U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date: 5/18/15	Facility/Unit: Indian Point Unit 3
Region: I X II ☐ III ☐ IV ☐	Reactor Type: W X CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results		
RO/SRO-Only/Total Examination Values	_____ / _____ / _____	Points
Applicant's Scores	_____ / _____ / _____	Points
Applicant's Grade	_____ / _____ / _____	Percent

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000007EA204 Ability to determine or interpret the following as they apply to a reactor trip: If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP	
Importance:		4.4	4.6

Question: #1

The reactor was initially at 100% power. A transient has occurred which caused reactor protection setpoints to be reached; however, the reactor does NOT trip. All attempts to trip the reactor from the control room have failed. The crew has entered 3-FR-S.1; Response to Nuclear Power Generation / ATWS. The Reactor Operator is manually inserting control rods and an NPO has been dispatched to locally trip the reactor. Westinghouse ATWS Analysis for various Condition II transients provides the bases for the mitigation strategy implemented in 3-FR-S.1. What are the **next two actions** required by 3-FR-S.1 and what is the limiting Condition II transient requiring those actions?

a. Trip the Turbine within 90 seconds and Initiate Emergency Boration. Uncontrolled RCCA Bank Withdrawal.
b. Trip the Turbine within 90 seconds and Verify total AFW flow greater than 365 gpm. Loss of Normal Feedwater.
c. Trip the Turbine within 30 seconds and Initiate Emergency Boration. Uncontrolled RCCA Bank Withdrawal.
d. Trip the Turbine within 30 seconds and Verify total AFW flow greater than 686 gpm. Loss of Normal Feedwater.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the requirement for a turbine trip is 90 seconds. Also plausible for the student to believe that since we are concerned with a loss of subcriticality, that emergency boration is started earlier than verifying aux feed flows. Finally, an Uncontrolled RCCA Bank Withdrawal is a Condition II transient that is analyzed for an ATWS event.
b. Incorrect. Plausible because Plausible because the student may believe that the requirement for a turbine trip is 90 seconds. Also plausible because the two steps of verifying / tripping the turbine and verifying aux feed flow are the next two steps. However, FR-S.1 verifies AFW Flow > 686 gpm, the 365 gpm is the normal AFW Flow verification in

E-0; Reactor Trip or Safety Injection. Finally the Loss of Normal Feedwater is the ATWS event requiring Turbine Trip to maintain SG inventory.

c. Incorrect. Plausible because the first part of the distractor is correct regarding the turbine trip. Also plausible for the student to believe that since we are concerned with a loss of subcriticality, that emergency boration is started earlier than verifying aux feed flows. Finally, an Uncontrolled RCCA Bank Withdrawal is a Condition II transient that is analyzed for an ATWS event.

d. Correct. Based on a loss of normal feedwater ATWS event, it is important to trip the turbine within 30 seconds to maintain SG inventory and verify aux feed flow > 686 gpm within 60 seconds. See Westinghouse Owners Group Emergency Response Guideline for FR-S.1. (Steps 2 and 3 of 3-FR-S.1) (Note that Emergency Boration is step 4)

Technical References:	I3LP-ILO-EOPFRS
Proposed References to be provided:	None
Learning Objective:	Objective 11
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000038EA210 Ability to determine or interpret the following as they apply to a SGTR: Flowpath for charging and letdown flows	
Importance:		3.1	3.3

Question: #2

Given the following conditions:

- The plant has experienced a SGTR coincident with a loss of off-site power.
- All three Diesels started and loaded in an SI plus Blackout Mode.
- The Operators are currently implementing 3-E-3.
- The ruptured SG is isolated.
- The RCS has been cooled down and depressurized to less than the ruptured SG pressure.
- The High Head Safety Injection Pumps have been stopped and placed in AUTO.
- Pressurizer Level is 36%.

What is the current status of Charging and Letdown Flow in accordance with EOP procedure implementation?

a. A Charging Pump was started in E-0; Reactor Trip or Safety Injection and letdown is isolated.
b. A Charging Pump was started in E-0; Reactor Trip or Safety Injection and letdown in service.
c. A Charging Pump was started in E-3; Steam Generator Tube Rupture and letdown is isolated.
d. A Charging Pump was started in E-3; Steam Generator Tube Rupture and letdown is in service.

Answer: a

Explanation / Justification

a. Correct. A Charging Pump is running per step 11 of 3-E-0. Because the Diesels Loaded in SI plus Blackout Mode, the Component Cooling (CCW) pumps are not running. A Charging Pump can still be run however, with City Water providing pump cooling. However, Letdown will remain isolated because CCW has not been established (See Note at the beginning of 3-E-3, Attachment 2; Establishing Letdown – "Letdown should not be placed in service unless charging and CCW have been established."

b. Incorrect. Plausible because the first part of the distractor is correct and the student may believe because PZR Level is 36% (>29%) that letdown flow can be established per step 28 of 3-E-3. However, because component cooling has not been established, letdown will not be placed in service.
c. Incorrect. Plausible because the student may believe that the charging pump is not started until steps 13 and 14 of 3-E-3; Steam Generator Tube Rupture. Finally, it is also plausible because the second part of the distractor is actually true, because component cooling has not been established, letdown will not be placed in service.
d. Incorrect. Plausible because the student may believe that the charging pump is not started until steps 13 and 14 of 3-E-3; Steam Generator Tube Rupture. Also plausible because the student may believe because PZR Level is 36% (>29%) that letdown flow can be established per step 28 of 3-E-3. However, because component cooling has not been established, letdown will not be placed in service.

Technical References:	I3LP-ILO-EOPE30
	System Description 10.0; Engineered Safeguards
Proposed References to be provided:	None
Learning Objective:	Objectives 7, 10, & 16
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000402421 Steam Line Rupture – Excessive Heat Transfer: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	
Importance:		4.0	4.6

Question: #3

Given the following conditions:

- The Unit was operating at 100% Power when a Steam Line Break occurred downstream of the MSIVs.
- A Reactor Trip and Safety Injection were Automatically initiated,
- The MSIVs failed to close.
- The Operators are presently performing step 5 of 3-ECA-2.1; Uncontrolled Depressurization of All Steam Generators.
- An NPO reports that the 33 SG MSIV has just been closed locally.

The STA is monitoring Critical Safety Function Status Trees 30 minutes after the Reactor Trip and observes the following indications:

- Feedwater flow to each Steam Generator is 100 gpm
- All Steam Generator WR Levels are < 9%
- All Steam Generator Pressures are < 100 psig.
- RCS Pressure is 1150 psig.
- All RCS Cold Leg Temperatures are < 240°F
- Intermediate Range SUR = +.1 DPM

Based on the observations made by the STA, the CRS should direct the crew to:

a. Remain in ECA-2.1 until SI is terminated.
b. Transition to E-2; Faulted Steam Generator Isolation.
c. Transition to FR-P.1; Response to Imminent Pressurized Thermal Shock.
d. Transition to FR-S.1; Response to Nuclear Power Generation.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may recognize that 33 SG MSIV has been closed, but also remember that foldout page for ECA-2.1 states that; E-2 transition criteria does not apply while performing SI termination in ECA-2.1 (steps 5 through 16). However, this large steam line break has challenged RCS Integrity and a transition to FR-P.1 is required.
b. Incorrect. Plausible because the student may recognize that 33 SG MSIV has been closed and assume that 33 SG Pressure is now rising, meeting the E-2 transition criteria.
c. Correct. This large steam line break with failure of the MSIVs to close has challenged RCS Integrity and a transition to FR-P.1 is required because of meeting RED Path criteria per Figure F04-1. SRO candidate should be knowledgeable of the main temperature limits on the curve.
d. Incorrect. Plausible because the student may recognize that Intermediate Range SUR > zero in a transition to FR-S.1. However, although Subcriticality is higher in priority than Thermal Shock, the Subcriticality is only an Orange Path. The Red Path Thermal Shock Path will take priority.

Technical References:	I3LP-ILO-EOPE20
	Critical Safety Function Status Trees
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	Modified IP3 Questions # 2907 & # 24135
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000562418 Loss of Off-Site Power: Knowledge of the specific bases for EOPs	
Importance:		3.3	4.0

Question: #4

A major storm has caused a complete loss of off-site power. Con Ed reported that a widespread system blackout has occurred across Westchester County, restoration will take several days. The Reactor has tripped and all three EDGs have started. However, only 31 and 32 EDGs have loaded in a Blackout Mode. A fault exists on the 5A Bus. The plant is now stable with Natural Circulation Cooling established and the crew has transitioned to 3-ES-0.2; Natural Circulation Cooldown.

What RCS Cooldown restrictions exist and what are the bases for those restrictions?

a. Maintain RCS Cooldown rate < 60°F / HR due to having all CRDM Fans running. Before final RCS Depressurization, a soak time of 27 hours is required due to Indian Point being a top hat upper support plate plant.
b. Maintain RCS Cooldown rate < 25°F / HR due to **NOT** having all CRDM Fans running. Before final RCS Depressurization, a soak time of 27 hours is required due to Indian Point being a top hat upper support plate plant.
c. Maintain RCS Cooldown rate < 60°F / HR due to having all CRDM Fans running. Before final RCS Depressurization, a soak time of 29 hours is required due to Indian Point being a flat upper support plate plant.
d. Maintain RCS Cooldown rate < 25°F / HR due to **NOT** having all CRDM Fans running. Before final RCS Depressurization, a soak time of 29 hours is required due to Indian Point being a flat upper support plate plant.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may not remember that all four CRDM Fans are powered from MCC 38 which is associated with the faulted 5A Bus. 60°F / HR is the rate if all CRDM Fans are operating. Second part of distractor is correct, but for less than all CRDM Fans operating.
b. Correct. Bus 5A which feeds MCC 38 (CRDM Fan Power Supply) is faulted and de-energized, therefore no CRDM Fans are available and a cooldown restriction of 25°F / HR is required. Additionally, Indian Point is a top hat upper support plate plant which requires a 27 hour soak prior to the final RCS depressurization (see ES-0.1, attachment 2, step 20).

c. Incorrect. Plausible because Plausible because the student may not remember that all four CRDM Fans are powered from MCC 38 which is associated with the faulted 5A Bus. 60°F / HR is the rate if all CRDM Fans are operating. The second part of the distractor describes the soak time for a flat upper support plate plant, however, Indian Point is a top hat upper support plate design.
d. Incorrect. Plausible because Bus 5A which feeds MCC 38 (CRDM Fan Power Supply) is faulted and de-energized, therefore no CRDM Fans are available and a cooldown restriction of 25°F / HR is required. The second part of the distractor describes the soak time for a flat upper support plate plant, however, Indian Point is a top hat upper support plate design.

Technical References:	I3LP-ILO-EOPE00, 3-ES-0.2
	WOG ERG for ES-0.2
Proposed References to be provided:	None
Learning Objective:	Objectives 7 & 12
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 1 & 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000772225 Generator Voltage and Electric Grid Disturbances; Knowledge of the bases in Technical Specifications or limiting conditions for operations and safety limits	
Importance:		3.2	4.2

Question: #5

Given the following:

- The Unit is at 100% Power
- The System Operator has notified the control room that grid frequency is unstable

What plant protection is designed for grid instabilities resulting in lowering frequency, how is the protection achieved, and what is the bases for needing this protection?

a. A single Reactor Coolant Pump Bus Underfrequency condition > P-8 will directly trip the reactor at < 57.5 HZ. This ensures that protection is provided against violating the DNBR limit due to loss of flow in one or more RCS loops from a major network frequency disturbance.
b. A two out of four Reactor Coolant Bus Underfrequency condition will directly trip the reactor at < 57.5 Hz. This ensures that protection is provided anticipating an actual loss of RCS Flow Condition. This is required because an underfrequency condition will slow down the pumps and reduce the pump coastdown time and therefore reduce reactor heat removal capability.
c. A single Reactor Coolant Pump Bus Underfrequency condition > P-8 will cause the tripping open of the associated Reactor Coolant Pump Breaker at < 57.5 Hz. The Reactor will subsequently trip on a Loss of Reactor Coolant Flow condition. Above P-8, loss of flow in any RCS loop will actuate a reactor trip. This ensures that protection is provided against violating the DNBR limit due to loss of flow in one or more RCS loops.
d. A two out of four Reactor Coolant Bus Underfrequency condition will trip all four Reactor Coolant Pump Breakers at < 57.5 HZ. The Reactor will subsequently trip on the Reactor Coolant Pump Breaker Open Position Protection Trip. This ensures that protection is provided against violating the DNBR limit due to loss of flow in two or more RCS loops from a major network disturbance.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the coincidence for underfrequency is a single loop above P-8, similar to the low RCS loop flow trip. The student may also believe that underfrequency conditions are a direct reactor trip. The bases statement is actually correct for a single loop loss of flow condition.
b. Incorrect. Plausible because the underfrequency protection is a two out of four bus logic. The student may also believe that underfrequency conditions are a direct reactor trip. Finally, the bases statement is correct for an underfrequency condition.
c. Incorrect. Plausible because the student may believe that the coincidence for underfrequency is a single loop above P-8, similar to the low RCS loop flow trip. Also plausible because the underfrequency condition is not a direct trip and does cause the opening of the reactor coolant breakers, however underfrequency is a two out of four logic and the reactor coolant pump breaker open signal will trip the reactor, not loss of RCS loop flow as stated. The bases statement is actually correct for a single loop loss of flow condition.
d. Correct. A two out of four Reactor Coolant Bus Underfrequency condition will trip all four Reactor Coolant Pump Breakers at < 57.5 HZ. The Reactor will subsequently trip on the Reactor Coolant Pump Breaker Open Position Protection Trip. This ensures that protection is provided against violating the DNBR limit due to loss of flow in two or more RCS loops from a major network disturbance.

Technical References:	I3LP-ILO-ICRXP RPS Instrumentation TS Bases – B 3.3.1 #12 System Description 28.0; Overall Unit Protection
Proposed References to be provided:	None
Learning Objective:	Objective E-7
Question Source:	Modified IP3 Question # 25121
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 2

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	00WE11A201 Ability to determine or interpret the following as they apply to the (Loss of Emergency Coolant Recirculation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations	
Importance:		3.4	4.2

Question: #6

Given the following plant conditions:

- 31 RHR Pump is cleared and tagged for motor replacement
- A LOCA coincident with a complete loss of Off-Site Power has occurred
- Emergency Diesel Generators 31 and 33 have started and loaded in SI plus Blackout Mode
- Emergency Diesel Generator 32 has tripped and operators have been unable to restart.
- After starting 31 Recirc Pump per 3-ES-1.3; Transfer to Cold Leg Recirculation, the following alarm was received:
    - o Low Head Injection Line Low Flow
- RWST Level is presently 10 feet and lowering
- Containment Pressure is 25 psig and slowly lowering

Which of the following describes the required procedure transition and required actions based on present plant conditions?

a. Transition to 3-FR-Z-1; Response to High Containment Pressure. An Orange Path termination requires suspension of any E-set procedure in progress and transition to the required FRP. Operators will start available Containment Spray Pumps and Containment Fan Cooler Units as necessary.
b. Transition to 3-ECA-1.1; Loss of Emergency Coolant Recirculation. FRPs are NOT implemented during the performance of ECA-1.1 because the strategy for Containment Spray Pump Operation in the FRP does not take into account conserving RWST inventory like ECA-1.1 does. Operators will stop 32 Containment Spray Pump and operate three Containment Fan Cooler Units.
c. Remain in 3-ES-1.3; Transfer to Cold Leg Recirculation because FRPs are NOT implemented during the performance of ES-1.3 as per the NOTE that states; “FRPs should not be implemented until the transfer to cold leg recirculation has been completed.” 3-ES-1.3 will recognize the loss of recirculation pumps and align 32 RHR Pump for recirculation.

d. Transition to 3-ECA-1.3; Loss of Emergency Coolant Recirculation Caused By Sump Blockage. FRPs are NOT implemented during the performance of ECA-1.3 per guidance in OAP-012; EOP Users Guide. Operators will run one Containment Spray Pump and four Containment Fan Cooler Units.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that transition to the Orange Path FRP is appropriate as containment pressure is > 22 psig. Actual transition will depend on whether a containment spray pump is still running and which procedure transition is appropriate. In this case, ECA-1.3; loss of recirculation caused by sump blockage is appropriate and FRPs are not entered from ECA-1.3.
b. Incorrect. Plausible because the student may believe that because the present plant configuration leaves them with no recirculation capability, then transition to ECA-1.1 is appropriate. Also plausible because the second statement is partially true with respect to the strategy of RWST conservation, however, FRPs are entered from ECA-1.1. Finally, the operators will not stop 32 Containment Spray Pump because it is not running due to the loss of 32 EDG.
c. Incorrect. Plausible because FRPs are not implemented during the performance of ES-1.3; Transfer to Cold Leg Recirculation. Also plausible because if the recirculation pumps were not available, ES-1.3 would use Attachment 3 to align RHR for recirculation. However, indication of the "Low Head Injection Line Low Flow" alarm is a foldout page transition to ECA-1.3 as an indication of sump blockage. Additionally due to the loss of 6A bus (32 EDG trip), 32 RHR Pump is also unavailable.
d. Correct. The indication of the "Low Head Injection Line Low Flow" alarm is a foldout page transition to ECA-1.3 as an indication of sump blockage. FRPs are NOT implemented during the performance of ECA-1.3 per guidance in OAP-012; EOP Users Guide. Operators will run one Containment Spray Pump and four Containment Fan Cooler Units. (See ECA-1.3, steps 2 & 3). Note that step 2.a. will attempt to start all available FCUs (only Fan Cooler 35 is unavailable), however per table (step 3.C.) "all" FCUs are not available so one Containment Spray pump is still required whether three or four FCUs are running.

Technical References:	I3LP-ILO-EOPC10 480 Volt AC Power Distribution Simplified Drawing
Proposed References to be provided:	None
Learning Objective:	Objective 5
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	0000332225 Loss of Intermediate Range Nuclear Instrumentation: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	
Importance:		3.2	4.2

Question: #7

Given the following plant conditions:

- Plant Startup is in progress IAW 3-POP-1.3; Plant Startup from Zero to 45% Power.
- At 4% Power, N35 Intermediate Range Channel was reading significantly higher than its expected current limit, was declared inoperable and the channel was removed from service IAW 3-SOP-NI-001.
- At 8% Power, N36 Intermediate Range Channel has just failed low.

What actions are required per Technical Specifications and what are the bases for those actions?

a. Power operations may continue based on power level being greater than 5% (MODE 1). In MODE 1, Technical Specification 3.0.3 in not applicable for the Intermediate Range Neutron Flux High Reactor Trip. Greater than 5% power the P-6 interlock has already been met and the Power Range Neutron Flux - Low Setpoint Reactor Trip provides protection from a reactivity addition accident.
b. Immediately suspend operations involving positive reactivity additions and reduce THERMAL POWER to < P-6 within 2 hours. At least one Intermediate Range Channel is required to be operable > P-6 and less than P-10 when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup. Below P-6, the Source Range Neutron Flux Trip provides backup core protection for reactivity accidents.
c. Enter Technical Specification 3.0.3 due to both Intermediate Range Channels being inoperable. Action shall be initiated within 1 hour to place the unit in Mode 3 within 7 hours. In MODE 3, the Intermediate Range Neutron Flux Trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition.
d. Within 1 hour, verify interlock (P-6) is in the required state for the existing unit conditions and because the P-6 interlock has already been met, power shall be raised to greater than P-10 within the next 2 hours. Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux Reactor Trip will be blocked making this function no longer

necessary during a power accession. Above the P-10 setpoint, the Power Range Neutron Flux – High Setpoint Trip provides core protection for a rod withdrawal accident.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because student may believe that TS 3.0.3 would apply for both channels being inoperable, but also believe that applicability is only during startup < 5% power. The P-6 interlock has already been met and the Power Range Neutron Flux - Low Setpoint Reactor Trip does provide primary protection from a reactivity addition accident.
b. Correct. Technical Specification 3.3.1, Action F is applicable if less than one channel is operable. The bases are as stated for TS 3.3.1, Action F.
c. Incorrect. Plausible because student may believe that TS 3.0.3 would apply for both channels being inoperable. Mode 3 bases statement is correct for Mode 3.
d. Incorrect. Plausible because this is the correct stated bases for Technical Specification 3.3.1, Action M if one of more channels of (P-6) interlock are inoperable. The fact that the interlock is in the correct state greater than P-6 setpoint and not required is also correct.

Technical References:	I3LP-ILO-ICEXC Technical Specification 3.3.1, Actions F & M TS 3.3.1 Bases Document
Proposed References to be provided:	None
Learning Objective:	Objectives E-7, 8, & 9
Question Source:	Modified IP3 Question # 8739 (added bases info)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 2

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	0000362123 Fuel Handling Incidents: Ability to perform specific system and integrated plant procedures during all modes of plant operation	
Importance:		4.3	4.4

Question: #8

Given the following conditions:

- Refueling Operations are in progress, Full Core Off-Load is being performed.
- A spent fuel assembly has been latched in the manipulator mast and the assembly has just been lifted clear of the reactor vessel flange.
- Visual Observations indicate Refueling Cavity and Spent Fuel Pool Level are both lowering.
- These local visual indications also indicate Refueling Cavity and Spent Fuel Pool are both below Technical Specification Requirements.

The refueling SRO has entered 3-AOP-FH-1; Fuel Damage OR Loss of SFP / Refueling Cavity Level. Which of the following describe the correct initial procedural actions based on the above information?

a. Immediately suspend movement of all irradiated fuel assemblies in accordance with Technical Specifications 3.7.14 and 3.9.6, evacuate all personnel from Containment, initiate Containment Ventilation Isolation (Purge & Pressure Relief Valves Closed), close the fuel transfer canal gate valve and initiate level makeup.
b. Place the suspended fuel assembly in the containment upender, lower and send to the Fuel Storage Building, then close the fuel transfer canal gate valve, and evacuate all personnel from the Fuel Storage Building (FSB) and Containment VC).
c. Close the fuel transfer canal gate valve, place the suspended fuel assembly back in the reactor vessel and evacuate non-essential personnel form the Fuel Storage Building (FSB) and Containment (VC).
d. Place the suspended fuel assembly in the containment upender in the vertical position, evacuate non-essential personnel from the Fuel Storage Building (FSB) and Containment (VC), close the fuel transfer canal gate valve and initiate level makeup.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may be focused on the need to immediately suspend movement of irradiated fuel assemblies per the tech spec actions, however, as stated in the tech spec bases; "this does not preclude movement of a fuel assembly to a safe position." Also plausible because if the assembly is left in the manipulator crane mast with level continuing to lower, actions from the "damaged" fuel assembly section requiring Containment Ventilation Isolation and evacuation are very plausible. The need to makeup is also a subsequent step and therefore additionally plausible.
b. Incorrect. Plausible because depending on how fast level is lowering, the upender would be a second choice per the procedure (See Attachment 2, step 2.5), however, the upender would never be sent back to the FSB. Transfer cart needs to be on containment side to facilitate gate valve closure. The remaining steps are correct, except that all personnel are not evacuated, only "non-essential". Plausible because, all personnel are evacuated for a "damaged" assembly.
c. Correct. See steps 4.27 – 4.30 in body of procedure, Attachments 1 & 2.
d. Incorrect. Plausible because depending on how fast level is lowering, the upender would be a second choice per the procedure (See Attachment 2, step 2.5), however, the upender would be then lowered to the "horizontal" position. Closing the gate valve is correct and the need to makeup is also a subsequent step and therefore additionally plausible.

Technical References:	I3LP-ILO-AOPFH1
	3-AOP-FH-1
Proposed References to be provided:	None
Learning Objective:	Objectives B & E
Question Source:	Modified – IP3 Question # 18590
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5 & 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000060AA202 Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: The possible location of a radioactive-gas leak, with the assistance of PEO, health physics and chemistry personnel	
Importance:		3.1	4.0

Question: #9

Chapter 14 of the FSAR discusses the Accidental Release of Waste Gases as a potential event. The accidental release of waste gases is analyzed assuming a rupture of tanks that accumulate significant quantities of radioactive gases during operation. Waste Gas Operation Procedural precautions and limitations, along with ODCM Limiting Conditions for Operations have been established to ensure analysis assumptions are met.

To ensure a rupture of a Unit 3 Waste Gas Decay Tank remains within analyzed limits, operations personnel will limit the pressure of the gas decay tank to _____ psig and the total curie activity of the gas decay tank to ≤ _______ curies. Chemistry will ensure the curie content restriction is met by a grab sample and analysis prior to a controlled release, but Radiation Monitor _______ is used to alert operators of the potential for reaching the tank curie limit before sampling has occurred. The bases for the curie radioactive content ensures that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed _____ Rem.

a. 110 psig, 50,000 curies, R-20, 0.5
b. 95 psig, 29,761 curies, R-14, 0.5
c. 95 psig, 50,000 curies, R-14, 1.5
d. 110 psig, 29,761 curies, R-20, 1.5

Answer: a

Explanation / Justification

a. Correct. See System Description for Gaseous Waste Disposal System, section 3.1. Also see ODCM Spec D 3.2.6 and associated bases statement. Also see 3-SOP-WDS-002; Gaseous Waste Disposal System Operation Precautions and Limitations.
b. Incorrect. Plausible because the total curie content number is Unit 2's limit for the same ODCM spec. Student may believe that gas decay tanks swap at 95 psig. 0.5 Rem is

correct total dose number and the student could believe that R-14; Plant Vent Radiogas Monitor would be the appropriate.

c. Incorrect. Plausible because the 50,000 curie content number is correct. Student may believe that gas decay tanks swap at 95 psig. 1.5 Rem is the thyroid dose limit and the student could believe that R-14; Plant Vent Radiogas Monitor would be the appropriate.
d. Incorrect. Plausible because the total curie content number is Unit 2's limit for the same ODCM spec. Also 110 psig and R-20 are correct, while 1.5 Rem is the thyroid dose limit.

Technical References:	I3LP-ILO-GWR001 ODCM D3.2.6 & Bases 3-SOP-WDS-002 P&Ls System Description 5.2, section 3.1, page 23
Proposed References to be provided:	None
Learning Objective:	Objectives 11 & 12
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 2

NOTE:
Waste Gas Operation Procedural precautions and limitations, along with ODCM Limiting Conditions for Operations have been established to ensure analysis assumptions are met. These conditions are ensured by operations personnel verifying proper pressure limits are maintained, chemistry conducting periodic grab samples, and health physics monitoring local radiation and airborne contamination monitors.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000074EA202 Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Availability of main or auxiliary feedwater	
Importance:		4.3	4.6

Question: #10

Operators are presently implementing 3-FR-C.1; Response to Inadequate Core Cooling, Given the following conditions:

- Core Exit Thermocouples are 1250°F and slowly rising
- RVLIS Full Range = 30% and slowly lowering
- All Auxiliary Feedwater Flow has been lost
- It will take approximately one hour to restore a source of feedwater
- SG WR Levels are 24%, 23%, 22%, and 25% respectively.
- NO support systems are available for the RCPs
- Attempts to start High Head Safety Injection Pumps were unsuccessful

What recovery and mitigation strategy will be implemented **next** and why?

a. Start Reactor Coolant Pumps (RCPs) because in 3-FR-C.1; Response to Inadequate Core Cooling, RCPs are started even if support systems are unavailable. Starting the RCPs will provide forced two phase flow through the core and temporarily improve core cooling.
b. Immediately transition to 3-FR-H.1; Response to Loss of Secondary Heat Sink due to the Red Path on Heat Sink. This procedure will help expedite the restoration of a source of feedwater while monitoring for the need to initiate Bleed and Feed. The RCPs are to remain stopped in FR-H.1.
c. Open all PRZR PORVs, PORV Block Valves and Reactor Vessel Head Vents to depressurize the RCS in accordance with 3-FR-C.1; Response to Inadequate Core Cooling. The RCPs are not started due to insufficient water level in the steam generators to protect the steam generator tubes from creep rupture.
d. Depressurize ALL Intact SGs to 175 psig in accordance with 3-FR-C.1; Response to Inadequate Core Cooling. The rapid secondary depressurization has been shown to be the most effective way to reduce RCS pressure, allowing the SI accumulators to inject. The RCPs are not started because without proper support conditions, damage will occur to the RCPs, preventing any future use.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because if high head safety injection has not been established and a secondary heat sink (AFW) cannot be established, FR-C.1, step 9 RNO ends you to step 18 which determines whether RCPs should be started. The NOTE before step 18 even states; "Normal conditions are desired but NOT required for starting the RCPs. The remaining statement is correct and from the WOG ERG Bases for the NOTE. However, step 18 also requires SG NR Level > 9%, which we do not have.
b. Incorrect. Plausible because a RED Path for Heat Sink does exist. However, we are implementing a higher priority Functional Restoration Procedure (FRP) in 3-FR-C.1; Inadequate Core Cooling.
c. Correct. If high head safety injection has not been established and a secondary heat sink (AFW) cannot be established, FR-C.1, step 9 RNO ends you to step 18 which determines whether RCPs should be started. However, step 18 also requires SG NR Level > 9%, which we do not have. The WOG ERG Bases states that "RCPs are only started in this step if there is sufficient water level in their associated steam generator to protect the steam generator tubes from creep rupture." Step 18, RNO has you open the PORVs, PORV Block Valves, and the Reactor Vessel Head Vents to depressurize the RCS. See Step 18.b RNO.
d. Incorrect. Plausible because the next step in FR-C.1 (step 19) (after the RCP decision step 18) is to depressurize all intact SGs to atmospheric Pressure. However, the proper **next** mitigation strategy (in order) is to depressurize the RCS first using PORVs and Head Vents.

Technical References:	I3LP-ILO-EOPFRC
	3-FR-C.1 and WOG ERG Bases
Proposed References to be provided:	None
Learning Objective:	Objectives 9 & 15
Question Source:	Modified (IP3 Questions # 24661 & # 12314)
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	003000A201 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPs; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off	
Importance:		3.5	3.9

Question: #11

Given the following plant conditions:

- 100% Reactor Power
- All Control Systems are in AUTO
- RCP NO. 1 SEAL RETURN HIGH LOW FLOW (COMMON) is in alarm
- 33 RCP #1 Seal return flow has been trending upward for the last three days, presently stable at 5.1 gpm
- 33 RCP Seal Inlet Temperature has been stable at 150°F
- The Reactor Operators are recording RCP Seal Return Flow Data per 3-AOP-RCP-1; Reactor Coolant Pump Malfunction

The ATC Reactor Operator has just notified the CRS that #1 Seal Return flow has risen to the maximum indicated flow on the digital recorder. 33 RCP Seal Inlet Temperature is also slowly rising. Which of the following describes how the CRS should direct the crew to respond?

a. Initiate a normal Reactor Shutdown IAW 3-POP-2.1; Operation at Greater Than 45% Power to ensure the 33 RCP can be shut down within 8 hours.
b. Trip the Reactor, Stop 33 RCP, Initiate E-0, Close the associated Spray Valve, and when 33 RCS Loop Flow stabilizes at 20-30%, then Close Seal Return Valve 261C.
c. Trip the Reactor, Stop 33 RCP, Close the associated Spray Valve, when the pump stops rotating close Seal Return Valve 261C and then, Initiate E-0.
d. Continue to monitor 33 RCP Seal Parameters and if Seal Inlet Temperature rises to > 185°F, then Initiate a normal Reactor Shutdown IAW 3-POP-2.1; Operation at Greater Than 45% Power to ensure the 33 RCP can be shut down within 8 hours.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the #1 Seal Leak-off Flow did not exceed 6 gpm. It is also plausible because various RCP abnormal conditions (vibrations, temperatures, etc.) require a shutdown within 8 hours based on vendor manual recommendations.
b. Correct. Step 4.1 IAAT conditions include the following; "#1 seal return flow > 6gpm (5.99 gpm digital). Student needs to recognize that the maximum indicated flow on the digital recorder is 5.99 gpm. See steps 4.2 through 4.9 and then step 4.12 to close affected RCP Seal Return Valve. Note that E-0 is "initiated" after stopping affected RCP.
c. Incorrect. Plausible because the majority of the distractor is correct. See above, but note that E-0 is "initiated" after stopping affected RCP.
d. Incorrect. Plausible because various RCP abnormal conditions (vibrations, temperatures, etc.) require a shutdown within 8 hours based on vendor manual recommendations.

Technical References:	I3LP-ILO-AOPRCP 3-ARP-009 (pages 38-44) 3-AOP-RCP-1 3-SOP-RCS-001 System Description 1.3
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	Modified IP3 Question # 23459
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	0080001217 Component Cooling Water: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	
Importance:		4.4	4.7

Question: #12

Given the following conditions:

- Plant is operating at 100% power
- Annunciator, PROCESS MONITOR HIGH RAD is in alarm
- Annunciator, THERMAL BARRIER COOLING WATER RETURN HIGH TEMP is in alarm
- Annunciator, THERMAL BARRIER CCW HEADER LOW FLOW is in alarm
- AC-FCV-625 indicates closed

Based on the above plant indications, what procedure does the CRS enter, what are the first mitigations steps, and why are those steps taken?

a. The CRS enters 3-AOP-LICCW-1; Leakage into CCW System and directs mitigation through Attachment 1. Attachment 1 directs the crew to momentarily select OPEN on AC-FCV-625 to determine the affected RCP by observing which RCP has the low thermal barrier delta p.
b. The CRS enters 3-AOP-RCP-1; Reactor Coolant Pump Malfunction. The procedure directs the crew to momentarily select OPEN on AC-FCV-625 to determine the affected RCP by observing which RCP has the low thermal barrier delta p.
c. The CRS enters 3-AOP-CCW-1; Loss of Component Cooling and directs mitigation through Attachment 9. Attachment 9 directs the crew to Split CCW Headers to prevent the spread of contamination and assist in exposing the affected RCP when the AC-FCV-625 is subsequently momentarily opened.
d. The CRS enters 3-AOP-LICCW-1; Leakage into CCW System and directs mitigation through Attachment 9. Attachment 9 directs the crew to Split CCW Headers to prevent the spread of contamination and assist in exposing the affected RCP when the AC-FCV-625 is subsequently momentarily opened.

Answer: a

Explanation / Justification

a. Correct. These alarms and the fact that the AC-FCV-625 is closed are all indications of a RCP Thermal Barrier leak. Therefore, the CRS would enter AOP-LICCW, Leakage into CCW System and utilize Attachment 1 to determine which RCP thermal barrier heat exchanger is leaking and isolate it. Attachment 1 directs the crew to momentarily select OPEN on AC-FCV-625 to determine the affected RCP by observing which RCP has the low thermal barrier delta p. See bases document for AOP-LICCW, step 1.2; "determines the affected RCP by momentarily reinitiating CCW flow and observing which RCP has the low thermal barrier delta p."
b. Incorrect. Plausible because the student may believe that RCP Thermal Barrier leaks are handled in AOP-RCP; RCP Malfunctions. Additionally plausible because the remainder of the mitigation strategy and bases are true.
c. Incorrect. Plausible because the student may believe that RCP Thermal Barrier loss is covered by AOP-CCW; Loss of CCW. Second part of distractor is an action from AOP-LICCW to split the headers if the PROCESS MONITOR HIGH RAD is in alarm and it is caused by R17 (CCW High Activity). The student can easily infer that the alarm was caused by R17, then AOP-LICCW actually has an attachment 9 that splits the headers and the bases document for step 4.5 of the procedure states; "Splitting the headers would help prevent the spread of contamination and assist in exposing the affected component."
d. Incorrect. Plausible because AOP-LICCW is the proper mitigating procedure and the second part of distractor is an action from AOP-LICCW to split the headers if the PROCESS MONITOR HIGH RAD is in alarm and it is caused by R17 (CCW High Activity). The student can easily infer that the alarm was caused by R17, then AOP-LICCW actually has an attachment 9 that splits the headers and the bases document for step 4.5 of the procedure states; "Splitting the headers would help prevent the spread of contamination and assist in exposing the affected component." However, the NOTE before step 4.5 states that "it is NOT necessary to split headers if leak has been identified and is isolable." Clearly based on the given information, a RCP Thermal Barrier Leak exists and splitting the headers does not help the mitigation strategy of Attachment 1 of the procedure. The CRS knowing this would not elect to use Attachment 9 and split the headers.

Technical References:	I3LP-ILO-AOPLIC 3-ARP-010 3-ARP-009 3-AOP-LICCW-1 & Bases
Proposed References to be provided:	None
Learning Objective:	Objectives 1, 2, & 3
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	039000A204 Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump	
Importance:		3.4	3.7

Question: #13

The plant has been operating at steady state 100% power for the last 90 days. The ATC Reactor Operator monitoring the control board notices the following indications:

- Reactor Power is 100.5% and slowly rising
- RCS T_{avg} is 569°F and slowly lowering
- Rods are at their fully withdrawn position of 230 steps
- A red light is lit for Group 1 Steam Dump Valves on the flight panel (FCF)

The CRS has entered 3-UC-1; Uncontrolled Cooldown. What is the appropriate CRS crew direction for these indications IAW the procedure?

a. Trip the Reactor, Close the MSIVs, and GO TO 3-E-0; Reactor Trip or Safety Injection.
b. Stop any inward rod motion, stop any turbine load increase in progress, initiate a dilution to restore T_{avg} to 1.5°F of program, and dispatch an operator to manually isolate all three steam dump valves in Group 1.
c. Place rods in Manual, stop any turbine load increase in progress, reduce turbine load to maintain reactor power ≤ 100%, stop any boration in progress, maintain turbine load as necessary to maintain T_{avg} within 4°F of T_{ref}, and dispatch an operator to manually isolate affected valves.
d. Trip the Reactor, Close the MSIVs, and INITIATE 3-E-0; Reactor Trip or Safety Injection, while continuing with the implementation of Attachment 1; Investigation Checklist of 3-UC-1.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because this would be the action for a locally un-isolable steam leak. This is actually steps 4.1 through 4.4 of the procedure. However, control board indications

show a malfunctioning / open high pressure steam dump valve, this could be isolated locally.

b. Incorrect. Plausible because the first two actions are correct per steps 4.6 & 4.8 of the procedure, however the procedure does not direct restoring T_{avg} via a dilution. Although this is still plausible as raising temperature to program will add negative reactivity. Student may also believe that isolation of all three group 1 valves is necessary based on the indication. However, the indicator for group 1 turning red may only be one valve.
c. Correct. Information in the stem indicates a malfunctioning / open high pressure steam dump valve and therefore this would be locally isolable. Rods are placed in Manual per step 4.7 due to rods being at 230 steps, then step 4.8 stops any turbine load increases in progress, step 4.9 reduces turbine load to maintain reactor power $\leq$ 100%, step 4.10 stops any boration in progress, and step 4.15 maintains turbine load as necessary to maintain T_{avg} within 4°F of T_{ref} (because rods are in manual). Finally, step 4.19 RNO closes any affected valves.
d. Incorrect. Plausible because this would be the action for a locally un-isolable steam leak, except that 3-UC-1 states GO TO 3-E-0, not INITIATE. But also plausible because a number of AOPs do use the term INITIATE so that supplemental actions can be completed. In this case the student would believe that the malfunctioning steam dump valve would still require isolation via 3-UC-1, Attachment 1.

Technical References:	I3LP-ILO-AOPUC1
	3-AOP-UC-1 & Bases
	System Description 18.1
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

SRO: Requires knowledge of specific content, verses knowledge of the procedure's overall mitigative strategy or purpose.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	061000A204 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: pump failure or improper operation	
Importance:		3.4	3.8

Question: #14

During a plant startup, feedwater control has just been transferred to the Low Flow Bypass Feed Regulation Valves with 31 MBFP in service. Testing was done on 32 MBFP an hour ago that left the BFP No. 32 Turbine Trip and Reset on Panel FAF indicating a Red Flag and Green Light.

While aligning the AFW system for standby IAW 3-SOP-AFW-001; Auxiliary Feedwater System Operation, the Reactor Operator stops 31 AFW Pump and places the switch in Auto. Which of the following describes the response of 31 AFW pump and any required response?

a. 31 AFW Pump will re-start after a time delay due to the flag not being matched on 32 MBFP. Since this is an un-planned actuation of a safeguards system, an 8 hour notification to the NRC IAW 10CFR50.72 will be required.
b. 31 AFW Pump will re-start after a time delay due to the flag not being matched on 32 MBFP. Since the safeguard system actuation was not caused by an actual loss of feedwater, no immediate notification to the NRC is required, however the procedural error is reportable as a 60 day LER IAW 10CFR50.73.
c. 31 AFW Pump will not start because the condition of the flag not being matched on 32 MBFP existed before 31 AFW pump switch was placed in Auto. Technical Specification LCO 3.3.2, Table 3.3.2-1, Function 6e, Condition I is required to be entered due to an inoperable Auxiliary Feedwater actuation signal associated with the trip of a Main Boiler Feed Pump.
d. 31 AFW Pump will not start because the auto-start timer had previously timed out following testing of 32 MBFP. Technical Specification LCO 3.3.2, Table 3.3.2-1, Function 6e, Condition I is <u>NOT</u> required to be entered because the Auxiliary Feedwater actuation signal associated with the trip of a Main Boiler Feed Pump is already in its tripped state.

Answer: a

Explanation / Justification

a. Correct. This is how the system will operate and is discussed in a Caution in SOP-AFW-001 prior to the steps to place AFW pumps in Auto. This event would be reportable because it is a safeguards actuation based on a valid signal. The exception of being part of a pre-planned activity covered by procedures would not apply. This was an error by the operator and was therefore not planned.
b. Incorrect. Incorrect because the signal is valid regardless of whether or not feedwater flow was lost. It is plausible that a candidate could believe the rules are as stated in this choice. Also plausible that the student could believe that procedural actions are planned and therefore not immediately reportable. However, this was an error by the operator and was therefore not planned. Student may believe that although it was not immediately reportable an LER is warranted.
c. Incorrect. Incorrect because the system does not respond this way. Plausible because the Technical Specification is applicable if the MBFP was in service.
d. Incorrect. Incorrect because the system does not respond this way. Plausible because the technical specification is not applicable because the 32 MBFP is not in service and already tripped.

Technical References:	I3LP-ILO-AFW001 3-SOP-AFW-001 System Description 21.2
Proposed References to be provided:	None
Learning Objective:	Objective 5a
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 5

SRO justification:

The question cannot be answered based solely on system knowledge and/or overall purpose of the applicable procedure. The SRO candidate must assess plant conditions and then select the procedure to address the event (IP-SMM-LI-108; reportability and/or Technical Specification applicability)

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	0620002440 AC Electrical Distribution System: Knowledge of SRO responsibilities in emergency plan implementation	
Importance:		2.7	4.5

Question: #15

Given the following conditions:

- The Unit is at 100% power.
- The Shift Manager is incapacitated due to an illness.

A major grid disturbance has resulted in a Reactor Trip and a Complete Loss of Offsite Power. All 138KV and 13.8KV feeds to Unit 3 are unavailable. Con Ed has stated that grid restoration times are presently unknown, but greater than 4 hours. All available equipment has functioned per design and Natural Circulation Cooling has been established.

As the Control Room Supervisor classify the event, and state what personnel can relieve you as the Emergency Director.

a. UNUSUAL EVENT; Loss of all offsite AC power to 480 V safeguards buses for ≥ 15 minutes. The On-Call Emergency Director or the Plant Operations Manager.
b. ALERT; Loss of all offsite AC power to 480 V safeguards buses for ≥ 15 minutes. The On-Call Emergency Director or the Plant Operations Manager.
c. ALERT; Loss of all offsite AC power to 480 V safeguards buses for ≥ 15 minutes. The On-Call Emergency Director only.
d. UNUSUAL EVENT; Loss of all offsite AC power to 480 V safeguards buses for ≥ 15 minutes. The On-Call Emergency Director only.

Answer: a

Explanation / Justification

a. Correct. In accordance with IP-EP-120; Emergency Classification this event would be classified as an UNUSUAL EVENT (SU1.1). However, since references are not being provided, the basic SRO knowledge of the Emergency Plan and the classification thresholds should allow the student to answer this question. UE events are the lowest category and indicate a **potential** degradation of the level of safety. ALERT events involve actual or **substantial** degradation of the level of safety. In this case, a loss of of-site power has occurred, but this is within design analysis (accidents analyzed with a loss of of-site power)

and more than minimum safeguards equipment is available with all three AC buses energized from the diesel generators. Additionally the plant is stable with natural circulation established. Therefore, no more than an UNUSUAL EVENT classification is justified. With regard to who can relieve the CRS, IP-EP-210; Central Control Room and IP-EP-120 both state; "... until relieved by the On-Call Emergency Director or other qualified Emergency Director (Plant Operations Manager)."

b. Incorrect. Plausible because the student may believe that because they have lost all off-site power an ALERT is warranted. However, again the plant is stable and the minimum design requirements are presently met (two AC buses energized from the diesel generators). Actually all three AC buses are energized from the EDGs. Second part of distractor regarding Emergency Director relief is correct.
c. Incorrect. Plausible because the student may believe that because they have lost all off-site power an ALERT is warranted. However, again the plant is stable and the minimum design requirements are presently met (two AC buses energized from the diesel generators). Actually all three AC buses are energized from the EDGs.
d. Incorrect. Plausible because the first part of the distractor is correct, UNUSUAL EVENT is the proper classification. However, with regard to who can relieve the CRS, IP-EP-210; Central Control Room and IP-EP-120 both state; "... until relieved by the On-Call Emergency Director or other qualified Emergency Director (Plant Operations Manager)."

Technical References:	IPEC Emergency Plan (Classification Definitions) IP-EP-120; Emergency Classification IP-EP-210; Central Control Room IP-EP-AD13; EAL Bases Tech Spec 3.8.1 & Bases
Proposed References to be provided:	None
Learning Objective:	N/A
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A #	002000A204 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of heat sinks	
Importance:		4.3	4.6

Question: #16

Given the following conditions:

- 33 Auxiliary Feedwater Pump is cleared and tagged for motor replacement.
- PORV PCV-455C is inoperable due to seat leakage and PORV Block Valve RC-MOV-535 is closed.
- The Reactor has tripped from 100% power due to a loss of main feedwater.
- 480 VAC Bus 3A supply breaker has tripped open due to a bus fault.
- 32 Auxiliary Feedwater Pump has tripped on overspeed.
- Operators have transitioned to 3-FR-H.1; Response to Loss of Secondary Heat Sink.
- Initial attempts to restore a secondary heat sink have been unsuccessful and the operators have initiated “Bleed and Feed” in accordance with 3-FR-H.1; Response to Loss of Secondary Heat Sink.

Based on present plant conditions and procedural guidance for “Bleed and Feed”, what mitigative actions will be directed by the CRS and will they be successful?

a. Two Charging Pumps running, Three HHSI Pumps running, and both PORV & Associated Block Valves open. Continue attempts to establish secondary heat sink in any SG. Bleed and Feed will be successful, sufficient feed and bleed flow exists to permit RCS heat removal.
b. Two Charging Pumps running, Two HHSI Pumps running, one PORV & Associated Block Valve open, All Reactor Vessel Head Vent Valves open. GO TO Attachment 3, Establishing Feedwater Flow from Secondary Plant. Bleed and Feed may not be successful, if core decay heat exceeds RCS bleed and feed heat removal capability, significant RCS inventory depletion will occur.
c. Two Charging Pumps running, Two HHSI Pumps running, and both PORV & Associated Block Valves open. Continue attempts to establish secondary heat sink in any SG.

Bleed and Feed will be successful, sufficient feed and bleed flow exists to permit RCS heat removal.

d. Two Charging Pumps running, Three HHSI Pumps running, one PORV & Associated Block Valve open, All Reactor Vessel Head Vent Valves open. GO TO Attachment 3, Establishing Feedwater Flow from Secondary Plant. Bleed and Feed may not be successful, if core decay heat exceeds RCS bleed and feed heat removal capability, significant RCS inventory depletion will occur.

Answer: a

Explanation / Justification

a. Correct. Due to the loss of 3A 480 VAC Bus, 32 Charging Pump and 31 ABFP are unavailable, therefore procedurally two charging pumps and three HHSI pumps will be running. The HHSI pumps are on 480VAC Buses 2A, 5A, & 6A. Both PORVs and their associated Block Valves are available because TS 3.4.11 only requires the PORV Block Valve be closed with power available to it for a leaking PORV. The power supplies for the PORV Block Valves are 5A and 6A, not 3A. In accordance with the procedure (3-FR-H.1) and its bases two charging pumps and three HHSI pumps are adequate (feed) to allow opening both PORVs and their associated Block valves (Bleed). Bleed and Feed will be successful, sufficient feed and bleed flow exists to permit RCS heat removal.
b. Incorrect. Plausible because the student may believe that 480VAC Bus 3A supplied both a Charging Pump and a HHSI Pump. Also plausible for the student to believe that power was not available to the one PORV Block Valve, making that PORV not available. Both PORVs and their associated Block Valves are available because TS 3.4.11 only requires the PORV Block Valve be closed with power available to it for a leaking PORV. The power supplies for the PORV Block Valves are 5A and 6A, not 3A. Finally the remainder of the distractor is correct for a condition with only one PORV open.
c. Incorrect. Plausible because the student may believe that 480VAC Bus 3A supplied both a Charging Pump and a HHSI Pump. The HHSI pumps are on 480VAC Buses 2A, 5A, & 6A. Also plausible because the remainder of the distractor is correct, both PORVs are available and the Bleed and Feed would be successful.
d. Incorrect. Plausible because due to the loss of 3A 480 VAC Bus, 32 Charging Pump and 31 ABFP are unavailable, therefore procedurally two charging pumps and three HHSI pumps will be running. The HHSI pumps are on 480VAC Buses 2A, 5A, & 6A. Also plausible for the student to believe that power was not available to the one PORV Block Valve, making that PORV not available. Both PORVs and their associated Block Valves are available because TS 3.4.11 only requires the PORV Block Valve be closed with power available to it for a leaking PORV. The power supplies for the PORV Block Valves are 5A and 6A, not 3A. Finally the remainder of the distractor is correct for a condition with only one PORV open.

Technical References:	I3LP-ILO-EOPFRH WOG ERG for FR-H.1 3-FR-H.1 and Deviation Document Technical Specification 3.4.11 & Bases
Proposed References to be provided:	None
Learning Objective:	Objective 7.0
Question Source:	New

Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A #	039000A204 Malfunctions or operations on the HRPS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: LOCA condition and related concern over hydrogen	
Importance:		3.5	3.9

Question: #17

The Emergency Operating Procedures place Hydrogen Recombiners in service if measured hydrogen concentration is between ____ % by volume and ____ % by volume. This is because a hydrogen burn becomes a threat to containment integrity when hydrogen concentration exceeds ____ % by volume in dry air.

a. 0.5, 2.4, 3.0.
b. 2.0, 3.0, 3.0.
c. 2.4, 3.0, 4.0.
d. 0.5, 4.0, 4.0.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because hydrogen recombiners are placed in service greater than 0.5% hydrogen concentration by volume in dry air. Also 3-SOP-CB-007; H_2 Recombiner Operation states that they should be placed in service prior to reaching 2.4% by volume to provide adequate time to prevent H_2 concentration from reaching 3.0% by volume.
b. Incorrect. Plausible because student may believe that 2% is the threshold for placing recombiners in service and that the recombiner design capability of 3% is also the lower flammability limit.
c. Incorrect. Plausible because 3-SOP-CB-007; H_2 Recombiner Operation states that they should be placed in service prior to reaching 2.4% by volume to provide adequate time to prevent H_2 concentration from reaching 3.0% by volume. Also plausible because 4% is the lower flammability limit.
d. Correct. 3-ES-1.3; Transfer to Cold Leg Recirculation, states to place the Hydrogen Recombiners in service between 0.5% and 4.0% hydrogen concentration by volume in dry air. Also the lower flammability limit is 4.0%.

Technical References: I3LP-ILO-VCPAH2
3-SOP-CB-007

	3-ES-1.3 & Bases
	System Description 10.9
Proposed References to be provided:	None
Learning Objective:	Objective 5.0
Question Source:	Modified Bank Questions # 9083 & 9089
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 5

SRO: Requires knowledge of specific content, verses knowledge of the procedure's overall mitigative strategy or purpose.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A #	0720001241 Area Radiation Monitoring: Knowledge of EOP entry conditions and immediate action steps	
Importance:		4.6	4.8

Question: #18

Given the following conditions:

- A small break LOCA resulted in a Safety Injection.
- ECCS operating problems caused some core damage.
- The operating crew restored ECCS flow and is now in 3-ES-1.2; POST-LOCA COOLDOWN AND DEPRESSURIZATION.
- The STA has reviewed the Critical Function Status Trees and has informed the CRS that there is no Red or Orange Path Condition.
- However, the STA has stated that a Yellow Condition exists due to Containment Radiation Levels and states that Functional Restoration Procedure 3-FR-Z.3; Response to High Containment Radiation Level is the appropriate procedure.

What is the Containment Radiation Level that causes the Containment Status Tree to be in a Yellow Condition and is the CRS required to implement the actions of 3-FR-Z.3?

a. ≥ 2 R/HR. No, the CRS is allowed to decide whether or not to implement any Yellow condition FRP.
b. ≥ 3 R/HR. Yes, the CRS will implement the actions of 3-FR-Z.3 concurrently with 3-ES-1.2 or after completion of 3-ES-1.2, but they must complete the actions of 3-FR-Z.3 before the EOP network is exited.
c. ≥ 3 R/HR. No, the CRS is allowed to decide whether or not to implement any Yellow condition FRP.
d. ≥ 2 R/HR. Yes, the CRS will implement the actions of 3-FR-Z.3 concurrently with 3-ES-1.2 or after completion of 3-ES-1.2, but they must complete the actions of 3-FR-Z.3 before the EOP network is exited.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the "Alert" setpoint for Containment High Range Area Radiation Monitors R-25 and R-26 is 2 R/HR. Also plausible because the second part of the distractor is correct per OAP-012, step 4.3.18.
b. Incorrect. Plausible because in accordance with 3-F-0.5; Containment Status Tree, if Containment Radiation is ≥ 3 R/HR a Yellow Path condition exists. The Status Tree states to GO TO 3-FR-Z.3 for those radiation conditions. Also plausible because although the student knows that Yellow Conditions don't require immediate transition to the FRP from the procedure in effect, they may believe that all valid Yellow conditions must be resolved prior to exiting the EOP network.
c. Correct. In accordance with 3-F-0.5; Containment Status Tree, if Containment Radiation is ≥ 3 R/HR a Yellow Path condition exists. Also see OAP-012, step 4.3.18; "The CRS/SM is allowed to decide whether or not to implement any YELLOW condition FRP."
d. Incorrect. Plausible because the "Alert" setpoint for Containment High Range Area Radiation Monitors R-25 and R-26 is 2 R/HR. Also plausible because although the student knows that Yellow Conditions don't require immediate transition to the FRP from the procedure in effect, they may believe that all valid Yellow conditions must be resolved prior to exiting the EOP network.

Technical References:	I3LP-ILO-EOPFRZ
	3-FR-Z.3 & Deviation Document
	3-F-0.5; Containment Status Tree
	OAP-012; EOP Users Guide
Proposed References to be provided:	None
Learning Objective:	Objectives 6 & 8
Question Source:	Modified Bank (Questions # 17079 & # 2997)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 4 & 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		COO
	K/A #	1940002135 Knowledge of the fuel-handling responsibilities of SROs	
Importance:		2.2	3.9

Question: #19

In accordance with 3-REF-001-GEN, SECTION 1.1; Refueling Administration, which of the following refueling activities requires that the Refueling SRO be present and in a location that allows direct observation?

I. Reactor Vessel Head Stud De-tensioning and Tensioning.
II. Any movement of the Reactor Vessel Head.
III. Any movement of the Reactor Vessel Lower Internals.
IV. Control Rod Latching, Unlatching and Drag testing.
V. Any movement of Fuel into or out of the Reactor Vessel.
VI. Draining Refueling Cavity Level with fuel in the Reactor Vessel.

a. I, II, III, IV, V.
b. II, III, IV, V.
c. II, IV, V. VI
d. I, IV, V.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the refueling procedure requires RSRO oversight of all refueling activities. This is also plausible because stud de-tensioning and tensioning determines MODE change activities as well.
b. Correct. See 3-REF-001-GEN, SECTION 1.1, STEP 1.4.1.2.F. RSRO is not required to be present and in a location that allows direct observation or stud tensioning or de-tensioning.
c. Incorrect. Plausible because the student may believe that only the activities involving core alterations or core cooling (draining cavity) require a RSRO present. Student may believe that the internals are a special case required by a separate radiation protection procedure because of special radiological implications (extreme high dose) and therefore a refueling SRO may not be necessary. Student may believe the lower internals don't require a RSRO because the fuel is already removed.
d. Incorrect. Plausible because the student may believe that only the activities involving core alterations or a MODE change require a RSRO present. Student could also believe that the RSRO was not required for "ALL" (ANY) vessel head and internals moves.

Technical References:	I3LP-ILO-FHD001 3-REF-001-GEN, Section 1.1
Proposed References to be provided:	None
Learning Objective:	Objectives P (E-16) & Q (E-17)
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		COO
	K/A #	1940001215 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	
Importance:		2.9	3.9

Question: #20

Given the following conditions:

- The Unit is in MODE 1.
- The Shift is manned to the minimum composition in accordance with EN-OP-115; Conduct of Operations.
- The Shift has 4 hours remaining.
- The RO (ATC) has become ill and must leave the site for emergency medical treatment.

Which ONE (1) of the following describes the requirements regarding the shift composition and the MINIMUM required action in this situation IAW EN-OP-115 and Technical Specifications?

a. The RO may leave the site immediately. A replacement must arrive within 1 hour.
b. The RO may leave the site immediately. A replacement must arrive within 2 hours.
c. Responsibilities of the RO may be turned over to the BOP for the remainder of the shift.
d. The CRS may assume the responsibilities of the RO. The STA may perform the duties of the CRS until normal shift relief.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that Tech Spec 5.2.2.b requires replacement within 1 hour. Both Tech Spec 5.2.2.b and EN-OP-115 require replacement within 2 hours.
b. Correct. Both Tech Spec 5.2.2.b and EN-OP-115, Attachment 9.4 require replacement within 2 hours.
c. Incorrect. Plausible because the student may believe that Tech Spec 5.2.2.b requires replacement within 4 hours. Both Tech Spec 5.2.2.b and EN-OP-115 require replacement within 2 hours.
d. Incorrect. Plausible because this may be possible if a field supervisor is available and is both SRO Licensed and STA qualified. However, the question stem states minimum composition IAW EN-OP-115, which states that a Field Support Supervisor is not required (0).

Technical References:	I3LP-ILO-ADMIN? Technical Specification 5.2.2.b EN-OP-115, Attachment 9.4
Proposed References to be provided:	None
Learning Objective:	Objective?
Question Source:	Bank – Question # 16705
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 1

SRO knowledge: The required actions or not meting administrative controls listed in Technical Specifications (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements).

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		EC
	K/A #	1940002237 Ability to determine operability and/or availability of safety related equipment	
Importance:		3.6	4.6

Question: #21

The plant is at 100% power with the following configuration:

- 1-2-3 is the Essential Service Water Header
- 35 Service Water Pump (SWP) is tagged out for maintenance
- 32 Component Cooling Water Pump (CCWP) is tagged for maintenance

At 1200, 32 EDG is declared inoperable when its pre-lube pump is found off and its lube oil and jacket water temperatures are both below operability values.

In accordance with the Technical Specifications, what must be done?

a. No additional equipment needs to be declared inoperable. 32 EDG must be returned to an operable status within 72 hours or a shutdown is required.
b. Immediately declare 33 CCWP and 36 SWP inoperable. Be in MODE 3 in 7 hours and MODE 5 in 37 hours.
c. Immediately declare 33 CCWP, 33 SWP, and 36 SWP inoperable. 32 EDG must be returned to an operable status within 72 hours or a shutdown is required.
d. 33 CCWP and 36 SWP must be declared inoperable within 4 hours of when 32 EDG was declared inoperable. Once the pumps are declared inoperable, then be in MODE 3 in 6 hours and MODE 4 in 12 hours.

Answer: a

Explanation / Justification

a. Correct. LCO 3.7.9 states that three pumps are required for essential service water header and that two pumps are required for nonessential header. In this case we lose one essential header pump and two nonessential header pumps, both of these being 72 hour action statements. This is the same as the EDG spec and therefore we would not "cascade" nor are we below design bases requirements. LCO 3.7.8 states that two component cooling loops shall be operable, each loop consisting of a single pump and single heat exchanger. Therefore, with only one loop operable, this would be in a 72 hour action statement as well. Again bounded by the Diesel spec and within design bases requirements. Minimum design bases equipment is met with 2 EDGs, 2 essential service water pumps, 1 nonessential

service water pump, and 1 component cooling water pump. Therefore only the 72 hour EDG action statement time is required as redundant equipment remains operable.

b. Incorrect. Plausible because student may believe that redundant equipment is not operable and that Tech Spec 3.0.3 is applicable. They may believe that they are below minimum equipment required as defined in OAP-034.
c. Incorrect. Plausible because these three pumps are supported by the 32 EDG for an emergency power supply. The student may believe that if the emergency power supply is inoperable, then the pumps are automatically declared inoperable and that we do "cascade" tech specs and enter all three technical specifications. Still plausible for the student to believe the 72 hour action time, because we are still meeting minimum design requirements.
d. Incorrect. Plausible because the student may believe that redundant equipment is inoperable and that EDG Tech Spec actions A3 and F apply. Tech Specs are not being provided as a reference, so student may believe that actions can't be complied with and therefore MODE 3 in 6 hours and MODE 4 in 12 hours per 3.8.1.F.

Technical References:	I3LP-ILO-EDSEDG
	Tech Spec 3.7.8 & Bases
	Tech Spec 3.7.9 & Bases
	Tech Spec 3.8.1 & Bases
	OAP-034; Safety Function Determination Process
Proposed References to be provided:	None
Learning Objective:	Objectives 7 & 8
Question Source:	Bank – Question # 24834
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 2

NOTE: SRO level because requires application of required actions and bases knowledge. Also actions are greater than 1 hour and require knowledge of below the line information.

No references provided because knowledge of operability requirements (minimum design bases), Tech Spec LCOs, and Tech Spec Bases are enough to answer question.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		RC
	K/A #	1940001235 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	
Importance:		4.4	4.6

Question: #22

Given the following conditions:

- The Unit is at 100% Power.
- R15 SJAE EXH has actuated on the Radiation Monitoring Control Cabinet (RMCC)
- 3-AOP-SG-1; Steam Generator Tube Leak has been entered and is presently being implemented by the crew.
- R-15; Air Ejector Exhaust Gas Activity Monitor has a rising trend.
- R-19; Steam Generator Blowdown Liquid Activity Monitor has a rising trend.
- Chemistry has been requested to perform 0-CY-2450; Primary to Secondary Leak.
- The crew has initiated Attachment 1 of 3-AOP-SG-1 and has determined an initial estimated Primary-to-Secondary Leakrate of 30 gpd.

Listed below are the first (3) three estimated leakrate determinations recorded by the crew on Attachment 2; Leak Rate Log.

	Current Leakrate (gpd)
Estimate # 1	30 gpd
Estimate # 2	32 gpd
Estimate # 3	35 gpd

How often does the estimated leakrate need to be calculated and what radiation monitors are being used to calculate the estimated leakrate?

a. 15 minutes. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-19; Steam Generator Blowdown Liquid Activity.
b. 2 hours. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-19; Steam Generator Blowdown Liquid Activity.
c. 2 hours. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-63A&B; Gross Failed Fuel Detector.
d. 15 minutes. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-63A&B; Gross Failed Fuel Detector.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because first part of distractor is correct, Attachment 2 of 3-AOP-SG-1 states; "If leak increases ≥ 30 gpd but < 75 gpd take data & determine leakrate every 15 minutes." Also plausible because the student may believe that R-19; SG Blowdown is used in addition to R-15; Air Ejector for determining the estimated leakrate.
b. Incorrect. Plausible because if the leakrate is ≥ 30 gpg but < 75 gpd and stable for 1 hour (≤ 5 gpd increase in 1 hour), then the procedure has you take the readings every 2 hours. Because the initial estimate is ≥ 30 gpd, 15 minute readings are required and the rate of increase is actually 8 – 12 gpd/hr. Also plausible because the student may believe that R-19; SG Blowdown is used in addition to R-15; Air Ejector for determining the estimated leakrate.
c. Incorrect. Plausible because if the leakrate is ≥ 30 gpg but < 75 gpd and stable for 1 hour (≤ 5 gpd increase in 1 hour), then the procedure has you take the readings every 2 hours. Because the initial estimate is ≥ 30 gpd, 15 minute readings are required and the rate of increase is actually 8-12 gpd/hr. Also plausible because the second part of the distractor is correct, R-15 and R-63A&B are used in the estimated leakrate calculation.
d. Correct. Attachment 2 of 3-AOP-SG-1 states; "If leak increases ≥ 30 gpd but < 75 gpd take data & determine leakrate every 15 minutes." The readings provided have therefore been taken every 15 minutes and the delta leakrate would then be 8-12 gpd/hr and also above the threshold for taking data every 2 hours. Finally in accordance with Attachment 1, estimated leakrate calculation method 1, R-15 and R-63A&B are used in the estimated leakrate calculation.

Technical References:	I3LP-ILO-AOPSG I3LP-ILO-RMSPRM 3-AOP-SG-1 & Bases 3-ARP-040 3-SOP-RM-10 System Description 12.0
Proposed References to be provided:	None
Learning Objective:	Objective 2 Objective B (2)
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 4 & 5

NOTE: SRO knowledge based on the analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures. SRO knowledge of content of the procedure versus knowledge of the procedure' overall mitigative strategy or purpose and when to implement attachments and appendices, including how to coordinate these items with procedure steps.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		RC
	K/A #	1940002311	
		Ability to control radiation releases	
Importance:		3.8	4.3

Question: #23

The Unit is in MODE 5 following planned shutdown into a scheduled refueling outage. The Unit had a continuous run at power for the last 6 months. The Outage Manager has requested Operations initiate a release permit for a VC Purge in accordance with 3-SOP-WDS-013. Chemistry has completed VC grab sample and provided operations with both Noble Gas and Iodine Activity numbers.

Shift Operations has determined that the Calculated Release Rate for Noble Gas is greater than the IP-SMM-CY-001 Annual Release Rate. It has been determined that the Purge will require the use of the Instantaneous Release Rate Limit. Who is required to authorize the use of this limit in accordance with 3-SOP-WDS-013?

a. Unit Operations Manager.
b. Chemistry Manager.
c. General Manager – Plant Operations.
d. Radiation Protection Manager.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the Unit Operations Manager is required to authorize the use of the Quarterly Average Release Rate Limit.
b. Incorrect. Plausible because the student may believe that since the RPM has approval authority over the Radioactive Effluents Control Program (IP-SMM-CY-001) that he would be authorized to provide approval for using a higher release rate.
c. Correct. See step 2.2 and step 4.3.9 of 3-SOP-WDS-013.
d. Incorrect. Plausible because in accordance with the Radioactive Effluents Program Procedure (IP-SMM-CY-001), step 6.2.1.4; "the Instantaneous Release Rate Limit is a SITE limit, and applies to the total release rate from all operating units on site." Based on this definition, the student may believe that using the Instantaneous Limit requires Site Vice President's approval.

Technical References:	I3LP-ILO-GWR001
	3-SOP-WDS-013
	IP-SMM-CY-001
Proposed References to be provided:	None

Learning Objective:	Objectives 6e & 10h
Question Source:	Modified Bank – Question # 24958
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 4

NOTE: SRO knowledge based knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose. Topic is related to 55.43 (b) (4); process for gaseous/liquid release approvals, i.e., release permits.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		EOP-EP
	K/A #	1940002438	
		Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required	
Importance:		2.4	4.4

Question: #24

Given the following conditions:

- A Site Area Emergency has been declared due to a LOCA outside Containment.
- The LOCA is into the PAB building and a pathway to the environment exists.
- Limited makeup to the RWST is available.
- An operator has volunteered to go to the PAB building to locally isolate the leak.
- The EOF has been fully staffed and operational for an hour.
- This action will result in a significant reduction in the offsite dose, protecting a large population.

Using the Emergency Exposure Guidelines listed in the Indian Point Energy Center Emergency Plan and Form EP-6; Emergency Exposures Authorization, what is the maximum emergency exposure this volunteer operator may receive while performing this action and who may approve?

a. 10 Rem TEDE with SM approval.
b. 25 Rem TEDE with SM approval.
c. 10 Rem TEDE with ED approval.
d. 25 Rem TEDE with ED approval.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because 10 Rem is the dose limit for protecting valuable property and the student may believe that 25 Rem is only for **life saving** (**bolded** on Form EP-6). Also plausible because the Shift Manager is initially the Emergency Director (ED). However, he would have been relieved by the on duty Emergency Director (ED) at the EOF.
b. Incorrect. Plausible because 25 Rem is the correct dose limit for life saving or <u>the protection of large populations.</u> Also plausible because the Shift Manager is initially the Emergency Director (ED). However, he would have been relieved by the on duty Emergency Director (ED) at the EOF.

c. Incorrect. Plausible because 10 Rem is the dose limit for protecting valuable property and the student may believe that 25 Rem is only for **life saving** (**bolded** on Form EP-6). Also plausible because the second part of the distractor is correct.
d. Correct. See Form EP-6 and below technical references. "Volunteers may be authorized up to 25 Rem for life saving or the protection of large populations."

Technical References:	IP-EP-115, Forms Form EP-6; Emergency Exposure Authorization IPEC E-Plan, 13-01, Page K-1; Emergency Exposure Guidelines IPEC E-Plan, 13-01, Pages B-3 through B-6 IP-EP-210, Central Control Room IP-EP-220, Technical Support Center IP-EP-250, EOF
Proposed References to be provided:	None
Learning Objective:	N/A
Question Source:	Bank – Indian Point 2 – 2014 NRC Exam, #98
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 4

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		EOP-EP
	K/A #	1940002445 Ability to prioritize and interpret the significance of each annunciator or alarm	
Importance:		4.1	4.3

Question: #25

Given the following initial plant conditions:

- Unit is at 100% Power.
- Rod Power Supply M-G Set 32 is cleared and tagged for motor replacement.
- Individual Rod Position Indicating System is powered from backup power supply MCC-39.

A failure of the Station Auxiliary Transformer has just occurred. 32 and 33 EDGs have started and loaded in Blackout Mode. The Control Room receives a number of alarms; including the following annunciator alarms that the BOP Operator believes to be most relevant to the loss of the Station Auxiliary Transformer and requiring action:

- STATION AUX XFMR LOCKOUT RELAY TRIPPED
- 6900 V STATION AUX BREAKER TRIP 52ST5, 52ST6
- MOTOR CONTROL CENTER #39 52/MCC9 AUTO TRIP

Following the CRS's assessment of plant conditions and prioritization of the above annunciators, which of the following procedures should be implemented by the CRS?

a. 3-AOP-ROD-1; Rod Control and Indications System Malfunction.
b. 3-AOP-138KV-1; Loss of Power to 6.9KV Bus 5 and/or 6.
c. 3-ARP-011; MOTOR CONTROL CENTER #39 52/MCC9 AUTO TRIP.
d. 3-E-0; Reactor Trip or Safety Injection.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the loss of Bus 5A resulted in the loss of MCC-39 which was providing power to the IRPIs. This would cause all the IRPIs to be inoperable and require placing rod control in Manual immediately IAW Tech Spec 3.1.7.B. This action is taken per 3-AOP-ROD-1 step 4.136 and proceeding NOTE. However, although this is an immediate action per Tech Specs, step 4.11 of 3-AOP-138KV-1 will perform the same action and that procedure will also handle the bigger issue of loss of off-site power to buses 5A & 6A.

b. Correct. The CRS should recognize that given the existing plant conditions, 3-AOP-138KV-1 will provide the best overall mitigation. Step 4.11 of 3-AOP-138KV-1 will handle the immediate action required for rods in Manual per Tech Spec 3.1.7.B, but will also handle the bigger issue of loss of off-site power to buses 5A & 6A.
c. Incorrect. Plausible because the loss of Bus 5A resulted in the loss of MCC-39 which was providing power to the IRPIs. This would cause all the IRPIs to be inoperable and require placing rod control in Manual immediately IAW Tech Spec 3.1.7.B. Student may believe that prioritizing the loss of MCC-39 alarm response procedure is the highest priority action. However, this alarm response does not reference Tech Spec 3.1.7.B nor does it require rods to be placed in manual. This is more than likely because MCC-39 is the backup power supply to IRPIs. Again, Step 4.11 of 3-AOP-138KV-1 will handle the immediate action required for rods in Manual per Tech Spec 3.1.7.B, but will also handle the bigger issue of loss of off-site power to buses 5A & 6A.
d. Incorrect. Plausible because the student may believe that the above transient has resulted in a Reactor Trip or requires a Reactor Trip be initiated. The stem of the question states that the IRPIs are being powered from MCC-39, the student may believe that a Reactor Trip is required due to not having any IRPI indications. More plausible is that the student may believe that the plant has experienced a loss of power to both Rod Control M-G Sets. Note that 32 M-G set was tagged for motor replacement in the stem, the student may believe that 31 M-G Set is powered by either bus 5A or 6A. Actually, 32 M-G Set is powered from Bus 6A (already O/S) and 31 M-G Set is powered from Bus 2A which is being powered via the Unit Auxiliary Transformer (Main Generator).

Technical References:	I3LP-ILO-AOP138 and Bases
	3-ARP-011; PANEL SHF – ELECTRICAL
	3-AOP-ROD-1 and Bases
	3-SOP-RC-001
	System Description 27.1
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Answer Key

2015 U3 NRC Exam

1	**D**	26	**C**	51	**B**	1	**D**
2	**B**	27	**B**	52	**C**	2	**A**
3	**D**	28	**B**	53	**A**	3	**C**
4	**D**	29	**A**	54	**A**	4	**B**
5	**C**	30	**B**	55	**C**	5	**D**
6	**D**	31	**A**	56	**B**	6	**D**
7	**B**	32	**C**	57	**A**	7	**B**
8	**C**	33	**B**	58	**A**	8	**C**
9	**C**	34	**C**	59	**D**	9	**A**
10	**C**	35	**A**	60	**B**	10	**C**
11	**A**	36	**D**	61	**A**	11	**B**
12	**A**	37	**D**	62	**B**	12	**A**
13	**C**	38	**C**	63	**D**	13	**C**
14	**B**	39	**C**	64	**B**	14	**A**
15	**C**	40	**A**	65	**A**	15	**A**
16	**B**	41	**D**	66	**C**	16	**A**
17	**B**	42	**C**	67	**D**	17	**D**
18	**D**	43	**B**	68	**A**	18	**C**
19	**C**	44	**A**	69	**D**	19	**B**
20	**A**	45	**A**	70	**D**	20	**B**
21	**B**	46	**B**	71	**C**	21	**A**
22	**D**	47	**C**	72	**A**	22	**D**
23	**D**	48	**B**	73	**D**	23	**C**
24	**A**	49	**D**	74	**C**	24	**D**
25	**D**	50	**D**	75	**A**	25	**B**

U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information	
Name:	
Date: 5/18/15	Facility/Unit: Indian Point Unit 3
Region: I X II ☐ III ☐ IV ☐	Reactor Type: W X CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results		
Examination Value	________	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000008AK305 Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: ECCS termination or throttling criteria	
Importance:		4.0	4.5

Question: #1

The crew is performing ECCS reduction in ES-1.2, Post LOCA Cooldown and Depressurization. One SI pump has been stopped.

Which of the following describes how the required subcooling changes and the basis for the amount of subcooling required for securing the second SI pump?

a. Required Subcooling decreases due to decreased break flow after stopping the second pump.
b. Required Subcooling decreases to allow pressure to decrease sufficiently after stopping the second pump to place RHR cooling in service.
c. Required Subcooling increases to compensate for the anticipated void formation after the second pump is stopped.
d. Required Subcooling increases to ensure the RCS will remain subcooled after the second SI pump is stopped.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because break flow does decrease after stopping each pump; however, subcooling requirement increases.
b. Incorrect. Plausible because the most likely exit from ES-1.2 is starting an RHR pump in injection mode; however, subcooling requirement increases.
c. Incorrect. Plausible because required subcooling increases; however, no void formation is expected or should exist because the RCS is subcooled.
d. Correct because throttling SI flow is not practical, pumps must be stopped. As pumps are stopped, a step decrease in flow is experienced. RCS pressure will decrease to reach equilibrium with break flow and ECCS flow. RCS subcooling and pressure control must be maintained as the reduction takes place. As flow is reduced, a large allowance is needed.

Technical References: I3LP-ILO-EOPE10, Rev. 2

Proposed References to be provided:	None
Learning Objective:	Objective 15
Question Source:	Bank, IP2 2012 NRC Exam
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 5 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000009EA109	
		Ability to operate and monitor the following as they apply to a small break LOCA: RCP	
Importance:		3.6	3.6

Question: #2

Indian Point 3 has experienced a Reactor Trip and a Safety Injection. E-0 (Reactor Trip or Safety Injection) Immediate Operator Actions have been completed. The following conditions are observed:

- All Control Rods are fully inserted
- All Reactor Coolant Pumps are running
- RCS Pressure is 1185 psig and slowly lowering
- RCS Average Temperature is 547°F
- The hottest incore thermocouple is reading 570°F
- Containment Pressure is 2 psig and slowly rising
- 31 High Head Safety Injection Pump tripped on overcurrent
- 32 High Head Safety Injection Pump is tagged for motor replacement
- 33 High Head Safety Injection Pump is running

Analyze and describe the required Reactor Coolant Pump configuration by procedure and why?

a. All Reactor Coolant Pumps running, to delay the onset of inadequate cooling
b. All Reactor Coolant Pumps stopped, to prevent excessive core uncovery
c. All Reactor Coolant Pumps stopped, less than two HHSI pumps are running
d. All Reactor Coolant Pumps running, adequate RCS subcooling exists

Answer: b

Explanation / Justification

a. Incorrect. Plausible because if there were no HHSI pumps running, the RCPs are kept running to delay the onset of inadequate core cooling. At least one HHSI pump must be running to stop the RCPs. Student may believe that a minimum of 2 HHSI pumps are required.
b. Correct. At least one HHSI pump is running and subcooling is less than specified in table. The trip criteria purpose is to minimize high PCTs due to excessive core uncovery caused by continued RCP operation during a SBLOCA.

c. Incorrect. Plausible because the design bases minimum Engineered Safeguards Systems would be two HHSI pumps running and subcooling is less than specified in table.
d. Incorrect. Plausible because the student may believe that adequate subcooling exists. Question requires the use of steam tables to determine the amount of subcooling and knowledge of the subcooling required by the EOP.

Technical References:	I3LP-ILO-EOPE10, Rev. 2
Proposed References to be provided:	None
Learning Objective:	Objective 4.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000011EK101 Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: Natural circulation and cooling, including reflux boiling	
Importance:		4.1	4.4

Question: #3

During which of the following accident scenarios are natural circulation and reflux boiling not important.

a. A LOCA with leak rate such that SI flow is less than break flow and core boiling is occurring.
b. A LOCA with a leak rate such that the reactor coolant pressure stabilizes at a pressure higher than SG pressure.
c. A LOCA with break flow cooling less than decay heat and the enthalpy of the RCS is rising.
d. A LOCA with break flow cooling greater than decay heat and the pressure of the RCS is rapidly lowering.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because natural circulation and reflux cooling are required when ECCS flow and break flow are not adequate to remove decay heat. These conditions are indicative of a SBLOCA that requires supplemental heat removal via the steam generators to adequately cool the core.
b. Incorrect. Plausible because natural circulation and reflux cooling are required when ECCS flow and break flow are not adequate to remove decay heat. These conditions are indicative of a SBLOCA that requires supplemental heat removal via the steam generators to adequately cool the core.
c. Incorrect. Plausible because natural circulation and reflux cooling are required when ECCS flow and break flow are not adequate to remove decay heat. These conditions are indicative of a SBLOCA that requires supplemental heat removal via the steam generators to adequately cool the core.
d. Correct. A LBLOCA (design bases) will result in break flow cooling greater than decay heat, resulting in 1. Blowdown (rapid depressurization & core uncovery), 2. Refill (Accumulator injection), 3. Reflood (ECCS Flow), and 4. Long-term recirculation. Both break flow and ECCS flow are adequate for core cooling.

Technical References:	I3LP-ILO-EOPE10, Rev. 1
Proposed References to be provided:	None
Learning Objective:	Objective 1.0 & 2.0
Question Source:	Modified Bank,
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 10 & 14

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000015AK208 Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: CCWS	
Importance:		2.6	2.6

Question: #4

Indian Point 3 is at 100% Power. 3-AOP-RCP-1; Reactor Coolant Pump Malfunction has been entered due to elevated shaft vibration and rising lower motor bearing temperatures on 31 Reactor Coolant Pump. The cause of the elevated vibration and temperature readings are being investigated. The following conditions are presently observed:

- Two CCW Pumps are in service.
- 31 RCP Shaft Vibration has stabilized at 15 mils.
- 31 RCP Lower Motor Bearing Temperature is 185°F and rising at approximately 1°/hour.
- 32, 33, and 34 RCP vibration and motor bearing temperature readings are all normal.

Based on the present observed conditions, what actions are required In accordance with the AOP?

a. Trip the reactor, stop 31 RCP, and Initiate E-0; Reactor Trip or Safety Injection due to 31 RCP Shaft Vibration Readings ≥ 15 mils.
b. Initiate a normal plant shutdown (POP-2.1 Operation at Greater than 45% Power, POP-3.1 Plant Shutdown from 45% Power) and secure 31 RCP prior to reaching the high vibration limit.
c. Trip the reactor, stop 31 RCP, and Initiate E-0; Reactor Trip or Safety Injection due to 31 RCP Lower Motor Bearing Temperature Readings ≥ 185°F.
d. Initiate a normal plant shutdown (POP-2.1 Operation at Greater than 45% Power, POP-3.1 Plant Shutdown from 45% Power) and secure 31 RCP prior to reaching the high bearing temperature limit.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because IAAT Shaft Vibration is >15 mils and rising greater than 1.0 mils/hour, these action are taken IAW the AOP. However, shaft vibration is equal to 15 mils and has stabilized there.

b. Incorrect. Plausible because a plant shutdown is initiated if Shaft Vibration is > 15 mils and is trending ≥ 0.5 mils/hour rise over a 4 hour period. However, shaft vibration is equal to 15 mils and has stabilized there.
c. Incorrect. Plausible because at > 200°F, these actions are taken IAW the AOP.
d. Correct. Per the AOP; IAAT motor bearing temperature for the affected RCP is ≥ 185°F, and rising, then initiate plant shutdown with goal of stopping affected RCP prior to reaching 200°F motor bearing temperature. Operationally valid as the lower bearing CCW flow could be below the nominal 5 gpm and still not cause a total low bearing coolant flow alarm at 155 gpm. Low CCW flow to the lower bearing cooler alone could cause both elevated lower bearing temperatures and elevated shaft vibration.

Technical References:	I3LP-ILO-AOPRCP
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000022AK101 Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Consequences of thermal shock to RCP seals.	
Importance:		2.8	3.2

Question: #5

Operators have transitioned to 3-ECA-0.0; Loss of All AC Power due to the loss of all the 480V AC Safeguards Buses. Efforts to restore AC Power have been unsuccessful and the crew has determined the need to isolate RCP seal cooling per the procedure. How is RCP cooling isolated IAW ECA-0.0 and why is it isolated?

a. Close CH-MOV-222, Seal Water Return Isolation Valve only. To prevent hot RCS inventory from returning to the VCT.
b. Close both FCV-625, CCW Return from RCP Thermal Barrier and CH-MOV-222, Seal Water Return Isolation Valve. To ensure any RCS leakage from the seals remains in the containment.
c. Close CH-MOV-222, Seal Water Return Isolation Valve, FCV-625, CCW Return from RCP Thermal Barrier, and all four RCP Seal Injection (needle) Valves. To ensure that seal cooling remains isolated when power is restored and charging is re-established, this will prevent seal and shaft damage due to thermal shock.
d. Close all four RCP Seal Injection (needle) Valves and CH-MOV-222, Seal Water Return Isolation Valve. To prevent vapor binding due to potential elevated VCT and charging suction temperatures when power is restored and charging is re-established.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may believe that seal cooling need only be isolated on the return to ensure no leakage exits the containment. Student may believe that since component cooling system is intact, isolation is not required. This would help prevent hot RCS inventory from returning to the CVCS system via seal return.
b. Incorrect. Plausible because the student may believe that seal cooling need only be isolated on the return to ensure no leakage exits the containment.
c. Correct. If neither thermal barrier cooling nor seal injection exists (as would be the case on a loss of all 480V Vital AC Buses), the procedure has you isolate all four individual seal injection valves, seal return, and component cooling return from the thermal barrier.

Following restoration of AC power, it is desirable to keep the RCP seal cooling isolated to prevent thermal shock of the RCP seals and shaft.

d. Incorrect. Plausible because the student may believe that since component cooling system is intact, isolation is not required. A procedure caution states that "High VCT temperature, as indicated on seal water return TI-117, may cause vapor binding when charging pump is started."

Technical References:	I3LP-ILO-EOPC00, 3-ECA-0.0
Proposed References to be provided:	None
Learning Objective:	Objectives 8.0 & 11.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 3 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000025G2413	
		Loss of RHR System: Knowledge of crew roles and responsibilities during EOP usage	
Importance:		4.0	4.6

Question: #6

The unit is performing a plant cool down on RHR with 34 RCP operating. Current RCS Temperature is 275°F. PT-403 has just failed high. What is the effect of the failure on the plant and what actions are required to mitigate the effect?

a. AC-MOV-730 will close resulting in a loss of RHR. 3-AOP-RHR-1 will immediately sound the containment evacuation alarm, initiate containment closure, and stop the running RHR pump.
b. AC-MOV-731 will close resulting in a loss of RHR. 3-AOP-RHR-1 will immediately stop the running RHR pump, reopen AC-MOV-731, and restart the standby RHR pump.
c. AC-MOV-730 and AC-MOV-731 will close resulting in a loss of RHR. 3-AOP-RHR-1 will immediately stop the running RHR pump, reopen both AC-MOV-730 and AC-MOV-731, and restart the standby RHR pump.
d. AC-MOV-731 will close resulting in a loss of RHR. The operator should trip the running RHR pump in accordance with the alarm response procedure. 3-AOP-RHR-1 will direct the operator to maintain RCS temperature < 350°F using atmospheric or condenser steam dumps.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because these actions would be performed if the loss of RHR was to occur at ≤ 200°F in accordance with 3-AOP-RHR-1. Plant is in Mode 4 however, therefore RCS temperature is > 200°F and the steam generators are still available. Also PT-403 is interlocked with MOV-731 not 730, student may not remember which pressure transmitter is interlocked with which valve.
b. Incorrect. Plausible because PT-403 is interlocked with MOV-731. Also plausible because the steps make sense if restoring RHR is a priority. However, with PT-403 still failed high the valve will not reopen. Also, these steps are not taken if RCS temperature is > 200°F.
c. Incorrect. Plausible because the student may believe that both RHR suction valves close whether PT-402 or PT-403 fails. Also plausible because the steps make sense if restoring RHR is a priority.
d. Correct. PT-403 is interlocked with MOV-731. 3-ARP-010 directs the operator to trip the running RHR pump and directs the operator to 3-AOP-RHR-1. If RCS temperature is >

200°F, step 4.47 & 4.48 direct the operator to maintain RCS temperature < 350°F using atmospheric or condenser steam dumps.

NOTE: Reactor Operator fundamental knowledge of the valve interlocks and RHR pump operation plus comprehension understandings (question stem) regarding the fact that the unit is MODE 4 will allow the RO to answer this question.

Technical References:	I3LP-ILO-AOPRHR, 3-AOP-RHR-1 System Description 4.2, RHR 3-ARP-010 (Panel SGF)
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 5, 7, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000026AA203 Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition	
Importance:		2.6	2.9

Question: #7

The Unit was in MODE 3 at normal operating temperature and pressure when a large Component Cooling Water Leak occurred resulting in the implementation of 3-AOP-CCW-1; Loss of Component Cooling Water. The Component Cooling Water Headers were originally cross connected and actions were taken based on <u>both</u> surge tank levels lower than 5%.

Following leak identification and isolation, the Component Cooling System is being restarted in accordance with 3SOP-CC-001B; Component Cooling System Operation. What precautions and limitations are applicable during startup of the system?

a. ADJUST SW flow through both HXs to maintain CCW outlet temperature below 120°F while limiting the HX Outlet Isolation Valves to a maximum opening of 27.5° and 27° respectively.
b. ADJUST SW flow through both HXs to maintain CCW outlet temperature below 110°F while limiting the HX Outlet Isolation Valves to a maximum opening of 27.5° and 27° respectively.
c. ADJUST SW flow through both HXs to maintain CCW outlet temperature below 110°F while limiting the HX Outlet Isolation Valves to a maximum opening of 37.5° and 37° respectively.
d. ADJUST SW flow through both HXs to maintain CCW outlet temperature below 120°F while limiting the HX Outlet Isolation Valves to a maximum opening of 37.5° and 37° respectively.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because when placing RHR in service, then a CCW inlet temperature of 120°F is allowable for a period of 2 hours. The maximum opening of the valves is correct also. Unit is in MODE 3, RHR is not being placed in service.

b. Correct. The procedure states to adjust SW flow through the HXs to maintain CCW outlet temperature below 110°F, while limiting the HX outlet valves to a maximum opening of 27.5° and 27° respectively.
c. Incorrect. Plausible because the temperature requirement of 110°F is correct, but the valve opening limitation is incorrect.
d. Incorrect. Plausible because when placing RHR in service, then a CCW inlet temperature of 120°F is allowable for a period of 2 hours. The maximum opening of the valves is correct also. Unit is in MODE 3, RHR is not being placed in service. The valve opening limitation is also incorrect.

Technical References:	3-SOP-CC-001B, Rev. 33, I3LP-ILO-CCW001
Proposed References to be provided:	None
Learning Objective:	Objectives 1.0 & 2.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 4 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000027AK303 Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Actions contained in EOP for PZR PCS malfunction	
Importance:		3.7	4.1

Question: #8

Indian Point 3 has experienced a Reactor Trip and a Safety Injection. E-0 (Reactor Trip or Safety Injection) has been completed through step 13. Step 14 is checking the status of Pressurizer PORVs, Safeties and Spray Valves. The following conditions are observed:

- All Reactor Coolant Pumps are running
- RCS Pressure is 1700 psig and lowering
- Both PZR PORVs are closed
- PZR Safety Tailpipe Temperatures and Acoustic Monitors are normal
- Both PZR Spray Valves are opened

The ATC Operator has attempted to manually close the Spray Valves, but they remain full open. Describe the next action attempted by the procedure to mitigate these malfunctioning spray valves. If valve(s) NOT closed,

a. THEN STOP ALL RCP(s) to stop spray flow.
b. THEN STOP ANY RCP required to stop spray flow.
c. PULL CONTROLLER FUSE for failed spray valve(s) to stop spray flow.
d. THEN GO TO 3-E-1, LOSS OF REACTOR OR SECONDARY COOLANT.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because stopping the RCPs will stop the RCS depressurization caused by the failed open spray valves. However, the EOP deviation document states that the procedure reflects IP3's design by addressing removal of control power fuses. This addition promotes the ERG's intent for this substep's RNO and attempts to do so without forcing an RCP be removed from service.
b. Incorrect. Plausible because stopping the required RCPs will stop the RCS depressurization caused by the failed open spray valves. However, the EOP deviation document states that the procedure reflects IP3's design by addressing removal of control power fuses. This addition promotes the ERG's intent for this substep's RNO and attempts

to do so without forcing an RCP be removed from service. This step is also performed if the valves fail to close by removal of the control power fuses. Question asked for the "next" step after attempting to

c. Correct. The EOP deviation document states that the procedure reflects IP3's design by addressing removal of control power fuses. This addition promotes the ERG's intent for this substep's RNO and attempts to do so without forcing an RCP be removed from service.
d. Incorrect. Plausible because this is the action taken if the PORVs and their associated Block Valves cannot be closed.

Technical References:	I3LP-ILO-EOPE00, 3-E-0r4
Proposed References to be provided:	None
Learning Objective:	Objective 12.0
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 5, 7, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000029EK206 Knowledge of the interrelations between the and the following an ATWS: Breakers, relays, and disconnects	
Importance:		2.9	3.1

Question: #9

Given the following:

- Reactor Trip testing is in progress on Train “A”.
- Reactor Trip Breaker “A” is open.
- Reactor Trip Bypass Breaker “A” is closed.
- A transient occurs requiring a reactor trip.
- The RO attempts to manually trip the reactor but the reactor does NOT trip.

Which ONE of the following describes a malfunction that has contributed to the reactor trip failure?

a. Undervoltage Trip coils failed to energize; Shunt Trip coils failed to de-energize.
b. Undervoltage Trip coils failed to energize; Shunt Trip coils failed to energize.
c. Undervoltage Trip coils failed to de-energize; Shunt Trip coils failed to energize.
d. Undervoltage Trip coils failed to de-energize; Shunt Trip coils failed to de-energize.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the UV coils energize to trip and the Shunt coils de-energize to trip.
b. Incorrect. Plausible because the student may believe that the UV coils energize to trip.
c. Correct. UV coils de-energize to trip and the Shunt coils energize to trip (DC).
d. Incorrect. Plausible because the student may believe that the Shunt coils de-energize to trip.

Technical References:	I3LP-ILO-ICRXP
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	Bank (Indian Point Common)
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000054AA206 Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): AFW adjustments needed to maintain proper T-ave. and S/G level	
Importance:		4.0	4.3

Question: #10

Indian Point Unit 3 is at 100% when the 31 MBFP trips, operators take actions per AOP-FW-1; Loss of Feedwater. Which of the following describes the status of the Main and Auxiliary Feedwater Systems after the initial procedural actions are taken by the operators?

a. Main Feedwater (32 MBFP) is still in service as operators have reduced load to less than 650MWe. Auxiliary Feedwater is not in service.
b. Main Feedwater (32 MBFP) is still in service as operators have reduced load to less than 650MWe. AOP-FW-1 also starts both 31 and 33 ABFPs.
c. Main Feedwater is isolated due to a Feedwater Isolation Signal and the Auxiliary Feedwater System is feeding the Steam Generators at greater than 365 gpm flow until at least one SG NR level is greater than 9%.
d. Main Feedwater is isolated due to a Feedwater Isolation Signal and the Auxiliary Feedwater System is feeding the Steam Generators at greater than 365 gpm flow until all SG NR levels are greater than 14%.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because if initial power level is ≤ 80%, per attachment 1, load is reduced to < 650 MWe. However, the AOP also starts the motor driven aux feed pumps as well.
b. Incorrect. Plausible because if initial power level is ≤ 80%, per attachment 1, load is reduced to < 650 MWe. The AOP also starts the motor driven aux feed pumps as well.
c. Correct. Because power is > 80%, the AOP directs the operator to trip the reactor and go to E-O; Reactor Trip or Safety Injection. Once in ES-0.1; Reactor Trip Response, the operator would verify proper auxiliary feedwater flow > 365 gpm until at least one SG NR level is greater than 9%. The main and bypass FRVs receive a FW Isolation Signal (< 554°F & Trip Breakers open)
d. Incorrect. Plausible because power is > 80% and the AOP directs the operator to trip the reactor and go to E-O; Reactor Trip or Safety Injection. Once in ES-0.1; Reactor Trip Response, the operator would verify proper auxiliary feedwater flow > 365 gpm until at least

one SG NR level is greater than 9%, not 14%. 14% is plausible as it is the minimum level for adverse containment conditions. The main and bypass FRVs receive a FW Isolation Signal (< 554°F & Trip Breakers open)

Technical References:	I3LP-ILO-AOPFW1 I3LP-ILO-EOPE00 3-ES-0.1 3-AOP-FW-1
Proposed References to be provided:	None
Learning Objective:	Objective 3.0 and Objective 12.0 respectively
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 5, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000055EK102 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling	
Importance:		4.1	4.4

Question: #11

The unit experiences a reactor trip concurrent with a loss of off-site power. The Emergency Diesel Generators have either failed to start or tripped, a loss of all 480V buses has occurred.

Ten minutes after the loss of all 480V bus event, the following plant conditions exist:

- All S/G Pressures are approximately 1030 psig and stable
- RCS Pressure is 2185 psig and stable
- T-hot is approximately 575°F in all loops and lowering slowly
- Core Exit Thermocouples indicate approximately 580°F
- T-cold is approximately 550°F in all loops and stable

Based on the above conditions, what is the present status of RCS cooling and what procedural actions are required to attempt to mitigate the event if a vital 480V AC bus is not restored?

a. Natural Circulation exists. Initiate a rapid SG Depressurization to allow the RCS Accumulators to inject even if voiding exists in the reactor vessel head.
b. Natural Circulation exists. Initiate a rapid RCS Depressurization by opening both PORVs to allow the RCS Accumulators to inject.
c. Natural Circulation does not exist. Initiate a rapid SG Depressurization to allow the RCS Accumulators to inject even if voiding exists in the reactor vessel head.
d. Natural Circulation does not exist. Initiate a rapid RCS Depressurization by opening both PORVs to allow the RCS Accumulators to inject.

Answer: a

Explanation / Justification

a. Correct. Based on given plant conditions, natural circulation exists. Saturation temperature for the SG Pressure of 1030 psig is 550°F and the saturation temperature for RCS Pressure of 2185 psig is 649.5°F. Therefore all parameters are trending properly and RCS subcooling is > 40°F (69.5°F). Additionally, step 20 of 3-ECA-0.0 initiates a rapid SG depressurization even if voids form in the head (note prior to step 20).

b. Incorrect. Plausible because based on given plant conditions, natural circulation exists. See above. However, the Loss of all AC procedure does not initiate a rapid RCS depressurization using the PORVs. Plausible because the Red Path FR-H.1 does initiate a "Bleed and Feed" using PORVs.
c. Incorrect. Plausible because steam tables and simple calculations are required, student also needs to know EOP Natural Circulation criteria. Second part of distractor is correct, step 20 of 3-ECA-0.0 initiates a rapid SG depressurization even if voids form in the head (note prior to step 20).
d. Incorrect. Plausible because steam tables and simple calculations are required, student also needs to know EOP Natural Circulation criteria. However, the Loss of all AC procedure does not initiate a rapid RCS depressurization using the PORVs. Plausible because the Red Path FR-H.1 does initiate a "Bleed and Feed" using PORVs.

Technical References:	I3LP-ILO-EOPC00
Proposed References to be provided:	None
Learning Objective:	Objectives 3.0 & 11.0
Question Source:	New
Question Cognitive Level:	Comprehension / Analysis
10CFR Part 55 Content:	55.41 (b) 8 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000057G2411 Loss of Vital AC Electrical Instrument Bus: Knowledge of abnormal condition procedures	
Importance:		4.0	4.2

Question: #12

The plant is in a normal 100% power lineup with Channel B selected for feed flow and steam flow control. Which set of actions will be required per 3-AOP-IB-1, Loss of Power to an Instrument Bus, for a loss of 31 and 31A Instrument Busses?

I. Swap feed flow and steam flow to the A channel
II. Place MBFPs in Manual control
III. Place MFRVs in Manual control

a. I and II
b. I and III
c. II and III
d. I, II, and III

Answer: a

Explanation / Justification

a. Correct. 3-AOP-IB-1, step 4.6 places all steam flow and feedwater flow channel transfer switches in position “A”. Step 4.10 places all MBFPs in manual control.
b. Incorrect. Plausible because III (MFRVs in manual) is a required action for a loss of 34 Instrument Bus. Selection II not being required is plausible because only one MBFP is affected by this failure and it goes into track and hold, however the procedure specifies going to manual.
c. Incorrect. Plausible because III (MFRVs in manual) is a required action for a loss of 34 Instrument Bus. Selection I not being needed is plausible because the plant does not trip if this is not done, however the procedure specifies this action.
d. Incorrect. Plausible because III (MFRVs in manual) is a required action for a loss of 34 Instrument Bus.

Technical References:	I3LP-ILO-AOPIB1R1, 3-AOP-IB-1; Rev. 8
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	Bank (IP3)
Question Cognitive Level:	Knowledge

10CFR Part 55 Content: 55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000058AK302 Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of dc power	
Importance:		4.0	4.2

Question: #13

During the performance of ES-0.1, Reactor Trip Response, the operators are directed to manually open generator breakers 1 and 3 if they do not open automatically 30 seconds following a turbine trip. If a loss of DC control power occurred for generator breakers 1 and 3, the operator is directed by 3-AOP-DC-1, "Loss of a 125V DC Panel", to locally trip generator breakers 1 and 3 if a unit trip has occurred and the generator breakers 1 and 3 have not tripped.

Which of the following is the overriding concern?

a. Reverse power in the Main Transformer
b. Tripping of RCP's on Low Frequency
c. Motorizing the Main Generator
d. Unit Auxiliary Transformer damage

Answer: c

Explanation / Justification

a. Incorrect. Plausible because student may believe that the main transformer is the concern. Main transformer can send power either way. It is not a generator.
b. Incorrect. Plausible because student may believe that RCP pumping capability is the concern. Grid will maintain proper frequency.
c. Correct. Breakers must be opened to disconnect from grid.
d. Incorrect. Plausible because student may believe that the Unit Aux Transformer is the concern. UAT will not be damaged by being connected, not a generator.

Technical References:	I3LP-ILO-AOPDC1
Proposed References to be provided:	None
Learning Objective:	Objective 2.0
Question Source:	Bank – IP3 2006 NRC Exam
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	0000622139	
		Loss of Nuclear Service Water: Knowledge of conservative decision making practices	
Importance:		3.6	4.3

Question: #14

Unit 3 is operating at 100%. A low pressure condition exists in the Service Water System due to a leak on the essential service water header. The crew has entered 3-AOP-SW-1; Service Water Malfunction.

Given the following conditions:

- Essential Service Water Pressure is 55 psig
- No temperature alarms associated with essential service water header cooled components
- Leak isolation requires the entire Essential Service Water Header to be isolated
- Maintenance has estimated that the repair will take 24 hours

What operator actions should be taken in accordance with 3-AOP-SW-1?

a. Trip the reactor and Initiate E-0, continue in parallel with 3-AOP-SW-1 and transfer essential headers IAW 3-SOP-RW-005; Service Water System Operation. Following transfer of essential headers, isolate the leak.
b. Initiate plant shutdown to MODE 3, initiate transfer of essential headers IAW 3-SOP-RW-005; Service Water System Operation, and when the leak can be isolated, isolate leak.
c. Maintain Reactor Power, transfer essential headers IAW 3-SOP-RW-005; Service Water System Operation, and when leak can be isolated, isolate leak.
d. Initiate plant shutdown to MODE 3 and when the leak can be isolated, isolate leak. Non-essential service water header is required to remain operable in MODE 3, so the procedure does not initiate transfer of the essential headers.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because if service water pressure is < 50 psig and system cooling ability is affected, then operator is directed to trip the reactor and initiate E-0. Not required because essential service water header pressure is 55 psig. Remaining actions are IAW procedure.
b. Correct. Because service water pressure is > 50 psig, a reactor trip is not required. However, since isolation of the leak requires the entire Essential Service Water Header to

be isolated (shutdown), a plant shutdown to MODE 3 is initiated (see steps 4.30-4.35 of 3-AOP-SW-1). Because the essential header is the affected header, the procedure first initiates transfer of essential headers IAW 3-SOP-RW-005; Service Water System Operation.

c. Incorrect. Plausible because the student may believe that technical specifications allow inoperability of the essential header (one header) for greater than the 24 hour repair estimate. Student could believe it is a standard 72 hour action statement. It is actually a 12 action statement.
d. Incorrect. Plausible because the plant shutdown to MODE 3 is correct, but although technical specifications require the operability of both headers in MODE 3, the procedure does initiate the transfer of essential headers (Diesels, Instrument Air, Control Room Ventilation, etc.).

Technical References:	I3LP-ILO-SW001; 3-AOP-SW-1 Background document for 3-AOP-SW-1
Proposed References to be provided:	None
Learning Objective:	Objective 6.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7 & 10

NOTE: Conservative decision making enters into plant shutdown because of the requirement to isolate entire essential header. Conservative decision making also enters into plant shutdown because of the repair estimate by maintenance being 24 hours. This is greater than 12 action time. Trip not required because procedural IAAT condition is not met.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	000065AA205 Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing	
Importance:		3.4	4.1

Question: #15

The Unit is at 100% when a line break in the instrument air header in the PAB requires entry into 3-AOP-AIR-1; Air Systems Malfunction.

Which of the following plant conditions require that the operators trip the reactor and initiate E-0 (Reactor Trip or Safety Injection)?

a. Instrument Air Pressure < 90 psig.
b. Volume Control Tank Level lowers to 9%.
c. Multiple AOVs begin repositioning.
d. Pressurizer Level deviation of 5% from program.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because IAAT Instrument Air pressure is < 60 psig, the procedure requires tripping the reactor. Also another decision point exists at < 95 psig to send local operator to restore instrument air locally.
b. Incorrect. Plausible because IAAT VCT Level lowers to < 5%, the procedure requires tripping the reactor.
c. Correct. As per step 4.1 of the procedure, IAAT <u>any</u>; which includes "Multiple AOVs begin repositioning".
d. Incorrect. Plausible because charging pump speed is adjusted per the AOP, however this is not trip criteria.

Technical References:	I3LP-ILO-IA001, Rev. 3, 3-AOP-AIR-1, Rev. 4
Proposed References to be provided:	None
Learning Objective:	Objective 7.0
Question Source:	Modified - Bank U2
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00WE04EA130 Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment): Desired operating results during abnormal and emergency situations	
Importance:		3.8	4.0

Question: #16

The crew has completed the actions of ECA-1.2; LOCA Outside Containment. The following plant conditions now exist:

- Pressurizer level is 35% and slowly lowering.
- Steam Generator narrow range levels are 30%.
- Subcooling is 35°F.
- RCS pressure is 1400 psig and slowly lowering.

Which ONE (1) of the following procedures will be performed next?

a. ES-1.2, Post-LOCA Cooldown and Depressurization
b. ECA-1.1, Loss of Emergency Coolant Recirculation
c. E-0, Reactor Trip or Safety Injection
d. E-1, Loss of Reactor or Secondary Coolant

Answer: b

Explanation / Justification

a. Incorrect. Plausible because ECA-1.1 actions include initiating a cooldown to cold shutdown and depressurizing the RCS to minimize subcooling. Student may remember required actions and assume they could be accomplished with a transition to ES-1.2, Post-LOCA Cooldown and Depressurization.
b. Correct. After all potential leak isolations have been completed, ECA-1.2 checks if RCS Pressure is rising. The RNO is to transition to ECA-1.1, Loss of Emergency Coolant Recirculation because a leak outside containment still exists.
c. Incorrect. Plausible because ECA-1.2 can be entered from E-0 (step 22) and student may believe that the transition is back to procedure and step in effect, however there is no transition to E-0.
d. Incorrect. Plausible because a transition to E-1 is correct if the leak was isolated and RCS Pressure is rising.

Technical References:	I3LP-ILO-EOPC10, 3-ECA-1.2
Proposed References to be provided:	None
Learning Objective:	Objective 7.0

Question Source:	Modified IP Common Bank (IP2 2012 Exam)
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00WE05A102 Ability to operate and / or monitor the following as they apply to the (Loss of Secondary Heat Sink): Operating behavior characteristics of the facility	
Importance:		3.7	4.0

Question: #17

Given the following conditions:

- A reactor trip with Safety Injection (SI) occurs from 100% power
- All SG NR levels are offscale low
- Minimum AFW flow cannot be established
- The crew has entered FR-H.1; "Response to Loss of Secondary Heat Sink"
- Reactor Coolant System pressure is 300 psig and lowering
- Steam Generator pressures are all 600 psig and stable
- The operators are directed by FR-H.1 to return to procedure and step in effect

Based on this information, which ONE of the following statements correctly summarizes plant conditions?

a. Steam line rupture in progress and secondary heat sink is NOT required.
b. Large-break Loss of Coolant Accident in progress and secondary heat sink is NOT required.
c. Steam Generator Tube Rupture in progress and secondary heat sink is required.
d. Small-break Loss of Coolant Accident in progress and secondary heat sink is required.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that feedwater flow has purposely been reduced / isolated by procedural action. The first "Caution" in FR-H.1 states the following; "If total feedflow is less than 365 GPM due to operator action, this procedure should not be performed."
b. Correct. RCS pressure < non faulted SG indicates a larger LOCA, SGs no longer act as a heat sink; core decay heat is being removed by break flow.
c. Incorrect. Plausible because a SGTR is a small break LOCA and requires a secondary heat sink. The student may also believe that actions have been taken to reduce RCS pressure less than ruptured SG pressure.
d. Incorrect. Plausible because a SBLOCA does require a secondary heat sink.

Technical References:	I3LP-ILO-EOPFRH
Proposed References to be provided:	None
Learning Objective:	Objective 7.0
Question Source:	Bank, IP Common
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10 & 14

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	00WE12EK202	
		Knowledge of the interrelations between the (Uncontrolled Depressurization of all Steam Generators) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	
Importance:		3.6	3.9

Question: #18

During the performance of ECA-2.1, "UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS", the following condition exists:

- Cooldown rate of the RCS is greater than 100°F/hour

How is the operator directed to control feedwater flow?

a. Feedwater flow is terminated to all but a single S/G, preferably an intact S/G.
b. Feedwater flow is maintained at least 365 gpm total until a narrow range level is greater than 9% (14%).
c. Feedwater flow is terminated to all S/Gs until temperature stabilizes.
d. Feedwater flow is reduced to 100 gpm to all S/Gs with narrow range levels less than 9% (14%).

Answer: d

Explanation / Justification

a. Incorrect. Plausible because a similar strategy is used in FR-H.1 for feeding a hot dry S/G.
b. Incorrect. Plausible because 365 gpm is the minimum normal feedwater flow to ensure adequate heat sink if S/G level is less than 9% (14%) per 3-E-0, Reactor Trip or Safety Injection.
c. Incorrect. Plausible because feedwater flow is terminated (isolated) to individual faulted S/Gs in EOP 3-E-2, Faulted Steam Generator Isolation to limit energy blowdown to the containment.
d. Correct. 3-ECA-2.1 states in the RNO for step 2.b (cooldown rate < 100°F/HR) to reduce feed flow to 100 gpm to each SG. Caution also states; "A minimum feed flow of 100 gpm must be maintained to each SG with a narrow range level less than 9% (14%)."

Technical References:	I3LP-ILO-EOPE20, 3-ECA-2.1
Proposed References to be provided:	None
Learning Objective:	Objective 4.0
Question Source:	Bank (IP3 - # 24602)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000001AK102 Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: SUR	
Importance:		3.6	3.9

Question: #19

Given the following:

- The Reactor is critical at 10^{-8} amps
- The ATC operator withdraws control rods to continue raising power
- Controls rods continue to move after the In-Hold-Out switch is released

Which of the following identifies the INITIAL indications for the event in progress?

	SUR	PRZR Level	Tavg
a.	Increase	Increase	Increase
b.	Stable	Increase	Increase
c.	Increase	Stable	Stable
d.	Stable	Stable	Increase

Answer: c

Explanation / Justification

a. Incorrect. Plausible because these indications are correct if the continuous rod withdrawal occurred > POAH.
b. Incorrect. Plausible because candidate may believe that the increase in Tavg will offset the reactivity from the rods. Below the POAH Tavg and PRZR level will not increase
c. Correct. SUR will continue to increase and PZRZ Level and Tavg will not change below the POAH.
d. Incorrect. Plausible because the student must know the reactor is below the point of adding heat.

Technical References:	I3LP-ILO-AOP-ROD-1, 3-AOP-ROD-1
Proposed References to be provided:	None
Learning Objective:	Objective ?
Question Source:	Bank (IP3 Question # 25621)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 5 & 6

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000003AA203 Ability to determine and interpret the following as they apply to the Dropped Control Rod: Dropped rod, using in-core/excore instrumentation, in-core or loop temperature measurements	
Importance:		3.6	3.8

Question: #20

Given the following initial plant conditions:

- The operating crew is performing 3-POP-1.3, Plant Startup from Zero to 45% Power
- Indicated NIS Reactor Power is 25%
- Control Bank D is at 149 steps
- T_{avg} is 552°F

The Control Room then receives indication that the Control Bank D Rod located at core location H-8 has dropped into the core. This includes the following indications:

- The Rod Bottom Light for Location H-8 illuminates
- IRPI for Location H-8 indicates 0 inches
- All systems operate as designed in automatic

Which of the following additional plant indications will verify that the control rod has dropped and it is not an IRPI failure?

a. T_{avg} has lowered to 548°F and all four Power Range NIs indicate approximately 23% Reactor Power.
b. T_{avg} and Reactor Power have been restored to original values after an Auto Rod withdrawal.
c. T_{avg} has lowered to 548°F and an asymmetric power distribution is seen by the Power Range NIs, power in one channel lower than the remaining three channels.
d. Both T_{avg} and Reactor Power NIs indicate an asymmetric power distribution is seen by the Power Range NIs, power in one channel lower than the remaining three channels.

Answer: a

Explanation / Justification

a. Correct. Core location H-8 is the actual center of the core, therefore the power distribution will be symmetric, but will lower. Both RCS Temperature and NI power indications will lower.
b. Incorrect. Plausible because the student may believe that rod control will see the temperature deviation and restore it to program value. This cannot happen due to the designed rod stop (Rod Bottom Rod Stop at a minimum).
c. Incorrect. Plausible because in most conditions (rods not in the center of the core), the operator will see an asymmetric power distribution. One quadrant will indicate low, while the other channel will actually indicate higher than original levels.
d. Incorrect. Plausible because in most conditions (rods not in the center of the core), the operator will see an asymmetric power distribution. See FSAR section 14.1.3.1.

Technical References:	I3LP-ILO-ICROD
	System Description 16.1, Rod Control System
Proposed References to be provided:	None
Learning Objective:	Objective G
Question Source:	Modified Bank Question - # 25643
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 5, 6, 7, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000005AK201 Knowledge of the interrelations between the Inoperable / Stuck Control Rod and the following: Controllers and positioners	
Importance:		2.5	2.5

Question: #21

Given the following Unit 3 conditions:

- The Unit is at 85% reactor power with Control Bank "D" rods at 190 steps.
- A single control rod in Control Bank "D" is at 160 steps.
- The shift has entered 3-AOP-ROD-1; Rod Control and Indicating Systems Malfunction.
- The Rod was verified misaligned, I&C has corrected a blown "movable coil" fuse, and the control rod has been realigned IAW 3-AOP-ROD-1.
- During the implementation of the rod recovery per the procedure, poor procedure place keeping has resulted in the P/A converter not being reset.

After restoring the Rod Control System to Automatic, a plant transient occurs and rods begin to drive in.

Relative to ACTUAL Rod Insertion Limits, at what rod heights will the alarms "ROD INSERTION LOW LIMIT" and "ROD INSERTION LOW-LOW LIMIT" actuate?

a. Alarms will actuate 36 steps and 31 steps respectively ABOVE Rod Insertion Limit
b. Alarms will actuate 24 steps and 29 steps respectively BELOW Rod Insertion Limit
c. Alarms will actuate 20 steps and 29 steps respectively BELOW Rod Insertion Limit
d. Alarms will actuate 40 steps and 31 steps respectively ABOVE Rod Insertion Limit

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may confuse rod direction and P/A converter readings relative to the rod recovery. Would be correct if the misaligned rod was above the remainder of the rods in its group. Alarm set points from the actual COLR are correct (low 6 steps above limit & low-low 1 step above limit).
b. Correct. The step counter would have been reset to the stuck rod height and then the stuck rod withdrawn 30 steps in order to realign with the remainder of the bank. Because the P/A converter was not reset, the P/A converter is going to believe that Control Bank "D" is at 220 steps (190+30=220). The ARP 3-ARP-003, page 57 for "Rod Insertion Low Limit" states

that the alarm comes in at 6 steps or 3.75 inches above variable limit. Therefore the "low" alarm will come in (30-6=24) steps below the actual rod insertion limit. The "low-low" comes in 1 step above the variable limit (page 59). Therefore the "low-low" alarm will come in (30-1=29) steps below the actual rod insertion limit.

c. Incorrect. Plausible because the rod direction and P/A converter readings relative to the rod recovery are correct. Also plausible because the student may believe that the "low-low" insertion limit alarm is set to come in at 10 steps above the variable limit.
d. Incorrect. Plausible because the student may confuse rod direction and P/A converter readings relative to the rod recovery. Alarm for "low-low" insertion alarm is correct if the misaligned rod was above the remainder of the rods in its group. However, the "low" insertion alarm is not set to come in at 10 steps above the variable limit.

Technical References:	I3LP-ILO-AOPROD, 3-AOP-ROD-1
Proposed References to be provided:	None
Learning Objective:	Objective 2.0
Question Source:	Modified Bank (IP3 Question #20491)
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 6, 7, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000024AK301 Knowledge of the reasons for the following responses as they apply to Emergency Boration: When emergency boration is required	
Importance:		4.1	4.4

Question: #22

Which of the following conditions would require initiation of EMERGENCY BORATION in accordance with 3-ONOP-CVCS-3; Emergency Boration?

I. Panel SAF – Reactor Coolant System, ROD INSERTION LOW LIMIT Alarm.
II. Panel SAF – Reactor Coolant System, ROD INSERTION LOW-LOW LIMIT Alarm.
III. Any control rod greater than or equal to 20 STEPS with the Reactor Trip Breakers open.
IV. Uncontrolled RCS cooldown to less than 547°F with any control rod greater than 20 STEPS with the Reactor Trip Breakers open.
V. More than one control rod is greater than or equal to 20 STEPS with the Reactor Trip Breakers open.
VI. Uncontrolled RCS cooldown to less than 500°F with the Reactor Trip Breakers open.

a. I, IV, V
b. II, III, VI
c. III, IV, V
d. II, V, VI

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the alarm response for I3 requires a "normal" boration to clear the alarm, also IV would be correct if cooldown was less than 540°F, and V is correct. Incorrect.
b. Incorrect. Plausible because II is correct, student may believe that any rod stuck out greater than or equal to 20 steps is correct, and VI is correct.
c. Incorrect. Plausible because student may believe that any rod stuck out greater than or equal to 20 steps is correct, also IV would be correct if cooldown was less than 540°F, and V is correct. .
d. Correct. II, V, and VI are all correct. See 3-ES-0.1; Reactor Trip Response, step 5 and 3-ONOP-CVCS-3; Emergency Boration.

Technical References:	3-ES-0.1; Reactor Trip Response, 3-ONOP-CVCS-3; Emergency Boration, I3LP-ILO-ONPCVC I3LP-ILO-EOPE00
Proposed References to be provided:	None
Learning Objective:	Objective 1.0 & Objective 12.0 respectively
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 5, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	000059AA202 Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: The permit for liquid radioactive-waste release	
Importance:		2.9	3.9

Question: #23

In accordance with 3-SOP-WDS-014; Liquid Waste Releases, the Liquid Radioactive Waste Release Permit Form requires documenting the results of the Radiation Monitor Source Check. While completing the waste release permit for the release of 31 Waste Monitor Tank the R-18 Channel 1 failed its source check.

What actions are required in accordance with the procedure and the ODCM to allow the release authorization of the 31 Waste Monitor Tank?

a. As long as the radioactive liquid contained in the 31 Waste Monitor Tank is verified to be ≤ 10 Curies, excluding tritium and dissolved or entrained gases, then the release can be authorized.
b. As long as the volume of the 31 Waste Monitor Tank is verified by tank level indication to be ≤ 90% and the liquid waste discharge flow meter (FT-1064) is operable, then the release can be authorized.
c. As long as a satisfactory check-source test of R-18 Channel 2 has been completed, then the release can be authorized.
d. As long as two (2) independent samples have been taken, the release rate calculations independently verified, and the discharge valve alignment independently verified, then the release can be authorized.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because ODCM Spec D 3.1.4; Liquid Holdup Tanks states that outdoor liquid storage tanks (like 31 Monitor Tank) shall be limited to ≤ 10 Curies. However, the procedure and ODCM Spec 3.3.1 require independent samples, independently verified release rate calculations, and independently verified discharge valve alignments in order to authorize a release with an inoperable rad monitor.
b. Incorrect. Plausible because one way that liquid effluent curie content is limited is by limiting the volume of the tank. Student may believe that knowing the tank volume, activity, and

then discharge flow rate is enough to authorize discharge. ODCM and procedure require independent samples, calculations, and lineup verification.

c. Incorrect. Plausible because the student may believe that R-18 Channel 2 is utilized for liquid waste discharges. See NOTE prior to step 4.1.6 of 3-SOP-WDS-014, which states: "Do NOT perform a check-source check of R-18 Channel 2 as an inability to remove R-18 Channel 2 from service may occur after test. Channel 2 is NOT used."
d. Correct. See step 4.1.13 of 3-SOP-WDS-014 and ODCM Spec 3.3.1. The release may be authorized if independent samples, independently verified release rate calculations, and independently verified discharge valve alignments are completed.

Technical References:	I3LP-ILO-LWR001, 3-SOP-WDS-014 ODCM D 3.3.1, Radioactive Liquid Effluent Monitoring Instrumentation
Proposed References to be provided:	None
Learning Objective:	Objective 7
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7, 10, & 11

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	0000762128 High Reactor Coolant Activity: Knowledge of the purpose and function of major system components and controls	
Importance:		4.1	4.1

Question: #24

Given the following:

- Unit 3 is at 100% power
- R63A and B (Gross Failed Fuel Detectors) are in Alert
- Chemistry has confirmed that high RCS activity exists
- Chemistry has also confirmed that RCS activity is less than TS 3.4.16 limits
- 3-AOP-HIACT-1; High RCS Activity has been entered

Which of the following actions should be taken due to the rising RCS activity in accordance with 3-AOP-HIACT-1 and the reason for the actions?

a. Raise letdown flow to a maximum of 120 gpm to raise the rate of activity removal via the CVCS demineralizers.
b. Lower letdown flow to minimum (45 gpm) to minimize the amount of radioactive letdown that is flowing through the auxiliary building.
c. Conduct a Feed and Bleed of the RCS by diverting Letdown and initiating a continuous makeup to the VCT to rapidly reduce activity levels.
d. Place the Cation bed demineralizer in service and remove the mixed bed demineralizers in order to facilitate removal of fission products from the RCS.

Answer: a

Explanation / Justification

a. Correct. 3-AOP-HIACT-1 ensures letdown is aligned through the CVCS demineralizers and then raises letdown flow up to a maximum of 120 gpm. The background document for 3-AOP-HIACT-1 states; “Step 4.5 increases letdown flow to increase the rate of activity removal via the CVCS demins.”
b. Incorrect. Plausible because minimizing radiation levels in the auxiliary building is prudent from an ALARA perspective, however that will not help lower RCS activity levels.
c. Incorrect. Plausible because the student may believe that diverting the RCS will help lower RCS activity levels and permit waste processing activities.

d. Incorrect. Plausible because placing the Cation bed demineralizer in service is an action in the AOP, but from a system perspective the mixed bed demineralizers cannot be removed from service and still have flow through the Cation bed.

Technical References:	I3LP-ILO-RCSPSS, 3-AOP-HIACT-1
Proposed References to be provided:	None
Learning Objective:	Objectives 4d, 5, and 6
Question Source:	Modified Bank (IP3 Question # 24189)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	00WE07EA120 Ability to operate and / or monitor the following as they apply to the (Saturated Core Cooling): Operating behavior characteristics of the facility	
Importance:		3.2	3.7

Question: #25

A SATURATED CORE COOLING condition is identified during a review of the Critical Safety Function Status Trees following a small break loss of coolant accident. Operators have transitioned to 3-E-1; Loss of Reactor or Secondary Coolant. The following conditions are noted:

- No Safety Injection pumps are available
- Both RHR pumps are running
- No RCPs are running
- Both motor driven AFW pumps are running
- RCS Pressure is 1100 psig
- CETs are 700°F and slowly rising
- RVLIS is 35% and slowly lowering
- Steam Generator pressure is 1050 psig

Is a transition to a Functional Restoration procedure required? What indications are being monitored to verify adequate core cooling?

a. Yes, a transition to 3-FR-C.2; Response to Degraded Core Cooling is required. RVLIS level, Core Exit Thermocouples, and ECCS flow are monitored to verify adequate core cooling.
b. Yes, a transition to 3-FR-C.3; Response to Saturated Core Cooling is required. RVLIS level, Core Exit Thermocouples, RCP operating status, and RCS Subcooling are monitored to verify adequate core cooling.
c. No, a transition to a functional restoration procedure is not required. RVLIS level, RCP operating status, and AFW system status are monitored.
d. No, a transition to a functional restoration procedure is not required. RVLIS level, Core Exit Thermocouples, RCP operating status, and RCS Subcooling are monitored to verify adequate core cooling.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because if CETC temperatures were 715°F or RVLIS was 33%, then the transition would be to an Orange Path 3-FR-C.2. The second part of the answer is correct, but not all inclusive. Also ECCS flow will correct an inadequate core cooling situation, but is not a parameter monitored by CSFST 3-F-0.2; Core Cooling.
b. Incorrect. Plausible because the Critical Safety Function Status Tree will direct you to the Yellow Path based on the given plant conditions. However, Yellow conditions do not require immediate operator attention and therefore do not "require transition". The CRS/SM is allowed to decide whether or not to implement any Yellow condition FRP.
c. Incorrect. Plausible because Yellow conditions do not require immediate operator attention and therefore do not "require transition". No is correct, but AFW system status is not a parameter or indication monitored by CSFST 3-F-0.2; Core Cooling.
d. Correct. Yellow conditions do not require immediate operator attention and therefore do not "require transition". The CRS/SM is allowed to decide whether or not to implement any Yellow condition FRP. RVLIS level, Core Exit Thermocouples, RCP operating status, and RCS Subcooling are monitored by CSFST 3-F-0.2; Core Cooling.

Technical References:	I3LP-ILO-EOPFRC, 3-FR-C.3 OAP-012, EOP USERS GUIDE, Section 4.3
Proposed References to be provided:	None
Learning Objective:	Objective 11.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	00WE10K303 Knowledge of the reasons for the following responses as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	
Importance:		3.4	3.6

Question: #26

ES-0.3; "Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS), checks RVLIS Full Range greater than 70% after the RCS coodown and depressurization have been started.

Why is maintaining RVLIS Full Range greater than 70% important and what actions are required should RVLIS Full Range lower to less than 70%?

a. Ensures the core is always covered with water to prevent the precipitation of boron in the upper region of the core that could reduce natural circulation flow. If RVLIS lowers to less than 70%, manually ACTUATE SI and GO To E-0, Reactor Trip or Safety Injection.
b. Ensures the reactor vessel head is maintained full of water to aid in the cooldown of the head and to prevent formation of additional steam in the upper vessel head area. If RVLIS lowers to less than 70%, the RCS depressurization is stopped while the cooldown is continued.
c. Ensures reactor vessel level is maintained above the top of the hot legs to prevent the formation of steam in the hot legs and the steam potentially reaching the top of the Steam Generator U-tubes causing a disruption in natural circulation. If RVLIS lowers to less than 70%, re-pressurize the RCS.
d. Ensures the reactor vessel level is maintained above the active fuel region of the core to ensure heat transfer via reflux boiling while controlling void growth in the reactor vessel head. If RVLIS lowers to less than 70%, transition to FR-I.3, Response to Voids in Reactor Vessel.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may believe 70% RVLIS level is the level just above active fuel and boron precipitation is a concern following the injection phase during a

LOCA. Also plausible because SI initiation criteria exits in ES-0.3, but it is PRZR level cannot be maintained > 5%.

b. Incorrect. Plausible because the student may believe that 70% RVLIS levels indicate water in the reactor head region and that having level in that region may prevent additional steam formation. Finally, the second part of the answer is partially correct. The procedure would require that the depressurization be stopped while continuing with the cooldown, however re-pressurization to reduce the size of the void would also be required.
c. Correct. Ensures reactor vessel level is maintained above the top of the hot legs to prevent the formation of steam in the hot legs and the steam potentially reaching the top of the Steam Generator U-tubes causing a disruption in natural circulation. If RVLIS lowers to less than 70%, re-pressurize the RCS.
d. Incorrect. Plausible because if level does lower below the top of the hot legs, reflux boiling may be the only method of heat transfer available. The student may believe that ES-0.3 allows a rapid cooldown and depressurization, so therefore it will allow significant void growth to ensure an expedited cooldown and depressurization. Also the student may believe that the functional recovery procedure for voids is applicable. It actually states that if natural circulation is occurring and void growth is expected, do not enter.

Technical References:	I3LP-ILO-EOPE00, 3-ES-0.3
Proposed References to be provided:	None
Learning Objective:	Objective 12.0
Question Source:	Modified Bank Question # 24605
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	00WE14K201 Knowledge of the interrelations between the (High Containment Pressure) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	
Importance:		3.4	3.7

Question: #27

Select the statement below which describes the automatic containment spray actuation signal.

a. Containment Pressure of 22 psig sensed on 2 of 3 containment pressure detectors PT-948A, B, C.
b. Containment Pressure of 22 psig sensed by 2 of 3 containment pressure detectors from PT-948A, B, C **AND** 2 of 3 containment pressure detectors PT-949A, B, C.
c. Containment Pressure of 22 psig sensed by 2 of 3 containment pressure detectors from PT-948A, B, C **OR** 2 of 3 containment pressure detectors PT-949A, B, C.
d. Containment Pressure of 22 psig sensed on any 4 containment pressure detectors PT-948A, B, C and PT-949A, B, C.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe the coincidence for containment spray actuation is only 2/3 on one channel. Pressure setpoint is correct.
b. Correct. The signal is a coincident 2/3 from two separate channels. See System Description 10.2, page 5.
c. Incorrect. Plausible because the student may believe that for redundancy there are two sets of instruments, but only one channel required to initiate spray.
d. Incorrect. Plausible because the student may believe that for redundancy there are multiple instruments, but only any four are required to initiate spray.

Technical References:	I3LP-ILO-CS001
Proposed References to be provided:	None
Learning Objective:	Objective E-5
Question Source:	Bank Question # 23848
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003000A103 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP motor stator winding temperatures	
Importance:		2.6	2.6

Question: #28

Given the following conditions:

- The Reactor is in Mode 5
- RCS pressure is 350 psig
- RCS temperature is 190°F
- 34 RCP is in service
- RHR is in service

Which of the following conditions if associated with 34 RCP would require the immediate tripping of the pump?

a. Shaft vibration reading of 20 mils and stable.
b. Motor winding (stator) temp. on 4-point recorder of 255°F for 52 hours.
c. Shaft vibration reading of 15 mils and rising at 0.2 mils/hr.
d. #1 seal return flow reading 5.8 gpm.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because 3-AOP-RCP-1, STEP 4.1 requires stopping affected RCPs if Shaft Vibration is > 20 mils. In this case, vibration is stable at 20 mils.
b. Correct. 3-AOP-RCP-1, STEP 4.1 requires stopping affected RCPs if Motor winding (stator) temp on 4-point recorder > 250°F for > 50 hours.
c. Incorrect. Plausible because 3-AOP-RCP-1, STEP 4.1 requires stopping affected RCPs if Shaft vibration > 15 mils and increasing > 1.0 mils/hr. Also the IAAT for frame vibration uses increasing > 0.2 mils/hr.
d. Incorrect. Plausible because 3-AOP-RCP-1, STEP 4.1 requires stopping affected RCPs if #1 seal return flow > 6 gpm (5.99 gpm digital).

Technical References:	I3LP-ILO-AOPRCP
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 5 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004000K604	
		Knowledge of the effect of a loss or malfunction on the following CVCS components: Pumps	
Importance:		2.8	3.1

Question: #29

Given the following conditions:

- Unit is at 50% Power
- All Control Systems are in Automatic

Which of the following malfunctions will result in an increased actual NPSH available for the in service Charging Pump?

a. TCV-130, Non-Regen. Hx. CCW Outlet Flow Valve fails open.
b. Controlling Pressurizer Level Channel fails to 25% indicated level.
c. LT-112, VCT Level Channel fails full scale high.
d. TE-130, Non-Regen. Hx. Outlet temperature device fails full scale low.

Answer: a

Explanation / Justification

a. Correct. If the CCW control valve to the NRHX fails open, letdown temperature will lower which will result in a lower VCT temperature and therefore a higher actual NPSH available to the running Charging Pump.
b. Incorrect. Plausible because the student is required to understand PZR Level Control and how that will impact the charging and letdown system. A failed PZR Level (controlling channel) will result in a level deviation causing the level controller to raise charging pump speed to in turn raise charging flow. However, a rise in charging pump speed will result in a decrease in actual NPSH available to the running charging pump. Knowledge of NPSH relationships to temp, flow, pressure, level, and speed also required.
c. Incorrect. Plausible because the student is required to understand VCT level control and what will happen to actual level without operator action. The student may also believe that either makeup or swap over to the RWST occurs automatically. However, with the VCT level channel failed high, the LCV-112A will direct all letdown flow to the CVCS HUT and this will result in a lowering of VCT level and therefore a lowering of actual NPSH available to the running Charging Pump. With LT-112 failed high, no auto makeup or swapping to the RWST will occur.

d. Incorrect. Plausible because the student is required to understand how TE-130 is used for letdown temperature control and then how letdown temperature affects charging pump NPSH. If the TE-130 fails low, it sends a low temperature signal to the HTC-130 which controls the TCV-130 valve (CCW control valve to the NRHX) in the CCW system causing the TCV-130 to close. Closing the TCV-130 results in rising Letdown temperature, which then results in rising VCT temperature and a decrease in actual NPSH available to the running charging pump.

Technical References:	I3LP-ILO-CVC001 3-ARP-009 3-AOP-CVCS-1 System Description 3.0
Proposed References to be provided:	None
Learning Objective:	Objective 6.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005000K505 Knowledge of the operational implications of the following concepts as they apply to the RHRS: Plant response during "solid plant" pressure change due to the relative incompressibility of water.	
Importance:		2.7	3.1

Question: #30

Given the following plant conditions:

- Mode 5
- 31 RHR Pump is in shutdown cooling mode
- Low Pressure purification is in service
- The pressurizer is water solid
- RCS temperature is 166°F
- Letdown Pressure Control Valve PCV-135 is in automatic
- RCS pressure is being maintained at 275 psig

The 31 RHR Heat Exchanger Outlet Flow Control Valve (HCV-638) has just stroked from an initial position of 25% open to the full open position due to a circuit failure. How does the Letdown system initially respond to this valve failure?

a. PCV-135 OPENS, Letdown flow through HCV-133 will LOWER
b. PCV-135 CLOSES. Letdown flow through HCV-133 will LOWER
c. PCV-135 OPENS, Letdown flow through HCV-133 will RISE
d. PCV-135 CLOSES, Letdown flow through HCV-133 will RISE

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the higher RHR flow rate will cause an increase in RCS pressure during solid plant operations. A lower letdown flow rate is plausible if more RHR flow is directed to the RCS loops instead of the low pressure purification system.
b. Correct. RHR / RCS temperature will initially lower due to greater flow through the RHR Heat Exchanger. In a solid plant condition, this will result in a reduction in RCS pressure. PCV-135 will respond to the reduction in pressure by closing down in automatic. The

combination of this closing of PCV-135 and the rise in flow to the RCS loops will result in reduced letdown flow through the HCV-133.

c. Incorrect. Plausible because the student may believe that the higher RHR flow rate will cause an increase in RCS pressure during solid plant operations. Also plausible because if the PCV-135 opens, letdown flow through the HCV-133 may also rise.
d. Incorrect. Plausible because PCV-135 will respond to the reduction in pressure by closing down in automatic. Student may be confused on the subsequent flow rate through HCV-133, he may believe that the higher flow through the RHR HX will also result in higher letdown flow rate.

Technical References:	I3LP-ILO-RHR001 3-SOP-RHR-001 3-SOP-CVCS-012
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	Modified Bank – Seabrook 2013 NRC Exam – Q30
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 5 & 10

Exam Outline Cross Reference:	Level:	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	0060002123		
		Emergency Core Cooling: Ability to perform specific system and integrated plant procedures during all modes of operation		
Importance:		4.3	4.4	

Question: #31

Which of the following methods of limiting High Head Safety Injection (HHSI) into the RCS to ensure cold RCS over pressurization protection is the **preferred** method in accordance with integrated operating procedure 3-POP-3.3; Plant Cooldown – Hot to Cold Shutdown?

a. Place 31, 32, and 33 Safety Injection Pump Control Switches in PULL OUT and close specified SI MOVs from the control room.
b. RACK OUT Breakers associated with 31, 32, and 33 Safety Injection Pumps and close and remove power from specified SI MOVs.
c. RACK OUT Breakers associated with 31, 32, and 33 Safety Injection Pumps **and** Place 31, 32, and 33 Safety Injection Pump Control Switches in PULL OUT.
d. Close the specified SI MOVs associated with the Boron Injection Tank Flowpath and maintain one Safety Injection Pump available to be manually realigned to the RCS Cold Legs.

Answer: a

Explanation / Justification

a. Correct. Attachment 3, step 1.0, of 3-POP-3.3 states that placing the Control Switches in Pull Out and closing listed SI MOVs is the **preferred** method. The pump breakers are not racked out so that two safety injection pumps shall be maintained available for manual realignment to injection in Mode 4 to support a Hot Shutdown LOCA with Accumulators Isolated event.
b. Incorrect. Plausible because Method C (not the **preferred** method) has the Breakers racked out, but no valves are manipulated.
c. Incorrect. Plausible because this is exactly what is stated as Method C (not the **preferred** method).
d. Incorrect. Plausible because Attachment 3 footnote (1) states, Two safety injection pumps shall be maintained available for manual realignment to injection in Mode 4 to support ONOP-RCS-8; Hot Shutdown LOCA with Accumulators Isolated. Tech Spec LCO 3.4.12, Note 2 also states; “One HHSI pump may be made capable of injecting into the RCS as needed to support emergency boration or to respond to a loss of RHR cooling.” But again, not the **preferred** method.

Technical References:	I3LP-ILO-POP002, 3-POP-3.3 I3LP-ILO-ONPRCS, ONOP-RCS-8
Proposed References to be provided:	None
Learning Objective:	Objectives 4.0 and 2.0 respectively
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006000K506 Knowledge of the operational implications of the following concepts as they apply to ECCS: Relationship between ECCS flow and RCS pressure	
Importance:		3.5	3.9

Question: #32

A small break LOCA is in progress with the following conditions:

- All ECCS Pumps are operating except 31 SI Pump which failed to start.
- RCS Pressure is 1100 psig and stable.
- Pressurizer Level is 5% and stable.
- RCS Average Temperature is slowly lowering about 5°F/hour.
- No Charging Pumps are operating.

For this condition what is the approximate break flow?

a. 100 – 200 gpm
b. 300 – 500 gpm
c. 700 – 900 gpm
d. 1100 – 1300 gpm

Answer: c

Explanation / Justification

a. Incorrect. Plausible because student needs to know that design pressure and flow of the SI Pumps is 400 gpm @ 1100 psig. He then needs to know that only two pumps are operating in parallel, with something close to almost 2x flow depending on system resistance. For the leak to be in this range, then PZR Level would be rising.
b. Incorrect. Plausible because the student may confuse parallel and series pump operations.
c. Correct. Student needs to know that design pressure and flow of the SI Pumps is 400 gpm @ 1100 psig. He then needs to know that only two pumps are operating in parallel, with something close to almost 2x flow depending on system resistance.
d. Incorrect. Plausible because this would be the approximate flow range for all three SI pumps operating.

Technical References:	I3LP-ILO-SIS001
Proposed References to be provided:	None

Learning Objective:	Objective 3.0
Question Source:	Bank Question # 23949
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 8

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007000A102 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank pressure	
Importance:		2.7	2.9

Question: #33

Given the following:

- The plant is operating at 100% power.
- The following indications are reported by the Reactor Operator:
  - The PRESSURIZER RELIEF TANK LIQUID HIGH TEMPERATURE Annunciator has alarmed and PRT temperature is currently 135°F and slowly rising.
  - The PRESSURIZER RELIEF TANK HIGH PRESSURE Annunciator has alarmed and PRT pressure is currently 7 psig and slowly rising.
  - The field operator has reported that the PRT N^2 pressure regulator is set at less than 5 psig.

Responding to the Alarm Response Procedures for both alarms, the Reactor Operator is taking action in accordance with 3-SOP-RCS-007, Pressurizer Relief Tank Operations. Which of the following procedural actions will restore both alarmed conditions to their normal operating parameters and at what pressure are the PRT Rupture Disks designed to rupture to containment?

a. Venting the PRT directly to Containment atmosphere while filling the PRT to 90% will reduce both temperature and pressure. The Rupture Disk Design Pressure is 85 psig.
b. Spraying Primary Water into the PRT will reduce both temperature and pressure. The Rupture Disk Design Pressure is 100 psig.
c. Spraying Primary Water into the PRT will reduce both temperature and pressure. The Rupture Disk Design Pressure is 85 psig.
d. Venting the PRT directly to Containment atmosphere while filling the PRT to 90% will reduce both temperature and pressure. The Rupture Disk Design Pressure is 100 psig.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because venting will lower PRT pressure and adding water volume to the PRT will lower temperature. However, procedural guidance does not exist in 3-SOP-RCS-007 to vent the PRT directly to the Containment atmosphere. Filling and draining of the PRT would also be done within normal operating limits of 67 – 77%, 90% is a value used for purging the PRT. Additionally, Rupture Disk rupture pressure is not 85 psig.

b. Correct. The procedure states that both temperature and pressure control can be achieved by using the FILL and DRAIN PRT instructions under section 4.3, PRT Level Control. This section uses Primary Water Spray to control both temperature and pressure. Also plausible because 100 psig is the correct rupture disk rupture pressure

c. Incorrect. Plausible because the procedure states that both temperature and pressure control can be achieved by using the FILL and DRAIN PRT instructions under section 4.3, PRT Level Control. This section uses Primary Water Spray to control both temperature and pressure. Additionally, Rupture Disk rupture pressure is not 85 psig.

a. Incorrect. Plausible because venting will lower PRT pressure and adding water volume to the PRT will lower temperature. However, procedural guidance does not exist in 3-SOP-RCS-007 to vent the PRT directly to the Containment atmosphere. Filling and draining of the PRT would also be done within normal operating limits of 67 – 77%, 90% is a value used for purging the PRT. Additionally, Rupture Disk rupture pressure is 100 psig.

Technical References:	I3LP-ILO-RCSPZR
Proposed References to be provided:	None
Learning Objective:	Objective E-8
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 5, 7, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008000K401 Knowledge of the CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump	
Importance:		3.1	3.3

Question: #34

Given the following:

- The Unit is at 100% Power.
- CCW Pumps 32 and 33 are running. CCW Pump 31 is in Auto.
- 6.9 KV Bus 6 normal feeder inadvertently opens.

Which ONE of the following describes the AUTOMATIC operation of CCW Pumps for this event?

a. CCW Pump 31 starts when Bus 6A is de-energized and CCW Header Pressure lowers to 105 psig. CCW Pump 33 restarts 34 seconds after Bus 6A is re-energized.
b. CCW Pump 31 starts when Bus 6A is de-energized and CCW Header Pressure lowers to 100 psig. CCW Pump 33 does NOT automatically restart.
c. CCW Pump 31 starts when Bus 6A is de-energized and CCW Header Pressure lowers to 100 psig. CCW Pump 33 restarts 34 seconds after Bus 6A is re-energized.
d. CCW Pump 31 starts when Bus 6A is de-energized and CCW Header Pressure lowers to 105 psig. CCW Pump 33 does NOT automatically restart.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because student may believe that the standby CCW pump starts on low supply header pressure at 105 psig. Also plausible because the second part of the distractor is correct.
b. Incorrect. Plausible because the standby pump does start at 100 psig. Also plausible for student to believe that CCW Pumps do not load on a "Blackout" or single bus loss of voltage. CCW Pumps are not started for a "Blackout and SI" signal, but are started on a "Blackout" or single bus loss of voltage.
c. Correct. The standby CCW Pump starts on low supply header pressure (discharge) at 100 psig. The single bus loss of voltage sequence for Bus 6A starts the 33 CCW Pump at 34 seconds into the loading sequence.

d. Incorrect. Plausible because student may believe that the standby CCW pump starts on low supply header pressure at 105 psig. Also plausible for student to believe that CCW Pumps do not load on a “Blackout” or single bus loss of voltage. CCW Pumps are not started for a “Blackout and SI” signal, but are started on a “Blackout” or single bus loss of voltage.

Technical References:	I3LP-ILO-CCW001 System Description 4.1; Component Cooling
Proposed References to be provided:	None
Learning Objective:	Objective 4
Question Source:	Modified Bank – Indian Point 3 – Question # 24303
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7, 8 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010000K602 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR	
Importance:		3.2	3.5

Question: #35

The plant is operating at 100% power with the following conditions:

- Pressurizer Pressure Control is in AUTO
- Pressurizer Pressure Defeat Switch is in position "Defeat 3&4"
- Pressurizer Pressure Channel PT-455 fails full scale low
- **NO** operator action is taken for the event

Which of the following describes the Pressurizer Pressure Control System response?

a. All PZR Heaters energize, Spray Valves PCV 455A and 455B go full closed, and PORV RC-PCV-456 opens.
b. All PZR Heaters energize, Spray Valves PCV 455A and 455B go full closed, and both PORVs RC-PCV-455C and 456 open.
c. All PZR Heaters energize, Spray Valves PCV 455A and 455B then subsequently open to offset the Pressure rise, both PORVs RC-PCV-455C and 456 remain closed.
d. All PZR Heaters energize, Spray Valves PCV 455A and 455B then subsequently open to offset the Pressure rise, PORV RC-PCV-456 opens.

Answer: a

Explanation / Justification

a. Correct. PZR Control configuration is such that Channel I (PT-455) is the controlling channel and Channel II (PT-456) is the alarm channel. All heaters energize due to the failed low controlling channel, causing actual pressure to rise. Both spray valves close, and PORV RC-PCV-455C is inhibited from auto opening. If no operator action is taken, actual pressurizer pressure rises slowly until PZR PORV RC-PCV-456 opens (Ch II & Ch III).
b. Incorrect. Plausible because student may believe that only one channel is required to allow the PORVs to open. There may also be some misunderstanding of the configuration of the PZR PCS with the defeat switch in the "Defeat 3&4" position.
c. Incorrect. Plausible because the student may believe that the PZR Spray valves will respond to the pressure rise forgetting that the controller is selected to the failed channel.

d. Incorrect. Plausible because may believe that the PZR Spray valves will respond to the pressure rise forgetting that the controller is selected to the failed channel. Plausible because PORV RC-PCV-456 does open.

Technical References:	I3LP-ILO-ICPZRC I3LP-ILO-AOPINT System Description 1.4; Pressurizer & Pressurizer Relief Tank
Proposed References to be provided:	None
Learning Objective:	Objective 5.0 & 3.0 respectively
Question Source:	New
Question Cognitive Level:	Comprehension / Analysis
10CFR Part 55 Content:	55.41 (b) 5, 7, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012000K502	
		Knowledge of the operational implications of the following concepts as they apply to the RPS: Power density	
Importance:		3.1	3.3

Question: #36

Which of the following Reactor Protection System Trips ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded?

a. Overtemperature ΔT (OTΔT)
b. Power Range Neutron Flux - High
c. Steam Generator Water Level – Low Low
d. Overpower ΔT (OPΔT)

Answer: d

Explanation / Justification

a. Incorrect. Plausible because OTΔT ensures the design limit DNBR is met and the inputs to the setpoint include coolant temperature and axial power distribution.
b. Incorrect. Plausible because this trip ensures protection against a positive reactivity excursion leading to DNB.
c. Incorrect. Plausible because this trip protects against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes.
d. Correct. The OPΔT trip ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. See Tech Spec 3.3.1 bases.

Technical References:	I3LP-ILO-ICRXP
Proposed References to be provided:	None
Learning Objective:	Objective E-1
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	0120002145 Ability to identify and interpret diverse indications to validate the response of another indication	
Importance:		4.3	4.3

Question: #37

Which of the following malfunctions will cause the Overtemperature ΔT Channel Trip or Rod Stop alarm to annunciate?

a. PR Channel N41 failure high
b. Loop 32 T_{hot} microprocessor fails low
c. PT-456 fails high
d. Loop 31 T_{cold} RTD fails low

Answer: d

Explanation / Justification

a. Incorrect. Plausible because NIS does provide an input to OTΔT; however input is from delta flux. If the upper detector for N41 failed high, the alarm would annunciate.
b. Incorrect. Plausible because the candidate must determine how the failure will affect actual ΔT and how the calculated setpoint will be affected.
c. Incorrect. Plausible because candidate must recall that Channel 2 (PT-456) cannot be the controlling channel. If a controlling channel failed high, actual pressure would decrease and the alarm would annunciate when the first OTΔT channel setpoint was reached.
d. Correct. The failure causes the indicated ΔT to fail high. The alarm will annunciate when the calculated setpoint is less than the indicated ΔT. Setpoint should also decrease based on a higher $T_{avg.}$

Technical References:	I3LP-ILO-ICRXP System Description 28.0
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	Bank (IP3 – Question # 25208)
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013000K301 Knowledge of the effect that a loss or malfunctions of the ESFAS will have on the following: Fuel	
Importance:		4.4	4.7

Question: #38

Which of the following choices contains ONLY items that are described in 10 CFR 50.46 as Acceptance Criteria for Emergency Core Cooling Systems for light-water nuclear power reactors?

I. The calculated total amount of hydrogen generated shall not exceed .17 times the hypothetical amount that would be generated if all the fuel cladding were to react.
II. The calculated maximum fuel element cladding temperature shall not exceed 2000°F.
III. The calculated total oxidation of the cladding shall not exceed 0.17 times the total cladding thickness before oxidation.
IV. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
V. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
VI. The calculated total oxidation of the cladding shall not exceed 0.10 times the total cladding thickness before oxidation.
VII. The zirc-water reaction postulated to occur at 1800°F will not become self-sustaining under any circumstance.
VIII. The calculated total amount of hydrogen generated shall not exceed 0.01 times the hypothetical amount that would be generated if all the fuel cladding were to react.

a. I, II, VI, VII
b. II, III, IV, VIII
c. III, IV, V, VIII
d. I, V, VI, VII

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the answer includes hydrogen production, cladding reduction, and cladding temperature numbers. The numbers are incorrect for their individual acceptance criteria, but include some actual numbers from 50.46.
b. Incorrect. Plausible because III, IV, and VIII are correct. Student may believe the peak cladding temperature to be 2000°F.
c. Correct. See 10 CFR 50.46. Cladding – 17%, Hydrogen – 1%, Coolable Geometry, 2200°F.

d. Incorrect. Plausible because the answer includes hydrogen production, cladding reduction, and cladding temperature numbers. The numbers are incorrect for their individual acceptance criteria, but include some actual numbers from 50.46.

Technical References:	I3LP-ILO-ESS001
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	Modified Bank – Salem 2004 NRC Exam
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022000A405 Containment Cooling System - Ability to manually operate and/or monitor in the control room: Containment readings of temperature, pressure, and humidity system	
Importance:		3.8	3.8

Question: #39

Given the following plant conditions:

- Unit is at 100% Power
- RWST Temperature = 90°F
- Containment Temperature = 120°F

In accordance with Technical Specifications, Containment Pressure shall be maintained below what upper operating limit and why?

a. Containment pressure shall be maintained ≤ +1.5 psig to ensure that in the event of a DBA, the resultant containment accident pressure will remain below the containment design pressure.
b. Containment pressure shall be maintained ≤ +2.5 psig to maximize the effectiveness of the ECCS during the core reflood phase of a LOCA and ensure motor heating concerns are addressed.
c. Containment pressure shall be maintained ≤ +2.5 psig to ensure that in the event of a DBA, the resultant containment accident pressure will remain below the containment design pressure.
d. Containment pressure shall be maintained ≤ +1.5 psig to maximize the effectiveness of the ECCS during the core reflood phase of a LOCA and ensure motor heating concerns are addressed.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because if RWST Temperature was greater than 95°F and Containment Temperature was greater than 125°F, then this would be the correct answer.
b. Incorrect. Plausible because pressure shall be maintained ≤ +2.5 psig in accordance with TS LCO 3.6.4, but the bases is incorrect. Higher Containment Pressures (not limiting them) will cause higher containment backpressure and therefore increase the effectiveness of the ECCS during the reflood phase. Also motor heating concerns are associated with ensuring pressure is greater than the lower limit of -2.0 psig. ECCS effectiveness and motor heating

are discussed in the bases, but not with respect to limiting containment pressure to less than or equal to the upper operating limit.

c. Correct. If RWST temperature is ≤ 95°F and Containment temperature is ≤ 125°F, Containment pressure shall be ≥ -2.0 psig and ≤ +2.5 psig. See TS 3.6.4 LCO and bases.
d. Incorrect. Plausible because if RWST Temperature was greater than 95°F and Containment Temperature was greater than 125°F, the first part of this distractor is correct. Also plausible because ECCS effectiveness and motor heating are discussed in the bases, but not with respect to limiting containment pressure to less than or equal to the upper operating limit.

Technical References:	Technical Specification LCO 3.6.4 & Bases
	I3LP-ILO-VCVCB
Proposed References to be provided:	None
Learning Objective:	Objectives 7.0 & 9.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	000022K201 Knowledge of power supplies to the following: Containment cooling fans	
Importance:		3.0	3.1

Question: #40

Fan cooler Unit 31 is tagged out for maintenance. During a Safety Injection with a Loss of Off-Site Power, Emergency Diesel Generator 32 fails to start. Which of the following indicates the containment cooling equipment available?

a. 32, 33, and 34 Fan Cooler Units
   31 Containment Spray Pump
   Minimum safeguards equipment for containment cooling is satisfied.
b. 33 and 35 Fan Cooler Units
   31 Containment Spray Pump
   Minimum safeguards equipment for containment cooling is NOT satisfied.
c. 33 and 35 Fan Cooler Units
   31 and 32 Containment Spray Pumps
   Minimum safeguards equipment for containment cooling is satisfied.
d. 32, 33, and 34 Fan Cooler Units
   32 Containment Spray Pump
   Minimum safeguards equipment for containment cooling is NOT satisfied.

Answer: a

Explanation / Justification

a. Correct. #31FCU is tagged in the stem. 32 EDG feeds bus 6A which supplies 32 Containment Spray Pump and #35 FCU. All other equipment related to containment cooling remains available. Minimum design requirements are also met, 1 Containment Spray Pumps and 3 CFUs.
b. Incorrect. Plausible because the student may believe that #32 and #34 FCU are supplied by 32 EDG and the combination of one spray pump and only two FCUs does not satisfy minimum design requirements.
c. Incorrect. Plausible because the student may believe that 32 EDG does not supply a containment spray pump and two containment spray pumps does satisfy minimum design requirements.
d. Incorrect. Plausible because the available FCUs are correct, but the containment spray pump available is incorrect. The student may also believe more than three FCUs are required for minimum design requirements to be met.

Technical References:	I3LP-ILO-VCCARC I3LP-ILO-CS001
Proposed References to be provided:	None
Learning Objective:	Objectives 3.0, 4.0 and E-4
Question Source:	Modified Bank Question # 23864
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	000026K202 Knowledge of bus power supplies to the following: MOVs	
Importance:		2.7	2.9

Question: #41

The unit has experienced a large break LOCA and a Loss of Off-Site Power. Emergency Diesel Generator 33 has tripped. Containment pressure has reached the HI-HI setpoint and a Containment Spray actuation signal has occurred. What is the present status of the Containment Spray System?

a. Both Containment Spray Pumps are running and discharging to the spray header via their respective open pump discharge MOVs (866A and 866B).
b. Both Containment Spray Pumps are running, however only 31 Containment Spray Pump is discharging through its open pump discharge MOV (866A).
c. Only 31 Containment Spray Pump is running and discharging through its open pump discharge MOV (866A).
d. Only 32 Containment Spray Pump is running and discharging through its open pump discharge MOV (866B).

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the 33 EDG does not supply either of the Containment Spray Pumps and their associated MOVs.
b. Incorrect. Plausible because the student may believe that the 33 EDG does not supply either of the Containment Spray Pumps, however they may think that the 33 EDG feeds one of the MOVs through MCC-36A or MCC-36B.
c. Incorrect. Plausible because the student may believe that 32 Containment Spray Pump and its respective discharge MOV are supplied by the 33 EDG.
d. Correct. 31 Containment Spray Pumps and its respective discharge MOV (MCC-36A) are powered from the 5A Bus and the 33 EDG.

Technical References:	I3LP-ILO-CS001
Proposed References to be provided:	None
Learning Objective:	Objective E-4
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	000026A405 Ability to manually operate and/or monitor in the control room: Containment spray reset switches	
Importance:		3.5	3.5

Question: #42

A large break LOCA has occurred and containment pressure has risen to the automatic containment spray setpoint. Both Containment Spray Pumps start and 4 Fan Cooler Units start. The fifth FCU cannot be started. If containment pressure remains above 20 psig; which of the following describes when the Containment Spray Pumps are stopped?

a. Both Containment Spray Pumps are stopped when Recirc Switch 1 is taken to ON.
b. 31 Containment Spray Pump is stopped immediately after the RWST reaches 11.5 feet and 32 Containment Spray Pump is stopped when the RWST reaches 1.5 feet.
c. 32 Containment Spray Pump is stopped when Recirc Switch 1 is taken to ON and 31 Containment Spray Pump is stopped after the RWST reaches 1.5 feet.
d. 31 Containment Spray Pump is stopped when Recirc Switch 1 is taken to ON and 32 Containment Spray Pump is stopped after the RWST reaches 1.5 feet.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may believe that recirculation switch 1 stops both Containment Spray Pumps. Also Attachment 6 of ES-1.3 uses containment pressure of 16 psig or 5 FCU running to stop both Containment Spray Pumps after 4 hours. Stem includes information on containment pressure and number of FCU running.
b. Incorrect. Plausible because the entry condition for ES-1.3, Transfer to Cold Leg Recirculation is RWST level < 11.5 feet. RWST level of 1.5 feet is also the level at which operators stop the second containment spray pump when transferring to recirculation mode.
c. Correct. See ES-1.3, Transfer to Cold Leg Recirculation step 4.0 and Attachment 6, steps 1.0 & 2.0. Also see System Description 10.2, pages 4-6.
d. Incorrect. Plausible because the student may believe the order of stopping pumps is 31 first, followed by 32.

Technical References:	I3LP-ILO-EOPE10, 3-ES-1.3 System Description 10.2
Proposed References to be provided:	None
Learning Objective:	Objective 22.0

Question Source:	Bank IP3 – Question # 2435
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039000K406 Knowledge of MRSS (Main and Reheat Steam System) design feature(s) and/or interlock(s) which provide the following: Prevent reverse steam flow on a steam line break	
Importance:		3.3	3.6

Question: #43

The Main Steam Non-Return Checks Valves are:

a. Located upstream of the MSIVs and provide steam line isolation in the event of a downstream rupture. They are designed to limit the blowdown rate of the steam generators in the event of a steam line break.
b. Located downstream of the MSIVs and provide steam line isolation in the event of an upstream rupture. They are designed to preclude the blowdown of more than one steam generator inside containment.
c. Located upstream of the MSIVs and provide steam line isolation in the event of an upstream rupture. They are designed to preclude the blowdown of more than one steam generator inside containment.
d. Located downstream of the MSIVs and provide isolation in the event of a downstream rupture. They are designed to limit the blowdown rate of the steam generators in the event of a steam line break.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that they are installed upstream of the MSIVs and may believe they function similar to the MSIVs in that they open into the steam flow and steam flow assists in their closure. They may also believe they function like an “excess flow” check valve. The flow restrictor is designed to limit the blowdown rate in the event of a downstream rupture. The valve will limit the total blowdown, but not the rate.
b. Correct. They are located downstream of the MSIVs and provide steam line isolation in the event of an upstream rupture. They are designed to preclude the blowdown of more than one steam generator inside containment.
c. Incorrect. Plausible because everything in the statement is correct except the physical location of the check valve respective to the MSIVs.
d. Incorrect. Plausible because the location respective to the MSIVs is correct. Also plausible because the student may believe they function similar to the MSIVs in that they open into the steam flow and steam flow assists in their closure.

Technical References:	I3LP-ILO-MSS001 FSAR Section 14.2.5 System Description 18.0
Proposed References to be provided:	None
Learning Objective:	Objective 2.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059000A206 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW (Main Feedwater System); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of steam flow to MFW system	
Importance:		2.7	2.9

Question: #44

Given the following:

- The Plant is at 100% Power
- All Control Systems are in Automatic
- All Steam Flow and Feed Flow Channel Transfer Switches are selected to Channel A
- 31 SG Steam Flow Compensating Pressure Transmitter PT-419A fails to zero.

First predict the plant response to the failed pressure transmitter with **NO** operator action, then secondly describe what procedural operator actions would correct or mitigate the consequences of the failed pressure transmitter.

a. 31 SG Steam Flow Channel I fails low causing a steam flow / feed flow mismatch signal which then closes the Feed Regulating Valve reducing feed flow. Actual Steam Generator Level lowers causing a level program error, but without operator action, SG level lowers until the reactor trips due to low SG level. Mitigating actions would be taken in accordance with 3-AOP-INST-1. Operator may take manual control of the FRV and select the SG Transfer Switch to Steam Flow Channel II.

b. 31 SG Steam Flow Channel I fails low causing a steam flow / feed flow mismatch signal which then closes the Feed Regulating Valve reducing feed flow. Actual Steam Generator Level lowers causing a level program error with a 3 times gain which then overcomes the steam flow / feed flow mismatch signal. Because of the level program error gain, the Feed Regulating Valve opens restoring Steam Generator level to approximately 5% below program. Mitigating actions would be taken in accordance with 3-AOP-FW-1. Operator may take manual control of the FRV and select the SG Transfer Switch to Steam Flow Channel II.

c. 31 SG Steam Flow Channel I fails high causing a steam flow / feed flow mismatch signal which then opens the Feed Regulating Valve raising feed flow. Actual Steam Generator

Level rises causing a level program error, but without operator action, SG level rises until a turbine trip and feedwater isolation occurs on high SG level. With reactor power > 10%, the turbine trip causes a reactor trip. Mitigating actions would be taken in accordance with 3-AOP-INST-1. Operator may take manual control of the FRV and select the SG Transfer Switch to Steam Flow Channel II.

d. 31 SG Steam Flow Channel I fails high causing a steam flow / feed flow mismatch signal which then opens the Feed Regulating Valve raising feed flow. Actual Steam Generator Level rises causing a level program error with a 3 times gain which then overcomes the steam flow / feed flow mismatch signal. Because of the level program error gain, the Feed Regulating Valve closes restoring Steam Generator level to approximately 5% below program. Mitigating actions would be taken in accordance with 3-AOP-FW-1. Operator may take manual control of the FRV and select the SG Transfer Switch to Steam Flow Channel II.

Answer: a

Explanation / Justification

a. Correct. See SGWLC Lesson Plan and System Description. 3-AOP-INST-1 provides mitigating actions.
b. Incorrect. Plausible because the steam flow channel does fail low and the level channel error does have a 3X gain. However the negative signal from the failed low steam flow will actually still cause the overall signal to remain negative. Mitigating actions are correct, but wrong procedure.
c. Incorrect. Plausible because the student may believe if the compensating pressure channel fails to zero, the steam flow indication will fail high. Mitigating actions and the procedure are correct.
d. Incorrect. Plausible because the student may believe if the compensating pressure channel fails to zero, the steam flow indication will fail high. Mitigating actions are correct, but the procedure is incorrect.

Technical References:	I3LP-ILO-ICSGL
	System Description 21.1
	3-AOP-INST-1
Proposed References to be provided:	None
Learning Objective:	Objective 2.0 & 5.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061000K301 Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS	
Importance:		4.4	4.6

Question: #45

Given the following plant conditions:

- The Unit is at 100% Power
- All Control Systems are in Automatic
- All Systems are in their normal 100% Power configuration
- Control Rods are at their fully withdrawn position

Which of the following describes the effects of a spurious start of the 32 ABFP and why?

a. T_{ave} will lower, Reactor Power will rise due to the steam demand required to start the 32 ABFP.
b. T_{ave} will lower, Reactor Power will rise due to the injection of cold auxiliary feedwater from the 32 ABFP.
c. T_{ave} will lower, Reactor Power will rise due to both the steam demand required to start the 32 ABFP and the injection of cold auxiliary feedwater from the 32 ABFP.
d. T_{ave} and Reactor Power will both be restored to normal by the automatic rod control system.

Answer: a

Explanation / Justification

a. Correct. T_{ave} will lower, Reactor Power will rise due to the steam demand required to start the 32 ABFP. The normal operations lineup has the discharge feed regulating valves for the 32 ABFP aligned closed to prevent their injecting water.
b. Incorrect. Plausible because the student may believe that the discharge feed regulating valves for the turbine driven auxiliary feed pump are normally open as is the case for the motor driven pumps.
c. Incorrect. Plausible because the student may believe that the discharge feed regulating valves for the turbine driven auxiliary feed pump are normally open as is the case for the motor driven pumps.

d. Incorrect. Plausible because the student may believe that the steam demand for the 32 ABFP will effect turbine impulse pressure, lowering $T_{ref,}$ causing control rods to insert to restore temperature and power.

Technical References:	I3LP-ILO-AFW001
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 5, 6, & 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062000A304 Ability to monitor automatic operation of the ac distribution system, including: Operation of inverter (e.g., precharging synchronizing light, static transfer)	
Importance:		2.7	2.9

Question: #46

Failure of the 31 Static Inverter has caused the Automatic Static Transfer Switch to transfer to the alternate source of power. Electrical Maintenance has investigated and has requested that 31 Static Inverter be taken out of service. In accordance with 3-SOP-EL-002, Instrument Bus and Plant Computer Static Inverter Operation, how will Operations take the static inverter out of service?

a. Operations will verify that the static transfer switch is in the "Forward Transfer" position and then place the Maintenance Bypass Switch (MBS) in the BYPASS position.
b. Operations will verify that the static transfer switch is in the "Reverse Transfer" position and then place the Maintenance Bypass Switch (MBS) in the BYPASS position.
c. Operations will verify that the static transfer switch is in the "Reverse Transfer" position and the Maintenance Bypass Switch (MBS) is in the INVERTER position.
d. Operations will verify that the static transfer switch is in the "Forward Transfer" position and the Maintenance Bypass Switch (MBS) is in the INVERTER position.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the "Forward Transfer" position is correct. When the static transfer switch transfer to the alternate source of power, it is considered to have "reverse transferred". The MBS in the BYPASS position is correct.
b. Correct. The static transfer switch automatically transfers to reverse transfer on inverter failure. Then per the procedure the Maintenance Switch is placed in BYPASS. See procedure section 4.3 and also Caution before step 4.1.16. See lesson plan section II.D.
c. Incorrect. Plausible because the static transfer switch does automatically transfers to reverse transfer on inverter failure. Student may believe that the Maintenance Bypass switch is placed in the INVERTER position to take it out of service.
d. Incorrect. Plausible because the student may believe that the "Forward Transfer" position is correct. When the static transfer switch transfer to the alternate source of power, it is considered to have "reverse transferred". Student may believe that the Maintenance Bypass switch is placed in the INVERTER position to take it out of service. The MBS in the BYPASS position is correct.

Technical References:	I3LP-ILO-EDS118, 3-SOP-EL-002 System Description 27.5
Proposed References to be provided:	None
Learning Objective:	Objectives E-3 and E-9
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063000A201 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	
Importance:		2.5	3.2

Question: #47

Given the following:

- Unit 3 is operating at 100% power
- A ground on 34 Battery Charger causes a loss of power to 34 DC Power Panel.
- 34 Battery Charger is damaged by the ground
- 34 DC Power Panel and 34 Battery are NOT damaged

Which of the following is correct regarding the impact of this event and what actions, if any, can be taken?

a. 34 Instrument Bus will lose power. 3-AOP-IB-1; Loss of Power to an Instrument Bus, will address this by powering the bus from MCC-36C.
b. 34 Instrument Bus will lose power. 3-AOP-IB-1; Loss of Power to an Instrument Bus, will address this by powering the bus from MCC-36B.
c. 34 Instrument Bus will remain energized. 3-AOP-DC-1; Loss of a 125V DC Panel, will provide direction to re-energize 34 DC Panel using 35 Battery Charger.
d. 34 Instrument Bus will remain energized. 3-AOP-DC-1; Loss of a 125V DC Panel, will NOT allow re-energizing 34 DC Panel using 35 Battery Charger because the plant is above Mode 5.

Answer: c

Explanation / Justification

a. Incorrect because 34 Instrument Bus will remain energized. Plausible because 34 Instrument Bus used to not automatically switch to a backup power source. Additionally, there are two back up feeds; one from MCC-36B and the other from MCC-36C.

b. Incorrect because 34 Instrument Bus will remain energized. Plausible because 34 Instrument Bus used to not automatically switch to a backup power source. Additionally, there are two back up feeds; one from MCC-36B and the other from MCC-36C.
c. Correct. 34 Instrument Bus will remain energized. 3-AOP-DC-1 will utilize 35 Battery Charger. Instructions and P&Ls in 3-SOP-EL-003 are applicable.
d. Incorrect. Plausible because there are restrictions on using 35 Battery Charger above 200°F due to the potential for cross tying buses. 3-AOP-DC-1 will utilize 35 Battery Charger. Instructions and P&Ls in 3-SOP-EL-003 are applicable.

Technical References:	I3LP-ILO-AOPDC1, 3-AOP-DC-1 3-SOP-EL-003
Proposed References to be provided:	None
Learning Objective:	Objectives 1.0 & 2.0
Question Source:	Bank – Indian Point 3 – Question # 25118
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064000K405	
		Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Incomplete-start relay	
Importance:		2.8	3.2

Question: #48

An operator is manually starting the 31 Emergency Diesel Generator (EDG) for a Technical Specification Surveillance run. The operator depresses the start pushbutton which completes the start circuit and applies 125 VDC to the engine start relay coil. During this manual start, the diesel trips and the operator receives an Overcrank Alarm. The OVERCRANK alarm is actuated if the engine fails to attain speed within ____ seconds of receiving the start signal as sensed by ______.

a. 10, lube oil pressure at 30 psig
b. 15, jacket water pressure at 8 psig
c. 10, jacket water pressure at 8 psig
d. 15, lube oil pressure at 30 psig

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may remember the design start requirement of 10 seconds and also remember interlocks associated with lube oil pressure being 30 psig. Actually the Low Oil Pressure trip is 30 psig with a time delay of 20 seconds to allow for sufficient time for the oil pressure to increase to 30 psig. The 20 second timer is energized when jacket water pressure is > 8 psig as well.
b. Correct. The OVERCRACK alarm is actuated if the engine fails to attain speed within 15 seconds of receiving the start signal as sensed by jacket water pressure (set at 8 psig). See System Description 27.3, section 2.6.3.
c. Incorrect. Plausible because the student may remember the design start requirement of 10 seconds and the second part of the distractor is correct.
d. Incorrect. Plausible because the first part is correct. Also plausible because the student may remember interlocks associated with lube oil pressure and engine start. Actually the Low Oil Pressure trip is 30 psig with a time delay of 20 seconds to allow for sufficient time for the oil pressure to increase to 30 psig. The 20 second timer is energized when jacket water pressure is > 8 psig as well.

Technical References: I3LP-ILO-EDSEDG
System Description 27.3; Emergency Diesels

Proposed References to be provided:	None
Learning Objective:	Objectives 4 & 6
Question Source:	Modified – Indian Point 3 – Question # 23776
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	0640002132 Ability to explain and apply system limits and precautions	
Importance:		3.8	4.0

Question: #49

Given the following conditions:

- The Unit has experienced a Safety Injection and a Loss of Offsite Power
- The 32 EDG is carrying 1740 KW
- It is desired to start the 33 CCW pump
- The 33 CCW pump is a 220 KW load

Which of the following is the MINIMUM amount of load that must be reduced before starting the 33 CCW pump that would ensure the final diesel loading is at or below the CONTINUOUS run load limit for 32 EDG?

a. No load reduction required
b. 10 KW
c. 110 KW
d. 210 KW

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the continuous load limit is 2000 KW. 2000 KW is the ½ hour limit.
b. Incorrect. Plausible because the student may believe that the continuous load limit is 1950 KW. 1950 KW is the 2 hour limit.
c. Incorrect. Plausible because the student may believe that the continuous load limit is 1850 KW. 1850 is 100 KW below the 2 hour limit.
d. Correct. 1740 KW plus 220 KW is 1960 KW, which is 210 KW above the continuous load limit of 1750 KW. See 3-SOP-EL-015, P&L 2.3.

Technical References:	I3LP-ILO-EDSEDG 3-SOP-EL-001, Attachment 4 3-SOP-EL-015, P&L 2.3 and Attachment 3
Proposed References to be provided:	None
Learning Objective:	Objective 5

Question Source:	Bank – IP3 – Question # 1578 (slightly modified)
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7, 8, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073000K301 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radiative effluent releases	
Importance:		3.6	4.2

Question: #50

A loss of power has occurred to the following process radiation monitors, R-12; Containment Radiogas Monitor, R-14; Plant Vent Radiogas Monitor, and R-27; Wide Range Plant Vent Gas Monitor. Which of the following are common system responses to a loss of power to these three radiation monitors.

I. CHANNEL FAILURE Alarm
II. Containment Evacuation Alarm
III. Containment Purge Supply and Exhaust Valves will Close
IV. PAB Exhaust will be Diverted Through Charcoal Filters
V. Flow Control Valve (RCV-014) in the Waste Gas Discharge Line will Close
VI. Containment Pressure Relief Valves will Close

a. I, II, VI
b. II, III, IV
c. III, IV, V
d. I, III, VI

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the Plant Vent radiation monitors also cause the initiation of the containment evacuation alarm since they are a backup to the Containment monitor and provide containment isolation. I and IV are correct.
b. Incorrect. Plausible because the student may believe that the Plant Vent radiation monitors also cause the initiation of the containment evacuation alarm since they are a backup to the Containment monitor and provide containment isolation. III and IV are correct.
c. Incorrect. Plausible because III and IV are correct and two of the three monitors cause V to be correct.
d. Correct. The common system responses are a "CHANNEL FAILURE" ALARM due to the loss of power and all three cause containment ventilation isolation.

Technical References: I3LP-ILO-RMSPRM

Proposed References to be provided: None

Learning Objective:	Objective F
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7, 10, & 11

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078000A401 Ability to manually operate and/or monitor in the control room: SWS pumps	
Importance:		2.9	2.9

Question: #51

Given the following:

- The plant is at 100% power
- 31, 32, and 35 Service Water Pumps are running
- 33, 34, and 36 Service Water Pumps are in AUTO
- The Service Water Pump Mode Selector Switch is in the 1-2-3 position.
- The 31 Emergency Diesel Generator is tagged for maintenance.

Assuming no operator action, which of the following describes the status of the SW Pumps after a Reactor Trip coincident with a Loss of Off-Site power occurs?

a. 31, 33, 34, and 36 Service Water Pumps will be running
b. 31 and 33 Service Water Pumps will be running
c. Only 33 Service Water Pump will be running
d. 34, 35, and 36 Service Water Pumps will be running

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may assume that all service water pumps will start except those associated with 31 EDG, that being 32 and 35 SW Pumps.
b. Correct. Only 31 and 33 SW Pumps start because 32 is powered from 31 EDG which is tagged for maintenance. Only essential service water header pumps start (1-2-3).
c. Incorrect. Plausible because the student may believe that 31 EDG powers both 31 and 32 Service Water Pumps.
d. Incorrect. Plausible because the student may believe that the non-essential pumps or the pumps not selected will start on a non-SI blackout. Student may also believe that 31 EDG powers both 31 and 32 Service Water Pumps.

Technical References:	I3LP-ILO-SW001, System Description 24.0
Proposed References to be provided:	None
Learning Objective:	Objectives 2.0, 3.0, and 6.0
Question Source:	Modified Bank Question # 25927

Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078000K107 Knowledge of the physical connections and/or cause-effect relationships between SWS and the following systems: Secondary closed cooling water	
Importance:		2.5	2.3

Question: #52

During normal operating conditions, which of the following loads is cooled directly from the Conventional Plant Non-essential Service Water Header?

I. Main Boiler Feed Pump L.O. Coolers
II. Turbine Hall Closed Cooling
III. Instrument Air Closed Cooling
IV. Exciter Air Coolers
V. Hydrogen Coolers
VI. Heater Drain Pumps
VII. Component Cooling Heat Exchangers

a. I, IV, VI
b. II, III, VII
c. II, IV, V
d. I, V, VI

Answer: c

Explanation / Justification

a. Incorrect. Plausible because Main Boiler Feed Pumps and Heater Drain Pumps are secondary non-essential components. The Main Boiler Feed Pump pedestals and the Heater Drain Pumps are also both cooled by Turbine Hall Closed Cooling, but not service water directly. Exciter Air Coolers are a correct answer.
b. Incorrect. Plausible because Turbine Hall Closed Cooling is correct, the #33 Instrument Air Compressor is cooled by Turbine Hall Closed Cooling, and the Component Cooling Heat Exchangers are "Non-essential" loads but not on the conventional header.
c. Correct. Turbine Hall Closed Cooling, Exciter Air Coolers, and Hydrogen Coolers are all Conventional Non-essential Service Water Header loads.
d. Incorrect. Plausible because Main Boiler Feed Pumps and Heater Drain Pumps are secondary non-essential components. The Main Boiler Feed Pump pedestals and the Heater Drain Pumps are also both cooled by Turbine Hall Closed Cooling, but not service water directly. Hydrogen Coolers are a correct answer.

Technical References:	I3LP-ILO-SW001, System Description 24.0
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	Modified Bank Question #14315
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 4

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078000K103 Knowledge of the physical connections and/or cause-effect relationships between IAS and the following systems: Containment Air	
Importance:		3.3	3.4

Question: #53

With the Unit at 100%, a spurious Safety Injection has been initiated due to human error during maintenance functional testing. The operators have transitioned to 3-ES-1.1; SI Termination. In accordance with 3-ES-1.1; SI Termination, what actions are required from the control room to restore Instrument Air to Containment

a. Reset SI, Reset Phase A, Press Instrument Air Reset Pushbutton, Open IA-PCV-1228
b. Reset SI Only, valve IA-PCV-1228 will return to the pre-SI position Automatically
c. Reset SI, Reset Phase A, Reset Phase B, Open IA-PCV-1228
d. Reset Phase A Only, valve IA-PCV-1228 will return to the pre-SI position Automatically

Answer: a

Explanation / Justification

a. Correct. See 3-ES-1.1; SI Termination procedure steps 1 through 3. PCV-1228, supply to VC closes directly on a Phase A signal.
b. Incorrect. Plausible because the student may believe that because it was a spurious SI signal, only SI is required to be reset. Also plausible because original plant design provided for valves to automatically reset to their previous position upon reset of the containment Phase A isolation signal (System Description 10.7, page 4).
c. Incorrect. Plausible because the procedure resets Phase B if actuated, student may believe Phase B needs to be reset also. Student may also believe that PCV-1228 is a Phase B valve.
d. Incorrect. Plausible because PCV-1228, supply to VC closes directly on a Phase A signal. Also, plausible because original plant design provided for valves to automatically reset to their previous position upon reset of the containment Phase A isolation signal.

Technical References:	I3LP-ILO-EOPE10, 3-ES-1.1 System Description 10.7; Containment Isolation I3LP-ILO-IA001 System Description 29.2; Instrument Air
Proposed References to be provided:	None
Learning Objective:	Objectives 20.0 and 8.0 respectively

Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103000A205 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry	
Importance:		2.9	3.9

Question: #54

The unit is in Mode 1, 8% Reactor Power, preparing to synchronize the Turbine Generator in accordance with 3-POP-1.3; Plant Startup from Zero to 45% Power.

The control room has determined that there is a small RCS leak inside the containment. The Shift Manager has determined that an expedited emergency containment entry should be made to determine the leak location.

After discussion with Health Physics, Maintenance, and the Operations Manager, it is determined that three work parties will be needed to investigate the RCS leak.

In accordance with OAP-007; "Containment Entry and Egress", which of the following is the minimum number of people in each work party and the maximum number of personnel permitted in the containment under the established conditions.

a. 2, 12
b. 3, 12
c. 2, 24
d. 3, 24

Answer: a

Explanation / Justification

a. Correct. P&L 2.13 and the note prior to step 4.4.13; "If RX is critical in Modes 1 or 2, then the number of people in VC at any one time SHALL NOT exceed 12. P&L 2.22 also states; "When Containment Integrity is established in Modes 1, 2, 3, or 4, Then Entry Parties SHALL consist of at least 2 people, and individuals SHALL NOT separate into parties of less than two people."

b. Incorrect. Plausible because as per P&L 2.13 and the note prior to step 4.4.13; "If RX is critical in Modes 1 or 2, then the number of people in VC at any one time SHALL NOT exceed 12."
c. Incorrect. Plausible because the first part is correct per P&L 2.22 and because the student may remember that a 24 person guideline is the limit for VC entries when in Modes 3 or 4. They may assume the 24 person limit is for Modes 1-4 and not just Modes 3 & 4.
d. Incorrect. Plausible because the student may remember that a 24 person guideline is the limit for VC entries when in Modes 3 or 4. They may assume the 24 person limit is for Modes 1-4 and not just Modes 3 & 4.

Technical References:	I3LP-ILO-VCVCB, OAP-007
Proposed References to be provided:	None
Learning Objective:	Objective 8.0
Question Source:	Modified Bank Question #3755
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 9

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103000A301 Ability to monitor automatic operation of the containment system, including: Containment Isolation	
Importance:		3.9	4.2

Question: #55

Given the following conditions:

- The Unit was at 100% when a human error by an I&C Technician caused a Safety Injection Signal to be generated.
- The crew is performing 3-E-0, Reactor Trip or Safety Injection and has not yet transitioned to 3-ES-1.1, SI Termination.
- The BOP Operator is performing the actions of 3-RO-1, BOP Operator Action During Use of EOPs.

The following valves are identified as being in the OPEN position:

a. AOV-204B, Normal Charging Isolation
b. TCV-1104, Service Water Containment Cooling Full Flow Isolation
c. AC-AOV-791, Excess Letdown Heat Exchanger CCW Isolation
d. IV-AOV-1410, Isolation Valve Seal Water System Flow Control Valve

For the current plant conditions, which ONE of the following valves must be repositioned to place it in the required accident configuration?

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may believe that charging flow needs to be isolated for the spurious Safety Injection Signal, but 3-E-0 step 11 has the reactor operator establish charging flow and verify AOV-204B is open.
b. Incorrect. Plausible because the student may believe that the Containment Fan Cooling Units are not needed for the spurious safety injection signal. The TCV-1004 opens automatically from a safety injection signal.
c. Correct. Excess Letdown Heat Exchanger is considered a nonessential process and is isolated on a Phase A signal caused by the Safety Injection Signal. This valve should be closed. Student may think that because it is a CCW valve, it is only isolated by a Phase B Signal.

d. Incorrect. Plausible because the student may be confused on how the Isolation Valve Seal Water System operates and expect any valves associated with containment isolation to be closed.

Technical References:	I3LP-ILO-EOPE00, 3-E-0r4
Proposed References to be provided:	None
Learning Objective:	Objective 12.0
Question Source:	Modified Bank – IP2 Question # 24353
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001000K205 Knowledge of bus power supplies to the following: M/G sets (Control Rod Drive System)	
Importance:		3.1	3.5

Question: #56

Given the following conditions:

- The Unit is at 100% power.
- #31 Rod Drive MG Set is cleared and tagged for motor replacement.
- #32 Emergency Diesel Generator is cleared and tagged for a fuel line leak repair.
- A fault occurred in the Buchanan switchyard resulting in a loss of 138KV power.

What is the immediate impact on the plant?

a. The reactor will remain at approximately 100% power because #32 Rod Drive MG Set still energized from 480 Volt Bus 5A.
b. The reactor will trip because power is lost to the only remaining Rod Drive MG Set with the loss of 480 Volt Bus 6A.
c. The reactor will trip because power is lost to the only remaining Rod Drive MG Set with the loss of 480 Volt Bus 3A.
d. The reactor will remain at approximately 100% power because #32 Rod Drive MG Set still energized from 480 Volt Bus 2A.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that #32 Rod Drive MG Set is powered from 480 Volt Bus 5A via the 33 EDG.
b. Correct. #32 Rod Drive MG Set is powered from 480 Volt Bus 6A which is de-energized due to the loss of 138KV and #32 EDG being cleared and tagged.
c. Incorrect. Plausible because the student may believe that #32 Rod Drive MG Set is powered from 480 Volt Bus 3A. Also plausible because Bus 3A does not lose power as of the result of the switchyard fault.
d. Incorrect. Plausible because the student may believe that #32 Rod Drive MG Set is powered from 480 Volt Bus 2A. Also plausible because Bus 2A does not lose power as of the result of the switchyard fault. Bus 2A is actually the power supply for the #31 Rod Drive MG Set, but that MG Set is cleared and tagged.

Technical References:	I3LP-ILO-ICROD
Proposed References to be provided:	None
Learning Objective:	Objective D
Question Source:	Modified Bank Question # 25961
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011000K401 Knowledge of PZR LCS design feature(s) and/or interlock(s) which provide for the following: Operation of PZR heater cutout at low PZR level	
Importance:		3.3	3.7

Question: #57

The plant is at 100% power steady with the following conditions:

- All Control Systems are in Automatic.
- Pressurizer Level Channel Defeat Switch is selected to the "Defeat I" position.
- Backup Heater Group #32 is ON.

Subsequently, Pressurizer Level Channel LT-461 fails to zero. After five minutes, which of the following describes the response to the instrument failure assuming no operator action is taken?

a. Actual Pressurizer level rises, charging flow lowers, all Pressurizer Heaters de-energize.
b. Actual Pressurizer level rises, charging flow rises, all Pressurizer Heaters de-energize.
c. Actual Pressurizer level rises, charging flow rises, Modulating Heater Group OFF and Backup Heater Group #32 remains ON.
d. Pressurizer level remains on program, charging flow remains constant, Backup Heater Group #32 remains ON.

Answer: a

Explanation / Justification

a. Correct. With the PZR Level Channel Defeat Switch in the "Defeat I" position, LT-461 is the alarm channel and LT-460 is the controlling channel. When the alarm channel fails to zero, letdown isolates via LCV-460 causing a charging/letdown mismatch with actual pressurizer level rising. The controlling channel will see actual level greater than program level and lower charging flow. The alarm channel failing to zero will also de-energize all pressurizer heaters.
b. Incorrect. Plausible because the student may believe that LT-461 is the controlling channel based on the position of the channel defeat switch. This would be the correct response for the controlling channel. The controlling channel would see a low level and raise charging flow to compensate.
c. Incorrect. Plausible because the student may believe that LT-461 is the controlling channel. The student may also believe that because the Backup Heater Group #32 was in manual

"ON" that the failed level channel would not affect the operation of the heater. Incorrect understanding of the heater cut-out circuit.

d. Incorrect. Plausible because this would be the effect if, LT-461 was the defeated channel. The defeated channel has no input to the Pressurizer Level Control System.

Technical References:	I3LP-ILO-ICPZLV
Proposed References to be provided:	None
Learning Objective:	Objective E-5
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	014000K202 Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of power to the RPIS	
Importance:		3.1	3.6

Question: #58

Unit 3 is operating at 100% power with the IRPI System aligned to its backup power supply. Offsite power is lost, the plant remains on line and all equipment functions as designed. How does this affect power to the IRPIs and what Tech Spec actions are required, if any?

a. The IRPIs will lose power since the backup power supply is MCC-39. Rod Control must immediately be placed in Manual to comply with Technical Specification 3.1.7.B.
b. The IRPIs will lose power since the backup power supply is MCC-39. There are no applicable Technical Specification requirements associated with a loss of power to the IRPIs.
c. The IRPIs will have power because the backup power supply is 36C. If the IRPIs had been on normal power (MCC-39), they would have lost power, and Rod Control would have had to be placed in Manual to comply with Technical Specification 3.1.7.B.
d. The IRPIs will have power because the backup power supply is 36C. If the IRPIs had been on normal power (MCC-39), they would have lost power. However, there are no applicable Technical Specification requirements associated with a loss of power to the IRPIs.

Answer: a

Explanation / Justification

a. Correct. The IRPI backup power supply is MCC-39 which is fed from 480 Volt Bus 5A. Bus 5A will lose power due to the Loss of Off-Site Power; therefore power has been lost to the IRPI System. Technical Specification 3.1.7.B requires Control Rods in Manual with a completion time of immediately if more than one IRPI per group is inoperable. See Note prior to 3-AOP-ROD-1, step 4.136.
b. Incorrect. Plausible because the IRPI backup power supply is MCC-39 which is fed from 480 Volt Bus 5A. Bus 5A will lose power due to the Loss of Off-Site Power; therefore power

has been lost to the IRPI System. Student may not recognize that all IRPIs are inoperable due to loss of indication.

c. Incorrect. Plausible because the student may believe that MCC-36C is the backup power supply to the IRPI System. Plausible because MCC-36C is fed from 480 Volt Bus 2A which still has power following the Loss of Off-Site Power event. Technical Specification requirement is also true.
d. Incorrect. Plausible because the student may believe that MCC-36C is the backup power supply to the IRPI System. Plausible because MCC-36C is fed from 480 Volt Bus 2A which still has power following the Loss of Off-Site Power event. Student may not recognize that all IRPIs are inoperable due to loss of indication.

Technical References:	I3LP-ILO-ICRPI
Proposed References to be provided:	None
Learning Objective:	Objectives C, F, and H
Question Source:	Modified Bank Question # 25119
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	033000A302 Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Spent fuel leak or rupture	
Importance:		2.9	3.1

Question: #59

Design features of the Spent Fuel Cooling and Purification Loop ensure the fuel stored in the Spent Fuel Pool will not become uncovered as a result of any postulated loss of system integrity.

Which one of the following correctly states these design features?

a. The Spent Fuel Cooling Pump suction pipe has an anti-siphon hole, and the Spent Fuel Cooling Pump will automatically trip on Low Spent Fuel Level.
b. The Spent Fuel Cooling Pump suction pipe has an anti-siphon hole, and the return line outlet is above the fuel level.
c. The Spent Fuel Cooling loop does not have drains located below the level of the fuel, and the makeup flow capacity exceeds the System flow.
d. The Spent Fuel Cooling Pump suction pipe inlet is above the fuel level, and there is an anti-siphon hole on the return pipe.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the return line has a ¼ inch hole to prevent siphoning and the student may confuse suction and discharge piping design features.
b. Incorrect. Plausible because the return line has a ¼ inch hole to prevent siphoning and the student may confuse suction and discharge piping design features.
c. Incorrect. Plausible because there are no drain lines in the spent fuel pit.
d. Correct. The SFP cooling pump suction is approximately 6 feet below the surface of the pit (remember 23 feet of water is maintained above the fuel assemblies) and the SFP pump discharges into the pit approximately 7 feet above the top of the fuel assemblies. An anti-siphon hole in the discharge pipe prevents draining due to return line failure.

Technical References:	I3LP-ILO-SFP001
Proposed References to be provided:	None
Learning Objective:	Objective 7.0
Question Source:	Bank Question # 23900

Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	0340002140 Knowledge of refueling administrative requirements	
Importance:		2.8	3.9

Question: #60

In accordance with 3-REF-003-GEN, Fuel Movement Requirements – Core Reload, WHEN the transfer system is unattended, THEN the transfer cart

a. SHALL always be stored in the Fuel Storage Building (FSB) to facilitate closing of the fuel transfer systems gate valve.
b. SHALL always be stored in the Containment Building (VC) to facilitate closing of the fuel transfer systems gate valve.
c. SHALL always be stored in the Containment Building (VC) to ensure the UPENDER CLEAR interlock is met.
d. SHALL always be stored in the Fuel Storage Building (FSB) to ensure the UPENDER CLEAR interlock is met.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the transfer cart's home position is the FSB and that in order to close the gate valve, the cart must be positioned in the FSB.
b. Correct. 3-REF-003-GEN, precautions and limitations steps 2.21 and 2.33 both state that when the transfer system is left unattended, the transfer cart shall always be stored in the containment building to facilitate the closing of the fuel transfer systems gate valve.
c. Incorrect. Plausible because the containment is the transfer cart's home position and the procedure precaution states that it should be stored in its home position when the transfer system is left unattended. Also plausible because the UPENDER CLEAR interlock does exist and it ensures that the respective upender is clear of either the manipulator crane or the fuel handling crane. Student may believe that having this interlock made up allows manipulations in both respective buildings without use of the transfer system.
d. Incorrect. Plausible because the student may believe that the transfer cart's home position is the FSB. Also plausible because the UPENDER CLEAR interlock does exist and it ensures that the respective upender is clear of either the manipulator crane or the fuel handling crane. Student may believe that having this interlock made up allows manipulations in both respective buildings without use of the transfer system.

Technical References:	I3LP-ILO-FHD-001, 3-REF-003-GEN System Description 17.0

Proposed References to be provided:	None
Learning Objective:	Objective C (E-3)
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035000K113 Knowledge of the physical connections and/or cause-effect relationships between the S/Gs and the following systems: Condensate System	
Importance:		2.7	2.8

Question: #61

Operators are responding to a Loss of Secondary Heat Sink Red Path in accordance with 3-FR-H.1. The only feed source that is available is 31 Condensate Pump. RCS pressure is greater than Steam Generator Pressure. Steam Generator narrow range (NR) levels are off scale low and wide range (WR) levels are 45% and lowering. What actions are required to mitigate the event?

a. Depressurize at least one SG to less than 380 psig to establish feed flow. Closely monitor SG WR levels for Bleed and Feed criteria during the SG depressurization. Bleed and Feed is initiated if the average of the 3 lowest WR SG levels is less than 20%.
b. Depressurize all but one SG to less than 380 psig to establish feed flow. Closely monitor SG WR levels for Bleed and Feed criteria during the SG depressurization. Bleed and Feed is initiated if the average of all WR SG levels is less than 30%.
c. Depressurize at least one SG to less than 380 psig to establish feed flow. Closely monitor SG WR levels for Bleed and Feed criteria during the SG depressurization. Bleed and Feed is initiated if the average of all WR SG levels is less than 30%.
d. Depressurize all but one SG to less than 380 psig to establish feed flow. Closely monitor SG WR levels for Bleed and Feed criteria during the SG depressurization. Bleed and Feed is initiated if the average of the 3 lowest WR SG levels is less than 20%.

Answer: a

Explanation / Justification

a. Correct. See EOP 3-FR-H-1, Attachment 3; Establishing Feedwater Flow from Secondary Plant, steps 3 & 4. SG pressure must be lowered to less than 380# to allow a condensate pump to feed. Step states “at least one”. Bleed and Feed criteria is stated in step 2 of the procedure and foldout page as “average of the lowest WR SG levels less than 20% (30% adverse).”
b. Incorrect. Plausible because the student may believe that he needs to leave one SG untouched for future needs. Also plausible because the student may remember the adverse SG level of 30%. Containment conditions are normal.

c. Incorrect. Plausible because the first part of the distractor is correct and because the student may remember the adverse SG level of 30%. Containment conditions ae normal.
d. Incorrect. Plausible because the student may believe that he needs to leave one SG untouched for future needs. Also plausible because the second part of the distractor is correct.

Technical References:	I3LP-ILO-EOPFRH 3-FR-H.1 System Description 20.0; Condensate
Proposed References to be provided:	None
Learning Objective:	Objectives 4.0 & 6.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 4, 8, & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045000A106 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of secondary plant parameters following T/G trip	
Importance:		3.3	3.7

Question: #62

The following plant conditions exist:

- The Plant is operating at 100% power.
- All systems are lined up in their normal lineups.
- All control systems are in automatic.
- Main Generator output breakers 1 and 3 trip due to a pilot wire trip.

Which of the following describes the expected immediate plant response?

a. S/G pressure initially RISES as main turbine is lost, S/G levels initially LOWER due to shrink, feed flow initially RISES.
b. S/G pressure initially RISES as main turbine is lost, S/G levels initially LOWER due to shrink, feed flow initially LOWERS.
c. S/G pressure initially LOWERS as main turbine is lost, S/G levels initially LOWER due to shrink, feed flow initially RISES.
d. S/G pressure initially LOWERS as main turbine is lost, S/G levels initially RISE due to lower steam pressure, feed flow initially LOWERS.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may think that an ESF actuation for AFW will cause levels to initially rise, however that is not an immediate response. Steam Flows initially rapidly decrease due to the loss of the turbine causing initial feed flow to decrease as well.
b. Correct. S/G pressures rapidly increase to ~ 1000 psig due to the immediate drop in steam flow caused by the turbine trip. Continued steam dump system action will maintain SG pressure at ~ 1000 psig. S/G NR Levels decrease rapidly off-scale low due to the increase in SG pressure and the drop in SG recirculation flow causing shrink in the SG. Steam Flows

initially rapidly decrease due to the loss of the turbine causing initial feed flow to decrease as well.

c. Incorrect. Plausible because the student may believe that less steam production caused by the loss of turbine load will result in lower S/G pressures.
d. Incorrect. Plausible because the student may believe that less steam production caused by the loss of turbine load will result in lower S/G pressures.

Technical References:	I3LP-ILO-EOPE00, 3-E-0r4
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	Bank Question # 17368 IP3 NRC Exam 2006 Seabrook NRC Exam 2003
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	071000K305 Knowledge of the effect that a loss or malfunction of the Waste Gas Disposal System will have on the following: ARM and PRM systems	
Importance:		3.2	3.2

Question: #63

Given the following conditions:

- The Waste Gas System is aligned for normal operation
- 31 Large Gas Decay Tank is in-service
- 32 Large Gas Decay Tank is in standby
- R-20, Gas Decay Tank Activity Monitor is **NOT** in alarm

Which one of the following describes the system response (assuming no operator action) if the 31 Large Gas Decay Tank relief valve were to fail open?

a. Radiation monitor R-27, Wide Range Plant Vent Gas Activity Monitor will alarm causing the plant vent discharge valve, RCV-014 to close isolating the release.
b. Radiation monitor R-14, Plant Vent Gas Activity Monitor and R-27, Wide Range Plant Vent Gas Activity Monitor will both alarm causing the plant vent discharge valve, RCV-014 to close isolating the release.
c. Radiation monitor R-20, Gas Decay Tank Activity Monitor will alarm causing the plant vent discharge valve, RCV-014 to close isolating the release.
d. Radiation monitor R-14, Plant Vent Gas Activity Monitor and R-27, Wide Range Plant Vent Gas Activity Monitor may alarm, however the release will continue.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the R-27, Wide Range Plant Vent Gas Monitor is used to monitor Gas Decay Tank releases and when in alarm, it will isolate a normal gas decay tank release path via the RCV-014.
b. Incorrect. Plausible because both the R-14, Plant Vent Gas Activity Monitor and R-27, Wide Range Plant Vent Gas Activity Monitor are used to monitor Gas Decay Tank releases and when in alarm, they will isolate a normal gas decay tank release path via the RCV-014.
c. Incorrect. Plausible because the R-20, Gas Decay Tank Activity Monitor is used to monitor the radioactive gas moving through the Waste Gas System and accumulating in the in

service gas decay tank. However, there is no automatic isolation / actions associated with the monitor in alarm. It is used to warn the operators of the potential for reaching the tank curie limit of 19,400 curies.

d. Correct. Although activity in the in service gas decay tank may cause plant vent radiation monitors to alarm, the release via the relief valve will continue because the relief valve has a direct lease path to the plant vent downstream of the RCV-014. It is not capable of being isolated.

Technical References:	I3LP-ILO-GWR001 System Description 5.2; Gaseous Waste Disposal – Figure 5.2-4 System Description 12.0; Radiation Monitoring
Proposed References to be provided:	None
Learning Objective:	Objectives 9e & 11e
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7 & 13

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	079000A401 Knowledge of SAS (Station Air System) design feature(s) and/or interlock(s) which provide the following: Cross-connect with IAS (Instrument Air System)	
Importance:		2.9	3.2

Question: #64

Which of the following describes how Station Air is automatically cross-tied to the Instrument Air System?

a. When instrument air system pressure falls to 95 psig, PCV-1142 is automatically opened to admit air to the system. Supplied downstream of the heatless desiccant dryers, this air is **not** conditioned the same as the air normally supplied by the Instrument Air compressors.
b. When instrument air system pressure falls to 90 psig, PCV-1142 is automatically opened to admit air to the system. Supplied upstream of the heatless desiccant dryers, this air is conditioned the same as the air normally supplied by the Instrument Air compressors.
c. When instrument air system pressure falls to 95 psig, PCV-1142 is automatically opened to admit air to the system. Supplied upstream of the heatless desiccant dryers, this air is conditioned the same as the air normally supplied by the Instrument Air compressors.
d. When instrument air system pressure falls to 90 psig, PCV-1142 is automatically opened to admit air to the system. Supplied downstream of the heatless desiccant dryers, this air is **not** conditioned the same as the air normally supplied by the Instrument Air compressors.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the standby IA compressor starts at 95 psig. Student may also believe that the station air cross-tie line comes in downstream of the dryers.
b. Correct. When instrument air system pressure falls to 90 psig, PCV-1142 is automatically opened to admit air to the system. Supplied upstream of the heatless desiccant dryers, this air is conditioned the same as the air normally supplied by the Instrument Air compressors. See System Description 29.2, page 19, section 2.9.1. Also see lesson plan.
c. Incorrect. Plausible because the standby IA compressor starts at 95 psig. The discussion regarding conditioned air is correct.
d. Incorrect. Plausible because the valve opens at 90 psig, first part of distractor is correct. Student may believe that the station air cross-tie line comes in downstream of the dryers.

Technical References:	I3LP-ILO-IA001 System Description 29.2
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	086000K604 Knowledge of the effect of a loss or malfunction on the Fire Protection Systems following will have on the: Fire, smoke, and heat detectors	
Importance:		2.6	2.9

Question: #65

Given the following conditions and TRM references:

- The Unit is in Mode 4.
- A loss of 125VDC has caused a loss of power to the CO_2 Fire Protection Control Panels.
- All Diesel Generator Room(s) fire detectors and fire barriers are operable.
- The 31 Emergency Diesel Generator Wet Pipe Sprinkler System is isolated for maintenance to replace faulty sprinkler heads.

What are the required compensatory actions for the inoperable fire suppression equipment and can CO_2 suppression be manually discharged into the Diesel Generator Room(s)?

a. Establish a continuous fire watch and backup fire suppression equipment. Yes, on a loss of DC power, a CO_2 suppression discharge can be manually initiated.
b. Establish an hourly fire watch patrol with backup fire suppression equipment. No, a loss of DC power causes the master pilot valve to fail closed precluding a manual initiation of CO_2 suppression.
c. Establish an hourly fire watch patrol with backup fire suppression equipment. Yes, on a loss of DC power, a CO_2 suppression discharge can be manually initiated.
d. Establish a continuous fire watch and backup fire suppression equipment. No, a loss of DC power causes the master pilot valve to fail closed precluding a manual initiation of CO_2 suppression.

Answer: a

Explanation / Justification

a. Correct. Student is given various fire protection TRMs. TRM 3.7.A.2; Fire Protection Water Spray and Sprinkler Systems Action B, states if the Diesel Generator Building Water Spray System is inoperable, then establish a continuous fire watch and backup fire suppression equipment. Procedure 3-SOP-FP-003; Fire Protection CO_2 System Operation discusses operation with DC power unavailable.

b. Incorrect. Plausible because the student may believe that an hourly fire watch is acceptable because all the fire detectors and fire barriers are operable. If the 31 EDG sprinkler wasn't also inoperable, this could be true. The student may also see action A of TRM 3.7.A.2 and believe it is applicable. Also plausible because the student may believe that a loss of DC renders the CO_2 Fire Protection System completely unavailable, however, the master pilot valve fails open on a loss of DC power which enables manual actuation per 3-SOP-FP-003.
c. Incorrect. Plausible because the student may believe that an hourly fire watch is acceptable because all the fire detectors and fire barriers are operable. If the 31 EDG sprinkler wasn't also inoperable, this could be true. The student may also see action A of TRM 3.7.A.2 and believe it is applicable. Also plausible because the second part of the distractor is true, Procedure 3-SOP-FP-003; Fire Protection CO_2 System Operation discusses operation with DC power unavailable.
d. Incorrect. Plausible because the student may believe that a loss of DC renders the CO_2 Fire Protection System completely unavailable, however, the master pilot valve fails open on a loss of DC power which enables manual actuation per 3-SOP-FP-003. The first part of the distractor is correct.

Technical References:	I3LP-ILO-FPS001 System Description 29.6 3-SOP-FP-003
Proposed References to be provided:	**TRM 3.7.A.2; Fire Protection Water Spray and Sprinkler Systems, TRM 3.7.A.4; Fire Detection Systems, TRM 3.7.A.7; CO_2 Fire Protection System, and the respective bases documents.**
Learning Objective:	Objective 5.a
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7

NOTE:
Fire Protection Lesson Plan, I3LP-ILO-FPS001, Objective, 5.a states:
"Describe the effect that a loss or malfunction of the following will have on the Fire Protection system, a. Loss of Power to CO_2 System." Student needs to know that manual capability still exists and that fire detection equipment (fire, smoke, and heat detectors) are unaffected by the DC power loss. Automatic actuation actuated by at least two heat detectors, but auto actuation is effected by the loss of DC power.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	Conduct of Operations	
	K/A #	1940012129 Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	
Importance:		4.1	4.0

Question: #66

In accordance with OAP-019, Component Verification and System Status Control; Verification SHALL be performed when component manipulations are performed on certain safety systems to restore these systems to a functional or operable status. Which of the systems below require Verification in accordance with OAP-019?

I. Carbon Dioxide Fire Protection System
II. Condensate
III. Fire Pump House
IV. Gaseous Disposal System
V. Instrument Air
VI. Liquid Disposal System
VII. Station Air

a. I, III, V, VII
b. II, IV, V, VI
c. I, III, IV, VI
d. I, II, III, IV

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may believe that instrument air is an important safety system listed on Attachment 1, Safety Systems Requiring Verification, of OAP-019. All other listed systems are correct.
b. Incorrect. Plausible because two of the systems listed are correct and the student may believe that both instrument air and condensate are important safety systems as both are required to maintain the unit on-line. Loss of either system can lead to a reactor trip and plant transient.
c. Correct. All four systems listed are included on Attachment 1, Safety Systems Requiring Verification, of OAP-019.
d. Incorrect. Plausible because the student may believe that condensate is an important safety system listed on Attachment 1, Safety Systems Requiring Verification, of OAP-019. Loss of the condensate system can lead to a reactor trip and plant transient. All other listed systems are correct.

Technical References:	I3LP-ILO-Admin-COO?, OAP-019
Proposed References to be provided:	None
Learning Objective:	Objective (Generic K/A – Admin-COO?)
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	COO	
	K/A #	1940012144 Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation	
Importance:		3.9	3.8

Question: #67

Given the following conditions:

- The Unit is in a "De-Fueled" condition
- The Reactor Core has been completely off loaded to the Spent Fuel Pool
- Today is day 20 of the refueling outage
- A loss of Spent Fuel Cooling has just occurred
- The NPO reports from the field that Spent Fuel Pool Temperature is currently 140°F
- 3-AOP-SF-1; Loss of Spent Fuel Cooling has been entered

Given reference 3-AOP-SF-1, Attachment 1, determine the Spent Fuel Pool heat-up rate.

a. 5°F/HR
b. 7.3°F/HR
c. 8.2°F/HR
d. 12°F/HR

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may read graph incorrectly. The graph provides a time of 5 hours to reach 200°F from the current temperature of 140°F. Student may read the graph as 5°F/HR heat-up.
b. Incorrect. Plausible because the student may read the graph incorrectly. The graph provides a time of 8.2 hours to reach 200°F from the temperature of 100°F, not the reported 140°F current temperature. Student could then still use the 8.2 hours for the 60°F rise from 140°F and come up with 7.3°F/HR.

c. Incorrect. Plausible because the student may read the graph incorrectly. The graph provides a time of 8.2 hours to reach 200°F from the temperature of 100°F, not the reported 140°F current temperature. Student may then read the graph as 8.2°F/HR heat-up.
d. Correct. The graph provides a time of 5 hours to reach 200°F from the current temperature of 140°F. Student will note that it takes 5 hours for the pool to rise 60°F, therefore 12°F/HR.

Technical References:	I3LP-ILO-SFP001, 3-AOP-SF-1
Proposed References to be provided:	3-AOP-SF-1, Attach 1 (3-GRAPH-ACS-1F & 1G)
Learning Objective:	Objective 8a
Question Source:	Modified, IP3 Question # 18683
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	EC	
	K/A #	0000002223	
		Ability to track Technical Specification limiting conditions for operations	
Importance:		3.1	4.6

Question: #68

The Unit is operating at 100% when Reactor Engineering informs the Shift Manager that they have determined via the 31 EFPD Flux Map that the Heat Flux Hot Channel Factor ($F_q(Z)$) has exceeded its limit specified in the COLR.

Which of the following describes the required actions in accordance with Technical Specifications?

a. Within 15 minutes reduce THERMAL POWER ≥ 1% RTP for each 1% $F_q(Z)$ exceeds limit.
b. Within 60 minutes reduce THERMAL POWER to < 50% RTP.
c. Within 30 minutes reduce THERMAL POWER to < 90% RTP.
d. Within 15 minutes reduce THERMAL POWER ≥ 3% RTP for each 1% $F_q(Z)$ exceeds limit.

Answer: a

Explanation / Justification

a. Correct. Technical Specification LCO 3.2.1 states that $F_q(Z)$ shall be within the limits specified in the COLR. Action A states to reduce THERMAL POWER ≥ 1% RTP for each 1% $F_q(Z)$ exceeds limit within 15 minutes.
b. Incorrect. Plausible because the AFD Tech Spec reduces THERMAL POWER to < 50% within 30 minutes when the cumulative penalty time has exceeded 1 hour in the previous 24 hours. Student may confuse Power Distribution Technical Specification actions and only remember the 60 minutes of penalty time.
c. Incorrect. Plausible because the AFD Tech Spec reduces THERMAL POWER to < 90% within 15 minutes when AFD is outside the target band. Student may confuse Power Distribution Technical Specification actions.
d. Incorrect. Plausible because the 15 minute action time is correct per Tech Spec 3.2.1, but the 3% RTP for each 1% is the same reduction required by the QPTR Tech Spec for QPTR not within limit (after 2 hours). Student may confuse Power Distribution Technical Specification actions.

Technical References:	Technical Specification 3.2.1
	I3LP-ILO-ICNXC

Proposed References to be provided:	None
Learning Objective:	Objective E-3
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 2 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	Equipment Control	
	K/A #	1030002238 Knowledge of conditions and limitations in the facility license	
Importance:		3.6	4.5

Question: #69

In Modes 1, 2, 3, and 4 Containment Penetration flow paths with manual isolation valves may be intermittently opened under administrative control. What administrative controls are required to open a normally locked closed manual containment isolation valve while still complying with Technical Specifications and are there any time limitations regarding the opening on the valve

a. The administrative controls consist of using two operators to concurrently verify the position of the valve before and after manipulating the valve. The valve SHALL not remain open for greater than one hour.
b. The administrative controls consist of entering a "Tracking LCO" to document the time the valve is open. The valve SHALL not remain open for greater than four hours.
c. The administrative controls consist of entering the applicable Technical Specification Action Statement and prior to opening the valve, ensure that the penetration affected has been isolated by the use of one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve. The valve SHALL not remain open for more than 72 hours.
d. The administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. The valve SHALL only remain open as long as necessary to perform the intended function.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because in accordance with OAP-19 the valve may require concurrent verification. The student may believe that Tech Spec 3.6.1 is applicable during the valve opening (Containment inoperable), therefore allowing only 1 hour to comply with the specification.
b. Incorrect. Plausible because a "Tracking LCO may be applicable from a configuration control process requirement. Also plausible because Tech Spec 3.6.3 action A has a 4 hour completion time.
c. Incorrect. Plausible because the actions for an inoperable containment isolation valve are exactly what is stated. Also plausible because Tech Spec 3.6.3 action C has a 72 hour completion time.
d. Correct. OAP-019, step 4.1.1 and Technical Specification 3.6.3 BASES state; "The administrative controls consist of stationing a dedicated operator at the valve controls, who

is in continuous communication with the control room. The valve SHALL only remain open as long as necessary to perform the intended function."

Technical References:	I3LP-ILO-VCCIS OAP-019 Technical Specification 3.6.3 BASES
Proposed References to be provided:	None
Learning Objective:	Objective 4.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 7 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	EC	
	K/A #	1030002240 Ability to apply Technical Specifications for a system	
Importance:		3.4	4.7

Question: #70

Which one of the following individual RCS leakages will require a unit shutdown in accordance with Technical Specifications?

a. 0.5 gpm leak to containment atmosphere from an unknown source.
b. 8.0 gpm measured leakage to the PRT.
c. 0.8 gpm leakage past the inner reactor vessel flange o-ring.
d. 0.11 gpm primary to secondary leakage on 31 Steam Generator.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because student may believe any leakage to the containment atmosphere from an unknown source may require a plant shutdown because it could interfere with future leakage detection by the containment process radiation monitors.
b. Incorrect. Plausible because the student may not know that leakage to the PRT is considered identified leakage or they may believe that the tech spec only allows a maximum of 1 gpm identified leakage.
c. Incorrect. Plausible because the student may believe that reactor vessel flange leakage is considered pressure boundary leakage.
d. Correct. Technical Specification LCO 3.4.13 states that RCS operational LEAKAGE shall be limited to: 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG). 150 gallons per day is equal to .104 gpm. Or .11 gpm is equal to 158.4 gpd.

Technical References:	Technical Specification 3.4.13 I3LP-ILO-RCS001
Proposed References to be provided:	None
Learning Objective:	Objective 9.0
Question Source:	Modified Bank Question # 1671
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 3 & 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	RC	
	K/A #	1940012304 Knowledge of the radiation exposure limits under normal or emergency conditions	
Importance:		3.2	3.7

Question: #71

A plant worker has received 700 mrem TEDE dose for the current year. This worker has been assigned to work in a 200 mrem/hr radiation field.

How long will it take for this worker to reach the 10CFR20 limits and the Site Administrative limits for whole body dose without any dose extensions being authorized?

	10CFR20	SITE ADMIN
a.	21.5 hours	21.5 hours
b.	19.0 hours	19.0 hours
c.	21.5 hours	6.5 hours
d.	19.0 hours	1.5 hours

Answer: c

Explanation / Justification

a. Incorrect. Plausible because student may believe that both the site and federal 10CFR20 limit are the same, 5 Rem per year.
b. Incorrect. Plausible because the student may believe that the total administrative annual limit (including dose extensions) applies, 4.5 Rem per year. The student may also believe that since dose extensions allow up to 4.5 Rem per year that it is also the federal limit.
c. Correct. The individual is allowed up to 5 Rem per year per 10CFR20.1201 and up to 2 Rem per year without any dose extensions in accordance with EN-RP-201; Dosimetry Administration.
d. Incorrect. Plausible because student may believe that since dose extensions allow up to 4.5 Rem per year that it is also the federal limit. The student may also believe that personnel are limited to 1 Rem per year without any dose extensions. 1000 mrem is a limit if the individual's lifetime dose is greater than their age.

Technical References:	IOWKB-ILO-ADM00 EN-RP-201 10CFR20.1201
Proposed References to be provided:	None
Learning Objective:	Objectives 2,3,4 ?
Question Source:	Modified Bank Question # 24102

Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10 & 12

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	RC	
	K/A #	1940012315 Knowledge of the radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	
Importance:		2.9	3.1

Question: #72

Which of the following Area Radiation Monitors is one that is checked in 3-E-1; Loss of Reactor or Secondary Coolant, to determine if a LOCA Outside Containment is occurring and if transition to 3-ECA-1.2, LOCA Outside of Containment is required?

a. R-69; 54' Elevation Pipe Penetration Area Monitor
b. R-70; 80' Elevation Fan House Area Monitor
c. R-6; Sampling Room Monitor
d. R-34A; 31 CVCS HUT Monitor

Answer: a

Explanation / Justification

a. Correct. See 3-E-1, Loss of Reactor or Secondary Coolant, Attachment 1; Plant Status Evaluation, steps 5 and 6. R-64 through R-69 are the listed radiation area monitors used to determine if PAB radiation conditions are normal. Deviation document states that the ERG's intent is that radiation monitors be used to locate the source of the leakage into the PAB.
b. Incorrect. Plausible because R-70 is an area radiation monitor in the PAB at elevation 80'. It would not meet the intent of determining the source of the leak. It is not listed in 3-E-1, Attachment 1, step 5.
c. Incorrect. Plausible because R-6 is an area radiation monitor located on the 55' elevation of the PAB. It would not meet the intent of determining the source of the leak. It is not listed in 3-E-1, Attachment 1, step 5.
d. Incorrect. Plausible because R34A is an area radiation monitor located on the 33' elevation of the PAB. It would not meet the intent of determining the source of the leak. It is not listed in 3-E-1, Attachment 1, step 5.

Technical References:	I3LP-ILO-EOPE10, 3-E-1r4 & Dev. Document
Proposed References to be provided:	None
Learning Objective:	Objective 19.0
Question Source:	Modified Bank (Salem 2012 NRC Exam # 72)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 10 & 12

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	EOP-EP	
	K/A #	1940012412	
		Knowledge of general operating crew responsibilities during emergency operations	
Importance:		4.0	4.3

Question: #73

The crew has entered EOP 3-E-0; Reactor Trip or Safety Injection and the CRS has directed the BOP Operator to implement 3-RO-1; "BOP Operator Actions during the Use of EOPs." Which of the following describes conditions which will result in the termination of the 3-RO-1 procedure?

a. The BOP Operator has completed or initiated all the actions of 3-RO-1 **or** if 3-E-0; Reactor Trip or Safety Injection has been exited.
b. The BOP Operator has completed or verified completed all the actions of 3-RO-1 **and** 3-E-0; Reactor Trip or Safety Injection has been exited.
c. The CRS has transitioned to 3-E-1, Loss of Reactor or Secondary Coolant **and** SI Initiation has been verified.
d. The CRS has transitioned to 3-ES-1.1; SI Termination **or** the CRS has transitioned to 3-ECA-0.0, Loss of All AC Power

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the first statement is correct, completion of RO-1 is one exit point. However, 3-RO-1, step 16 Response Not Obtained directs the BOP Operator to continue with the procedure if 3-E-0 has been exited.
b. Incorrect. Plausible because the first statement is correct, however actions only need to be initiated. However, 3-RO-1, step 16 Response Not Obtained directs the BOP Operator to continue with the procedure if 3-E-0 has been exited.
c. Incorrect. Plausible because the student may believe that once diagnosis of the accident has been made, the transition procedure will continue any required verification steps.
d. Correct. See third bullet note prior to step 1 of 3-RO-1 which states that; "If CRS subsequently transitions to 3-ECA-0.0, Loss of All AC Power, then this procedure SHALL be terminated. Also see step 16 of 3-RO-1 to see that the procedure is terminated if the CRS has transitioned to 3-ES-1.1; SI Termination. Also see note prior to step 17.

Technical References:	I3LP-ILO-EOPE00, 3-E-0r4, 3-RO-1 & Dev Doc OAP-012; EOP Users Guide
Proposed References to be provided:	None
Learning Objective:	Objectives 11.0 & 13.0

Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	EOP-EP	
	K/A #	0000742423 Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations	
Importance:		3.4	4.4

Question: #74

Operators have responded to a design large break LOCA in containment. Operators have transitioned from E-0; Reactor Trip or Safety Injection to E-1; Loss of Reactor or Secondary Coolant. While in E-1, RWST Level lowers to less than 11.5 feet. At the same time that RWST Level lowers to less than 11.5 feet, an Orange Path Critical Safety Function for 3-FR-Z.1; Response to Containment High Pressure is announced by the STA. There is no other Red or Orange Path identified. Which procedure transition is required?

a. Transition immediately to 3-FR-Z.1; Response to Containment High Pressure. If an Orange Path condition arises, suspend any E-Set procedure in progress and transition to the FRP required by the Orange Path.
b. Perform 3-FR.Z.1; Response to Containment High Pressure in parallel with ES-1.3; Transfer to Cold Leg Recirculation. Only Red Path conditions require suspension of any E-Set procedure in progress.
c. Transition immediately to ES-1.3; Transfer to Cold Leg Recirculation. FRPs shall not be implemented prior to the completion of transfer to cold leg recirculation.
d. Perform 3-FR.Z.1; Response to Containment High Pressure in parallel with E-1; Loss of Reactor or Secondary Coolant. The only E-Set procedure that has priority over an FRP is ECA-0.0; Loss of All AC Power.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because for most conditions this statement is true. However certain procedures take priority over FRPs because of their treatment of specific initiating events. See step 4.4.8.2 of OAP-012; EOP Users Guide.
b. Incorrect. Plausible because the student may believe that only Red Path conditions require suspension of any E-Set procedures.
c. Correct. See step 4.4.8.2 of OAP-012; EOP Users Guide. The steps of ES-1.3; Transfer to Cold Leg Recirculation must be completed, even if challenges to a Critical Safety Function occur at this time, since these steps ensure core cooling.
d. Incorrect. Plausible because the student may not recognize that less than 11.5 feet in the RWST requires transition to ES-1.3; Transfer to Cold Leg Recirculation. The student may

also believe that only Red Path conditions require suspension of any E-Set procedures. ECA-0.0; Loss of All AC Power is not the only E-Set procedure that takes priority over FRPs.

Technical References:	I3LP-ILO-EOPROU OAP-012; EOP Users Guide
Proposed References to be provided:	None
Learning Objective:	Objective 12.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #	3	
	Group #	EOP-EP	
	K/A #	1940012430 Knowledge of events related to system operation/status that must be reported to internal organization or external agencies, such as the State, the NRC, or the transmission system operator	
Importance:		2.7	4.1

Question: #75

Given the following:

- Unit 3 is at 100% power
- The Unit 3 Shift Manager has just been notified by the Security Shift Supervisor that a hostile action is occurring within the Owner Controlled Area.

The Unit 3 Shift Manager confers with the Unit 2 Shift Manager and agrees that both units are affected. The _______ Shift Manager classifies the event and declares an ALERT.

Prior to initiating the ALERT notification, the Security Shift Supervisor informs both Unit Shift Managers that a hostile action is now occurring in the Protected Area. The event has been reclassified and declared a Site Area Emergency.

As the Control Room Communicator, you will

a. Unit 2, disregard previous classification and continue notification with highest current classification. Follow-up notification SHALL include details of all conditions/emergency classifications.
b. Unit 3, continue notification, then state at the end the following; "Changes in plant conditions indicate a potential for escalating the Emergency Classification."
c. Unit 3, disregard previous classification and continue notification with highest current classification. Follow-up notification SHALL include details of all conditions/emergency classifications.
d. Unit 2, continue notification, then state at the end the following; "Changes in plant conditions indicate a potential for escalating the Emergency Classification."

Answer: a

Explanation / Justification

a. Correct. IP-EP-210 states that both Shift Managers shall confer with each other for any event of condition which may affect both Units such as security or natural events. If it is agreed that both units are affected, then the Unit 2 Shift Manager SHALL classify and declare the emergency. Step 1.4.F also states that if plant condition/emergency classification changes prior to initiating notification, disregard previous classification and continue with highest current classification.
b. Incorrect. Plausible because student may believe that since the Unit 3 Shift Manager was the first one notified, he would assume the role for classification and declaration of the event. Also plausible because IP-EP-210, step 1.4.G tells the communicator to continue with the notification if the plant condition/emergency classification changes while performing notification. The question stem states that the ALERT notification had not been initiated yet.
c. Incorrect. Plausible because student may believe that since the Unit 3 Shift Manager was the first one notified, he would assume the role for classification and declaration of the event. Second part of distractor is correct.
d. Incorrect. Plausible because IP-EP-210, step 1.4.G tells the communicator to continue with the notification if the plant condition/emergency classification changes while performing notification. The question stem states that the ALERT notification had not been initiated yet. First part of distractor is correct.

Technical References:	IP-EP-210
Proposed References to be provided:	None
Learning Objective:	E-Plan Admin
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.41 (b) 10