subcooling will not decrease with increasing RCS pressure and stable RCS temperature.

Technical Reference(s): ECCS Detailed System Text (3.3), LP1413I (LOCA Transient Accident Analysis)

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine or interpret PZR Level as applied to a SBLOCA.

Question Source: Bank

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5/7

Learning Objective: L1413I02RO For any LOCA list or determine the two mechanisms of thermal hydraulic relationships, which determine the RCS pressure response to the accident.

L8034I07RO List the RCS pressure at which high, intermediate and low head injection will occur.

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

Proposed Question:

Given the following plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred.
- E-0, "Reactor Trip or Safety Injection " is being implemented.
- Power is lost to 4.16kV Bus E-5.
- Containment Pressure is 23 psig and increasing.
- RCS Temperature is 532 F and decreasing.
- RCS Pressure is 1320 psig and decreasing.
- All systems responded normally to actuation signals.

Which of the following correctly describes the relationship between these plant conditions and reactor coolant pump (RCP) status?

The RCPs should _____

A. be stopped because RCP cooling is lost.

B. be stopped because inadequate RCS subcooling margin exists.

C. NOT be stopped because adverse containment conditions do NOT exist.

D. NOT be stopped because with no power to Bus 5 there are no Safety Injection pumps operating.

Proposed Answer: A

Explanation of answer: Containment Pressure has exceeded 18 psig (Hi-3 setpoint) which causes Phase B isolation and subsequently isolates PCCW which cools the RCP motors and seals. Therefore procedural requirements have the operator secure RCPs when these conditions exist.

Explanation of distractors:

B- Incorrect because at 532 F saturation pressure is approximately 850 psig which is well above the required 40 F subcooling to trip RCP's. This is also based on a SBLOCA versus large break LOCA.

C- Incorrect because containment pressure is above 4.3 psig so therefore adverse conditions do exist, however there is no trip criteria at 4.3 psig.

D- Incorrect because Bus 6 is still available and therefore one SI pump is available. This is a plausible answer for a SBLOCA because without high head injection RCPs would not be tripped.

Technical Reference(s): E-0 step 6 and OAS page criteria. Westinghouse ERG for RCP trip criteria. E-0 ERG Step 14 background document.

Proposed references to be provided to applicants during examination: Steam Tables

K/A Topic Knowledge of the interrelationships between the RCPs and Large Break LOCA.

Question Source: Modified Bank #20884 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.3/10

Learning Objective: L1202I03RO State the basis for the following steps, notes or cautions contained in E-0. L1202I05RO State the RCP trip criteria found in E-0.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 5	Group #	1	1
	K/A #	010.A3.02	
	Importance Rating	3.6	3.5

Proposed Question:

Given the following conditions:

- The plant is operating at 100% power, steady state, with all controls in AUTOMATIC.
- The Pressurizer pressure channel (PT-455) selected for control fails LOW.

Based on these conditions, which of the following choices identifies how pressurizer temperature and pressurizer pressure will **INITIALLY** be affected with NO operator action?

	PZR TEMPERATURE	**PZR PRESSURE**
A.	INCREASES	INCREASES
B.	INCREASES	DECREASES
C.	DECREASES	INCREASES
D.	DECREASES	DECREASES

Proposed Answer: A

Explanation of answer: A failure of the controlling PZR pressure channel will cause PZR heaters to energize which will result in an increase in PZR temperature and therefore pressure due to the controlling channel failure. The spray valves will not open to mitigate this pressure increase, therefore the volume of water in the PZR will also expand causing PZR pressure and temperature to continue to rise in their saturated environment until the plant trips on high pressure and the PORV opens. The question stem states INITIALLY.
Explanation of distractors: each distractor is incorrect because they have the wrong combination of parameter directional changes for initial impact of this failure.

Technical Reference(s): L1406i, Instrument Failure Analysis

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to monitor automatic operation of the PZR PCS including PZR Pressure.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5/14

Learning Objective: L1406I02RO For the following plant instruments state where the parameter is sensed and list all uses for the associated channel.

L8027I06RO Describe the auto control functions of the PZR pressure and level control system.

L8027I14RO Explain the auto response of the plant including indications to the operator to a high or low failure of the PZR pressure and level control system.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 6	Group #	1	1
	K/A #	APE.022.AK3.03	
	Importance Rating	3.1	3.3

Proposed Question:

The following plant conditions exist:

- The plant has been operating at 100% power for 300 days.
- An NSO reports to the control room that charging line containment isolation valve CS-V142 has a significant body to bonnet leak.
- The Shift Manager directs the control room crew to isolate letdown and charging, and place excess letdown in service using the guidance of OS1202.02, "Charging System Failure".

Based on these conditions, why is excess letdown required?

A. To restore Volume Control Tank hydrogen control capability.

B. To maintain constant reactor coolant system inventory/Pressurizer Level.

C. To provide a means of purifying reactor coolant by aligning flow through the mixed bed demineralizers.

D. To ensure an adequate cooling water flowrate through the reactor coolant pump seals while charging flow is isolated.

Proposed Answer: B

Explanation of answer: Without normal letdown in this situation there is no charging flowpath to the regenerative HX, excess letdown is required to maintain PZR level constant as it is designed to remove 20 gpm. This amount is exactly equal to seal injection flow that leaks into the RCS. Without excess letdown, PZR level would increase to 92% and the plant would eventually trip.
Explanation of distractors:
A – incorrect because although the return flowpath for excess letdown is to the suction of the charging pumps, there is little impact on the hydrogen capability of the VCT.
C – incorrect because the return flowpath in on the RCP seal return line which bypasses the demineralizers.
D – incorrect because seal injection can be supplied with charging isolated regardless of status of excess letdown. As long as thermal barrier HXs are supplied by PCCW, seal cooling can be provided with no seal injection.

Technical Reference(s): OS1202.02, Charging System Failure, CS Detailed Text, LP8024

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the reasons for performance of lineup to establish excess letdown after determining the need to isolate letdown as applied to loss of coolant makeup.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.3/5/8

Learning Objective: L8024I09RO Determine the flowrate in the following flowpaths that would be required to maintain a constant PZR level for any charging pump combination when provided with the letdown flowrate.

L1445I11RO Summarize the major actions of OS1202.02.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 7	Group #	1	1
	K/A #	APE.025.AK2.02	
Proposed Question:	Importance Rating	3.2	3.2

While operating with the Reactor Coolant System (RCS) in a mid-loop condition, the running Residual Heat Removal (RHR) pump shows signs of cavitation.

Which of the following statements is correct regarding the standby RHR pump?

The RHR Pump should _____

A. NOT be immediately started because air entrainment could cause a loss of both RHR trains.

B. be immediately started because following a loss of RHR flow, an RCS pressurization may occur precluding gravity feed makeup.

C. be immediately started because under certain loss of RHR conditions, core uncovery or core voiding can occur within 15 to 20 minutes.

D. NOT be immediately started because starting an idle RHR pump under mid-loop conditions could cause an unacceptable reduction in core shutdown margin.

Proposed Answer: A

Explanation of answer: The ARG provides a clear guidance which includes industry experience of why operation of an RHR pump operating with air entrapment should be evaluated because it could lead to pump damage. The AOP requires once indication of cavitation are experienced that both RHR pumps are placed in PTL. Starting the other RHR pump could transfer the problem to the other pump leading to a complete loss of the system.
Explanation of distractors:
B – incorrect because although gravity makeup is a required action, starting the RHR pump is not an option immediately.
C – incorrect because again while it is plausible that core uncovery or voiding could occur in a relatively short period of time, it is not correct that the RHR pump be started in this plant condition.
D- incorrect because although it is correct that the pump should not be started it is not correct that SDM would be affected in this condition since it was already verified procedurally to meet TS plant conditions.

Technical Reference(s): OS1213.02 Loss of RHR at Reduced Inventory or Midloop Conditions.
ARG-1 for Loss of RHR At Midloop

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the interrelationships between loss of RHR system and LPI or Decay Heat Removal/RHR Pumps.

Question Source: Bank #1318

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.3/5/10

Learning Objective: L1705I05RO Summarize the major actions of OS1213.02.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 8	Group #	1	1
	K/A #	APE026A2.06	
	Importance Rating	2.8	3.1

Proposed Question:

In accordance with OS1212.01, "PCCW System Malfunction", which of the following is the time limit for tripping the reactor following a loss of Primary Component Cooling Water to the Reactor Coolant Pumps?

A. 2 minutes

B. 5 minutes

C. 10 minutes

D. 13 minutes

Proposed Answer: C

Explanation of answer: To prevent damage to the RCPs following a loss of PCCW it is procedurally required to trip the reactor and then RCPs following the E-0 immediate actions within 10 minutes to help preclude damage to the RCP's.

Explanation of distractors:

A,B – incorrect however plausible because they are both times associated with RCP coastdown to complete stop with and without RCPs running in the RCP malfunction abnormal. These numbers also are close enough to the other numbers such that they coincide well.

D - incorrect however plausible because it is the amount of time noted in the key cautions and notes of OS1201.01 that it will take seal water inlet temperature to approach 550 F following a loss of seal injection and thermal barrier flow.

Technical Reference(s): OS1212.01, PCCW System Malfunction

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine and interpret the length of time after loss of PCCW flow to a component before that component may be damaged.

Question History: Bank #16013

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.3/10

Learning Objective: L1445I01RO Identify the conditions requiring an emergency shutdown of an RCP as found in OS1212.01.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 9	Group #	1	1
	K/A #	012.K1.01	
	Importance Rating	3.4	3.7

Proposed Question:

Which one of the following supplies voltage for OPERATION of the SLAVE RELAYS in the Solid State Protection System (SSPS) output cabinets?

A. 120 VAC from redundant inverters in the SSPS cabinets.

B. 48 VDC from a power distribution bus in the SSPS cabinets.

C. 120 VAC Vital Instrument Power from vital station inverters.

D. 15 VDC from redundant power supplies in the SSPS cabinets.

Proposed Answer: C

Explanation of answer: PP-1A,B,C,D (vital station inverters) provide 125VAC vital power to RPS slave relays.
Explanation of distractors:
A – incorrect because although 120 VAC is the source of power it is not derived from inverters in the SSPS cabinets themselves.
B & D – incorrect, however plausible because these supply power via redundant internal power supplies to SSPS electronics and trip and actuation output signals respectively

Technical Reference(s): RP Detailed System Text.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the physical connections and/or cause effect relationships between the RPS and 120 VAC vital/instrument power system.

Question Source: Bank # 25192

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8056I07RO State the function of the master and slave relays in RPS.
L1186I08RO Describe the effect of a loss of EDE power panel 1A or 1B on SSPS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 10	Group #	1	1
	K/A #	EPE.029.EA1.15	
	Importance Rating	4.1	3.9

Proposed Question:

The following plant conditions exist:

- The crew is performing the actions of FR-S.1, "Response to Nuclear Power Generation/ATWS" initiated by a Loss of All Feedwater event.
- In accordance with step 8, the operator is checking steam generator levels:
    - Narrow range levels for all steam generators are off-scale low.
    - Wide Range levels for all steam generators are approximately 62%.
    - Total Emergency Feedwater Flow (EFW) is 750 gpm.

Based on these conditions, what action should the crew take to control EFW flow?

A. Control feed flow to maintain reactor coolant system temperature.

B. Manually align valves and start pumps as necessary to increase EFW flow to greater than 880 gpm.

C. Maintain 500 gpm total feedwater flow until at least one steam generator is greater than 65% wide range on at least two steam generators.

D. Maintain 500 gpm total feedwater flow until at least one steam generator is greater than 5% narrow range on at least one steam generator.

Proposed Answer: B

Explanation of answer: Step 8 of FR-S.1 has a continuous action step to maintain > 880 gpm if narrow range level in at least one S/G is <5% NR. This is to maintain secondary heat sink.
Explanation of distractors:
A – incorrect because the basis of step 8 is to ensure sufficient EFW flow for decay heat removal during this event not to maintain RCS temperature. In an ATWS event RCS will rise and FTC will help shutdown the reactor.
C and D are incorrect yet plausible because they are criteria to maintain heat sink during E-0 versus FR-S.1.

Technical Reference(s): FR-S.1, Response to Nuclear Power Generation/ATWS & associated ERG basis

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to operate and monitor AFW system as applied to an ATWS

Question Source: Bank # 23076

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1200I02RO Summarize the major actions of FR-S.1.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 11	Group #	1	1
	K/A #	EPE.038EA2.14	
	Importance Rating	3.3	4.6

Proposed Question:

Which of the following groups of initial conditions, assumptions, and failures were used in the Seabrook Steam Generator Tube Rupture Chapter 15 UFSAR Accident Analysis, resulting in the **MOST SEVERE** magnitude of radiological release?

RCS= Reactor Coolant System
S/G= Steam Generator
EFW= Emergency Feedwater Flow
ASDV= Atmospheric Steam Dump Relief Valve

	RCS PRESSURE	S/G LEVEL	EFW FLOW	ASDV FAILS
A.	HIGH	HIGH	MAXIMUM	OPEN
B.	HIGH	LOW	MINIMUM	OPEN
C.	LOW	HIGH	MAXIMUM	SHUT
D.	LOW	LOW	MINIMUM	SHUT

Proposed Answer: B

Explanation of answer: According to Seabrook UFSAR, the combination of higher RCS initial pressure, Lower S/G Level, minimum EFW flow with the ADSV failing open will result in our worst case SGTR event resulting in the worst radiological consequences.
Explanation of distractors:
A, D are plausible because they contain elements of the UFSAR analyzed assumptions.
C is incorrect because it contains none of the assumptions, however provides a good mix of the assumptions, which distinguishes between a less vs. more competent operator.

Technical Reference(s): L1414i SGTR TAA Lesson, Chapter 15.1 USFAR

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine or interpret the magnitude of atmospheric radioactive release if cooldown must be completed using steam dumps or if atmospheric relief's lift as they apply to a SGTR.

Question Source: Bank #24976

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.5

Learning Objective: L1414I04RO Describe the two limiting SGTR events analyzed in the Seabrook specific analysis to include.....

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 12	Group #	1	1
	K/A #	APE.040.GEN.2.4.4	
	Importance Rating	4.0	4.3

Proposed Question:

Given the following plant conditions:

PARAMETER:	CURRENT VALUE:	TREND:
Reactor power:	58%	Increasing
RCS pressure:	2235 PSIG	Decreasing
Auctioneered High Tavg:	569°F	Decreasing
Turbine power:	595 MWE	Decreasing
Containment pressure:	2 PSIG	Increasing

Based on the above plant indications, what event is occurring?

A. Steamline Break.

B. RCS Dilution Event.

C. Small Break RCS LOCA.

D. Steam Generator Tube Rupture.

Proposed Answer: A

Explanation: Reactor power is increasing, indicating positive reactivity event ("C" and "D" wrong). Electric load is decreasing, indicating loss of steam to the turbine. Also although a dilution event would add positive reactivity, it would result in an increase in RCS temperature vs. decrease. The fact that containment pressure is rising indicates the SLB is inside the containment ("A" correct, "B" wrong).

Technical Reference(s): E-2 ERG Description of SLB Event

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures as applied to Steam Line Rupture.

Question History: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5

Learning Objective: L12075I01RO Recognize symptoms or entry conditions for E-2.
L1202I09RO State the criteria as found in E-0 for transition to E-2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 13	Group #	1	1
	K/A #	022.GEN2.1.33	
	Importance Rating	3.4	4.0

Proposed Question:

Which of the following identifies the normal operating containment internal pressure band in accordance with Technical Specification 3.6.1.4, "Primary Containment Internal Pressure"?

A. 12.6 to 14.6 psia

B. 14.6 to 16.2 psia

C. 16.2 to 18.6 psia

D. 18.2 to 20.2 psia

Proposed Answer: B

Explanation of answer: ROs are required to know from memory < 1 hr required action statements. TS 3.6.1.4 is part of TS required logs and should be controlled within a narrow band so that we do not exceed containment pressure on design basis accidents. The band for Seabrook according to TS is 14.6 to 16.2 psia.
Explanation of Distractors: A,C, & D are incorrect because they do not correspond with TS required value, however they are plausible because of their close proximity to the actual value.

Technical Reference(s): TS 3.6.1.4

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to recognize indications for system operating parameters which are entry level conditions for Technical Specifications.

Question Source: Bank #24420

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.10

Learning Objective: L8038I05aRO State from memory and apply the LCO and actions for the following TS: 3.6.1.4

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 14	Group #	1	1
	K/A #	APE.058.AK1.01	
	Importance Rating	2.8	3.1

Proposed Question:

The following plant conditions exist:

- The plant is operating at 100% power.
- Following a crew brief, Technical Specification action statements have been entered to transfer 125 VDC Bus 11A to its normal battery supply in accordance with OS1048.13, "Vital Bus 11A Operation".
- Due to a human performance error, the Nuclear Systems Operator missed several procedure steps and did not verify that battery charger (1-EDE-BC-1A) was connected to DC Bus 11A.
- The alternate battery supply breaker was then opened.

Given these conditions what operational implications are a direct result of these actions?

A. Loss of Turbine Trip Control.

B. Loss of Normal Feedwater Control.

C. Loss of Emergency Diesel Generator Stop Capability.

D. Loss of Pressurizer Power Operated Relief Valve Control (Fails open).

Proposed Answer: B

Explanation of answer: These conditions are outlined as a caution in OS1048.13 and result in a Loss of Vital Bus 11A. As a result of this bus loss, Feedwater Control is lost and will result in a plant trip.
Explanation of distractors:
A – Incorrect because this would be a result of non-vital DC bus loss.
C – Incorrect because the loss of DC Bus 11A results in a loss of start capability vs. stop capability.
D - Incorrect because the PORV fails closed vs. open.

Technical Reference(s): OS1248.01, Loss of Vital 125VDC Bus, OS 1048.13, Vital Bus 11A Operation

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of battery charger equipment and instrumentation as applied to Loss of DC Power.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7/10

Learning Objective: L1189I15RO Describe the effects of losing Vital DC Power on the following equipment……

L8020I23RO Describe the importance of 125VDC power for the proper auto operation of the EDG and EPS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 15	Group #	1	1
	K/A #	APE.062.GEN.2.4.11	
	Importance Rating	3.4	3.6

Proposed Question:

The following plant conditions and sequence of events are provided:

- The plant is operating at 100% power in a normal at power configuration.
- A Tower Actuation (TA) signal is received on **BOTH** trains.
- The control room crew enters OS1216.01, "Degraded Ultimate Heat Sink" and verifies Tower Actuation is properly aligned.

Based on these plant conditions, what Service Water System loads have isolated?

A. EDG Heat Exchangers.

B. PCCW Heat Exchangers.

C. SCCW Heat Exchangers.

D. Excess Letdown Heat Exchangers.

Proposed Answer: C

Explanation of answer: With the conditions given, the operator must recognize that actuation of both TA trains will shut both SW-V-4 & 5, isolating both trains of SCCW cooling from Turbine Building loads. If prompt response is not taken once in the AOP, this will lead to EOP entry due to Turbine overheating concerns. With a normal TA system realignment, the cooling tower pumps have started and are providing cooling to the EDG's and PCCW systems which eliminates the other two plausible Service Water Cooled loads in choices A & B. Choice D is incorrect because the excess letdown heat exchanger is cooled by PCCW which is not isolated by this event.

Technical Reference(s): OS 1216.01, Degraded Ultimate Heat Sink, SW Detailed System Text. Detailed Systems Summary Setpoints Handout

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of abnormal operating procedures as applied to Loss of Nuclear Service Water.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.4/10

Learning Objective: L1193I02RO Summarize the major actions of OS1216.01.

L8037I13RO List the auto actions which will occur upon initiation of TA signal.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 16	Group #	1	1
	K/A #	APE.065.A1.03	
	Importance Rating	2.9	3.1

Proposed Question:

With the plant at 100% power, a significant instrument air leak occurs, and the following sequence of events occurs:

- Air pressure starts to drop at a Moderate rate and the crew enters OS1242.01, "Loss of Instrument Air".
- A Nuclear System Operator reports that the leak location has been found, and the leak can be isolated.
- The leak is isolated, and instrument air pressure recovers from 70 psig to 110 psig prior to meeting any reactor trip criteria.
- The Balance of Plant Operator notes that due to the loss of air the "C" Circulating Pump has tripped on High Screen Differential.

Which one of the following procedures will address this set of plant conditions?

A. OS1242.01, "Loss of Instrument Air".

B. OS1233.01, "Loss of Condenser Vacuum".

C. OS1234.02, "Condenser Tube or Tube Sheet Leak".

D. OS1290.02, "Response to Condensate or Feedwater Heater System Transient".

Proposed Answer: B

Explanation of answer: As demonstrated by actual plant events, the circ pump house is the first effected by a loss of IA. The degraded air supply prior to restoration caused a failure of the traveling screen DP instrument (bubbler), causing the circ pump to trip. The operator must equate this failure with a Loss of Condenser Vacuum, which would occur as a result of the circ water pump trip. Entry conditions for the drop in condenser vacuum as a result of the pump trip would be met and therefore it is expected that this procedure is entered. Failure to take action would result in a plant trip.
Explanation of distractors:
A – Incorrect because remaining in this procedure will not address this problem. It is more designed to provide reactor trip criteria based on loss of air versus once air pressure is restored.
C & D – Incorrect because although they might have some system association making them plausible, they are designed to handle other problems and would not be correct choices for entry.

Technical Reference(s): OS1242.01, Loss of Instrument Air & other procedure cover sheets for referenced procedures.

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to operate and/or monitor the restoration of systems served by instrument air when pressure is regained as they apply to the Loss of Instrument Air.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.4/10

Learning Objective: L1194I04RO Describe how a loss of IA will effect the following systems.....
L1194I02RO Summarize the major actions of OS1242.01.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 17	Group #	1	1
	K/A #	W/E05.EK3.3	
	Importance Rating	4.0	4.1

Proposed Question:

Given the following plant conditions and sequence of events:

- The crew is combating a Loss of Secondary Heat Sink in accordance with FR-H.1, "Loss of Secondary Heat Sink".
- Reactor Coolant System (RCS) pressure is 1200 psig.
- Bleed and feed has been established with all Emergency Core Cooling System pumps in service and both Pressurizer Operated \Relief Valves (PORVs) open.
- Reactor Coolant Pumps are NOT running.
- Wide range levels are < 5% in all steam generators.
- RCS hot leg temperatures on all loops are 560 F and slowly dropping.
- The crew is about to re-establish feedwater flow to the "D" steam generator.

Based on these conditions, which of the following describes the flow rate that should be established to the "D" steam generator and the reason for the flow rate?

A. Feed at the maximum rate to mitigate the potential for core damage.

B. Feed at the minimum rate to minimize thermal stress on steam generator components.

C. Feed at the maximum rate to depressurize the RCS and facilitate accumulator injection.

D. Feed at the minimum rate to minimize RCS cooldown rate and RCS pressurized thermal shock.

Proposed Answer: B

Explanation of answer: To comply with the caution prior to step 21 of FR-H.1, since feed and bleed has been established and RCS temperatures are dropping in a dry S/G, operators are directed to feed at a minimum rate.

Explanation of distractors:

A & C – incorrect because they prescribe feeding at the maximum rate which would be applicable if RCS temperature were increasing and the S/G was dry.

D - incorrect because the concern is steam generator damage versus cooldown/PTS which is plausible during other accidents.

Technical Reference(s): FR-H.1, Response to Loss of Secondary Heat Sink Caution prior to step 21 and associated background document.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the reasons for manipulation of controls required to obtain desired operating results during abnormal and emergency situations as applied to a Loss of Heat Sink.

Question Source: Bank #23077

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1211I.02RO Summarize the major actions of FR-H.1

L1211I03RO State the basis for the caution, steps, and notes from FR-H.1.

L1420I04RO Describe the effectiveness of the following analyzed techniques for maintaining adequate core cooling I response to a loss of all feedwater event: EFW flow restoration.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 18	Group #	1	1
	K/A #	W/E11.EK1.1	
	Importance Rating	3.7	4.0

Proposed Question:
A Loss of Coolant Accident has occurred and the crew is now responding to a Loss of Emergency Coolant Recirculation. The crew has initiated Reactor Coolant System cooldown to cold shutdown, and they have checked conditions to determine spray requirements.

The following plant conditions exist:

- Two containment building spray (CBS) pumps are running.
- CBS pump suctions are aligned to the Refueling Water Storage Tank (RWST).
- RWST level is 250,000 gallons.
- Containment Building Level is 3 feet.
- Containment pressure is 12 psig.

Based on these conditions, which of the following statements describes the actions necessary in accordance with ECA-1.1, "Loss of Emergency Coolant", to meet CBS system alignment requirements at this time?

A. Stop the two CBS pumps.

B. Stop one CBS pump, and leave running pump aligned to the RWST.

C. Stop one CBS pump, and align running pump to the containment sump.

D. Leave both CBS pumps running and align to the containment sump.

Proposed Answer: A
Explanation of answer: Based on ECA-1.1, with CBS pumps aligned to the RWST and containment pressure < 18 psig, no CBS pumps are needed to mitigate high containment pressure and the priority shifts to conserving RWST volume.
Explanation of distractors:
B- Step 11 of ECA-1.1 determines the minimum spray requirements based on containment pressure. Since containment pressure is not between 18 – 52 psig, this answer is incorrect.
C – Again this answer part 1 of this answer is plausible if containment pressure were between 18 –52 psig, aligning to the containment sump is incorrect.
D- This is plausible if containment pressure is > 52 psig, again aligning to the containment sump is incorrect.

Technical Reference(s): ECA 1.1, Loss of Emergency Coolant Recirculation and associated background.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of components, capacity, and function as applied to Loss of Emergency Coolant Recirculation.

Question Source: Bank #18888

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1209I02RO Summarize the major actions of ECA-1.1.

L1209I03RO State the basis for steps, notes and cautions of ECA-1.1.

L1212I08RO State the basis for the steps, notes and cautions of FR-Z.1: caution prior to step 2 (if ECA-1.1 is in effect, containment spray should be operated as directed in ECA-1.1 rather than step 2 below.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 19	Group #	2	2
	K/A #	APE.003.AK1.03	
	Importance Rating	3.5	3.8

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power at End of Life.
- Control Bank "D" rods are at 228 steps.
- ONE of the Bank "D" rods drops indicating it is on the bottom of the core.
- The reactor does NOT trip and no operator action is taken.

Based on these conditions, which of the following best represents the expected plant response to the dropped rod?

Initial Reactor Power Response	Final Reactor Power	Tavg
A. Drops	Same as Initial	Lower
B. Drops	Lower than Initial	Little Change
C. Little Change	Lower than Initial	Little Change
D. Little Change	Same as Initial	Lower

Proposed Answer: A

Explanation of answer: The AOP for dropped rod notes dropping reactor power as one of the entry conditions for a dropped rod. Reactor power initially prompt drops due to instantaneous insertion of negative reactivity into the core. The resultant drop in Tavg will insert positive reactivity into the core. Since no control rod motion will occur with rods full out, reactor power will return to near its original value with Tavg lower to offset the negative worth of the rod insertion. Power returns to a value slightly lower due to a lower Tavg and lower S/G pressure and also because the greater worth of the entire bank. Reactor power follows steam demand.
Explanation of distractors:
All distractors are plausible but incorrect because they contain an aspect of the correct answer with combinations that discriminate between competent and less competent operators.

Technical Reference(s): UFSAR Section 15.4 on dropped rods.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of the relationship of reactivity and reactor power to rod movement as applied to a Dropped Rod.

Question Source: Modified Bank #20449 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.1/5

Learning Objective: L1185I01RO Recognize the entry conditions for Dropped Rod.
L1410I05RO Describe the response of the RCS to a dropped rod accident.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 20	Group #	2	2
	K/A #	APE.028.AA1.08	
	Importance Rating	3.7	3.6

Proposed Question:

The following plant conditions exist:

- The plant is operating at 100% power.
- All Control Systems are in a normal alignment.
- Pressurizer (PZR) level channel LT-460 fails LOW

Based on these plant conditions, select the action, if any, that should be performed by the crew to reopen RC-LCV-460 (Letdown Isolation Valve)?

A. Select channel 459/461 for control, then open RC-LCV-460.

B. Select channel 461/460 for control, then open RC-LCV-460.

C. No action required, RC-LCV-460 will not close on this failure.

D. Select channel 459/461 for control, open CS-V145, then open RC-LCV-460.

Proposed Answer: A

Explanation of answer: Normally Channel LT-459 which is the controlling channel and pushbutton 459/460 is selected on the MCB. When LT-460 fails low, the associated letdown isolation valve receives a signal and isolates letdown. The crew will enter the AOP for this failure (OS1201.07) which directs an alternate channel to be selected and then restoration of letdown.

Explanation of distractors:

B- incorrect but plausible since this channel would be the incorrect channel.

C- incorrect but plausible if the student is not aware of the interlock associated with this valve. LT-459 provides an input to RC-LCV-459 and LT-460 provides an input to RC-LCV-460.

D- incorrect but plausible since it has the correct selector switch, but incorrect interlock. The interlock is plausible still however, because in order to open CS-V145, both RC-LCV-459 & 460 must be closed.

Technical Reference(s): RC PZR Press & Level Control Functional 1-NHY-509051/OS1201.07, PZR Level Failure AOP/PPLC Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to operate and/or monitor the selection of an alternate PZR level channel if one has failed as applied to PZR level control malfunctions.

Question Source: Bank #18971

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7/10

Learning Objective: L8027I05RO Describe the function of any PZR pressure or level control system, indication or control on the MCB.

L8027I06RO Describe the Auto control functions of the PZR Pressure and level control system.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 21	Group #	2	2
	K/A #	APE.032.GEN.2.1.20	
	Importance Rating	4.3	4.2

Proposed Question:

Given the following conditions:

- A reactor startup is in progress with the reactor just critical in accordance with OS1000.07, "Approach to Criticality".
- The operator has just stopped moving control rods.
- Power slowly increases above the P-6 setpoint.
- ONE source range (SR) nuclear instrumentation channel (N-31) fails LOW.
- Remaining power indications stabilize.

In accordance with Technical Specification 3.3.1 (provided), which of the following actions is required?

A. Block the source range, since it is not required above P-6.

B. Trip the reactor and enter E-0, " Reactor Trip and/or Safety Injection.

C. Conduct a reactor shutdown and restore both SR channels to operability prior to next startup.

D. Suspend all operations involving positive reactivity changes until both SR channels are restored to operability.

Proposed Answer: A

Explanation of answer: TS 3.3.1(instrumentation) establishes that above P-6, the SR NI are not required by TS and will shortly be de-energized by procedure. Since there are no TS implications, the startup may proceed.
Explanation if distractors:
B – incorrect because no entry conditions are met for a reactor trip. (plausible misconception)
C – incorrect because a plant shutdown is not required unless both SR channels are lost during reactor startup.
D – incorrect because SR channels can be blocked. Reactor power is above P-6. This statement would be correct if reactor power was < P-6.

Technical Reference(s): TS 3.3.1 & OS1000.07, Approach to Criticality

Proposed references to be provided to applicants during examination: TS 3.3.1 is provided as a reference.

K/A Topic: Ability to execute procedure steps as related to a Loss of Source Range NI.

Question Source: Modified Bank # 6926 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.6/10

Learning Objective: L8030I12RO Given a copy of TS locate applicable LCOs and list the applicable action statements for stated excore nuclear instrumentation channel failures.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 22	Group #	2	2
	K/A #	APE.059.AA2.05	
	Importance Rating	2.9	3.6

Proposed Question:

The following plant conditions exist:

- A release from the 'B' Waste Test Tank is in progress at 45 gpm in accordance with ON1018.09, WASTE TEST TANK 'B' DISCHARGE TO TRANSITION STRUCTURE.
- RDMS channel R-6509, (1LM621), Liquid Waste Tank To CW System, goes into HIGH alarm.

Based on these conditions, which of the following automatic actions occur?

A. 1-WL-FCV-1458-1, high capacity waste distillate to discharge structure valve closes and AUTO reopens when high radiation condition clears.

B. 1-WL-FCV-1458-1, high capacity waste distillate to discharge structure valve closes and must be MANUALLY reopened when high radiation condition clears.

C. 1-WL-FCV-1458-2, low capacity waste distillate to discharge structure valve closes and AUTO reopens when high radiation condition clears.

D. 1-WL-FCV-1458-2, low capacity waste distillate to discharge structure valve closes and must be MANUALLY reopened when high radiation condition clears.

Proposed Answer: B

Explanation of answer/distractors: 1458-1, is used when the release rate is greater than 20 gpm and 1458-2 is for releases at less than 20 gpm. When radiation monitor 1LM621 goes into high alarm the release is terminated by closing the discharge valve. In this case, it would be 1458-1.
A – incorrect because a modification was added to require manual operation prior to reopening of either valve so that an inadvertent release does not occur.

C & D – incorrect because the release rate 45 gpm, therefore, 1458-1 will be used and 1458-2 not.

Technical Reference(s): OS1252.01, Process or Effluent High Radiation UFSAR Figure 11.2.4, N1319i Test Tank & Discharges

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to operate/monitor the occurrence of automatic safety actions as a result of a high PRM system signal as applied to Accidental Liquid Radwaste Release.

Question Source: Bank #22622

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.5/10

Learning Objective: L1187I02RO Summarize the major actions of OS1252.01.

L8059I06RO Describe the auto actions that result when the below listed monitors reach their specified setpoints.....

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 23	Group #	2	2
	K/A #	W/E09.GEN.2.4.12	
	Importance Rating	3.4	3.9

Proposed Question:

The crew is verifying natural circulation in ES-0.1, "Reactor Trip Response".

The following conditions exist:

- Reactor Coolant System (RCS) Wide Range (WR) Thot is 580 F and stable.
- RCS WR Tcold is 555 F and stable.
- RCS pressure is 1920 psig and stable.
- Core Exit Thermocouples are 585 F and stable.
- Steam Generator (S/G) pressure is 1090 psig and stable.
- S/G WR level is 70% in all S/Gs and stable.
- Emergency Feedwater flow is 400 gpm to each S/G.

Based on these plant conditions, which one of the following describes the status of natural circulation which will determine subsequent crew operating responsibilities?

A. All conditions are met, natural circulation is established.

B. Natural circulation is NOT established, heat sink is insufficient.

C. Natural circulation is NOT established, subcooling is insufficient.

D. Natural circulation is NOT established, RCS delta T is insufficient.

Proposed Answer: A

Explanation of answer: All of the conditions of Attachment F of ES-0.1, so therefore natural circulation conditions are met.
Explanation of distractors:
B & D are incorrect because heat sink and DT although required for natural circulation to exist are not part of the criteria to verify these support conditions.
C is incorrect because subcooling is > 40 F based on calculations using steam tables.

Technical Reference(s): ES-0.1, Reactor Trip Response Attachment F

Proposed references to be provided to applicants during examination: Steam Tables/ Mollier Diagram

K/A Topic: Knowledge of general operating responsibilities during emergency operations.

Question Source: Bank #5299

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5/10

Learning Objective: L1225I04RO Summarize the major actions of ES-0.1.

L1225I08RO State the five parameters used to verify natural circulation flow as listed in attachment F of ES-0.1.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 24	Group #	2	2
	K/A #	W/E10.EA2.1	
	Importance Rating	3.2	3.9

Proposed Question:

The following plant conditions exist:

- The unit has tripped from 100% power when a switchyard failure caused a loss of offsite power.
- ES-0.2, "Natural Circulation Cooldown" is in progress to perform a natural circulation cooldown and depressurization of the reactor coolant system (RCS).

For which of the following situations should ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS) be used instead of ES-0.2?

A. The Safety Injection accumulators are unable to be isolated.

B. Normal Pressurizer Spray is unavailable for use in depressurizing the RCS.

C. NO Reactor Coolant Pumps will be able to be restarted prior to cooling down the plant to < 200 F.

D. The required high rate of plant cooldown and depressurization results in less than 100% level in the reactor vessel head.

Proposed Answer: D

Explanation of answer: In accordance with ES-0.2, the note prior to step 12 and also at step 15, the criteria has been met to transition to ES-0.3.
Explanation of distractors:
A is incorrect because during all natural circulation procedures, the SI accumulators are isolated.
B is incorrect because normal spray is unavailable, the procedure directs use of PORVs if aux spray not available.
C is incorrect because if RCPs are restarted in this procedure there is no longer a natural circulation issue and transition back to normal operating procedures is directed.

Technical Reference(s): ES-0.2, Natural Circulation Cooldown

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine and interpret facility operations and selection of appropriate procedures during abnormal and emergency operations as applied to Natural Circulation with Steam Void in Vessel with/without RVLIS.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.5/10

Learning Objective: L1225I06RO Summarize the major actions of ES-0.2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 25	Group #	2	2
	K/A #	W/E14.EK3.2	
	Importance Rating	3.1	3.7

Proposed Question:

Which of the following correctly describes why the Containment Functional Restoration Procedure (FR-Z.1) is **different** from all other Functional Restoration Procedures?

The Containment Functional Restoration Procedure _____

A. is entered on Unit Supervisor judgment based on plant conditions.

B. assumes both trains of containment building spray are operating to ensure a success path.

C. is considered dual function guideline because it addresses both orange and red path conditions within the same guideline.

D. provides no alternate method to reduce containment pressure other than to restore normal containment cooling methods.

Proposed Answer: D

Explanation of answer: Per lesson 1422i and background documents for FR-Z.1 there is no alternative success path to lower containment pressure if normal methods are not available. This makes FR-Z.1 unique.
Explanation of distractors:
A – incorrect because this statement is contrary to rules of usage according to OP 9.2. Containment FRP has higher than yellow path entry conditions which are required to be entered based on rules of usage. The FRP is not unique in this respect.
B - incorrect because like most EOPs they assume one train only with two being preferable. The FRPs are beyond design bases so therefore seeks to reduce containment pressure with whatever CBS pumps are available.
C – incorrect because FR-S.1, FR-P.1 are also dual function FRPs so therefore it is not unique in this respect.

Technical Reference(s): OP-9.2 EOP Users Guide, FR-Z.1, Response to High Containment Pressure, LP1422i

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the reasons for normal, abnormal and emergency operating procedures associated with High Containment Pressure.

Question Source: Bank #16072

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1422I04RO Describe the basic difference between the operator actions of FR-Z.1 and those of other functional restoration procedures.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 26	Group #	2	2
	K/A #	W/E15.EK1.3	
	Importance Rating	2.8	3.0

Proposed Question:

Given the following plant conditions:

- With the plant initially at 100% power, a large break LOCA occurs in Containment resulting in Phase "A" & "B" isolation.
- Based on Containment Building Level > 5 feet, the crew has entered FR-Z.2, "Response To Containment Flooding", and is trying to identify and isolate the sources of water to the CTMT sump per FR-Z.2 step 1.
- The Reactor Operator (RO) has been directed to check the Reactor Makeup Water (RMW), Primary Component Cooling Water (PCCW), and Fire Protection (FP) systems as potential water sources into Containment.

How will the RO check these three paths, and which of these paths, if any, should still need to be isolated from Containment?

A. The Containment isolation valves for all three systems can be verified closed on the "A" Main Control Board front section. All three paths should already be isolated.

B. The RMW and PCCW isolation valves can be verified on the Main Control Board. The FP path can only be verified locally. All three paths should already be isolated.

C. The Containment isolation valves for all three systems can be verified closed on the "A" Main Control Board front section. The RMW System still needs to be isolated.

D. The RMW and PCCW isolations can be verified on the Main Control Board. The FP path can only be verified locally and would require local operator action to isolate.

Proposed Answer: B

Explanation of answer: FR-Z.2, step 1 has the operator identify unexpected sources of water to the containment sump. Only the RMW and PCCW isolations can be operated from the MCB. It should be recognized that most containment isolation valves which are manual and cannot be operated from the control room are normally closed valves unless open under administrative controls. There is no control room indication of FP valves. Note that this scenario was encountered during the 2004 graded E-Plan exercise.

Explanation of distractors:

A – incorrect because although RMW and PCCW can be verified on the MCB, FP cannot.

C – incorrect because of the same reason above. Additionally, RMW-V-30 closed on Phase "A" isolation.

D – incorrect because the FP is already isolated. The first portion of the answer is plausible since it is correct.

Technical Reference(s): FR-Z.2, Response to Containment Flooding./OX0443.75, Establishing Containment Fire Protection.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of annunciators and condition indicating signals and remedial actions associated with Containment Flooding.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.4/5.10

Learning Objective: L1212I10RO Summarize the major actions of FR-Z.2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 27	Group #	2	2
	K/A #	W/E16.EK2.1	
	Importance Rating	3.0	3.3

Proposed Question:

FR-Z.3, "Response to High Containment Radiation Level", (step 2) checks if the containment recirculation filter should be placed in service. Why is containment pressure verified less than 18 psig?

A. A "P" signal will prevent the containment recirculation filter fans, FN-3A and FN-3B, from starting.

B. A "P" signal will prevent the containment recirculation filter system realignment to the Filter Mode.

C. The radioactive release associated with a containment pressure of 18 psig exceeds the limits of the recirculation filter capability.

D. Containment pressure greater than 18 psig could damage the recirculation filter dampers when they are realigned to the Filter Mode.

Proposed Answer: B

Explanation of answer: The P signal or Phase B isolation realigns dampers to recirculation Mode, bypassing the filter at 18 psig.
Explanation of distractors:
A – incorrect because the P signal is what actually starts the fans. The P signal occurs at 18 psig.
C – incorrect and could be a common misconception.
D – incorrect because although potentially plausible that this pressure could cause damage, the logics prevent this from happening because it is not possible to realign to filter Mode until interlock is satisfied.

Technical Reference(s): FR-Z.3, Response to High Containment Radiation, CHV Detailed System Text (4.1.3)

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the interrelationships between High Containment Radiation and components, functions of control and safety systems, including instrumentation, signals, interlocks, failure Modes, and automatic and manual features

Question Source: Bank #22288/ 2003 NRC Exam

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7/10

Learning Objective: L1212I12RO Summarize the major actions of FR-Z.3.

L8038I04RO Describe the response of the CAH components to the following protective actions signals.....

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 28	Group #	1	1
	K/A #	003.K1.10	
	Importance Rating	3.0	3.2

Proposed Question:

The following plant conditions exist:

- The plant restarted following a forced outage.
- Loop flow measurement determined the "A" Reactor Coolant Pump impeller has degraded such that its Reactor Coolant System (RCS) loop flow has DECREASED by 5% from its original value.
- The other three RCS loop flows remain UNCHANGED.

Based on these conditions, which one of the following would be a result of the decreased flow rate in the "A" loop?

A. Delta temperature in the "A" RCS loop at full power will be lower.

B. Demand on the pressurizer variable heaters at 2235 psig will be lower.

C. Steam pressure in the "A" Steam Generator at full power will be higher.

D. The reactor core will operate closer to Departure from Nucleate Boiling when at full power.

Proposed Answer: D

Explanation of answer: putting out the same MWt with a reduced flowrate means reduced heat transfer capabilities and therefore operation closer to DNB.
Explanation of distractors:
A - Delta T should actually be higher in this situation.
B - Loop A & C RCPs provide PZR spray. However, based on plant design, the "A" RCP is less effective than the "C" RCP and provides limited flow. Additionally under steady state/ 2235 psig conditions, the sprays are closed and bypasses provide minimal flow to keep the lines warm, so therefore not much impact on current draw from variable heaters.
C – Higher Delta T means Tcold should be lower and therefore steam pressure should be lower.

Technical Reference(s): L8138i, Core Thermal Limits

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the physical connections and/or cause effect relationships between the RCPS and the RCS.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.3/14

Learning Objective: L8138I03RO Describe factors that effect peaking and hot channel factors.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 29	Group #	1	1
	K/A #	004.K5.06	
	Importance Rating	3.0	3.3

Proposed Question:

Given the following plant conditions:

- The plant is operating at 90% power during Middle of Life.
- Primary Component Cooling Water (PCCW) flow through the letdown heat exchanger is DECREASED.
- Assume no automatic actions occur in the Chemical Volume and Control System (CVCS) and no operator actions are taken.

Based on these conditions, which of the following describes how reactor coolant system (RCS) temperature will be INITIALLY affected?

RCS Temperature _____

A. decreases because the CVCS demineralizers are releasing boron into the letdown flow.

B. increases because the CVCS demineralizers are absorbing boron from the letdown flow.

C. decreases because loop delta temperature is being decreased by hotter CVCS charging flow.

D. increases because loop delta temperature is being increased by hotter CVCS charging flow.

Proposed Answer: A

Explanation of answer: With less PCCW flow through the letdown HX's the RCS water is not cooled as much. Therefore the inlet temperature of water flowing through the letdown demineralizers is warmer which result in more boron being released into CVCS. This boron once completing the flowpath back into the RCS will result in a drop in RCS temperature which impacts reactivity.
Explanation of distractors:
B – incorrect because exactly the opposite is true. This distractor is a common misconception and describes the effect if PCCW cooling increased and CVCS water temperature drops.
C & D are both incorrect as the cold narrow range RCS temperature RTDs are upstream of charging tap to the loop.

Technical Reference(s): L8114i, Demineralizers & IX's

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of the concept of boron worth or IBW (reactivity, pcm/ppm)

Question Source: Bank #20895

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.1/5

Learning Objective: L8114I22RO Describe the demineralizer characteristics that can cause a change in boron concentration.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 30	Group #	1	1
	K/A #	005.K6.03	
	Importance Rating	2.5	2.6

Proposed Question:

Given the following plant conditions:

- The plant is being cooled down in preparation for refueling.
- Reactor Coolant System (RCS) temperature is 175°F.
- RCS pressure is 305 PSIG.
- The Pressurizer is solid.
- Charging flow control is in manual.
- Residual Heat Removal (RHR) is in the cooldown Mode, with the "A" Train in service.
- The "A" RHR heat exchanger suddenly develops a 50 gpm tube leak.

Based on these conditions and assuming no operator action is taken, what will be the result of this event?

A. Indicated RHR Pump flow decreases. Pressurizer pressure increases. RCS cooldown rate increases.

B. Indicated RHR Pump flow remains the same. Pressurizer pressure decreases. RCS cooldown rate decreases.

C. PCCW surge tank level will increase, until overflowing to the Primary Auxiliary Building exhaust plenum.

D. PCCW surge tank level will decrease, until the "A" PCCW pump trips, resulting in a loss of shutdown cooling.

Proposed Answer: B

Explanation of answer: RHR system pressure is higher than PCCW system pressure, causing an RCS leak into PCCW. Pump flow would remain the same after automatic system adjustment. The system flow control valve is auto set to maintain around 3500 gpm. Downstream of the HX the increased flow would be sensed and the bypass valve would open to accommodate the increased flowrate to maintain overall system flowrate constant. Since more flow is now bypassing the RHR HX, the RCS cooldown will not be as effective. With a solid plant and charging in manual a 50 gpm loss from the RCS will result in decreasing RCS pressure

Explanation of distractors:

A - incorrect because pump flow would increase and therefore all parameters are exactly opposite of what would happen.

C - incorrect because although PCCW head tank increases and overflows, the overflow goes to the liquid waste system versus the PAB exhaust system. The tank is vented to the PAB exhaust via an unvalved line where it is directed to the plant vent.

D - incorrect because RHR system pressure is higher than PCCW system pressure which is around 60 to 80 psig, therefore leakage will be into the PCCW system and head tank level will increase.

Technical Reference(s): RHR Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the effect of a loss or malfunction on the heat exchanger will have on the RHRS.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7/14

Learning Objective: L8033I07RO Describe the RHR HX outlet temp/flow control scheme.
L8033I11RO Describe the flowpath through the RHR system during plant cooldown.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 31	Group #	1	1
	K/A #	006.K6.03	
	Importance Rating	2.5	2.6

Proposed Question:

Given the following plant conditions:

- The plant has sustained a Large Break Loss of Coolant Accident.
- Cold leg Emergency Core Cooling System flow on SI-FI-917 is 900 gpm.
- Residual Heat Removal flow is 4000 gpm in EACH train.
- Containment Building Spray flow is 3000 gpm in EACH train.
- Safety Injection Pumps both FAILED to start.

Based on these conditions and assuming Refueling Water Storage Tank was at its Technical Specification MINIMUM level when the accident occurred, approximately how much time will pass before initiation of swap-over to cold leg recirculation?

A. 15 minutes

B. 24 minutes

C. 30 minutes

D. 44 minutes

Proposed Answer: B

Explanation of answer: TS mimimum volume is 477,000 gallons. Swapover to cold leg recirculation occurs at about 120,000. Approximately 357,000 gallons will be pumped into containment at the following flowrates: SI Flow = 0, Charging Flow = 900 gpm, RHR flow = 8000 gpm, CBS flow = 6000 gpm. Total flow = 14,900 gpm. 357,000/14900 = 24 minutes

Technical Reference(s): TS 3.5.4, CBS & SI Detailed System Text

Proposed references to be provided to applicants during examination: Calculator

K/A Topic: Knowledge of the effect of a loss or malfunction of safety injection pumps will have on the ECCS.

Question Source: Modified Bank #23134 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.8

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 32	Group #	1	1
	K/A #	007.K3.01	
	Importance Rating	3.3	3.6

Proposed Question:

Given the following plant conditions and sequence of events:

- The plant is in Mode 1 at 100% power.
- One of the Pressurizer Operated Relief Valves (PORVs) is leaking past its shut seat.
- Primary Relief Tank (PRT) cooling does NOT function as designed.

If no operator action is taken, which of the following statements describes an expected outcome of this situation?

A. PRT pressure rises causing rupture disks to rupture to containment atmosphere.

B. PRT pressure rises causing the PRT relief valves to lift to containment atmosphere.

C. PRT level rises causing water to be diverted to containment sump via overflow lines.

D. PRT level rises causing water to be diverted to containment sump via sparging valves.

Proposed Answer: A

Explanation of answer: Without the normal cooling system available, the PRT level would rise and pressure in the tank would continue to increase until rupture disks blowout at 91 psig to containment atmosphere.
Explanation of distractors:
B - incorrect because the PRT has rupture disks not relief valves.
C – incorrect because although it is plausible that level will rise, there is no divert to the containment sump.
D - incorrect because again, although it is plausible that level will rise and the sparging lines are located inside the tank and do not interface externally with containment atmosphere.

Technical Reference(s): PZR & PRT Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the effect that a loss or malfunction of the PRTS will have on the containment.

Question Source: Modified Bank #24018 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.3/8

Learning Objective: L8022I11RO Explain how the PRT is protected from overpressurization.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 33	Group #	1	1
	K/A #	008.K2.02	
	Importance Rating	3.0	3.2

Proposed Question:

The following plant conditions exist:

- The Reactor Operator is in the process of swapping running Primary Component Cooling Water (PCCW) pumps in the "A" PCCW Loop.
- Both "A" & "C" PCCW pumps are running when a loss of off-site power occurs.
- All system function as designed.

Based on these conditions, what will be the status of Train "A" PCCW pumps at the completion of the emergency power sequencing?

A. Both "A" and "C" PCCW pumps tripped.

B. Both "A" and "C" PCCW pumps running.

C. Only the "A" PCCW pump is running and the "C" PCCW pump is tripped.

D. Only the "C" PCCW pump is running and the "A" PCCW pump is tripped.

Proposed Answer: D

Explanation of answer: The "C" PCCW pump is electrically the preferred pump. Therefore any loss of off-site power and subsequent re-energization of Bus 5 for the "A" Train will result in the "C" pump starting. The purpose of this configuration is to prevent the "A" EDG from overloading during initial starting and sequencing. The "A" pump gets locked out until RMO is reset by the operator.
Explanation of distractors: Each distractors seeks to cover the other various combinations of pumps starts or lack of starts and are incorrect because of how the interlock is configured as discussed above.

Technical Reference(s): CC Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the bus power supplies to the CCW pump, including emergency backup.

Question Source: Modified Bank #20230 Original question attached to reference.

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.5/8

Learning Objective: L8036I06RO Describe the interaction between the EPS and the CC pumps.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 34	Group #	1	1
	K/A #	008.A4.07	
	Importance Rating	2.9	2.9

Proposed Question:

Which of the following describes the automatic operation of the Primary Component Cooling Water (PCCW) System in response to PCCW "B" head tank level decreasing to 42%?

A. Train "A" and "B" Waste Processing Building PCCW isolation valves close.

B. Train "A" and "B" Waste Processing Building (WPB) Containment isolation valves close.

C. Train "B" PCCW Radiation Monitor isolation valve and "B" WPB isolation valves close.

D. Train "B" PCCW Radiation Monitor isolation valve and "B" Containment isolation valves close.

Proposed Answer: C

Explanation of answer: According to the OAS page of OS1212.01 and the CC Detailed System Text, the "B" train PCCW Radiation Monitor and WPB valves will isolate off of a "B" Train signal at 42%.
Explanation of distractors: The combination of train related signals tends to be a common misconception for this isolation and is the basis for all of the distractors.
A & B – incorrect because the Train A PCCW components are unaffected by a low head tank level in Train B PCCW head tank.
D – incorrect because the containment isolation valves go closed on a Lo-Lo head tank level of 36% vs. the 42% stated in the stem of the question.

Technical Reference(s): OS1212.01, PCCW Malfunction OAS Page, CC Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to manually operate and/or monitor control of minimum level in the CCWS surge tank in the control room.

Question Source: Bank # 23128

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8036I12RO Describe the auto responses of each CC loop to the following signals.....lo tank level.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 35	Group #	1	1
	K/A #	APE.015/017.AK2.07	
	Importance Rating	2.9	2.9

Proposed Question:

The following plant conditions exist:

- The plant is operating at 40% power.
- Alarms for "D" Reactor Coolant Pump (RCP) seal leakoff flow are received.
- "D" RCP Total Leakoff Flow indicates 9.0 gpm and increasing.

Based on these conditions, which of the following actions should be taken by the operating crew according to OS1201.01, "RCP Malfunction"?

A. Trip the 'D' RCP, take manual control of 'D' SG feedwater control, and commence a plant shutdown.

B. Feed 'D' SG to 60-70% narrow range, trip 'D' RCP, close the #1 seal leakoff valve after the pump has stopped, and commence a plant shutdown.

C. Trip the reactor and go to E-0, "Reactor Trip or Safety Injection". After step 4 of E-0, stop 'D' RCP and close the #1 seal leakoff valve after the pump has stopped.

D. Trip 'D' RCP, close the #1 seal leakoff valve after the pump has stopped, trip the reactor, and perform the immediate actions of E-0, "Reactor Trip or Safety Injection".

Proposed Answer: B

Explanation of answer: B is correct because it gives the appropriate series of actions based on a seal leakoff problem when the plant is < P-8 (50%). Isolation of seal return valve after tripping the RCP is one of those actions that is an important interrelationship which is vital to prevent further RCP degradation.
Explanation of distractors:
A – incorrect because it gives the sequence of actions for shutting down an RCP due to high vibrations.
C – incorrect because the plant is < P-8 (50%) setpoint, therefore a reactor trip is not required.
D – incorrect because a reactor trip is not necessary when the plant is < 50% power, where a normal reactor shutdown will be commenced.

Technical Reference(s): OS 1201.01, RCP Malfunction

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the interrelationship between RCP malfunctions and RCP seals.

Question Source: Bank # 20453

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5/10

Comment: Note that this question was swapped with K/A #5 for exam design purposes.

Learning Objective: L1181I03RO Summarize the major actions of OS1201.01.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 36	Group #	1	1
	K/A #	010.K6.03	
	Importance Rating	3.2	3.6

Proposed Question:

Given the following plant conditions:

- The plant is operating at 100% steady state power, with all systems in AUTOMATIC.
- The output of the Pressurizer Master Pressure Controller fails to 40%.

Based on these conditions, what effect will this have on Pressurizer heaters and spray valves?

A. Backup heaters energize, and spray valves open.

B. Backup heaters remain off, and spray valves open.

C. Backup heaters energize, and spray valves remain closed.

D. Backup heaters remain off, and spray valves remain closed.

Proposed Answer: C

Explanation of answer: The master pressure controller output varies from 0 to 100%, with 40% demand acting as if PZR pressure is low. A low failure of the master pressure controller will turn on all PZR heaters in auto and inhibit spray valves in auto.
Explanation of distractors: All combinations of heaters and sprays in the distractors are incorrect.

Technical Reference(s): PPLC Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the effect of a loss or malfunction of the PZR sprays and heaters will have on the PZR PCS.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8027I06RO Describe the auto control functions of the PZR pressure and level control system.

L8027I14RO Explain the auto response of the plant to a high or low failure of the PZR press or level instrument.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 37	Group #	1	1
	K/A #	APE.027.AK1.02	
	Importance Rating	2.8	3.1

Proposed Question:

The following plant conditions exist:

- The unit is at 100% power, steady state conditions.
- A malfunction of the Master Pressurizer Pressure Controller has caused Reactor Coolant system pressure to deviate 80 psig from its normal setpoint.

Given these conditions, which one of the following items will be **MOST** impacted by this malfunction?

A. Volume Control Tank Level.

B. Pressurizer Relief Tank Parameters.

C. Pressurizer Vapor Space Temperature.

D. Overpower Delta Temperature Trip Setpoints.

Proposed Answer: C

Explanation of answer: Since the PZR operates in a saturated condition, a pressure change will result in a change in saturation temperature. Saturation temperature of 2235 psig (NOP) is 652.7 F. An 80# decrease would drop this value to about 648 F. An 80 # increase would raise value to 658 F.
Explanation of distractors:
A - Incorrect because a small change of 80# would have only a minimal impact on charging and letdown flowrate, plus auto level control maintaining PZR level constant means VCT level will be constant.
B - Incorrect because an increase of 80 # is not high enough to lift the PORV so therefore there is no effect on PRT
D – Incorrect because OPDT doesn't use pressure input to calculate the trip setpoint (OTDT does however- common misconception)

Technical Reference(s): PZR Detailed System Text

Proposed references to be provided to applicants during examination: Steam Tables

K/A Topic: Knowledge of the operational implications of the concept of expansion of liquids as temperature increases as applied to PZR Pressure Control Malfunctions.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5/14

Learning Objective: L8027I14RO Explain the auto response of the plant to a high or low failure of the PZR press or level instrument.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 38	Group #	1	1
	K/A #	013.K5.02	
Proposed Question:	Importance Rating	2.9	3.3

Which of the following describes the logic required to initiate the ESF actuations generated by the following signals?

	Containment HI-1 (SI)	Containment HI-2 (MSLI)	Containment HI-3 (CBS/P/CVI)
A.	2 of 3	2 of 3	2 of 3
B.	2 of 4	2 of 4	2 of 3
C.	2 of 3	2 of 4	2 of 4
D.	2 of 3	2 of 3	2 of 4

Proposed Answer: D

Explanation of answer: RP system text, lesson plan, and functional diagram all describe the actuation coincidence for ESF actuations based upon containment pressure. Only D contains the correct combination of system logic.
Explanation of Distractor: Note distractors are combinations of the plausible answers.

Technical Reference(s): RP Detailed System Text, Functional Diagram 1-NHY-50904

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of the safety system logic and reliability as applied to ESFAS.

Question Source: Bank

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8057I08RO For each of the following ESFAS signals, describe how the signal is generated, reset, and blocked.....

Examination Outline Cross-reference:	Level	RO	SRO

	Tier #	2	2
Question # 39	Group #	1	1
	K/A #	013.A4.03	
	Importance Rating	4.5	4.7

Proposed Question:

The following plant conditions exist:

- The plant has sustained a Steam Line Break.
- The reactor has been manually tripped due to INCREASING power range indication.
- Subsequent to the plant trip, the following plant parameters indicate as follows:

  - "A" S/G Pressure is 800 psig and STABLE
  - "B" S/G Pressure is 760 psig and DECREASING
  - "C" S/G Pressure is 800 psig and STABLE
  - "D" S/G Pressure is 800 psig and STABLE
  - RCS Pressure is 1880 psig and DECREASING
  - Containment Pressure is 6.1 psig and INCREASING

Based on these plant conditions and assuming NO additional actions have been taken and all safeguards systems function as designed, which of the following describes the Engineering Safeguard Features Actuations that should have occurred, in addition to Safety Injection?

A. Containment Isolation Phase "A" ONLY.

B. Containment Isolation Phase "A", and Main Steam Isolation ONLY.

C. Containment Isolation Phase "A", Main Steam Isolation, and Containment Spray ONLY.

D. Containment Isolation Phase "A", Main Steam Isolation, Containment Spray, and Containment Isolation Phase "B".

Proposed Answer: B

Explanation of answer: A SLB has occurred in Containment. Containment pressure has increased sufficiently to satisfy the logic for Safety Injection and MSI at 4.3 psig. The fact that the A,C, & D S/G pressures are stable indicates MSI was successful in isolating the SLB to B S/G inside containment. Phase "A" isolation is generated in conjunction with an SI signal

Explanation of Distractors:

A – incorrect because MSI also occurred based on containment pressure > 4.3 psig.

C & D – incorrect because containment pressure did not satisfy the required logic for CBS to occur. The same signal which generates CBS also generates Phase B isolation which would also not occur.

Technical Reference(s): IS & RP Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to manually operate and/or monitor ESFAS initiation in the control room.

Question Source: Bank #23200

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7

Comments:	Minor editorial modifications to bank question.
Learning Objective:	L8057I08RO For each of the following ESFAS signals, describe how the signal is generated, reset, and blocked.....
	L8057I10RO For each of the following ESFAS signals list the actions auto initiated by the signal.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 40	Group #	1	1
	K/A #	022.K4.03	
	Importance Rating	3.6	4.0

Proposed Question:

Given the following plant conditions:

- CGC-V14 and CGC-V28, Containment Structure Purge Isolation Valves are open for surveillance testing.
- All systems are operating as designed.

Based on these conditions, which of the following choices correctly identifies the system response to an automatically generated "T" signal?

A. CGC-V14 and CGC-V28 BOTH CLOSE.

B. CGC-V14 and CGC-V28 BOTH remain OPEN.

C. CGC-V14 remains OPEN and CGC-V28 CLOSES.

D. CGC-V14 CLOSES and CGC-V28 remains OPEN.

Proposed Answer: A

Explanation of answer: It is important that the operator is aware of the potential for releasing airborne activity from containment to the outside environment and should take appropriate actions if the situation warrants. In the case of containment purge, if a "T" signal is generated both valves (CGC-V14 & CGC-V28) will close since they are in parallel paths. (common misconception)
Explanation of distractors:
B,C & D are incorrect because their combinations do not warrant a correct answer.

Technical Reference(s): CHV Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of CCS design feature(s) and/or interlock(s), which provide for automatic containment isolation.

Question Source: Bank #24422

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8038I12RO Describe the Response of the CGC components to a T signal.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Question # 41	Group #	1	1
	K/A #	APE.054.AA1.04	
	Importance Rating	4.4	4.5

Proposed Question:

The following plant conditions exist:

- A Total Loss of Feedwater has occurred.
- The crew is performing FR-H.1, "Response to Loss of Secondary Heat Sink" and are currently at step 5, "Try to Establish SUFP flow to at least one Steam Generator (S/G)".
- Containment conditions are normal.
- Reactor Coolant Pumps are tripped.
- Pressurizer Pressure is 2385 psig.

If pressurizer pressure is greater than or equal to 2385 psig **due to a loss of secondary heat sink**, then bleed and feed cooling should be immediately initiated.

Which of the conditions below support the initiation of bleed and feed cooling using this criteria?

A. SG wide range levels are all 50%
RCS Thot is 570 F and slowly rising
RCS Tcold is 560 F and relatively constant

B. SG wide range levels are all 40%
RCS Thot is 570 F and relatively constant
RCS Tcold is 560 F and relatively constant

C. Two SG wide range levels are 50%, the other two are 5%
RCS Thot is 570 F and relatively constant
RCS Tcold is 560 F and relatively constant

D. Two SG wide range levels are 7%, the other two are 5%
RCS Thot is 570 F and slowly rising
RCS Tcold is 570 F and slowly rising

Proposed Answer: D

Explanation of answer: With the onset of natural circulation (due to securing RCPs) RCS pressure continues to rise and may reach the PORV setpoint. The key to determining if RCS pressure rise is due to loss of heat sink as opposed to natural circulation is loop DT. Loop DT is expected to be large for natural circulation and small for loss of heat sink since there is no heat transfer. Additionally, there is insufficient inventory in the S/G's (3 < 26% WR) which warrants Bleed and Feed conditions. (common misconception)
Explanation of distractors:
A – Incorrect because it does not meet S/G water level criteria & more reflects natural circulation conditions.
B – Incorrect because adverse containment is not applicable and a larger DT more reflects natural circulation.
C – Incorrect because a third S/G would need to be below 26% and a larger DT more reflects natural circulation

Technical Reference(s): FR-H.1 Operator Action Summary, FR-H.1 Background Document

Proposed references to be provided to applicants during examination: None

K/A Topic:	Ability to operate and/or monitor HPI, under total feedwater loss conditions as applied to Loss of Main Feedwater.
Question Source:	Bank #20656
Question Cognitive Level:	Higher
10 CFR Part 55 Content:	55.41.10/14
Learning Objective:	L1211I03RO State the basis for steps, notes, and cautions in FR-H.1

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 42	Group #	1	1
	K/A #	026.A1.01	
Proposed Question:	Importance Rating	3.9	4.2

A Large Break Loss of Coolant Accident occurred. The following specific plant conditions exist:

- Highest Containment Pressure = 35 psig and decreasing.
- RWST Level = 480,000 gallons and decreasing.
- "A" Containment Spray Pump (CS-P-9A) Suction Pressure = 60 psig and decreasing.
- "A" Containment Spray Pump (CS-P-9A) Discharge Pressure = 120 psig and decreasing.
- "B" Containment Spray Pump (CS-P-9B) Suction Pressure = 60 psig and decreasing.
- "B" Containment Spray Pump (CS-P-9B) Discharge Pressure = 265 psig and decreasing.

Based on the above parameters you have determined that which of the following choices is correct?

	Pumping Normally	**Containment Pressure**
A.	CS-P-9A	Did NOT exceed design.
B.	CS-P-9A	Did exceed design.
C.	CS-P-9B	Did NOT exceed design.
D.	CS-P-9B	Did exceed design.

Proposed Answer: C

Explanation of answer: Based on detailed system text and LP information, the design pressure of the CBS pump is 350 psig. As the CBS pumps do not have indicators, it is important (and is the source of common misconceptions) that the candidates be able to discern proper CBS operating characteristics. A computer alarm comes in at 300 psig to warn the operator. A low discharge pressure alarm of 62 psig is also available. With the RWST full the static head of that tank should be about 60 psig which is felt on the suction of the CBS pump. Design containment pressure is 52 psig.
Explanation of distractors:
B & D – are incorrect based on containment pressure not exceeding design alone.
A – design pressure not exceeded is correct, however, the combination of discharge pressure choices makes this choice incorrect.

Technical Reference(s): CBS Detailed System Text, LP8035 CBS System

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict and/or monitor changes in parameters associated with operating the CSS controls including Containment Pressure.

Question Source: Modified Bank #22830 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7/8

Learning Objective: L8035I10RO State the design flowrate and approximate head for the CBS pumps.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 43	Group #	1	1
	K/A #	039.K5.08	
	Importance Rating	3.6	3.6

Proposed Question:

Given the following plant conditions:

- The plant is operating at 12% power.
- Control Bank D rods are at 150 steps withdrawn in MANUAL.
- Main turbine is steady at 800 RPM during turbine startup.
- Main Steam Dumps are set for 1050 psig in the STEAM PRESSURE Mode of control.

Given these conditions, if the main steam dump AUTO setpoint is adjusted to 1040 psig, what effect will this have on Tavg and reactor power assuming NO other operator action?

A. Tavg will RISE, Reactor Power will RISE.

B. Tavg will LOWER, Reactor Power will RISE.

C. Tavg will RISE, Reactor Power will LOWER.

D. Tavg will LOWER, Reactor Power will LOWER.

Proposed Answer: B

Explanation of answer: When the steam dump setpoint is adjusted downward, the control system will automatically attempt to control steam header pressure at the new setpoint by opening the steam dump valves further to reduce pressure. Tc will lower to the new saturation temp for the new lower steam pressure. Tave will lower, and positive reactivity added will cause reactor power to rise. Reactor power follows steam demand.

Technical Reference(s): Reactor Theory Fundamentals, SD Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of the effect of steam removal on reactivity as applied to MRSS.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.1/6

Learning Objective: L8047I06RO Describe the two Modes of operation of the SD system.
L8128I21RO Explain the relationship between reactor power and steam flow.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 44	Group #	1	1
	K/A #	059.A3.02	
	Importance Rating	3.2	3.3

Proposed Question:

Which one of the following describes the functional relationship (with respect to controlling steam generator (S/G) levels) between the Main Feedwater Pumps (MFPs) and the Main Feedwater Regulating Valves (MFRVs) when the unit is ramping from 50% to 100% power?

A. The MFPs maintain a variable differential pressure across the MFRVs, while the MFRVs throttle to maintain a constant S/G water level.

B. The MFPs maintain a constant differential pressure across the MFRVs, while the MFRVs throttle to maintain a variable S/G water level.

C. The MFPs maintain a variable differential pressure across the MFRVs, while the MFRVs throttle to maintain a variable S/G water level.

D. The MFPs maintain a constant differential pressure across the MFRVs, while the MFRVs throttle to maintain a constant S/G water level.

Proposed Answer: A

Explanation of answer: The design of the SGWLC system is to maintain a constant level in the S/Gs at all power levels. The MFW control system however varies the programming to maintain an optimum DP across the MFRVs.
Explanation of distractors:
B,C, & D – All incorrect because the combination of answers makes them so.

Technical Reference(s): LP8046, SGWLC

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to monitor automatic operation of the MFW, including programmed levels of the S/G.

Question Source: Bank #22280

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8046I05RO Describe how an auto feed pump control signal is generated.

L8046I06RO Explain how each of the following process parameters are used in SGWLC.....

L8046I03RO Describe how an auto MFRV control signal is generated.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 45	Group #	1	1
	K/A #	061.K4.14	
	Importance Rating	3.5	3.7

Proposed Question:

Given the following plant conditions and sequence of events:

- The reactor has tripped due to a Main Steam Line Break inside containment from the "C" Steam Generator (S/G).
- Emergency Feedwater (EFW) flow to the "C" S/G is 500 gpm and INCREASING.
- EFW flow to the "A, B, & D" S/Gs is 220 gpm each and STABLE.
- All systems are automatic and functioning as designed.

Based on these conditions, which of the following describes the expected response regarding EFW flow to the "C" S/G?

A. EFW control valves will auto close when flow reaches 525 gpm.

B. EFW flow control valves will modulate to maintain flow at setpoint.

C. EFW flow will be limited to 525 gpm by the venturi in the EFW piping.

D. EFW flow will be limited to 750 gpm by the differential pressure across the flow orifice.

Proposed Answer: A

Explanation of answer: By design downstream of each EFW headers flow venturi there is a flow sensing orifice. DP across this orifice is sensed by two train specific transmitters. When 525 gpm is sensed a signal is sent to its associated MOV flow control valve to isolate feedwater to the effected faulted S/G. The isolation should only occur to the first S/G without operator action. This is to prevent cascading isolation of EFW to all S/Gs. The function of the flow venturi and orifice is a common misconception which is used in the design of the distractors.
Explanation of distractors:
B – incorrect because these particular valves have no modulation feature such as that of the MFRVs. (plausible misconception)
C – incorrect because the venture limits flow to a faulted header to a maximum of 750 gpm if both FCV fail to close.
D – incorrect because the orifice function is to sense a flow of 525 to close the FCV MOV when this high flow is sensed.

Technical Reference(s): L8045, EFW SYSTEM EFW Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of AFW design feature(s) and/or interlock(s) which provide for AFW automatic isolation.

Question History: Modified Bank # 23143 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8045I07RO State the functions of the flow venture and the flow orifice in each S/G EFW supply line.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 46	Group #	1	1
	K/A #	061.GEN 2.1.32	
Proposed Question:	Importance Rating	3.4	3.8

Given the following plant conditions:

- A reactor trip from 100% power has occurred.
- The Balance of Plant Operator is throttling Emergency Feedwater (EFW) flow to all Steam Generators (S/G).

Based on these conditions, which of the following describes the system design limits used to minimize the potential for Loss of Heat Sink when throttling EFW?

A. Throttle or isolate flow to each S/G using Train "A" EFW flow control valve.

B. Throttle or isolate flow to each S/G using Train "B" EFW flow control valve.

C. Throttle or isolate flow to TWO S/Gs using Train "A" EFW flow control valve, and to TWO S/Gs using the Train "B" EFW flow control valve.

D. Throttle EFW flow to each S/G by using ONE EFW flow control valve. Isolate EFW flow to each S/G with the OPPOSITE train EFW flow control valve.

Proposed Answer: C

Explanation of answer: According to EFW system text and the USFAR, it is a precaution to use opposite train EFW control valves so that a loss of power will not interrupt flow to all S/Gs at one time. This would result in a loss of potential flow control to only two S/Gs vs. all four.
Explanation of distractors: All are incorrect because the combination of responses result in a loss of flow to all S/Gs with a loss of all AC power.

Technical Reference(s): EFW Detailed System Text, USFAR Section 6.8

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank / 1998 Seabrook NRC RO #92

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.8/10

Learning Objective: L8045I06RO Describe the train separation requirements for the EFW flow control valves and the basis for these requirements.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 47	Group #	1	1
	K/A #	062.A2.09	
	Importance Rating	2.7	3.0

Proposed Question:

Given the following plant conditions:

- The Unit is at 100% steady state power.
- Multiple VAS alarms are received related to the Unit Auxillary Transformer (UAT).
- The operating crew concludes that an Overcurrent condition has caused the **86UB lockout device** to trip for the "A" Train UAT(s).

Based on these conditions, which of the following correctly predicts the outcome and procedure that the crew should enter?

A. Buses 1, 3, & 5 will have power, the crew should enter E-0, "Reactor Trip or Safety Injection".

B. Buses 1, 3, & 5 will have NO power, the crew should enter E-0, "Reactor Trip or Safety Injection".

C. Buses 1, 3, & 5 will have power, the crew should respond using VAS and Abnormal procedures ONLY.

D. Buses 1, 3, & 5 will have NO power, the crew should respond using VAS and Abnormal procedures ONLY.

Proposed Answer: B

Explanation of answer: The 86UB device senses overcurrent (unique to UAT's only). Once actuated, the turbine will trip, the main generator output breaker will open, the exciter breaker will open, the UAT associated busses supply breakers will open (in this case 13.8 kV for Bus 1, 4160 kV for Bus 3 & 5). Additionally 345 kV PCB's 11 and 163 will open and the affected UAT cooing fans will trip. Since the reactor is above P-9, the turbine trip will cause the reactor to trip. Entry into E-0 in this case is required. The 86UB device once actuated will prevent transfer from the UAT to the RAT backup supply of power, so therefore Bus 1, 3, & 5 will remain de-energized.
Explanation of distractors:
A – incorrect because transfer to RAT power supply will not occur. E-0 is correct procedural reference.
C – incorrect because again the busses will have no power and additionally, although it is plausible the crew could use VAS procedures and eventually AOPs, E-0 would be the overriding procedure once the reactor trips.
D - incorrect because although it is plausible that the busses will have no power, the procedural references are incorrect for the reasons stated above.

Technical Reference(s): 13.8 kV Detailed System Text, L8012I 13.8 kV Lesson

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict the impacts of the consequences of exceeding current limitations on the AC distribution system and based on these predictions, use procedures to correct, control, or mitigate the consequences of these malfunctions or operations.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5/7/8

Learning Objective: L8012I08RO Describe how an electrical fault within a UAT/RAT will effect the station electrical distribution system alignment.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 48	Group #	1	1
	K/A #	062.GEN2.4.1	
	Importance Rating	4.3	4.6

Proposed Question:

Given the following plant conditions and sequence of events:

- The plant is operating at 100% Power.
- "A" Emergency Diesel Generator (EDG) is Danger Tagged out of service for oil pump replacement.
- A **Loss of Offsite Power** occurs.

Based on these conditions, which of the following choices is the correct course of action for the operating crew?

A. Verify reactor and turbine trip in accordance with ECA-0.0, "Loss of All AC Power".

B. Verify reactor trip, turbine trip, and restoration of power to Bus 6 in accordance with E-0, "Reactor Trip or Safety Injection".

C. Verify reactor trip and Bus 6 restoration of emergency power in accordance with OS1246.01, "Loss of Offsite Power - Plant Shutdown".

D. Verify reactor trip, turbine trip, and restoration of power to Bus 6 in accordance with OS1246.02, Loss of a Vital Unit Substation or MCC".

Proposed Answer: B

Explanation of answer: Based on given set of plant conditions, a loss of offsite power will result in a reactor trip due to loss of RCS flow. Therefore E-0 immediate actions must be carried out which include verifying reactor & turbine trip & ensuring power to AC Emergency busses making choice B correct. The operator must distinguish between a loss of offsite power and complete loss of AC power and recognize what entry conditions are required. These are immediate action steps and therefore are required to be performed from memory.
Explanation of distractors:
A – incorrect because a complete loss of AC power did not occur, so therefore it would not be correct to direct entry into ECA-0.0. Direct entry into ECA-0.0 is allowed if a complete loss of onsite and offsite power occurs. In the case of this question, the stem does not state the status of Bus 6 EDG which is assumed to start and load onto bus 6. The operator cannot assume this did or did not happen, it must be checked before direct entering ECA-0.0.
D – incorrect although plausible because the reactor does trip. EOP's take precedence over AOPs when the reactor trips.
C – incorrect although plausible because it contains all of the aspects of the correct answer with the improper procedural reference because EOPs take precedence over AOPs and the reactor/turbine trip is not verified in the AOP.

Technical Reference(s): E-0 immediate actions, ECA-0.0 immediate actions.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of EOP entry conditions and immediate action steps as applied to AC Distribution system.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1202I01RO State symptoms and entry conditions for E-0.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 49	Group #	1	1
	K/A #	063.K3.02	
	Importance Rating	3.5	3.7

Proposed Question:

Which of the following functions of Diesel Generator 1A is lost if 125 VDC bus 11A is de-energized?

A. Engine operating parameter indication.

B. All Engine protective tripping capability.

C. Normal and emergency engine start circuits.

D. Control of DG-1A supply and exhaust fans.

Proposed Answer: C

Explanation of answer: Loss of Vital Bus 11A results in Loss of DG-1A start capability. The test start and emergency start relays cannot be energized with Loss of DC control power.
Explanation of distractors: The control/protection features for a diesel provided from 125vdc is a source of common misconceptions and helps frame the structure of the distractors.
A – incorrect because the EDG will still trip on mechanical overspeed.
B – incorrect because engine operating parameters such as speed ad coolant flows and temperatures can be read on DG instrument panel on the end of the engine.
D - incorrect because control power for the supply & exhaust fans is from MCC-521 versus Bus 11A

Technical Reference(s): OS1248.01 Loss of A Vital DC Bus/DAH Print 310928

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the effect that a loss or malfunction of the DC Electrical Distribution System will have on components using DC Control power.

Question Source: Bank

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7

Learning Objective: L1189I15RO Describe the effects of losing vital DC power on the following equipment---EDGs.

L8020I23RO Describe the importance of 125VDC power for the proper auto operation of the EDG and EPS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 50	Group #	1	1
	K/A #	064.A2.04	
	Importance Rating	2.7	3.0

Proposed Question:

Given the following plant conditions and sequence of events:

- The plant is in Mode 5.
- The crew is making preparations to enter Mode 6 to commence core off-load.
- A Loss of off-site power occurs.

Based on these conditions, which of the following best describes how the Emergency Diesel Generator (EDG) will be unloaded?

Once paralleled with an off-site source, _____

A. open the EDG output breaker ONLY, using the guidance of ECA-0.0, Loss of All AC Power".

B. open the EDG output breaker ONLY, using the guidance of OS1246.01, "Loss of Offsite Power Plant Shutdown".

C. lower KVARS to less than 200 lagging and lower load to 75 to 125 KW, and open the EDG output breaker using the guidance of ECA-0.0, "Loss of All AC Power".

D. lower KVARS to less than 200 lagging and lower load to 75 to 125 KW, and open the EDG output breaker using the guidance of OS1246.01, "Loss of Offsite Power Plant Shutdown".

Proposed Answer: D

Explanation of answer: Since the plant is in Mode 5, OS1246.01 is the applicable procedure during these plant conditions. ECA-0.0 is not applicable in Mode 5 or 6. Attachment C of this procedure states that prior to opening the output breaker the EDG should be unloaded to <200 KVARS and load to 75 to 125 KW.
Explanation of distractors:
A – incorrect because both the methodology and procedure reference are incorrect. Opening the EDG output breaker without unloading could cause equipment damage.
B - incorrect because of methodology, procedural reference is correct.
C – incorrect because although procedural reference is correct the methodology is incorrect.

Technical Reference(s): OS1246.01, Loss of Offsite Power Plant Shutdown, Attachment C

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict the impacts of unloading the EDG prior to securing and based on these predictions, use procedures to correct, control, or mitigate the consequences of this malfunction/operation.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1192I09RO Recognize symptoms or entry conditions for OS1246.01
L1192I10RO Summarize the major actions of OS1246.01.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 51	Group #	1	1
	K/A #	073.A1.01	
	Importance Rating	3.2	3.5

Proposed Question:

Which of the following describes the automatic response of the Waste Gas System to a high radiation reading at the Waste Gas Compressors?

A. Waste Gas Inlet radiation monitor (RM-6503) alarms & WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve, opens.

B. Waste Gas Inlet radiation monitor (RM-6503) alarms & WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve, closes.

C. Waste Gas Outlet radiation monitor (RM-6504) alarms & WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve, opens.

D. Waste Gas Outlet radiation monitor (RM-6504) alarms & WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve, closes.

Proposed Answer: D

Explanation of answer: Attachment A of OS1252.01 provides a listing of all Seabrook Process and Effluent Radiation Monitors as well as automatic functions. For a Waste Gas alarm, RM-6504-1 senses outlet waste gas compressor radiation and functions to close WG-FV-1602.
Explanation of distractors:
A – incorrect because inlet radiation vs outlet and the isolation valve closes vs. opens.
B – incorrect because inlet radiation vs outlet, valve closing is plausible.
C- incorrect because valve closes vs opens, outlet radiation is plausible.

Technical Reference(s): OS1252.01, Process or Effluent High Radiation

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict and/or monitor changes in parameters associated with operating the PRM system controls including radiation levels.

Question Source: Bank #19439

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.12/13

Learning Objective: L8064I07RO Describe the auto actions associated with the Waste Gas radiation monitors upon a high radiation signal.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 52	Group #	1	1
	K/A #	073.A4.01	
	Importance Rating	3.9	3.9

Proposed Question:

The following plant conditions exist:

- A discharge of the Waste Test Tank (WTT) "A" is in progress.
- A high effluent alarm condition occurs on the radiation monitor associated with WTT "A" effluent.

Based on these conditions, which of the following describes the required action **and** associated time requirement in accordance with CP 4.1, "Effluent Surveillance Program"?

A. Resample, 15 minutes.

B. Resample, immediately.

C. Isolate the release, 15 minutes.

D. Isolate the release, immediately.

Proposed Answer: D

Explanation of answer: CP-4.1 states that when a monitor indicates a high level and/or trip, the release shall be terminated immediately. Since this is an immediate action, it is prudent that an operator should know this information closed book without reference to protect the health and safety of the public.
Explanation of distractors: all other combinations of potential answers are incorrect.

Technical Reference(s): CP-4.1 Section 4.2.

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to manually operate and/or monitor effluent release in the control room.

Question Source: Bank #26160

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.12/13

Learning Objective: L1517I10RO Describe the immediate action which must be performed should a process rad monitor identify an alarm condition while a release is in progress.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 53	Group #	1	1
	K/A #	076.K2.01	
	Importance Rating	2.7	2.7

Proposed Question:

Given the following plant conditions and sequence of events:

- Service Water Pumps P-41A and P-41B are running.
- A loss of 345 kV Switchyard has occurred.
- 4160 VAC Bus 5 has been de-energized due to a Bus differential relay actuation.
- Control room operators have initiated a MANUAL safety injection.
- All safety systems are automatically functioning as designed.

Based on these conditions, which of the following choices identifies the Service Water Pump(s) that will be running TWO minutes after the Safety Injection initiated with no operator action?

A. P-41B ONLY.

B. P-41D ONLY.

C. P-41C and D.

D. P-41B and D.

Proposed Answer: A

Explanation of answer: With Bus 5 de-energized, the "A" & "C" SW pumps will not have power, and therefore will not start. With a loss of off-site power, both EDG's will start on under voltage, however, only the Bus 6 EDG will repower its associated bus since Bus 5 has an electrical fault which will prevent its re-energization. Since only P-41B was running, it will restart at the appropriate time in the reloading sequence (HR8) well before the 2 minute point.
Explanation of distractors: In addition the interface of the EPS with the preferred SW pump operation is a common misconception and frames the structure of the distractors.
B- incorrect because both pumps were not running prior to loss of power and SI. This is a plausible distractor because if both were running P-41D would be the preferred pump.
C – incorrect because P-41C is powered from Bus 5 which has no power.
D – incorrect because only one of these pumps will reload automatically unto the bus without operator action. Both pumps are powered from Bus 6.

Technical Reference(s): SW & EDG Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the bus power supplies to Service Water.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8037I06RO Describe the interlocks and auto start/stop or open/closed features associated with the following components---SW pumps.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 54	Group #	1	1
	K/A #	078.K4.02	
	Importance Rating	3.2	3.5

Proposed Question:

Which of the following describes the operation of the Service Air isolation valves, SA-V92 and SA-V93, during an Instrument Air leak?

A. AUTOMATICALLY CLOSE at 90 psig decreasing, AUTOMATICALLY REOPEN above 93 psig INCREASING.

B. AUTOMATICALLY CLOSE at 80 psig decreasing, AUTOMATICALLY REOPEN above 83 psig INCREASING.

C. AUTOMATICALLY CLOSE at 90 psig decreasing, resets to allow MANUAL OPENING above 93 psig INCREASING.

D. AUTOMATICALLY CLOSE at 80 psig decreasing, resets to allow MANUAL OPENING above 83 psig INCREASING.

Proposed Answer: C

Explanation of answer: LP 8023i states that SA-V-92 close < 90 psig and reset > 93 psig. The reset allows manual opening via control switch as specified in ON1242.01.
Explanation of distractors:
A – incorrect because SA-V-92 & 93 do not reopen automatically when pressure is regained.
B & D – incorrect because the listed set-points are inaccurate. Additionally, B is incorrect because valves do not automatically reopen.

Technical Reference(s): ON1242.01, Loss of IA, L8023i Compressed Air Lesson

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the IAS design feature(s) and/or interlock(s) which provide for cross over to other air systems.

Question Source: Bank/ 1998 NRC Exam RO# 34

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.4

Learning Objective: L8023I16RO Describe the impact on plant operation if both IA loops within either compressed air system should become depressurized.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 55	Group #	1	1
	K/A #	103.K3.02	
	Importance Rating	3.8	4.2

Proposed Question:

Which one of the following conditions concerning the Containment System would exceed a Limiting Condition for Operation?

A. In Mode 1, an automatic containment isolation valve is OPEN and fully OPERABLE.

B. In Mode 1, a manual containment isolation valve is OPEN under administrative control.

C. In Mode 2, an automatic containment isolation is SHUT and declared INOPERABLE.

D. In Mode 2, a manual containment isolation valve is SHUT under administrative control.

Proposed Answer: C

Explanation of answer: Per TS 3.6.3, in Modes 1-4, each containment isolation valve shall be OPERABLE.
Explanation of distractors:
A – incorrect because automatic containment isolation valves may be opened as long as they are capable of carrying out their automatic protective functions. Since the valve is operable it can perform its safety function.
B – incorrect because there is an exception at the bottom of the page which allows a containment isolation valve to be open under administrative control.
D – incorrect because it meets no criteria for any LCO. The valve is shut, mentions nothing about operability and is under administrative control.

Technical Reference(s): TS 3.6.1.1 & 3.6.3

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the effect that a loss or malfunction of the containment system will have on Loss of containment.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.9

Learning Objective: L8038I05aRO State from memory an apply the LCO and actions for the following TS's: 3.6.1.1

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 56	Group #	2	2
	K/A #	002.K3.03	
	Importance Rating	4.2	4.6

Proposed Question:

Consider each of the following situations separately during Mode 1 operations with NO operator action.

Over the next 20 minutes, which of the following situations would be expected to raise containment humidity?

A. A sixth Containment Air Handling Fan is started.

B. A #1 seal failure is experienced on a Reactor Coolant Pump.

C. RCS Loop #3 Flow Transmitter (RC-FT-434) low side develops a leak.

D. Letdown Isolation Valve (CS-V-145) inadvertently shuts due to loss of air.

Proposed Answer: C

Explanation of answer: This was an actual event at Seabrook following last refueling outage (OR09). A high temperature coolant leak from a pressurized system when exposed to atmospheric pressure will flash to steam and rapidly increase humidity levels in the containment.
Explanation of distractors:
A – is incorrect because more cooling/heat transfer will actually reduce humidity levels.
B – is incorrect because #1 seal failure could result in high leakoff, however the increased flow will still go to the VCT via seal return line. A portion of the high leakoff might go to #2 seal which is hard piped to the RCDT.
D – is incorrect because letdown flow would isolate which would increase PZR level until the high level reactor trip occurred at 92%. However it would be a long term of no operator action until PORV's lifted and filled the PRT which would eventually rupture and raise containment humidity levels.

Technical Reference(s): L8021i RCS Lesson

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the effect that a loss or malfunction of the RCS will have on the containment.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.5

Learning Objective: L8021I07RO Describe how flow is measure in the RCS loop.

L8021I27RO Describe the principles of operation of the three seals in the RCP seal package.

L1180I04RO Entry conditions for an RCS leak.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 57	Group #	2	2
	K/A #	014.A2.02	
	Importance Rating	3.1	3.6

Proposed Question:

Given the following plant conditions and sequence of events:

- The plant is at 100% power.
- Power is lost to MCC-531.
- Soon afterwards, the Reactor Protection system generates an automatic reactor trip signal resulting in the reactor trip breakers opening.
- The US directs entry into E-0, "Reactor Trip or Safety Injection".

Based on these conditions, which of the following predictions and resultant actions should be performed by the operating crew?

A. Rod bottom lights are lit, decreasing reactor power is verified by the reactor operator and the crew should proceed on to verify turbine trip.

B. Rod bottom lights are NOT lit, the crew should exit the procedure and go to FR-S.1, "Response to Nuclear Power Generation/ATWS", Step 1.

C. Rod bottom lights are NOT lit, but reactor trip is verified by reactor trip breaker open indication and neutron flux levels decreasing. The crew should proceed on to verify turbine trip.

D. Rod bottom lights are lit, the primary side operator should attempt a manual trip using the manual trip switches. Reactor trip is then verified so the crew should proceed on to verify turbine trip.

Proposed Answer: C

Explanation of answer: DRPI indication only is lost. Without MCC-531, rod bottom lights cannot be verified. Positive trip indication is provided by reactor trip breaker position and neutron flux levels. The crew should proceed on to verify turbine trip in E-0. These steps are immediate actions and required to be known from memory.
Explanation of distractors:
A – is incorrect because rod bottom lights are not lit with a loss of MCC-531. Upon reactor trip there is no transfer of power which reenergizes this indication. The remainder of the question is correct and therefore makes it plausible.
B - is incorrect because positive trip indication is provided by reactor trip breaker position and neutron flux levels. The crew should remain in E-0 not transition to FR-S.1.
D - is incorrect because the stem of the question indicates that power is lost to MCC-531 de-energizing DRPI indication so rod bottom lights would not be lit. If rod bottom lights were lit an attempt to manually trip the reactor would not be required.

Technical Reference(s): E-0, Step 1, LP8032i pg 16

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict the impacts of Loss of Power to RPIS and based upon those predictions, use procedures to correct, control, or mitigate the consequences.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content:	55.41.6/10
Learning Objective:	L8032I06RO State the source of power for the DRPI data cabinets and MCB display.
	L1202I03RO State the basis for the following steps, notes, or cautions contained in E-0 IA steps.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 58	Group #	2	2
	K/A #	017.GEN.2.4.24	
	Importance Rating	3.3	3.7

Proposed Question:

The following plant conditions exist:

- The operating crew is responding to FR-C.1, "Inadequate Core Cooling".
- The crew is unable to re-initiate Emergency Core Cooling (ECCS) flow.
- Steam Generator (S/G) depressurization proved ineffective.
- All core exit thermocouples (CETs) indicate >1100 F.
- S/G narrow range levels are all 30% and increasing.
- Reactor Vessel Level Indicating System (RVLIS) indicates 35% and slowly decreasing.
- Reactor Coolant Pump (RCP) seal injection startup criteria cannot be established.

Based on these conditions, which of the following actions are required by FR-C.1?

A. Do not damage RCPs by starting; continue attempts to reestablish ECCS flow.

B. Start RCPs one at a time in any available RCS loop, until CETs are less than 1100 F.

C. Do not damage RCPs by starting; continue attempts to reestablish ECCS and secondary heat sink.

D. Start RCPs one at a time in any available RCS loop, continue operation of one RCP until CETs are less than 725 F.

Proposed Answer: B

Explanation of answer: According to FR-C.1, if CETs are indicating > 1100 F and narrow range S/G water level is > 5% (to avoid S/G tube creep), RCPs should be started one at a time in an attempt to reduce CETs < 1100 F.
Explanation of distractors:
A & C are incorrect because the procedure does direct starting RCPs if there is adequate secondary inventory.
D – incorrect because the procedure loops the operator around until a yes answer to CETs <1100 F is answered or no RCPs are available or until insufficient water is available in the S/Gs.

Technical Reference(s): FR-C.1, Response to Inadequate Core Cooling.

Proposed references to be provided to applicants during examination: Steam Tables

K/A Topic: Knowledge of Loss of Cooling Water Procedures related to in-core temperature monitoring.

Question Source: Bank # 26618

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1227I03RO State the basis for the following steps notes, or cautions from FR-C.1: step 18

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 59	Group #	2	2
	K/A #	028.K2.01	
	Importance Rating	2.5	2.8

Proposed Question:

The hydrogen recombiners (CGC-MM-284A/B) are powered from which of the following power supplies?

A. MCC-521 and 621.

B. MCC-131 and 142.

C. MCC-211 and 261.

D. Unit Substation 53 and 63.

Proposed Answer: A

Explanation of answer: The hydrogen recombiners are installed inside the containment and are powered from safeguards MCC 521 and 621 according to the HV Detailed system text.
B - incorrect because these are turbine building 21' and water treatment non-safety related MCC's
C – incorrect because these are non-safety related MCCs.
D- incorrect because it is not the correct power supply, however plausible since it is safety related.

Technical Reference(s): CHV Detailed System Text, LP8038I Containment CHV

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the bus power supplies to the hydrogen recombiners.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7

Learning Objective: L8038I07RO Describe the operation of the Hydrogen Recombiners.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 60	Group #	2	2
	K/A #	033.A1.01	
	Importance Rating	2.7	3.3

Proposed Question:

The following plant conditions exist:

- The plant is in MODE 1.
- Spent fuel pool water level LO alarms are received in the control room.
- Control room operators enter OS1215.07, "Loss of Spent Fuel Pool Cooling" as directed by alarm response procedures.
- Spent Fuel Pool Level is 21.5 feet and slowly decreasing.

Based on these conditions, which of the following describes a source of **EMERGENCY** makeup to the Spent Fuel Pool?

A. Refueling Water Storage Tank (RWST).

B. Chemical Volume Control System (CVCS).

C. Demineralized Water Storage Tank (DWST).

D. Secondary Component Cooling Water (SCCW).

Proposed Answer: A

Explanation of answer: Answer is directly from procedure. RWST is the source of water in an emergency situation. Attachment A also allows CST gravity feed and Fire Protection system fill.
Explanation of distractors:
B & C – incorrect because these sources are specified for normal makeup methods only.
D – incorrect because SCCW is not an alternative for makeup to the SFP.

Technical Reference(s): OS1215.07

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict and/or monitor changes in parameters associated with SFP cooling system operating the controls including spent fuel pool water level.

Question Source: Modified Bank # 23150 Original question attached to reference.

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1192I08RO List the emergency makeup water sources available to the SFP and the preferred order of use IAW attachment A of OS1215.07.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 61	Group #	2	2
	K/A #	041.A4.07	
	Importance Rating	2.9	3.0

Proposed Question:

The following plant conditions exist:

- The Plant is operating at Full Power.
- Control Rods are Full Out in AUTO.
- A Condenser Steam Dump Valve fails open and a Nuclear System Operator (NSO) is immediately dispatched to locally isolate this steam dump.
- Assume no other operator action occurs.

Given these conditions, what indications will you have in the control room that the NSO has successfully isolated the Condenser Steam Dump?

	Reactor Coolant System Tavg	**Reactor Power**
A.	INCREASES	INCREASES
B.	DECREASES	DECREASES
C.	INCREASES	DECREASES
D.	DECREASES	INCREASES

Proposed Answer: C

Explanation of answer: The failure of the condenser steam dump valve will result in an increase in steam demand which will increase reactor power and result in a drop in Tavg. As the control rods are in the full out position they will not restore RCS temperature. When the operator shuts the steam dump, steam demand will drop and reactor power will decrease. Since not as much energy is being extracted from the RCS, Tavg will increase. There is no valve indication that lets the control room crew know that this valve is shut and this is an operationally relevant situation.
Explanation of distractors: All other combinations are incorrect.

Technical Reference(s): L8124i, Reactivity Coefficients

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to manually operate and/or monitor in the control room remote gagging of stuck open relief valves as related to Steam Dumps.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.1/5

Learning Objective: L8128I21RO Explain the relationship between reactor power and steam flow.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 62	Group #	2	2
	K/A #	045.K4.13	
	Importance Rating	2.6	2.8

Proposed Question:

The following plant conditions exist:

- The plant is operating at 18% power with plant conditions established to support Turbine Overspeed Testing.
- The crew will be performing this test using ON1431.08, "Main Turbine Overspeed Test".
- All Pre-job briefings have occurred and permission has been granted to commence testing.
- The Balance of Plant Operator depresses and holds the OVERSPEED TEST pushbutton.
- Assume all systems function normally as designed.

What Main Turbine Generator System design feature(s), will occur as a result of this action?

When Turbine speed reaches _____

A. 110%, the turbine will automatically trip.

B. 105%, the turbine will automatically trip.

C. 110%, the turbine will be automatically runback.

D. 105%, the turbine will be automatically runback.

Proposed Answer: A

Explanation of answer: Based on procedural reference and lesson plan, the design of the turbine control system is to automatically trip the turbine once the turbine reaches 110% of 1800 RPM. This test is accomplished by holding the Overspeed test pushbutton. Once the turbine automatically trips the pushbutton is released.
Explanation of distractors:
B – incorrect because the turbine trips at 110% vs 105% of rated speed. It is correct that the turbine will auto trip. The 105% value is plausible because during normal operations if turbine speed reaches 105% of rated speed, the control valves will be driven closed by the negative output signal generated by the speed control unit.
C – incorrect because the turbine will auto trip. It is correct that the turbine trips at 110%.
D – Incorrect because again it is an auto trip vs auto runback and trips at 110% vs 105%.

Technical Reference(s): ON1431.08, MT Overspeed Test, L8049, EHC System Lesson, EHC Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the MT/G system design feature(s) and/or interlock(s), which provide for overspeed protection.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.4/8

Learning Objective: L8049I13RO State any interlock associated with the EHC pushbuttons and describe their functions.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 63	Group #	2	2
	K/A #	056.K1.03	
	Importance Rating	2.6	2.6

Proposed Question:

Which one of the following provides water to the shaft seal system for the "A" Main Feedwater Pump?

A. "A" MFP discharge.

B. Steam seal condensate.

C. Condensate storage tank.

D. Condensate pump discharge.

Proposed Answer: D

Explanation of answer: This question was used on the 2003 NRC exam and has been validated to be technically correct.
Explanation of distractors: All distractors are incorrect, however, plausible since they are system related.

Technical Reference(s): FW Detailed System Text pg 8

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the physical connections and/or cause-effect relationships between the condensate system and MFW.

Question Source: Bank / 2003 NRC RO#36

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.4

Learning Objective: L8042I04RO List the 6 components provided sealing or cooling water by the condensate system.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 64	Group #	2	2
	K/A #	068.K6.10	
	Importance Rating	2.5	2.9

Proposed Question:

Given the following list of Process/Effluent Radiation Monitors:

1. R-6509, "Waste Test Tank Discharge Monitor".
2. R-6514, "Waste Liquid Test Tank Inlet Monitor".
3. R-6505, "Condenser Air Evacuation Discharge Monitor".
4. R-6519, "Steam Generator Blowdown Flash Tank Discharge Monitor".
5. R-6516, "Primary Component Cooling Water Loop "A" Activity Monitor".

Which of these radiation monitors if a power failure occurred would result in an automatic system isolation?

A. 1 and 5.

B. 2 and 3.

C. 2 and 4.

D. 3 and 5.

Proposed Answer: C

Explanation of answer: According to OS1252.01 as well as System Lesson L8059i, several process or effluent radiation monitors have automatic control functions associated with them. #1,2 & 4 above fall into this category. A loss of the radiation monitor will result in these control functions occurring since they are fail safe in the alarm condition. C is the only correct combination of radiation monitors listed which provide these automatic functions. These monitors are all associated with the Liquid Radwaste System. This is operationally relevant because operators should be aware of system isolation status if high radiation should occur to protect the health and safety of the general public.
Explanation of distractors: Any other combination of these detectors is either partially correct or does not have an automatic control function associated with it. All choices are plausible in that they are radiation monitors listed in Attachment A of OS1252.01.

Technical Reference(s): OS1252.01, Process or Effluent High Radiation, RDMS Detailed System Text Table 4.2.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the effect of a loss or malfunction on the radiation monitors will have on the liquid radwaste system.

Question Source: Modified Bank #23206 Original question attached to reference.

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.7/13

Learning Objective: L8059I06RO Describe the auto actions (control signals) that result when the below listed monitors reach their alarm setpoints.....

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Question # 65	Group #	2	2
	K/A #	086.K5.04	
	Importance Rating	2.9	3.5

Proposed Question:

Which of the following choices correctly identifies the applicable fire type/class, fire/safety hazard and associated fire protection for the 13.8 kV Unit Auxiliary Transformers?

	Fire Type/Class	Fire/Safety Hazard	Fire Protection
A.	Class A	Explosion	Halon System
B.	Class B	Explosion	Dry-Pipe, Open-Head Sprinkler System
C.	Class C	Electrocution	Halon System
D.	Class D	Electrocution	Dry-Pipe, Open-Head Sprinkler System

Proposed Answer: B

Explanation of answer: B is correct because the UAT(s) contain transformer insulating oil which is flammable. The concern is that an internal electrical fault could produce a high temperature before protective systems could de-energize the UAT causing explosion. Therefore, the transformer is surrounded by a dry-pipe, open-head sprinkler system which will spray water on the UATs when actuated to quickly reduce temperature. According to the definitions of fire types/classes in the Fire Protection Manual, a fire involving energized electrical equipment (Class C) is classified according to construction of components within, so therefore, Class B is most correct because it is a Class III liquid having a flash point above 140 F.
Explanation of distractors:
A - is incorrect but plausible because the fire hazard is an explosion, however, it would not be classified as a type A fire nor is Halon an applicable fire protection system in this case since the UATs are located outside.
C - is incorrect but highly plausible since it could be classified as a Class C fire and electrocution is an applicable safety hazard. The Halon system is what makes this choice clearly incorrect. Since fire protection is applicable system knowledge, this distractor is relevant.
D - is incorrect but again highly plausible because the hazard and fire protection is applicable and valid. However, a Class D fire is clearly incorrect since it does not involve burning of magnesium, sodium etc.....

Technical Reference(s): Fire Protection Manual Glossary, 13.8 kV Detailed System Text.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the operational implications of the hazards to personnel as a result of fire type and methods of protection as applied to fire protection system.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.4

Learning Objective: L8012I10RO Describe the UAT/RAT fire protection schemes.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 66	Group #	1	1
	K/A #	GEN 2.1.1	
	Importance Rating	3.7	3.8

Proposed Question:

In accordance with OP 9.2, Emergency Operating Procedure User's Guide of the Operation Management Manual (OPMM), which of the following is an example of when "skill of the operator" can be used? (assume in all cases permission has been granted from the Unit Supervisor whether correct or incorrect)

A. Lining up for a normal dilution to the Reactor Coolant System.

B. Manual initiation of Safety Injection when the automatic actuation setpoint is exceeded.

C. Carrying out immediate action of E-0, "Reactor Trip or Safety Injection" and all equipment functions as designed.

D. Starting an Emergency Diesel Generator and closing its respective breaker after one failed attempt onto its respective bus during a Loss of Offsite Power event.

Proposed Answer: B

Explanation of answer: OP 9.2 specifically uses this choice as an example of when skill of the operator can be used.
Explanation of distractors:
A – incorrect because besides it being a strictly controlled reactivity evolution, it is not a simple task as defined by OP 9.2. Additionally, the system is not deviating from its design function, so there would be no need for skill of the operator. Procedure use would be required.
C – incorrect because although the operator performs these from memory, this is not skill of the operator related because everything is functioning as designed.
D – incorrect because although a component is not functioning as designed (ie: the EDG should have started), it is not a simple in nature and skill of the operator is not to be used for tasks that are beyond routine. The IA's allow one attempt for an emergency start but do not reference closing breakers. If a breaker does not close initially, this goes into troubleshooting and is beyond skill of the operator.

Technical Reference(s): OP 9.2 OPMM

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1505I23RO Describe a Skill of the operator task and when they might be used.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 67	Group #	1	1
	K/A #	GEN 2.1.7	
	Importance Rating	3.7	4.4

Proposed Question:

The time is 0300 and the plant is in MODE 3 at 555 F, following a trip from 100% power.

The following events occur:

- The Reactor Operator reports that Tavg has begun decreasing at a rate of approximately 1° F/min.
- The Balance of Plant Operator reports that the low setpoint safety valve on the "C" SG has apparently failed partially open.
- An Nuclear Systems Operator is dispatched and visually confirms one of the SG Safety Valves on the "C" SG is passing steam to the atmosphere.
- The Reactor Operator confirms an RCS cooldown rate of 5°F in the last 5 minutes.

Based on these conditions and assuming constant cooldown rate and NO operator action is taken in response to the stuck open safety valve, which of the following conditions will exist by time 0400?

1. Main Steamline isolation.
2. The Tech. Spec. for RCS Cooldown Rate will be exceeded.
3. The actual amount by which the reactor is shutdown will decrease (Shutdown Margin).
4. A PTS challenge to reactor vessel integrity will occur.

A. 1 and 4

B. 2 Only

C. 3 Only

D. 2 and 4

Proposed Answer: C

Explanation of answer: As Tavg is reduced due to the stuck open safety valve, positive reactivity will be inserted into the core from the negative MTC. With rods fully inserted and no change in RCS boron concentration, the actual amount that the reactor is shutdown will decrease over the next hour.
A - is incorrect because with a cooldown rate of 60 F/hr, Tavg will be 495 F by 0400. Steam pressure for this value is 636 psig which is above the MSLI value of 585. Even if the rate of cooldown was enough to actuate the rate compensated portion of MSLI, the answer is still incorrect because A PTS challenge is not a concern as further explained below.
B - is incorrect because the RCS cooldown rate is approximately 60 F/hr. Tech. Specs. limit the RCS cooldown rate to a maximum of 100F/hr in any one hour period. This limit will not be exceeded.
D - is incorrect because the Integrity Critical Safety Function Status Tree (F-0.4) will remain Green over the next hour because the rate of RCS cooldown is < 100F/hr and RCS temperatures will be approximately 495 F at the end of the one hour period.

Technical Reference(s):	UFSAR Chapter 15.1/2, Reactor Theory Fundamentals, TS 3.4.9.2, FR-P.1 Status Tree
Proposed references to be provided to applicants during examination:	Steam Tables, Calculator
K/A Topic:	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Question Source:	Bank
Question Cognitive Level:	Higher
10 CFR Part 55 Content:	55.41 & 55.45
Learning Objective:	L1411I06RO State the core parameter of the most concern for the stuck open steam generator safety valve accident. L8128I28RO Evaluate the change in SDM due to changes in plant parameters.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 68	Group #	1	1
	K/A #	GEN 2.1.28	
	Importance Rating	3.2	3.3

Proposed Question:

Which of the following choices contains ONLY items that are described in 10 CFR 50.46 as acceptance criteria for Emergency Core Cooling System (ECCS) which form the basis for this systems component/control function and design?

I. The total oxidation shall not exceed 17% of the total cladding thickness.
II. The calculated maximum fuel element cladding temperature shall not exceed 2200 F.
III. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
IV. The calculated total amount of hydrogen generated from the zirc-water reaction shall not exceed 0.01 times the amount that would be generated if all the cladding surrounding the fuel were to react.
V. The calculated total oxidation of the cladding shall nowhere exceed 0.03 times the total cladding thickness before oxidation.
VI. After any successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core.
VII. The zirc-water reaction postulated to occur at 1800 F, will not become self-sustaining under any circumstance.

A. I, III, V, VII

B. II, III, IV, VII

C. III, IV, V, VI

D. I, II, III, IV

Proposed Answer: D

Explanation of answer: 10CFR46 paragraph b delineates the acceptance criteria for a LOCA. Seabrook UFSAR contains the same criteria. The only answer which contains ONLY correct criteria is answer D. All other distractors contain either V or VII, neither of which are correct.

Technical Reference(s): UFSAR Section 6.3, Pg 27, ECC Detailed System Text, Pg 1

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.2/6

Learning Objective: L8034I03RO List the five ECCS Acceptance criteria for LOCA's.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 69	Group #	2	2
	K/A #	GEN 2.2.12	
	Importance Rating	3.0	3.4

Proposed Question:

According to surveillance procedure OX1401.02, "RCS Steady State Leak Rate Calculation", which of the following parameters is an input into the calculation for IDENTIFIED RCS leakage?

A. Pressurizer Level

B. Containment Sump Level.

C. Integrated Makeup Total.

D. Pressurizer Relief Tank Level.

Proposed Answer: D

Explanation of answer: This procedure is used by operations every 12 hours to determine sources of RCS leakage. Common misconceptions surrounding sources of IDENTIFIED versus UNIDENTIFIED leakage have in the past led to inaccuracies in these calculations. The surveillance specifies the difference between the two variables in order to comply with Technical Specifications. According to Form B, the PRT is a source of identified RCS leakage.
Explanation of distractors:
All distractors are incorrect however plausible since they are referenced in OX1401.02 as sources of UNIDENTIFIED RCS leakage. The candidate must be able to distinguish between the two sources of which all distractors are credible unidentified sources.

Technical Reference(s): OX1401.02, TS Section 1 Definitions

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of surveillance procedures.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.5/10

Learning Objective: L8010I10RO State the functions of the surveillance requirements.

L8010I03RO Define from memory the following terms as they are found in section 1 of TS----identified & unidentified leakage.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 70	Group #	2	2
	K/A #	GEN 2.2.23	
	Importance Rating	3.6	3.8

Proposed Question:

The following plant conditions exist:

- The plant is in Mode 6.
- Fuel movements between containment and the fuel storage building are in progress.

Which of the following conditions would require core alterations to be suspended?

A. Direct communications between control room and refueling machine are lost.

B. Direct communications between control room and outage control center are lost.

C. Direct communications between refueling machine and fuel storage building are lost.

D. Direct communications between refueling machine and outage control center are lost.

Proposed Answer: A

Explanation of answer: TR 25-3.9.5 requires direct communications maintained between the control room and personnel at the refueling station. At Seabrook we require both RO and SRO candidates to know from memory the less than 1 hour TS. This TS has an immediate action to suspend core alterations if communications is lost.
Explanation of distractors:
All distractors are plausible incorrect answers in that they are areas in which the ability to communicate related to refueling is applicable.

Technical Reference(s): Technical Requirements 25

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to track limiting conditions for operation.

Question Source: Modified Bank #26135 Original question attached to reference.

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.10

Learning Objective: L8060I10 Given a plant situation apply the TS LCO action statements & basis related to refueling operations.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 71	Group #	3	3
	K/A #	GEN 2.3.1	
Proposed Question:	Importance Rating	2.6	3.0

The following radiological conditions exist for a room in the plant:

- General dose rate levels 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.

Based on these conditions what is the radiological posting required for this room and who can authorize an individual to exceed Federal Annual TEDE limits while working in this room during a non-emergency situation?

A. High Radiation Area; Station Director.

B. Technical Specification Locked High Radiation Area; Station Director.

C. High Radiation Area; Nobody can authorize exceeding Federal Limits.

D. Technical Specification Locked High Radiation Area; Nobody can authorize exceeding Federal Limits.

Proposed Answer: D

Explanation of answer: In accordance with TS Admin Controls section 6.11.2, areas accessible to personnel with radiation levels greater than 1000 mr/hr @ 30 cm from the radiation source or from any surface that the radiation penetrates shall be provided with locked doors to prevent unauthorized entry. Additionally, according to the radiation protection manual, approval and extension of approvals are applicable only up to the extent of a workers federal limits of 5000 mrem.
Explanation of distractors: All distractors are combinations of plausible incorrect answers or incorrect conditions.

Technical Reference(s): TS 6.11.1 & 6.11.2, RP 5.1, Pg

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.12/13

Learning Objective: L1525I09RO Define the area designations.....
L1525I13RO State the required approvals for exposure limit upgrades.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 72	Group #	3	3
	K/A #	GEN 2.3.4	
	Importance Rating	2.5	3.1

Proposed Question:

The following plant conditions exist:

- A Site Area Emergency has been declared due to a Loss of Coolant Accident (LOCA) outside containment.
- A pathway to the environment exists.
- Limited makeup to the Refueling Water Storage Tank is available.
- An operator is sent into the area of the leak for local isolation.
- The action has all of the required approvals.
- The action will result in a significant reduction in offsite dose, protecting a large population.

The operator's current TEDE dose is as follows:

- Exposure for the year: 200 mrem
- Current total lifetime exposure: 1200 mrem

Based on these conditions, what is the maximum emergency exposure this operator may receive while performing this action?

A. 4800 mrem TEDE

B. 10000 mrem TEDE

C. 23600 mrem TEDE

D. 25000 mrem TEDE

Proposed Answer: D

Explanation of answer: Emergency exposure limits for life saving or protection of large populations is 25 Rem.
Explanation of distractors:
A – incorrect however plausible since this dose would bring the workers dose to 5 Rem which is the maximum TEDE dose allowed per year (non-emergency situation).
B – incorrect however plausible since this is the emergency dose limit for protection of valuable property.
C - incorrect however plausible since this would bring the worker dose to 25 R for the year, which is the emergency limit, but this limit is independent of previous dose.

Technical Reference(s): ER 4.3, Pg 5-7, Figure 2 & 3.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.12/13

Learning Objective: L1525I13RO State the required approvals for exposure limit upgrades.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 73	Group #	3	3
	K/A #	GEN 2.3.11	
	Importance Rating	2.7	3.2

Proposed Question:

The following plant conditions exist:

- The crew is reducing containment pressure in accordance with OS1023.69, "Containment On-Line Purge (COP) System Operation".
- COP Exhaust Containment Isolation Valves COP-V-3 and COP-V-4 have been opened.
- The crew is establishing COP flow through COP-V-8, COP Exhaust Throttle Valve (Coarse Control).
- RM-6527A-1 and RM-6527A-2, Train "A" COP go into HIGH alarm.
- Assume all systems function as designed.

Based on these conditions, which of the following choices describes how the control room crew will control the radiological release?

A. Control room operators must ensure COP-V-4 automatically closes to stop the release.

B. Control room operators must ensure COP-V-3 and COP-V-8 automatically close to stop the release.

C. Control room operators must ensure COP-V-4 and COP-V-8 automatically close to stop the release.

D. Control room operators must manually close COP-V-3 and COP-V-4 since no automatic actions will occur.

Proposed Answer: A

Explanation of answer: COP Valves (3 & 4) receive an automatic CVI signal to close when high radiation is sensed.
Explanation of distractors:
B – incorrect because COP-V-8 does not receive an auto signal nor does it need to be closed to stop the release.
C – incorrect because again COP-V-8 does not receive a CVI signal. Also in this case COP-V-4 is a Train "B"valve and will not receive a signal.
D - incorrect because COP-V-3 will auto close. It is plausible that COP-V-4 will not receive an auto signal since it is in the "B" Train. Only one valve needs to close to isolate the release.

Technical Reference(s): CHV Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to control radiation releases.

Question Source: Bank #22062

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.41.7/13

Learning Objective: L8059I06RO Describe the auto actions (control signals) that result when the below listed monitors reach their alarm setpoints.....

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 74	Group #	4	4
	K/A #	GEN 2.4.48	
	Importance Rating	3.5	3.8

Proposed Question:

Given the following plant conditions and sequence of events:

- The plant is operating at 75% power.
- A manual boration is directed by the Unit Supervisor.
- All control systems are in AUTOMATIC.
- The Reactor Operator then inadvertently opens the emergency boration valve (CS-V-426) instead of the boration flow control valve (CS-FCV-110A) and then turns the BLENDER MODE START Switch to START which successfully starts a boric acid pump.

Based on these conditions, which of the following symptoms would indicate that CS-V-426 is inadvertently opened instead CS-FCV-110A? (assume the operator took no other action)

A. Tavg remains steady, the makeup batch flow indicator controller (CS-FIQ-111) does NOT sense system flow, control rods begin to step out.

B. Tavg begins to decrease, the makeup batch flow indicator controller (CS-FIQ-111) does sense system flow, control rods begin to step in.

C. Tavg begins to decrease, the makeup batch flow indicator controller (CS-FIQ-111) does NOT sense system flow, control rods begin to step out.

D. Tavg begins to decrease, the makeup batch flow indicator controller (CS-FIQ-111) does sense system flow, control rods begin to step out.

Proposed Answer: C

Explanation of answer: Opening CS-V-426 is the method used in the Rapid Boration AOP and in FR-S.1 to injects large quantities of boric acid directly from the RWST or BWST to the CCP suction. This negative reactivity also causes RCS temperature to drop. As a result of dropping RCS temperature with control rods in normal they will begin to step out once Tavg is 1.5 F is lower than Tref. The impact on the RMUW controls is that it is essentially bypassed and will not sense the boric acid flow, therefore the controllers do not count down the amount of boric acid selected to be added.
Explanation of distractors:
A – incorrect because Tavg drops vs remaining steady. All other portions of answer are plausible.
B – incorrect because rods step out with lowering Tavg. Additionally, the controller does not sense flow.
D – incorrect because the controller does not sense flow. All other portions of answer are plausible,

Technical Reference(s): RMU Detailed System Text, L8025i RMUW System

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question Cognitive Level:	Higher
10 CFR Part 55 Content:	55.41.1/5/6
Learning Objective:	L8025I12RO Describe the operations of the ABB boric acid makeup flow, total makeup flow, and makeup batch flow controllers. L8124I11RO Describe the effects of boration/dilution on reactivity.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Question # 75	Group #	4	4
	K/A #	GEN 2.4.49	
Proposed Question:	Importance Rating	4.0	4.0

The following plant conditions exist:

- The reactor failed to trip on demand.
- The crew has entered FR-S.1,"Response to Nuclear Power Generation/ATWS".
- The Reactor Operator is manually driving control rods inward.
- The Balance of Plant Operator tripped the turbine and noted #2 stop valve indicates 10% open.

Based on these conditions, what procedurally driven action must the crew perform next?

A. Manually runback the turbine.

B. Immediately open the generator breaker.

C. Close the Main Steam Isolation Valves and bypass valves.

D. Open the generator breaker when generator ouput is ZERO Mwe.

Proposed Answer: A

Explanation of answer: Refer to immediate step #2 of FR-S.1. The first item to perform is a turbine runback if the turbine does not trip.
Explanation of Distractors:
B - incorrect because the RNO says to open the output breaker after verifying turbine Mwe at ZERO.
C – incorrect because it is after running back the turbine.
D – incorrect because although a correct step again it is out of order and performed last.

Technical Reference(s): FR-S.1, Step 2

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Question Source: Bank #22661

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.41.10

Learning Objective: L1200I01RO List from memory IA steps including RNOs.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 76	Group #		1
	K/A #	EPE.009.EA2.04	
	Importance Rating		4.0

Proposed Question:

The following plant conditions exist:

- A reactor trip and safety injection have occurred.
- The crew is responding to a Small Break Loss of Coolant Accident (LOCA).
- All Reactor Coolant Pumps are tripped.
- The crew is depressurizing the Reactor Coolant System (RCS) in accordance with Step 11 of ES-1.2, "Post LOCA Cooldown and Depressurization".
- A PORV is used to depressurize the RCS until Pressurizer level is > 25% (50% adverse).

Based on these conditions, as the depressurization occurs what is the expected trend of PZR level and what adverse operating condition may INITIALLY occur as a result?

A. DECREASING PZR Level; Uncovering PZR heaters.

B. INCREASING PZR Level; Water solid conditions in the PZR.

C. INCREASING PZR Level; Upper head region voiding may occur.

D. DECREASING PZR Level; Upper head region voiding may occur.

Proposed Answer: C

Explanation of answer: The caution prior to commencing depressurization in ES-1.2 to refill the PZR, states that a head void may occur as indicated by a increasing PZR level as water in transferred from the RCS to the PZR.
Explanation of distractors:
A – incorrect because PZR will increase during depressurization versus decrease. It is plausible that if the operator does not understand this concept and they believe PZR level will drop that the heaters would become uncovered which is an undesirable condition.
B – incorrect because although it is correct that PZR level will increase and it is plausible that eventually the PZR would go solid, that this would not INITIALLY occur.
D – incorrect because PZR level will increase vs. decrease. The reason is plausible and tests whether the student correctly understands the important concept.

Technical Reference(s): ES-1.2, Pg 9, WOG B/G Document for ES-1.2 caution prior to step 11.

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine or interpret PZR level as applied to a SBLOCA.

Question Source: Modified Bank #4971 Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.5

Learning Objective: L1204I03RO Summarize the major action of ES-1.2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 77	Group #		1
	K/A #	APE015/017.AA2.11	
	Importance Rating		3.8

Proposed Question:

The following plant conditions exist:

- A Loss of Coolant Accident has occurred.
- Currently the crew is performing the steps of FR-C.1, "Response to Inadequate Core Cooling" and preparations are being made to start the "C" Reactor Coolant Pump (RCP).
- Core Exit Thermocouples are approximately 1205 F.
- Steam Generators (S/G) are depressurized to 85 psig.
- Reactor Coolant System pressure is 1400 psig and increasing.
- "C" S/G level is 50% Narrow Range.

Based on these conditions, WHICH of the following choices must the Unit Supervisor require prior to directing a start of the "C" RCP?

	ADEQUATE S/G INVENTORY	**LIFT OIL INTERLOCK**	**SEAL DP > 220 PSID**
A.	Satisfied	Required	Required
B.	Satisfied	Required	NOT Required
C.	NOT Satisfied	Required	NOT Required
D.	NOT Satisfied	NOT Required	NOT Required

Proposed Answer: A

Explanation of answer: Per the note prior to step 18 of FR-C.1, normal conditions in this situation are desired but not required to start RCPs. However, the starting interlocks/permissives must be met which are the lift oil pump and minimum seal differential pressure. Additionally, the SRO should be aware of the S/G tube creep issue of starting RCPs when inadequate inventory in the S/G is available with high RCS temperatures (in this case >1100F) which can lead to S/G tube ruptures. Whether the containment is adverse or not, with S/G NR Level > 25%, S/G inventory is satisfied.
Explanation of distractors:
All distractors are credible since they contain combinations which are similar and are designed to discriminate between competent supervisors.

Technical Reference(s): FR-C.1, Pg 10, RC Detailed System Text

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine and interpret when to jog RCPs during ICC as applied to RCP malfunctions.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.1/5

Learning Objective: L1227I03RO State the basis for the following notes, steps, cautions from FR-C.1.
L8021I33RO Describe the MCB RCP controls including control interlocks.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 78	Group #		1
	K/A #	APE.022.GEN.2.4.44	
	Importance Rating		4.0

Proposed Question:

The following plant conditions exist:

- The Short Term Emergency Director declared a Site Area Emergency at 1215 due to a Loss of make-up capability to the Reactor Coolant System suffered as a result of an armed intruder attack.
- The initial report to the state and local government was completed at 1217.

If an upgrade to a general emergency was declared at 1230, the Protective Action Recommendation (PAR) to the state must be given by which of the following times?

A. 1242

B. 1245

C. 1300

D. 1327

Proposed Answer: B

Explanation of answer: PAR must be made within 15 minutes of declaration of a GE.
Explanation of distractors: All distractors are plausible but incorrect as they are intervals of the given time in the stem.

Technical Reference(s): ER-1.1 pg 3 and ER-1.2 pg 1 of 6

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of emergency plan protective action recommendations.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.1

Learning Objective: L1509I19RO State the time requirement related to the following: PARS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 79	Group #		1
	K/A #	APE027.AA2.15	
	Importance Rating		4.0

Proposed Question:

The following plant conditions and sequence of events:

- Seabrook is operating at 50% power with all control systems in AUTOMATIC.
- The controlling Pressurizer (PZR) Pressure Channel slowly fails high.
- The Reactor Operator reports pressure is 1940 psig and slowly dropping.
- The Unit Supervisor enters OS1201.06, "PZR Pressure Instrument/Component Failure".
- Attempts to close RC-PCV-455A (PZR Spray Valve) are unsuccessful.

Based on these conditions, which of the following actions should you direct as Unit Supervisor?

A. Commence a power reduction, and raise charging flow to compress the PZR bubble.

B. Trip the reactor, initiate safety injection, and enter E-0, "Reactor Trip or Safety Injection", while concurrently tripping the "A" RCP.

C. Energize ALL PZR heaters, commence a rapid power reduction in anticipation of tripping the reactor and stopping ALL Reactor Coolant Pumps.

D. Trip the reactor, when immediate actions of E-0, "Reactor Trip or Safety Injection", Stop the "C" RCP and up to two more RCPs as necessary to stop RCS pressure drop.

Proposed Answer: D

Explanation of answer: A failure of the controlling pressurizer channel high will cause spray valves to open and RCS pressure to drop. According to step 2 of OS1206.01, if spray control has failed and PZR pressure drops to < 1945 psig, the reactor should be tripped and after IA's, the "C" RCP should be stopped. Base on Salem OE, the abnormal has been changed to reflect the fact that additional RCPs may need to be secured to stop the RCS pressure drop.

Explanation of distractors:

A – incorrect because although it might be plausible to reduce power, it is not procedurally driven and because pressure is so low would be conservative to trip the reactor iaw OS1206.01.

B – incorrect although plausible because tripping the reactor is correct and recently Seabrook has aligned ourselves with the industry with regard to initiating SI after tripping the reactor. However, in this case, SI is not required until <1800 psig, so this action would be incorrect. Additionally, the "A" RCP is associated with the "B" Spray Valve (human factoring)

C – incorrect because this is non-procedurally driven and non-conservative in this situation. Additionally, stopping all RCPs is not desirable or correct as forced circulation is better than natural circulation.

Technical Reference(s): OS1201.06 step 2.

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine and interpret actions to be taken if PZR pressure instrument fails high as applied to PZR pressure control malfunctions.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.5

Learning Objective: L1182I05RO Summarize the major actions for OS1201.06.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 80	Group #		1
	K/A #	APE.056.GEN.2.4.7	
	Importance Rating		3.8

Proposed Question:

The following plant conditions exist:

- A Loss of Off-Site Power event is in progress.
- The crew is preparing to perform a cooldown using ES-0.2, "Natural Circulation Cooldown".
- Only one CRDM fan is available.

How does having only one CRDM fan impact the natural circulation cooldown?

A. The crew will go to ES-0.3, "Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS)".

B. The crew will maintain the natural circulation cooldown rate LESS THAN 50 F/hr, and subcooling GREATER THAN 50 F ONLY.

C. The crew will maintain the natural circulation cooldown rate BETWEEN 30 - 50 F/hr, and subcooling BETWEEN 100 – 130 F ONLY.

D. The crew will maintain the natural circulation cooldown rate BETWEEN 30 - 50 F/hr, and subcooling BETWEEN 100 – 130 F and NOT depressurize the RCS for 88 hours.

Proposed Answer: D

Explanation of answer: With less than 2 CRDM fans available, cooling of the upper head and subsequent potential for head voiding is a concern. Therefore additional restrictions are placed on the cooldown and depressurization. D correctly depicts the requirements of ES-0.2 when <2 CRDM fans are running.
Explanation of distractors:
A – incorrect because entry into ES-0.3 is required only if the cooldown and depressurization rate are performed at a rate where a steam void may form in the vessel head. Procedurally there is no transition to this procedure based on insufficient CRDM fans.
B – incorrect because, although these are requirements of ES-0.2, they do not apply to this situation.
C – incorrect however plausible. It is not completely correct however because an 88 hour soak is also required to ensure the heat is transferred across the vessel head, therefore only D contain all the elements of a correct answer.

Technical Reference(s): ES-0.2, Natural Circulation Cooldown.

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of EOP based mitigation strategies.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.1/5

Learning Objective: L1225I14RO State the basis for all procedure steps found in ES-0.2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 81	Group #		1
	K/A #	W/E11.GEN.2.1.9	
	Importance Rating		4.0

Proposed Question:

With the plant at 100% power, the following sequence of events occurs:

- A Loss of Coolant Accident (LOCA) outside containment occurs, resulting in a reactor trip and safety injection.
- The crew is responding using ECA-1.2, "LOCA Outside Containment".
- RWST level is 325,000 gallons and slowly decreasing.
- PZR Level is 65% and slowly decreasing.
- While attempting to isolate the break, the final valve the crew attempts closing to isolate the break is the Residual Heat Removal Train "B" cross-connect valve (RH-V-21).
- After closing RH-V-21, the Reactor Operator reports RCS pressure is still slowly decreasing.

Based on these conditions, which procedure will you as Unit Supervisor transition to from ECA-1.2?

A. ES-1.1, "SI Termination".

B. ES-1.3. "Transfer to Cold Leg Recirculation"

C. E-1, " Loss of Reactor or Secondary Coolant".

D. ECA-1.1, "Loss of Emergency Coolant Recirculation".

Proposed Answer: D

Explanation of answer: D is correct because after isolation of RH-V-21, it should be determined that the break is not isolated because RCS pressure continues to drop and is not increasing. Therefore step 5 of ECA-1.2 requires transition to ECA-1.1.
Explanation if distractors:
A – is incorrect but plausible because if the break was isolated the crew would go to E-1 and eventually to ES-1.1 to terminate SI.
B – incorrect because because during a LOCA outside containment, a loss of recirculation capability exists.
C – incorrect but plausible since step 5 directs transition to E-1 if the break is isolated.

Technical Reference(s): ECA-1.2, Step 5

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to direct personnel activities inside the control room.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.5

Learning Objective: L1209I05RO Summarize the major actions of ECA-1.2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 82	Group #		2
	K/A #	APE.005.GEN.2.1.7	
	Importance Rating		4.4

Proposed Question:

Given the following plant conditions and sequence of events:

- The unit is operating at 100% power at End of Life.
- The Reactor Operator (RO) manually inserts Control Bank "C" during quarterly Rod Operability testing.
- The RO reports that one control rod in Bank "C", Group 1, failed to move inward.
- The affected rod is indicating 228 steps withdrawn on the Digital Rod Position Indication (DRPI) display.
- The Control Bank "C" Group 1 and Group 2 step counters both indicate 215 steps.
- The affected rod is being realigned in accordance with OS1210.06, "Misaligned Control Rod".

What action must be taken if the affected rod will NOT move?

A. Enter OS1202.04, "Rapid Boration" and perform required actions.

B. Trip the reactor and enter E-0, "Reactor Trip or Safety Injection" and perform required actions.

C. Realign the control rods in Control Bank "C", Group1, by withdrawing the unaffected Control Bank "C", Group 1 control rods to 228 steps.

D. Place all lift coil disconnect switches for the affected bank to ROD CONNECTED and enter OS1210.02, "Failure of Control Rod or Rod Bank to Move" and perform required actions.

Proposed Answer: D

Explanation of answer: Misaligned Control Rod (OS1210.06) is entered when one or more rod position indications are not in agreement with the other rod group indications of the same group by 12 steps or more. Step 6c directs the operator to insert the affected rod to the same DRPI position as the group. If this is unsuccessful, the RNO directs entry into OS1210.02 after placing all lift coil switches for affected bank to rod connected.

Explanation of distractors:

A – incorrect because more than one rod would have to fail to insert during a normal shutdown for this to be applicable. It is plausible because it is a caution up front in this procedure.

B – incorrect because more than one rod would have to be misaligned by greater than 48 steps for this to apply. It is plausible because it is mentioned in step 2 of OS1210.06.

C – incorrect because this would be the action taken if the misaligned rod is low versus high which is plausible as it is directed by step 7 of this procedure.

Technical Reference(s): OS1210.06/OS1210.02

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation as related to an Inoperable/Stuck Control Rod.

Question Source: Bank #18561

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.2/5/6

Learning Objective: L1185I10RO Summarize the major actions of OS1210.06.

L1185I12RO For rods misaligned, both high and low as compared to affected bank position summarize the basic recovery method employed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 83	Group #		2
	K/A #	APE.037.AA2.16	
	Importance Rating		4.3

Proposed Question:

Given the following plant conditions and sequence of events:

- The crew has determined that a steam generator (S/G) tube leak exists in both the "A" and "C" S/Gs.
- While processing through OS1227.02, "S/G Tube Leak", the crew determines that they are unable to maintain Pressurizer Level with two Centrifugal Charging pumps running and decide to trip the reactor and actuate safety injection.
- After proceeding through E-0, "Reactor Trip or Safety Injection", the Unit Supervisor transitions to E-3, "S/G Tube Rupture, identifies both "A" and "C" S/Gs are ruptured and proceeds to the step 7, "Initiate RCS Cooldown".
- The Reactor Operator reports S/G Pressures as follows:
    - "A" S/G is 935 psig
    - "B" S/G is 915 psig
    - "C" S/G is 1125 psig
    - "D" S/G is 1100 psig

Based on these conditions, and using the attached step from E-3, what is the target temperature that you will select for the RCS cooldown?

A. 470 F

B. 480 F

C. 485 F

D. 495 F

Proposed Answer: A

Explanation of answer: The referenced step requires the cooldown be performed based on the lowest ruptured S/G pressure. Both A & C are leaking. A S/G has the lowest pressure and so conservatively, the required temperature based on 900 psig is 470 F. In addition the student must understand that interpolation is not allowed.
Explanation of distractor:
B – incorrect however plausible if the operator does not understand that the temperature is based on pressure equal to or less than. It is always conservative to cooldown to the lower value to ensure sufficient subcooling prior or depressuization.
C – incorrect
D – incorrect however plausible because the operator might choose the higher vs lower S/G pressure. This value corresponds to the higher pressure in the "C" S/G.

Technical Reference(s): E-3, Step 7

Proposed references to be provided to applicants during examination: E-3, Step 7

K/A Topic: Ability to determine and interpret the pressure at which to maintain RCS during S/G cooldown as applied to a SGTL.

Question Source:	Bank
Question Cognitive Level:	Higher
10 CFR Part 55 Content:	55.43.1/5
Learning Objective:	L1205I07RO Using the RCS cooldown table in E-3, determine CET based on steam generator pressure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 84	Group #		2
	K/A #	APE.068.AA2.01	
	Importance Rating		4.3

Proposed Question:

The following plant conditions and sequence of events exist:

- The plant is operating at 100% power.
- As the Work Control Supervisor you are observing Remote Safe Shutdown System Monthly Channel Check Surveillance (OX1400.01) being performed on CP-108A.
- The Licensed Operator performing the surveillance notes that Wide Range (WR) Steam Generator (S/G) Water Level in the "A" S/G is reading slightly higher than that indicated on the Main Control Board.
- A plant announcement is made that a Control Room Evacuation is required.
- Actions in accordance with OS1200.02, "Safe Shutdown and Cooldown from the Remote Safe Shutdown Facilities" are being performed.

Using OX1400.01, was "A" WR S/G water level at CP-108A normal prior to control room evacuation and what is the expected level trend immediately following control room evacuation?

	Level Prior to Evacuation	**Level Trend Following Evacuation**
A.	Level was Normal	Trending Down
B.	Level was NOT Normal	No Change in Level
C.	Level was Normal	No Change in Level
D.	Level was NOT Normal	Trending Up

Proposed Answer: A

Explanation of answer: Normal S/G Pressure at 100% power is just under 1000 psig. According to OX1400.01, Figure 1, Remote Safe Shutdown Level indication will be at 100% at 1000 psig. The equivalent MCB Level Indication will be just slightly lower. OS1200.02 requires a plant trip just prior to control room evacuation which results in S/G Shrink and a resultant drop in S/G water level until EFW starts and S/G water levels increase.
Explanation of distractors:
All distractors are combinations of plausible conditions. In B this is an incorrect answer with an incorrect set of conditions. In C this is a plausible correct answer with an incorrect condition. In D this is also a plausible incorrect answer with an incorrect condition.

Technical Reference(s): OX1400.01, Remote Safe Shutdown System Monthly Channel Check
OS1200.02, "Safe Shutdown and Cooldown from the Remote Safe Shutdown Facilities"

Proposed references to be provided to applicants during examination: Applicable portions of OX1400.01

K/A Topic: Ability to determine and interpret S/G level as applied to Control Room Evacuation.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.1/5

Learning Objective: L8210I05RO State the basis of caution, notes, steps of OS1200.02.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Question # 85	Group #		2
	K/A #	W/E03.EA2.1	
	Importance Rating		4.2

Proposed Question:

Given the following plant conditions:

- The unit has tripped and experienced a safety injection.
- While performing ES-1.2, "Post LOCA Cooldown and Depressurization", an ORANGE path condition was noted for the Core Cooling Critical Safety Function.
- FR-C.2, "Response to Degraded Core Cooling" was entered in response to this condition.
- While performing steps of this procedure, the Shift Manager/Shift Technical Advisor reports a RED path condition exists for Core Cooling Critical Safety Function
- Simultaneously a RED path conditions exists for the Containment Critical Safety Function.
- NO other abnormal conditions were noted.

Based on these plant conditions, which of the following is the appropriate action for the Unit Supervisor to take?

A. Complete actions of FR-C.2, and then transition to FR-Z.1.

B. Complete actions of FR-C.2, and then transition to FR-C.1.

C. Stop performing FR-C.2, and immediately transition to FR-Z.1.

D. Stop performing FR-C.2, and immediately transition to FR-C.1.

Proposed Answer: D

Explanation of answer: Per OP-9.2, Rules of usage, when a Red path is encountered, immediately initiate the FRP. Because FR-C.1 is a higher priority than FR-Z.1, the US should proceed to FR-C.1 vs. FR-Z.1.
Explanation of distractors: All distractors are not in accordance with rules of usage, however, plausible if these rules are not understood.

Technical Reference(s): OP-9.2, Section 4.3

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to operate and/or monitor facility conditions and selection of appropriate procedures during abnormal and emergency operations as applied to LOCA cooldown and depressurization

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.5

Learning Objective: L1196I03RO State the priority to be used when given multiple challenged CSFs.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 86	Group #		1
	K/A #	003.GEN.2.2.25	
	Importance Rating		3.7

Proposed Question:

Technical Specification 3.4.1.1 requires all Reactor Coolant loops be in operation in Modes 1 and 2. Which of the following describes the basis for this Technical Specification?

A. To ensure uniform differential temperature across each loop.

B. To maintain DNBR above 1.30 during all normal operations and anticipated transients.

C. To ensure even distribution of flow through the core to prevent exceeding excessive KW/FT limits.

D. To ensure maximum reactor coolant system flow so that boron concentration in all loops is uniform with the reactor vessel and pressurizer.

Proposed Answer: B

Explanation of answer: The plant is designed to operate with all reactor coolant loops in Modes 1 & 2 and to maintain DNBR above 1.30 during all normal and anticipated transients. All distractors are plausible bases for other specifications or precautions for operations.

Technical Reference(s): TS 3.4.1.1 and associated basis

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the bases in TS for LCO and safety limits associated with reactor coolant pumps.

Question Source: Bank #3357

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.2

Learning Objective: L8021I10SRO Describe the basis for TS 3.4.1.1 that requires all RCS loops to be in operation in Modes 1 & 2.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 87	Group #		1
	K/A #	004.A2.06	
Proposed Question:	Importance Rating		4.3

Given the following plant conditions:

- A reactor startup is in progress using OS1000.07, "Approach to Criticality".
- During the first control bank rod withdrawal, VAS alarm; D5972 (Boron Dilution Monitor Train "B") is received.
- The operator reports that the "A" Boron Dilution Monitor is also in alarm.

Given these conditions, what action, if any, should be taken by the Unit Supervisor (US)?

A. Initiate rapid boration in accordance with OS1202.04, "Rapid Boration".

B. Stop the startup and recalculate the estimated critical position using RS1735, "Reactivity Calculations".

C. The alarms may be reset or left in the alarm condition at the discretion of the US and continue with the approach to criticality using OS1000.07.

D. Place the alarm BLOCK switches at the bottom of the Nuclear Instrument cabinet to BLOCK and continue with the approach to criticality using OS1000.07.

Proposed Answer: C

Explanation of answer: OS1000.07 allows these alarms to be left in alarm or reset during the startup. The operator should understand the meaning and impact of these alarms.
Explanation of distractors:
A – incorrect however plausible if the operator does not understand the meaning of the alarm which is really based on monitoring inadvertent dilutions during shutdown conditions. A rapid boration would be the appropriate action to take in a non-startup situation neutron count-rate increase situation.
B – Incorrect but plausible since other actions within the startup procedure would require this action, such as indications of premature criticality etc.
D – incorrect because the alarms are turned off automatically when neutron countrate exceeds 10000 cps.

Technical Reference(s): OS1000.07, Pg 8 Note

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict the impacts of inadvertent boration/dilution on the CVCS and based on those predications, use procedures to correct, control, or mitigate the consequences.

Question Source: Bank #26239

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.5/6

Learning Objective: L1162I03RO Summarize the major steps of OS1000.07.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 88	Group #		1
	K/A #	013.A2.06	
Proposed Question:	Importance Rating		4.0

Given the following plant conditions and sequence of events:

- On UA-50, a "T signal, Train A or Train B Actuation" hardwire alarm is received.
- Concurrently VAS point D7906, "T Signal Actuation" is received.
- All systems function as designed.

Based on these conditions, what procedure should you enter as the Unit Supervisor and what impact would this event cause if no operator action were to occur?

	Procedural Reference	**Predicted Plant Impact**
A.	OS1205.01, "Inadvertent Phase "A" Containment Isolation"	Automatic reactor trip on low PZR level.
B.	OS1205.01, "Inadvertent Phase "A" Containment Isolation"	Automatic reactor trip on high PZR level.
C.	OS1290.01, "Response to HELB Systems Actuation or Malfunction"	Automatic reactor trip on low PZR level.
D.	OS1290.01, "Response to HELB Systems Actuation or Malfunction"	Automatic reactor trip on high PZR level.

Proposed Answer: B

Explanation of answer: These alarms are consistent with the entry conditions for OS1205.01. Either train of isolation will cause letdown to isolate and pressurizer level to increase to the high PZR level reactor trip setpoint.
Explanation of distractors:
A – incorrect because the reactor trips on high vs. low PZR level. It is plausible that the US should enter OS1205.01.
C/D – incorrect because the HELB procedure is not the correct procedure to enter and although letdown will also isolate, PZR will go high vs. low.

Technical Reference(s): OS1205.01 & OS1290.01

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict the impacts of inadvertent ESFAS actuation and based on predications, use procedures to correct, control, or mitigate the consequences.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.5

Learning Objective: L1181I15RO Summarize the major actions of 1205.01.
L1181I14RO Recognize entry conditions for OS1205.01.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 89	Group #		1
	K/A #	059.A2.07	
	Importance Rating		3.3

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- Both Main Feedwater Pumps trip.
- All other Safeguards systems perform as designed.

Based on these plant conditions, which of the following choices correctly states impact and correct mitigation strategy?

A. Plant will trip directly on AMSAC actuation circuitry, Emergency Feedwater will automatically actuate, safety limits are exceeded, enter E-0, "Reactor Trip or Safety Injection".

B. Plant will trip directly on low-low steam generator level, Emergency Feedwater will automatically actuate, NO safety limits are exceeded, enter E-0, "Reactor Trip or Safety Injection".

C. Plant will runback to lower power level within the capacity of Emergency Feedwater which will automatically actuate, safety limits will be exceeded, enter OS1231.03, "Turbine Runback-Setback".

D. Plant will setback to lower power level within the capacity of Emergency Feedwater which will automatically actuate, NO safety limits will be exceeded, enter OS1231.03, "Turbine Runback-Setback".

Proposed Answer: B

Explanation of answer: A loss of both MFPs from 100% power will result in a direct reactor trip on Lo-Lo S/G Level. This trip is designed to provide protection such that no safety limits are exceeded. The correct procedure to mitigate this ANS Condition II event is E-0 which is designed to verify actuation of EFW system to ensure continued heat sink.
Explanation of distractors:
A – incorrect however plausible because the AMSAC system is a backup system designed to provide protection during an ATWS event in the event of a complete loss of feedwater flow. It is also true that EFW will auto actuate since this is part of AMSAC design, no safety limits are exceeded and E-0 is the correct procedure to enter.
C – incorrect however plausible because a runback will occur, however at power levels greater than 85%, the plant cannot sustain a loss of 1 MFP. With both MFP lost, the runback will be overshadowed by loss of S/G inventory and subsequent reactor trip at about the same time EFW auto start occurs. The runback procedure would be correct if the reactor trip did not occur. It is not true that a safety limit will be exceeded.
D – incorrect for most of the reasons above, however even more plausible because it is true that safety limits will not be exceeded.

Technical Reference(s): UFSAR Section 15.2, Decrease in Heat Removal by Secondary System, L1400I Reactor Trips and Bases.

Proposed references to be provided to applicants during examination: None

K/A Topic:	Ability to predict the impacts of tripping of MFW pump turbine on MFW and based on those predictions, use procedures to correct, control, or mitigate the consequences of the loss of MFW pump turbine.
Question Source:	Modified Bank #16317 Original question attached to reference.
Question Cognitive Level:	Higher
10 CFR Part 55 Content:	55.43.1/5
Learning Objective:	L1400I02RO, Describe the basis for each of the reactor trips and the purpose of all permissive and control interlocks. L8056I18RO Given any Reactor trip signal, state an accident or unsafe condition the trip is designed to provide protections against.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 90	Group #		1
	K/A #	061.GEN.2.1.32	
	Importance Rating		3.8

Proposed Question:

The following plant conditions exist:

- The reactor has tripped from 80% power.
- The crew has transitioned to OS1000.11 "Post Trip to Hot Standby"

Which of the following describes the conditions that must be met to BYPASS the Startup Feedwater Pump (SUFP) LOW suction trip?

A. CST makeup capability is unavailable and level is less than 230,000 gallons.

B. CST has been shifted to the lower tap and level is less than 230,000 gallons.

C. CST makeup capability is unavailable and the SUFP is the only method to feed steam generators.

D. CST has been shifted to the lower tap and the SUFP is supplying steam generators via EFW cross-connect.

Proposed Answer: D

Explanation of answer: SRO's are required to know the basis of TS of which 3.7.1.2 and 3.7.1.3 apply. OS1000.11 requires that the SUFP low suction trip may be bypassed only if the CST has been shifted to the lower tap and the SUFP is supplying the S/Gs via cross connect. The Seabrook CST is undersized and <215,000 gallons will cause a SUFP trip if realignment is not made. Realignment to the lower tap and via the cross connect extends the amount of NPSH to complete plant cooldown if necessary. Explanation of distractors: All distractors are plausible since they either contain some combination of the correct answer or they relate to precaution 3.4 which is a different issue.

Technical Reference(s): OS1000.11, pg 4 Precaution 3.3/TS 3.7.1.2/3 Bases

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #24485

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.1/2

Learning Objective: L8062I20RO Describe two actions which must be taken outside the control room in order to use the SUFP as a backup to the EFW pumps.

L1169I03RO Describe the precautions, limitations, and setpoints for OS1000.11.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 91	Group #		2
	K/A #	001.GEN.2.1.22	
Proposed Question:	Importance Rating		3.3

In accordance with OS1000.07, "Approach to Criticality", when is Mode 2 declared?

A. When the reactor is declared critical.

B. When the reactor trip breakers are closed.

C. When the operators commence control bank withdrawal.

D. When the operators commence shutdown bank withdrawal.

Proposed Answer: A

Explanation of answer: OS1000.07 defines Mode 2 declaration when the reactor is critical. As defined by TS definitions, Mode 2 is when Keff is greater than or equal to .99. This is vague and further defined in the startup procedure to provide consistency and conciseness.
Explanation of distractors: All distractors are plausible since they are times or specific items which occur during the reactor startup. Additionally they are items related to the control rod drive system which is part of the K/A.

Technical Reference(s): OS1000.07, Pg 6 Note

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to determine Mode of Operation.

Question Source: Bank # 15682

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.1/2/6

Learning Objective: L1162I03RO Summarize the major steps of OS1000.07.
L8010I03RO Define form memory the following terms as found in TSs: Operational Mode.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 92	Group #		2
	K/A #	015.A2.02	
Proposed Question:	Importance Rating		3.5

The following plant conditions exist:

- Power Range channel N44 was removed from service and its associated bistables tripped by Instrument & Controls (I & C) in accordance with abnormal OS1211.04, "Power Range NI Instrument Failure".
- Due to a seal problem, the crew has just performed a downpower to 47% and removed the "D" RCP from service.
- Power Range channel N41 begins to drift and is currently reading 52%.

Which of the following should be subsequently performed by the crew as directed by the Unit Supervisor?

A. Verify reactor trip and enter E-0, "Reactor Trip or Safety Injection".

B. Continue with the power reduction and notify I & C to check power range channel N41.

C. Perform calorimetric calculation to correct N41 within 2 hours or be in HOT STANDBY within 6 hours.

D. Declare power range channel N41 inoperable and within one hour make preparations to be in MODE 3 within the next 6 hours.

Proposed Answer: A

Explanation of answer: 2 PRNIs are now above P-8, reactor should trip on low RCS flow (1 of 4) - verify reactor trip and go to E-0.
Explanation of distractors:
All distractors are incorrect because the reactor should have tripped.

Technical Reference(s): OS1211.04, RPS Detailed System Text pg 35, TS Table 3.3-1

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict the impacts of faulty or erratic operation of detectors or compensating components on the NIS and based on these predictions, use procedures to correct, control or mitigate the consequences.

Question Source: Bank #22037/2003 NRC #126

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.1/2/5

Learning Objective: L8030I14RO Given the number of operable channels associated with the Rx trip system interlocks, state whether or not it is less than the minimum number of channels required to be operable and if so the required action.

L8030I12RO Given a copy of TS, locate applicable LCOs and list applicable action statements for stated NI channel failures.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Question # 93	Group #		2
	K/A #	034.A2.01	
	Importance Rating		4.4

Proposed Question:

According to Seabrook's Updated Final Safety Analysis (UFSAR), Chapter 15 Accident Analysis, which of the following choices results in the **most limiting fuel handling accident** and what actions would you as a Unit Supervisor take to mitigate the consequences of this event?

A. A dropped spent fuel assembly within an open containment 80 hours following reactor shutdown, perform actions of OS1215.06, "Fuel Handling Accident".

B. A dropped new fuel assembly within the Fuel Storage Building 120 hours following reactor shutdown, perform actions of OS1215.06, "Fuel Handling Accident".

C. A dropped spent fuel assembly within an open containment 120 hours following reactor shutdown, perform actions of OS1252.02, "Airborne High Radiation".

D. A dropped Spent Fuel Cask within the Fuel Storage Building 80 hours following reactor shutdown, perform actions of OS1252.02, "Airborne High Radiation".

Proposed Answer: A

Explanation of answer: According to Chapter 15, section 15.7, the most limiting fuel handling accident is a dropped spent fuel assembly in an open containment 80 hours following shutdown which is the earliest time when spent fuel can be first moved from the reactor vessel. The correct mitigation strategy is OS1215.06, "Fuel Handling Accident".

Explanation of distractors:

B – incorrect but plausible because it was also analyzed in the UFSAR. This accident is not the bounding accident and therefore less limiting. Impacts are less than that of a containment fuel assembly drop. Once again this is the correct procedure.

C – incorrect but plausible because it is the correct accident, however, with more hours since shutdown, it is less limiting. It is possible that entry conditions for this procedure could be met making it even more plausible.\

D- incorrect but plausible because it is one of the analyzed accident in the UFASR. The consequences of this event have minimal impact.

Technical Reference(s): UFSAR Chapter 15.7 Fuel Handling Accidents

Proposed references to be provided to applicants during examination: None

K/A Topic: Ability to predict the impacts of a Dropped Fuel Element on the Fuel Handling System and based on those predictions, use correct procedures to control, correct, or mitigate the consequences of the Dropped Fuel Element.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.1/4/5/7

Learning Objective: L8060I03RO Describe the radiological hazards associated with spent fuel or new fuel.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 94	Group #		1
	K/A #	GEN.2.1.12	
	Importance Rating		4.0

Proposed Question:

With the Plant at 100% power the following conditions exist:

- The Emergency Diesel Generator (EDG) Monthly Surveillance was successfully performed 30 days ago for the "A" EDG.
- Preparation is being made to perform the Monthly Surveillance for the "A" EDG.

Due to a fuel oil leak, the "B" EDG becomes inoperable. Engineering determines that the potential exists for this to be a common Mode failure with the "A" EDG. Engineering will be contacting the vendor, and estimates the common cause failure possibility will be resolved within 48 hours.

Using the Technical Specifications provided, what action, if any, is required with regard to the "A" Emergency Diesel Generator?

A. Do not start the "A" EDG until the "B" EDG is OPERABLE.

B. Restore the "A" EDG to OPERABLE within 2 hours or be in at least HOT STANDBY within the next 6 hours.

C. Start the "A" EDG to satisfy LCO ACTION requirements. Load the "A" EDG to 5600 to 6100 kW for at least 60 minutes.

D. Start the "A" EDG as required by the LCO ACTION requirements. The "A" EDG does not need to be loaded onto the Emergency Bus.

Proposed Answer: C

Explanation of answer: LCO ACTION 3.8.1.1.b action b applies. This requires Surveillance 4.8.1.1.1.a. to be performed, and 4.8.1.1.2a.5 which is to start the "A" EDG and load to between 5600 – 6100 KW.
Explanation of distractors:
A – incorrect but plausible, since the "A" EDG run would not be required if the common cause failure concern could be resolved within 24 hours.
B - incorrect but plausible, since this action is required for two inoperable EDGs, and engineering has reported that the potential exists for a common Mode failure.
D- incorrect but plausible, since the "A" EDG is required to be started, however because it was determined to be a common Mode failure it must be loaded unto its respective bus to satisfy operability requirements of the TS.

Technical Reference(s): Technical Specification 3.8 Electrical Power Systems Pgs 1-4

Proposed references to be provided to applicants during examination: TS-3.8 pages 1-4

K/A Topic: Ability to apply technical specifications for a system.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.2

Learning Objective: L8020I24RO Given a copy of TS apply the TS associated with the EDG system and describe their bases.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 95	Group #		1
	K/A #	GEN.2.1.25	
	Importance Rating		3.1

Proposed Question:

Given the following plant conditions:

- The plant has been shutdown for 10 days.
- Reactor Coolant System (RCS) temperature is 120 F.
- RCS water level is at the (-) 73.5inch elevation with nozzle dams installed.
- The crew has just experienced a loss of the running Residual Heat Removal (RHR) pump and has entered OS1213.02, "Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions".
- A representative from Reliability and Safety Engineering is not available.

Based on these conditions and using the attached reference material, determine the MINIMUM time to boiling?

A. 10 minutes

B. 13 minutes

C. 17 minutes

D. 22 minutes

Proposed Answer: C

Explanation of answer: Two Figures are provided. The correct figure OS1213.02-4 versus 5 must be used. 10 days is 240 hours at an initial RCS temperature of 120 F. the closest answer is about 17 minutes.
Explanation of distractors:
The other times chosen are based on common errors which could occur, such as in choice D which would be plausible if the incorrect figure is referenced or choice B if they mistakenly used 120 hours vs. degrees.

Technical Reference(s): Figure OS1213.02-04/02-05, Time to Boiling vs. Time after Shutdown for RCS water level at Midloop

Proposed references to be provided to applicants during examination: Figure OS1213.02-04/02-05

K/A Topic: Ability to obtain and interpret station reference materials such as graphs, monographs, and tables, which contain performance data.

Question Source: Bank

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.5/6

Learning Objective: L1705I05RO Summarize major actions of OS1213.02.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 96	Group #		2
	K/A #	GEN.2.2.22	
	Importance Rating		4.1

Proposed Question:

Which of the following statements describe the required actions in accordance with Seabrook Technical Specifications if Reactor Coolant System (RCS) pressure reaches 2785 psig while in Mode 2?

A. Restore compliance and be in Mode 3 within 1 hour. Notify NRC within 1 hour. Notify the Site Vice-President and NSARC within 24 hours.

B. Reduce RCS pressure to below safety limit within 5 minutes. Notify NRC within 1 hour. Notify the Site Vice-President and NSARC within 24 hours.

C. Restore compliance and be in Mode 3 within 1 hour. Notify the NRC as soon as possible. Submit a safety limit violation report to NSARC within 1 hour.

D. Reduce RCS pressure to below safety limit within 5 minutes. Notify the NRC as soon as possible. Submit a safety limit violation report to NSARC within 1 hour.

Proposed Answer: A

Explanation of answer: If the RCS safety limit is exceeded TS requires compliance restored and be in Mode 3 within 1 hour. Additionally under the admin section of TS (6.6), it is required to notify the NRC within 1 hour and also inform NSARC and the site VP within 24 hours.
Explanation of distractors:
B – incorrect because although pressure should be promptly restored, the TS only specifies an hour to be in Mode 3 and restore compliance, however the other portions of the answer are plausible. In the bases of TS, if the plant is in Mode 3-5, then pressure must be restored within 5 minutes. The five minutes in these Modes is based on colder temperatures and brittle conditions.
C – incorrect because the notification times are incorrect, however the first part of the answer is plausible.
D – incorrect because of reasons previously stated and this does not contain any of the requirements of the TS.

Technical Reference(s): TS 2.1.2 & 6.6

Proposed references to be provided to applicants during examination: None

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

Question Source: Modified Bank #16007 Original question attached to reference.

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.1/2

Learning Objective: L8010I04RO State from memory the required actions if a safety limit is exceeded.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 97	Group #		2
	K/A #	GEN.2.2.25	
	Importance Rating		3.7

Proposed Question:

Which of the following statements is the bases for Technical Specification 2.1.2, "Reactor Coolant System (RCS) Pressure Safety Limit"?

A. To protect the integrity of the reactor fuel, which prevents the release of radionuclides contained in the RCS to the containment atmosphere.

B. To protect the integrity of the reactor fuel, which prevents the main steam safety valves and reactor protective system actuation from occurring.

C. To protect the integrity of the RCS piping and components, which prevents the release of radionuclides contained in the RCS to the containment atmosphere.

D. To protect the integrity of the RCS piping and components, which prevents the main steam safety valves and reactor protective system actuation from occurring.

Proposed Answer: C

Explanation of answer: Answer C is correct - 2.1.2 Basis
Explanation of distractors:
B – incorrect because both the first part and second part of the answer are incorrect. The answer describes the other safety limit and states the incorrect reason.
C –incorrect because protecting the fuel is the other safety limit at Seabrook. The second part of the answer is plausible as it is the correct reason why we protect the RCS pressure boundary.
D – incorrect because the MSSV's and RPS are designed to prevent safety limits from being exceeded. The first part of the answer is plausible.

Technical Reference(s): TS 2.1.2 bases

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the bases in technical specifications for limiting conditions for operations and safety limits.

Question Source: New

Question Cognitive Level: Lower

10 CFR Part 55 Content: 55.43.2

Learning Objective: L8010I05RO List and describe two safety limits and the protection afforded by each.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 98	Group #		3
	K/A #	GEN.2.3.10	
	Importance Rating		3.3

Proposed Question:

The following plant conditions exist:

- A General Emergency has been declared.
- The Wide Range Gas Monitor (WRGM) is in HIGH Alarm.
- Upper wind direction indicates 45 degrees.
- Offsite Dose Calculation determines PAR Group A is applicable only.

Based **ONLY** on these conditions, use the Emergency Response Manual (ER-1.2D/G) to determine the correct actions to guard against personnel exposure?

Schiller Station Activation	**State Recommendation**
A. REQUIRED	Evacuate 2 mile radius and 5 miles downwind- Shelter all others
B. NOT REQUIRED	Evacuate 2 mile radius and 5 miles downwind- Shelter all others
C. REQUIRED	Evacuate 5 mile radius and 10 miles downwind- Shelter all others
D. NOT REQUIRED	Evacuate 5 mile radius and 10 miles downwind- Shelter all others

Proposed Answer: B

Explanation of answer: Based on ER-1.2D and G, the required outcome of the given set of conditions results in Schiller Station not being activated and Sheltering and Evacuation occurring in the 2-5 mile radius towns only.
Explanation of distractors:
All combinations of distractors are incorrect however plausible since they contain some aspect of valid reponses that would apply given a different set of initial conditions.

Technical Reference(s): Emergency Response Manual ER-1.2D/G

Proposed references to be provided to applicants during examination: ER-1.2D and ER-1.2G

K/A Topic: Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

Question Source: New

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.4/5

Learning Objective: L1509I15SR Given a copy of ER-1.2, determine PAR for a GE using PAR group A & B.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 99	Group #		4
	K/A #	GEN.2.4.11	
	Importance Rating		3.6

Proposed Question:

The plant is operating at 75% power when the following alarms are received:

- D4433 LTOP Train "B" ARMED
- D4434 LTOP Train "A" 100# FROM ACTUATION

Which of the following is the proper abnormal operating procedure to enter in response to this condition?

A. OS1201.08, "Tavg Delta T Instrument Failure" due to loop 2 failing low.

B. OS1247.01, "Loss of a Vital Instrument Panel PP-1A, PP-1B, PP-1C, PP-1D" due to loss of power to PP-1B.

C. OS1201.06, "PT-455-458 PZR Pressure Instrument Failure" due to pressurizer pressure channel P-456 failing high.

D. OS1201.09, "RCS Wide Range Pressure or Temperature Instrument Failure", due to RCS loop 2 wide range Th failing low.

Proposed Answer: D

Explanation of answer: Auctioneered low Th is an input into the PORV control logic. OS1201.09 Attachment A provides a simplified diagram of this logic.
Explanation of distractors:
A – incorrect because RCS loop narrow range temperature channels do not feed PORV control logic.
B – incorrect because loss of PP-1B would result in the opposite train alarms.
C – incorrect because failure high of PZR pressure channel P-456 will not generate the stated conditions.

Technical Reference(s): OS1201.09, Attachment A

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.5

Learning Objective: L1182I07RO Recognize symptoms or entry conditions for OS1201.09.
L1182I08RO Summarize major actions for OS1201.09.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Question # 100	Group #		4
	K/A #	GEN.2.4.15	
	Importance Rating		3.5

Proposed Question:

The following conditions exist:

- An Anticipated Transient Without Scram event occurs.
- PRIOR to the Short Term Emergency Director (STED) evaluating the Emergency Plan, the crew locally trips the reactor.
- The STED determines the plant **was** in a Site Area Emergency and determines that the Site Area Emergency has cleared and the plant is **now** in an Unusual Event.

Which of the following describes how the initial emergency plan notifications should be carried out?

A. Perform notifications at the Unusual Event level, on initial NRC notification inform them that a higher classification level had existed.

B. Perform notifications at the Site Area Emergency level, during initial NRC notifications inform them that the higher classification has cleared.

C. Perform notifications at the Unusual Event level, during follow-up NRC notifications inform them that a higher classification level had existed.

D. Perform notifications at the Site Area Emergency level, during follow-up NRC notifications inform them that the higher classification cleared when the lower one clears.

Proposed Answer: A

Explanation of answer: According to the SSER, ER 1.1 states that if emergency related indications are received and reduced in severity such that the emergency classification went from an earlier higher to a current lower level, the lower level emergency should be declared, therefore the UE is the correct call above. Also this paragraph states that state and NRC notifications shall be made IAW ER 1.2. ER 1.2 states that if an emergency that was terminated or reclassified to a lower level prior to NRC notification, the initial notification shall include the fact that the higher emergency classification existed prior to initial notification.

Explanation of distractors:

B – incorrect because it is declared at a UE vs. SAE. The initial notification portion of the answer is plausible.

C - incorrect because it is on initial vs. follow-up notifications. The UE portion of the answer is plausible.

D – incorrect and not plausible but a consistent mix with the choices.

Technical Reference(s): SSER ER-1.1 pg 4, ER 1.2, pg 4

Proposed references to be provided to applicants during examination: None

K/A Topic: Knowledge of the communication procedures associated with EOP implementation.

Question Source: Modified from 2003 NRC exam. Original question attached to reference.

Question Cognitive Level: Higher

10 CFR Part 55 Content: 55.43.1/5

Learning Objective: L1509I11SR Discuss the actions required if emergency related indications are received and reduced in severity, such that the emergency classification went from an earlier higher level to a current lower level.