CLIFFS NUCLEAR POWER PLANT

2012 NRC

INITIAL LICENSED

OPERATOR

SRO WRITTEN EXAM

KEY

76. 2.4 - Emergency Procedures / Plan (2.4.11)

Unit-1 is performing a reactor startup at 300 MWD/MTU. Critical data has been recorded and reactor power stabilized at the POAH with Group 4 CEAs at 90 inches.

The TBV controller, 1-PIC-4056, output signal fails to 10% in automatic resulting in a plant cooldown. The RO monitoring the reactor reports the following:

- Reactor power is below 10E-1% and continuing to lower
- SUR is negative
- RCS T_{COLD} is 530 °F and lowering slowly

As the CRS, which **ONE** of the following actions would you direct the crew to perform?

A. Withdraw Regulating Group CEAs to restore RCS T_{COLD}.

B. Trip the reactor and implement EOP-0.

C. Place the TBV controller in manual at 0% output.

D. Fully insert Regulating Group 4 CEAs in manual sequential.

Answer: B

Answer Explanation:

A. **Incorrect** - OP-2 states the following precaution: _Primary plant anomalies caused by secondary plant transients are rarely, if ever, successfully mitigated by adding positive reactivity, especially by withdrawing CEAs. Do **NOT** use CEAs to control RCS temperature without an approved procedure_. Events have occurred in the industry where CEAs have been withdrawn to reestablish critical conditions. Conditions indicate the reactor has gone subcritical and AOP-7K Section IV Actions require a reactor trip and implement EOP-0.

B. **Correct** - Per AOP-7K, which is entered due to overcooling event and plant is in MODE 2, this is the correct action based on reactor conditions provided.

C. **Incorrect** - Although this is part of the recovery action to restore from overcooling event, conditions indicate the reactor has gone subcritical and AOP-7K Section IV Actions require a reactor trip and implement EOP-0.

D. **Incorrect** - OP-2 directs with conditions of reactor above, to FULLY insert ALL regulating CEAs not just Group 4. However, an overcooling event has occurred and actions of AOP-7K are required and operators will trip the reactor and implement EOP-0.

2012 NRC SRO EXAM MASTER KEY

Question 76 (Q97042)			
Topic:	Actions required when Rx goes subcritical from overcooling event in Mode 2		
Tier/Group:	3		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.11 - Knowledge of abnormal condition procedures.		
SRO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given an overcooling event in progress, determine and implement the applicable actions to mitigate the event per plant operating procedures.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	AOP-7K, Overcooling Event in Modes 1 and 2; OP-2, Plant Startup from Hot Standby to Minimum Load		
Comments:	None		

77. 037 - Steam Generator Tube Leak (2.4.18)

During a Steam Generator Tube Rupture event, Pressurizer level is maintained or lowered to between 101 to 120 inches if backfill of the RCS is anticipated.

Which **ONE** of the following describes the basis for maintaining Pressurizer level in this band?

A. Minimizes the loss of primary fluid to the secondary.

B. Minimizes the potential for a Pressurized Thermal Shock Event.

C. Ensures RCS Pressure and inventory control is established.

D. Allows additional inventory to be added to the RCS with minimal impact.

Answer: D

Answer Explanation:

A. **Incorrect** - This is the basis for maintain subcooling at the low end of the band.

B. **Incorrect** - The basis for throttling minimizes the potential for a Pressurized Thermal Shock Event.

C. **Incorrect** - This is a basis for minimum Pressurizer level (101 inches) and minimum subcooling. "If backflow from the affected S/G is anticipated, maintaining a lower pressurizer level will allow additional inventory to be added to the RCS with minimal impact."

D. **Correct** - As stated in the EOP-6 Technical Basis document, "If backflow from the affected S/G is anticipated, maintaining a lower pressurizer level will allow additional inventory to be added to the RCS with minimal impact".

2012 NRC SRO EXAM MASTER KEY

Question 77 (Q26581)			
Topic:	EOP-6 Technical Basis		
Tier/Group:	1/2		
K/A Info:	037 - Steam Generator Tube Leak 2.4 - Emergency Procedures / Plan 2.4.18 - Knowledge of specific bases for EOPs.		
SRO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given EOP-6, determine the basis for maintaining PZR level band prior to backfill into the RCS.		
10 CFR Part 55 Content:	55.43(b)(1)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI 2006-2 Remediation EOP/AOP Basis exam (10/08)		
Technical references:	EOP-6 Step I. 4 and Technical Bases		
Comments:	None		

78. 034 - Fuel Handling (K1.04)

Given the following conditions on Unit-1:

- Core Reload is in progress per FH-305, Core Alterations
- The Refueling Machine Operator is inserting a new fuel assembly into the core with current hoist readout at 200 inches
- Refueling Control Room Operator reports an unexpected increase in count rate on two of the four wide range NI channels
- Audible count rate in the containment is rising
- You are the Fuel Handling Supervisor

Which **ONE** of the following designates your required action?

A. Notify the Shift Manager immediately.

B. Observe behavior of the affected NIs.

C. Withdraw the assembly from the core.

D. Stop insertion and allow counts to stabilize.

Answer: C

Answer Explanation:

A. **Incorrect** - These are the actions per FH-305 for a sustained rising count rate, on two or more NIs, after an assembly has been inserted.

B. **Incorrect** - This is a partial action per FH-305 being taken for a single wide range NI channel that may be unreliable. Question stem states 2 of 4 channels have increased unexpectedly.

C. **Correct** - This is the proper action to take as stated in FH-305 for an unexpected increase in count rate on more than one wide range NI channel.

D. **Incorrect** - Stopping insertion is prudent but FH-305 requires that fuel assembly be withdrawn.

2012 NRC SRO EXAM MASTER KEY

Question 78 (Q26673)			
Topic:	Inadvertent dilution during Core Alts		
Tier/Group:	2/2		
K/A Info:	034 - Fuel Handling • K1 - Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling System and the following systems: • K1.04 - NIS		
SRO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine the proper location for a fuel assembly during an Inadvertent Dilution in Modes 3, 4, 5 or 6.		
10 CFR Part 55 Content:	55.43(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOR 11-6F Biennial written exam (12/12)		
Technical references:	FH-305, Core Alterations		
Comments:	None		

79. 015/17 - RCP Malfunctions (2.4.21)

Which **ONE** of the following conditions challenges the Core and RCS Heat Removal safety function during EOP-0 *and* which Optimal Recovery procedure should be entered?

A. 1-CVC-506-CV (RCP Bleed-Off Inboard Isol) fails closed due to a broken airline; EOP-1, Reactor Trip.

B. Unable to start any Component Cooling Pump due to loss of power effects; EOP-2, Loss Of Offsite Power/Loss Of Forced Circulation.

C. 11A RCP middle seal and vapor seals failed and 11A RCP was secured; EOP-5, Loss of Coolant Accident.

D. RCS pressure lowers to 1350 PSIA with containment parameters normal; EOP-6, Steam Generator Tube Rupture.

Answer: B

Answer Explanation:

A. **Incorrect** - Inboard RCP bleed-off isolation failing closed. No requirement to secure ALL RCPs as bleed-off RV lifts in containment to maintain a flowpath with RCPs operating. To enter EOP-1, ALL safety functions are complete (met). Core and RCS Heat Removal would be met as at least one RCP is operating in a loop with a S/G available.

B. **Correct** - Per EOP-0 Vital Auxiliaries if unable to start a CCW pump all RCPs must be secured. Core and RCS Heat Removal requires at least one RCP operating in a loop with a S/G available for heat removal and NO RCPs would be operating.

C. **Incorrect** - Two RCPs are secured due to trip strategy in EOP-0 but Core and RCS Heat Removal per EOP-0 is complete (met) as at least one RCP is operating in a loop with a S/G available. A loss of the vapor seal results in an RCS leak to the containment.

D. **Incorrect** - Core and RCS Heat Removal would be met as at least one RCP is operating in a loop with a S/G available. Two RCPs would be tripped based on SIAS actuation. HPSI Pumps are not injecting flow into the RCS so a cooldown is not occurring at this pressure value.

2012 NRC SRO EXAM MASTER KEY

Question 79 (Q97048)			
Topic:	RCP Malfunctions		
Tier/Group:	1/1		
K/A Info:	015/017 - RCP Malfunctions • 2.4 - Emergency Procedures / Plan • 2.4.21 - 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.		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given plant conditions, assess the status of Core and RCS Heat Removal safety function.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	EOP-0 and EOP-0 Diagnostic Flowchart 1C07-ALM, Chemical & Volume Control Alarm Manual, window F-07		
Comments:	None		

80. 038 - Steam Generator Tube Rupture (2.1.19)

A Unit-1 reactor trip occurred due to a loss of offsite power and the immediate actions of EOP-0 have been performed. The following conditions exist on SPDS:

- SIAS actuation
- Pzr pressure and level are continuing to lower
- Both S/G pressures are 850 PSIA and stable
- 11 S/G level is at (-) 110 inches and rising at 2 inches per minute
- 12 S/G level is at (-) 70 inches and rising at 5 inches per minute
- Containment pressure is 0.3 psig and stable

Which **ONE** of the following optimal recovery procedure recommendations would you make to the Shift Manager?

A. EOP-6, Steam Generator Tube Rupture

B. EOP-5, Loss of Coolant Accident

C. EOP-4, Excess Steam Demand

D. EOP-2, Loss of Offsite Power/Loss of Forced Circulation

Answer: A

A. **Correct** - Based on S/G level trends with containment pressure normal, both S/G pressures normal, and RCS pressure and level trends EOP-6 is appropriate procedure to recommend.

B. **Incorrect** - Although SIAS has actuated, the Containment pressure is normal and S/G levels are mismatched and trends are different which when diagnosed EOP-5 would not be recommended.

C. **Incorrect** - Although SIAS has actuated, S/G pressures are at 850 PSIA and stable. Upon the trip, the MSIVs were shut due to loss of power to 2nd Stage MSR source MOVs resulting in a slight cooldown. Based on these trends EOP-4 would not be recommended.

D. **Incorrect** – Although a loss of offsite power did occur, there are also indications to support a SGTR and EOP-6 addresses a SGTR coincident with a loss of offsite power. RCS pressure and level trends along with S/G level trends support a SGTR is occurring. EOP-2 would not be recommended.

Question 80 (Q26058)			
Topic:	Assessment of SGTR using SPDS		
Tier/Group:	1/1		
K/A Info:	038 –Steam Generator Tube Rupture 2.1 - Conduct of Operations • 2.1.19 - Ability to use plant computers to evaluate system or component status.		
SRO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	Using SPDS assess EOP-0 Safety Function status and using the EOP-0 Diagnostic flowchart determine applicable EOP to enter.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2008 Plant Computer, SPDS (01/09)		
Technical references:	EOP-0 Safety Function Status Checks and Diagnostic Flowchart		
Comments:	None		

81. 057 - Loss of Vital AC Inst. Bus (AA2.03)

Given the following conditions on Unit 2:

- Reactor power is 100%.
- A loss of Instrument Bus 2Y02 has occurred.

(1) Which **ONE** of the following component responses is observed and
(2) What actions would you direct as the Unit CRS?

A. (1) ONLY Two (2) TCBs open;
(2) Refer to alarm manual to determine cause and required corrective actions.

B. (1) Two trip paths de-energize resulting in a reactor trip;
(2) Implement EOP-0, Post-Trip Immediate Actions.

C. (1) ONLY Four (4) TCBs open and RPS Channel B is deenergized;
(2) De-energize RPS Channel B in preparation for power restoration.

D. (1) ESFAS Actuation Logic Cabinet BL and Sensor Cabinet ZD deenergize;
(2) De-energize Actuation Logic Cabinet BL and Sensor Cabinet ZD for power restoration.

Answer: C

Answer Explanation:

A. **Incorrect** - Loss of a single 120V AC Vital instrument bus opens 4 TCBs.

B. **Incorrect** - Loss of a single 120V AC Vital instrument bus opens 4 TCBs (two trip paths de-energize) but does not trip the reactor. Sometimes, two trip paths deenergizing will result in a reactor trip.

C. **Correct** - This is response observed in the control room. Alarm manual would be referenced as part of crew response directing them to AOP-7J which provides direction for de-energizing the RPS channel.

D. **Incorrect** – Logic cabinet referenced is correct. Sensor cabinet ZD is powered from 2Y01.

2012 NRC SRO EXAM MASTER KEY

Question 81 (Q97052)			
Topic:	Loss of Vital AC Inst. Bus effect to RPS		
Tier/Group:	1/1		
K/A Info:	057 - Loss of Vital AC Inst. Bus • AA2 - Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: • AA2.03 - RPS panel alarm annunciators and trip indicators		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the expected response of RPS upon a loss of a 120V Vital AC Instrument Bus with respect to final condition of Trip Path Relays and TCBs.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	None		
Technical references:	AOP-7J-2, Loss Of 120 Volt Vital AC or 125 Volt Vital DC Power		
Comments:	Modified from Q20182		

82. 054 - Loss of Main Feedwater (AA2.03)

Unit-1 was operating at 100% power when a plant transient caused a reactor trip. EOP-0, Post-Trip Immediate Actions, was implemented and the following conditions were observed:

- CEA #1indicates fully withdrawn
- Amber lights are energized for all other CEAs except CEA # 52, whose green light is energized
- 11 S/G Pressure at 920 PSIA
- 12 S/G Pressure at 800 PSIA
- 11 S/G level at (-)115 inches
- 12 S/G level at (-)165 inches
- Condenser vacuum at 19.5 inches Hg
- Containment pressure at 0.8 PSIG
- No automatic safety system actuations have occurred

Which action is the **first** required by the operating crew (assuming standard safety function hierarchy is used) and which EOP-0, Post-Trip Immediate Actions, block step would direct this action?

A. Shut the MSIVs as directed by "Ensure Turbine Trip".

B. Borate the RCS to 2300 PPM as directed by "Verify the Reactivity Control Safety Function is Satisfied".

C. Start an AFW Pump as directed by "Verify the Core and RCS Heat Removal Safety function is satisfied".

D. Place all Containment Air Coolers (CACs) in pull-to-low and open the Emergency Outlet valves for the operating CACs as directed by "Verify the Containment Environment Safety Function is Satisfied".

Answer: C

Answer Explanation:

A. **Incorrect** - "Verify the Core and RCS Heat Removal Safety function is Satisfied" provides direction to shut the MSIVs should S/G pressure drop to 800 PSIA. "Ensure Turbine Trip" does provide guidance to shut the MSIVs, but the guidance is based on turbine valve failures, turbine speed and loss of power effects.

B. **Incorrect** - Boration of the RCS is required only if "more than one CEA is not fully inserted". The EOP-0 basis document states "A CEA is considered fully inserted if the rod drop light (amber) or the lower electrical limit light (green) is energized.

C. **Correct** - Main Feedwater flow has been lost due to the SGFPs tripping on low condenser vacuum and is directed by "Verify the Core and RCS Heat Removal Safety function is satisfied".

D. **Incorrect** - The "Verify the Containment Environment Safety Function is Satisfied" does not direct placing the CACs in pull-to low.

Question 82 (Q97050)			
Topic:	EOP-0, Post-Trip Immediate Actions, hierarchy to initiate AFW		
Tier/Group:	1/1		
K/A Info:	054 - Loss of Main Feedwater • AA2 - Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): • AA2.03 - Conditions and reasons for AFW pump startup		
SRO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	EOP-0, Post-Trip Immediate Actions		
Comments:	None		

83. CE/E02 - Reactor Trip Recovery (2.2.42)

Following a plant trip from 100% power, Pressurizer (Pzr) level lowered to 90 inches before recovering. The crew implemented EOP-1, Reactor Trip, as ALL safety functions were met.

The Pzr level final acceptance criteria in EOP-1, Reactor Trip, has a band of 130 to 180 inches and trending to 160 inches.

Which **ONE** of the following choices below is (1) the basis for this band and (2) an administrative post-trip action requirement?

A. Allows some tolerance from the normal band assuming a standard reactor trip with charging and letdown isolated;
Entry into the T. S. LCO for the Pzr being inoperable because two emergency banks of Pzr heaters deenergized when Pzr level fell below 101 inches.

B. Actual level outside this band means it is challenging the Pressure and Inventory Control safety function;
Entry into the T. S. LCO for the Pzr being inoperable because Pzr level fell below the minimum operating band, following the trip.

C. Allows some tolerance from the normal band assuming a standard reactor trip with charging and letdown remaining in service;
Recording backup Charging Pump(s) start/stop times per EN-1-115, Recording of Plant Transients/Operational Cycles.

D. Ensures that pressurizer heaters remain covered and allows a band of (+) or (-) 25 inches from programmed pressurizer level;
Recording occurrence of the reactor trip per EN-1-115, Recording of Plant Transients/Operational Cycles.

Answer: B

Answer Explanation:

A. **Incorrect** – Isolation of charging and letdown in EOP-1, indicate something more than a standard reactor trip has occurred; Although the heaters are deenergized when level is below 101 inches (an interlock), the LCO for Pzr being inoperable based on emergency heaters is not entered as power remained to emergency heater banks defined in tech specs during this event.

B. **Correct** - This is per EOP-1 basis Step IV.D; EOP Att. 13 states ensure any LCOs that have NOT been met during the event are entered AND all appropriate log entries have been made. Pzr level went below minimum operating level of 133 inches per LCO 3.4.9 and this entry is required.

C. **Incorrect** - Actual level outside this band means it is challenging the Pressure and Inventory Control safety function. The backup Charging pump(s) will operate to return Pzr level to the specified band. Although charging pumps start and stop, no entries per EN-1-115 are required since charging was never lost based on stem statement that all safety functions were met.

D. **Incorrect** - Programmed level at 0% power is 160 inches, so high limit is only (+) 20 inches but lower limit is (-) 30 inches from program. The reactor trip transient log entry is required per EN-1-115.

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Question 83 (Q97051)			
Topic:	Basis for Pzr level in EOP-1, Reactor Trip and admin Post-trip actions		
Tier/Group:	1/1		
K/A Info:	CE E02 - Reactor Trip Recovery • 2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	Tech Spec LCO 3.4.9 Pressurizer EOP-1, Reactor Trip and Technical Bases EOP Att. 13, Administrative Post-Trip Actions EN-1-115, Recording Of Plant Transients/Operational Cycles		
Comments:	None		

84. CE/E05 - Excess Steam Demand (EA2.1)

With Unit-1 at 100% power, TBV-3942 failed open resulting in a reactor trip. EOP-0, Post-Trip Immediate Actions, was implemented and alternate actions taken as required due to a fault on 14 4KV Bus.

Given the following parameters in EOP-0:

- RCS Boration in progress due to loss of power effects (LOPE)
- 11 4KV Bus is energized from offsite
- All 125VDC bus voltages indicate 124 VDC
- Radiation Levels External to Containment (RLEC) alternate actions were taken due to loss of power effects
- PRZR pressure is 1950 PSIA and slowly lowering
- PRZR level is 70 inches and slowly lowering
- T_{COLD} is 516°F and slowly lowering
- 11 S/G pressure is 780 PSIA and continues to lower
- 12 S/G pressure is 880 PSIA and slowly rising
- 13 AFW pump is operating to restore S/G levels
- 11 S/G level is minus (-) 150 inches and lowering
- 12 S/G level is minus (-) 110 inches and rising
- Containment pressure is 1.5 PSIG and rising
- Containment temperature is 140°F and rising
- Containment RMS is unchanged

Which **ONE** of the following will be implemented based on plant parameters and conditions?

A. EOP-8, Functional Recovery Procedure

B. EOP-6, Steam Generator Tube Rupture.

C. EOP-5, Loss of Coolant Accident

D. EOP-4, Excess Steam Demand Event

Answer: D

Answer Explanation:

A. **Incorrect** - Based on T_{COLD} and S/G pressure/levels lowering an ESDE is occurring. There is only one event occurring so EOP-8 is not required to be entered. Plausible based on multiple degraded parameters.

B. **Incorrect** - Based on T_{COLD} and S/G pressure/levels lowering an ESDE is occurring. A SGTR can be eliminated based on S/G level and pressure responses. Plausible based on Pzr pressure and rising SG level.

C. **Incorrect** - Based on T_{COLD} and S/G pressure/levels lowering an ESDE is occurring. A LOCA can be eliminated based on containment RMS response. Plausible based on Pzr pressure and level and containment parameters.

D. **Correct** - Based on T_{COLD} and S/G pressures lowering an ESDE is occurring. LOCA and SGTR can be eliminated based on S/G level and pressure responses.

Question 84 (Q26689)			
Topic:	EOP-4 Excess Steam Demand		
Tier/Group:	1/1		
K/A Info:	CE/E05 Excess Steam Demand • EA2 - Ability to determine and interpret the following as they apply to the (Excess Steam Demand) • EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations		
SRO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given plant conditions and/or parameters, determine which optimal recovery procedure is the correct one for the condition/parameters given.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2008 AOP/EOP exam (04/10)		
Technical references:	EOP-0 Diagnostic Flowchart and EOP-4, Excess Steam Demand Event entry conditions		
Comments:	None		

85. 2.1 - Conduct of Operations (2.1.20)

Unit-2 is operating at 60% power when a loss of 4KV Bus 22 occurs.

(1) What effect does this condition have on plant operation?
(2) What is the correct action to address this condition?

A. (1) Loss of 22 and 23 Condensate Pumps;
(2) Commence a Rapid Power Reduction, to lower Condensate Header flow to less than 8000 GPM.

B. (1) Loss of lube oil to both SGFPs;
(2) Trip the Reactor, implement EOP-0.

C. (1) Loss of 21 and 22 Condensate Booster Pumps;
(2) Trip the Reactor, implement EOP-0.

D. (1) Loss of 21 and 22 Condensate Booster Pumps;
(2) Commence a Rapid Power Reduction, to lower Condensate Header flow to less than 8500 GPM.

Answer: D

Answer Justifications:

A. **Incorrect** - 22 and 23 Condensate Pps are powered from 4KV Bus 23 and remain in operation. Stated actions would be correct for a loss of 4KV Bus 23.

B. **Incorrect** - Each SGFP has an Oil Pp powered from MCC-206 and one powered from MCC-216; therefore lube oil will not be lost with a loss of MCC-206 (22 4KV bus).

C. **Incorrect** - The listed loads are in fact lost. Tripping the Reactor and implementation of EOP-0 would be correct actions if Reactor power were greater than 70%.

D. **Correct** - 21 and 22 Condensate Booster Pps are lost necessitating a power reduction to get Condensate Header flow to less than the capacity of a single Condensate Booster Pp.

2012 NRC SRO EXAM MASTER KEY

Question 85 (Q97053)			
Topic:	Loss of 22 4KV Bus effects		
Tier/Group:	3		
K/A Info:	2.1 – Conduct of Operations 2.1.20 - Ability to interpret and execute procedure steps.		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	AOP-7I-2, Loss Of 4kv, 480 Volt Or 208/120 Volt Instrument Bus Power		
Comments:	None		

86. 024 - Emergency Boration (AA2.05)

Unit-2 enters AOP-2A due to an RCS leak which required a reactor trip.

Given the following post-trip conditions:

- Prior to the trip, RCS boron was 813 PPM
- Both BAST concentrations are 7.25%
- CEAs 38 and 46 are stuck at 120 inches withdrawn
- The Pressurizer emptied in EOP-0
- SIAS was verified in EOP-0
- RCS pressure is 1380 PSIA and continuing to lower
- The Crew transitioned to the appropriate Optimal Recovery Procedure fifteen (15) minutes after entering EOP-0

Fifteen (15) minutes after requested, Plant Chemistry reports the RCS boron sample result is 1100 ppm.

Which **ONE** of the following represents: (1) The status of the boron concentration for Shutdown Margin (SDM) and (2) The required action for the existing plant conditions?

A. (1) Present boron concentration meets required SDM;
(2) Align Charging pump suction to the RWT.

B. (1) Present boron concentration is below required SDM;
(2) Borate until BAST volume or Charging Pp run time requirement is met.

C. (1) Present boron concentration is below required SDM;
(2) Borate until SDM requirement is met.

D. (1) Present boron concentration meets required SDM;
(2) Align Charging pump suction to VCT after SIAS has been reset.

Answer: C

Answer Explanation:

A. **Incorrect** – The status of boron concentration is incorrect. However, based on question stem two CEAs are stuck out and NEOP-23 Fig. 2-II-A.5 requires a boron concentration of ≥ 2300 PPM.

B. **Incorrect** – The present boron concentration does not meet the requirement of SDM for two stuck CEAs. Using Fig. 1 of OI-2B determines gallons of boric acid needed to reach 2300 PPM are **8,812**. Applying EOP-5 requirements for BAST volume or charging pump run times adds the following:

- 134 inches X 58.8 gallons / inch (Fig. 2 of OI-2C provided) = 7879 gallons
- 60 minutes X 132 gallons/minute = 7920 gallons

Second part is plausible if examinee fails to recognize that two stuck CEAs requires ≥ 2300 PPM for SDM.

C. **Correct** - Boration during the LOCA must continue until boron concentration is ≥ 2300 PPM per NEOP-23 Fig. 2-II-A-5 for two stuck CEAs. Using Fig. 1 of OI-2B determines gallons of boric acid needed to reach 2300 PPM are **8,812**. Applying EOP-5 requirements for BAST volume or charging pump run times adds the following:

- 134 inches X 58.8 gallons / inch (Fig. 2 of OI-2C provided) = 7879 gallons
- 60 minutes X 132 gallons/minute = 7920 gallons

D. **Incorrect** - EOP-5 does not direct continue to borate until SIAS is verified and reset. There are specific criteria to meet required SDM for 2 stuck CEAs. If SIAS is verified and reset, one of the paths to realign charging pump suction to is the VCT.

2012 NRC SRO EXAM MASTER KEY

Question 86 (Q97063)			
Topic:	SDM requirement for SGTR and two stuck CEAs		
Tier/Group:	1/2		
K/A Info:	024 - Emergency Boration • AA2 - Ability to determine and interpret the following as they apply to the Emergency Boration: • AA2.05 - Amount of boron to add to achieve required SDM		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	OI-2B, Figure 1(Boration Volume (RCS Not On SDC) OI-2C, Figure 2 Boric Acid Storage Tank volume EOP-5, Step IV.H. Commence RCS Boration NEOP-23, Figures 2-II.A.1 & 2-II.A.3		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/ Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	None		
Technical references:	NEOP-23, Figs. 2-II.A.1, Soluble Boron Concentration versus burnup NEOP-23, Figs. 2-II.A.3: Shutdown Boron Concentration for All Rods In NEOP-23 Fig. 2-II.A.5: Shutdown Boron Concentration for More than One CEA Stuck EOP-5 Step H and Technical Bases		
Comments:	Modified from Q25815		

87. 028 - Pzr Level Control Malfunction (AA2.08)

Unit-1 is at 75% power with T_{AVE} at 558 °F when 12 Hot Leg RTD, TE-121X, fails high.

Reactor Regulating System (RRS) channel selector switch, 1-HS-5600, is selected to RRS-X.

Which **ONE** of the following (1) describes the impact of the instrument failure on the Pressurizer (Pzr) level control system and (2) is the direction provided to the RO?

A. (1) Pzr level setpoint increases, all Charging Pumps start, letdown flow goes to minimum;
(2) Place the appropriate (S1 or S2) switch to off on RRS channel X and Y.

B. (1) RRS channel X removes the failed TE from the Pzr level setpoint calculation;
(2) Use OI-7, Reactor Regulating System, to determine failed TE actions.

C. (1) Pzr level setpoint decreases, selected Charging Pump remains in operation, letdown flow goes to maximum;
(2) Place the appropriate (S1 or S2) switch to off on RRS channel X.

D. (1) Pzr level control shifts from Remote-Auto to Local-Auto. Charging and letdown operate based on the Local-Auto setpoint.
(2) Place RRS channel selector switch, 1-HS-5600, to RRS-Y position.

Answer: A

Answer Explanation:

A. **Correct** - The Pzr level setpoint is generated from a T_{AVE} signal between 30 and 95% power. At 75% T_{AVE} is ~558 °F. The failed TE causes T_{AVE} to fail to its maximum value. This results in the Pzr level control system sending a signal to start all charging pumps and reduce L/D to minimum. It is necessary to place the S2 switch in both RRS channels to off to remove failed TE input.

B. **Incorrect** - As stated above, Pzr level setpoint increases; OI-7 is the correct procedure to reference per the alarm manual response.

C. **Incorrect** - Setpoint does not lower and placing S1 or S2 switch to off in Channel X only does not remove failed input that still exists in channel Y.

D. **Incorrect** - Switching to Channel Y without removing the failed TE input will not return Pzr level setpoint to the proper value.

2012 NRC SRO EXAM MASTER KEY

Question 87 (Q97054)			
Topic:	Failed TE input to RRS		
Tier/Group:	1/2		
K/A Info:	028 - Pressurizer Level Control Malfunction AA2 - Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: • AA2.08 - PZR level as a function of power level		
SRO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given the following conditions, determine as an RO/CRO and/or direct as the SRO the following actions needed: a. Pzr level response to failure of TE input to RRS and actions per OI-7, Reactor Regulating System operation.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	None		
Technical references:	OI-7, Reactor Regulating System Alarm Response Manual 1C05, window D-40		
Comments:	Modified from Q14428		

88. 2.1 - Conduct of Operations (2.1.35)

U-2 is in a Refueling Outage and is currently being defueled. The Refueling Machine operator has just begun lowering a fuel assembly from the core into the upender. A freshly burned Fuel Assembly is in the Inspection Stand in the Spent Fuel Pool (SFP). The Containment Outage Door (COD) is in place and is open for equipment move in.

A large truck carrying scaffold has backed into the COD and caused damage which prevents dogging the COD shut.

Which **ONE** of the following correctly describes the required actions?

A. Place the fuel assembly in a safe location, suspend movement of irradiated fuel assemblies within the Containment, and Install the Equipment Hatch with a minimum of 4 bolts.

B. Place the fuel assembly in a safe location, suspend movement of irradiated fuel assemblies within the Containment and the SFP, and Install the Equipment Hatch with a minimum of 4 bolts.

C. Install the Equipment Hatch with at least 4 bolts, within the Time to Boil, or place the fuel assembly in a safe location and suspend movement of irradiated fuel assemblies within the Containment.

D. No actions are required if the Equipment Hatch is available to be installed in less than the Time to Boil.

Answer: A

Answer justification:

A. **Correct** - Per AOP-4A, Loss of Containment Closure and T.S. 3.9.3.

B. **Incorrect** - T.S. 3.9.3 specifies “suspend movement of irradiated fuel assemblies within containment”. These actions are not extended to Spent Fuel Pool activities with irradiated fuel.

C. **Incorrect** - The fuel assembly must be placed in a safe location and movement of irradiated fuel assemblies within containment must be suspended until the equipment hatch is installed with a minimum of a least 4 bolts.

D. **Incorrect** - The fuel assembly must be placed in a safe location and movement of irradiated fuel assemblies within containment must be suspended until the equipment hatch is installed with a minimum of a least 4 bolts.

Question 88 (Q97055)			
Topic:	Loss of Containment Integrity during Fuel Handling		
Tier/Group:	Generic Knowledge and Abilities		
K/A Info:	2.1 - Conduct of Operations • 2.1.35 - Knowledge of the fuel-handling responsibilities of SROs.		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Exam Bank History:	None		
Technical references:	AOP-4A, Loss of Containment Integrity T.S. 3.9.3, Containment Penetrations		
Comments:	None		

89. 005 - Inoperable/Stuck Control Rod (2.4.11)

Given the following conditions on Unit-1:

- The RO is performing STP-O-29, CEA Free Movement Test, when a Shutdown Group CEA cannot be withdrawn from 127.5 inches, after insertion
- Electrical Maintenance determines the CEA is mechanically stuck
- System Engineering has declared CEA untrippable

Which **ONE** of the following actions is required based on the report from Electrical Maintenance?

A. Perform a rapid shutdown per OP-3, Appendix B, Rapid Power Reduction; upon turbine trip, borate the RCS at ≥ 40 GPM of at least 2300 PPM until SDM is met.

B. Insert the remaining CEAs within the group to realign with the stuck CEA to clear the CEA Motion Inhibit (CMI) while maintaining power level.

C. If unable to realign the CEA after two hours, then trip the reactor and implement EOP-0, Post Trip Immediate Actions.

D. Shutdown and place the unit in Mode 3 within 6 hours per OP-3, Normal Power Operation.

Answer: D

Answer Explanation:

A. **Incorrect** - These are actions required per AOP-1B for two or more untrippable CEAs and subsequent boration required.

B. **Incorrect** - Examinee recognizes realigning other CEAs in group with stuck CEA will remove CEA group deviation. However, this does not clear the CMI. CMI remains in because Regulating Group CEAs are unable to move (MIRG) as Shutdown CEAs are less than 129 inches withdrawn.

C. **Incorrect** - These are the actions required when two or more CEAs are misaligned by > 15 inches within their group.

D. **Correct** - For a single untrippable CEA, AOP-1B directs the plant be placed in MODE 3 within 6 hours, per OP-3.

2012 NRC SRO EXAM MASTER KEY

Question 89 (Q97056)			
Topic:	Actions for untrippable CEA (stuck)		
Tier/Group:	1/2		
K/A Info:	005 – Inoperable/Stuck Control Rod 2.4 - Emergency Procedures / Plan • 2.4.11 - Knowledge of abnormal condition procedures.		
SRO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	AOP-1B, CEA Malfunctions		
Comments:	None		

90. 2.2 - Equipment Control (2.2.18)

Due to emergent equipment issues, the GS-Shift Operations directs a change be to the Safe Shutdown Summary Schedule (S4).

Per NO-1-103, Conduct of Lower Mode Operations, which **ONE** of the following satisfies the S4 change review requirements?

A. The Shutdown Safety Review Board.

B. The designated SRO and a second independent SRO.

C. Outage Management Outage Specialist and the GS-Shift Operations.

D. The designated SRO and the GS-Shift Operations.

Answer: B

Answer justification:

A. **Incorrect** - The Shutdown Safety Review Board (SSRB) is comprised of one SRO, one Senior Leadership team member and a member from the PRA group or Engineering.

B. **Correct** - per NO-1-103, the review must be performed by an SRO appointed (designated) by the GS-SO and a second independent SRO.

C. **Incorrect** - per NO-1-103, the review must be performed by an SRO appointed (designated) by the GS-SO and a second independent SRO. Since the GS-SO is directing the change he would not be considered an independent SRO reviewer.

D. **Incorrect** - per NO-1-103, the review must be performed by an SRO appointed (designated) by the GS-SO and a second independent SRO. Since the GS-SO is directing the change he would not be considered an independent reviewer.

Question 90 (Q97060)			
Topic:	Approval of a change to the S4		
Tier/Group:	Generic Knowledge and Abilities		
K/A Info:	2.2 - Equipment Control • 2.2.18 - Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous use		
Exam Bank History:	Last used in LOR Session quiz - 1/11		
Technical references:	NO-1-103, Conduct of Lower Mode Operations		
Comments:	None		

91. 004 - CVCS (A2.17)

Unit-2 has entered the appropriate Optimal Recovery Procedure for a Loss of Coolant Accident. The following conditions exist:

- HPSI and LPSI pumps are in Pull To Lock to meet throttling criteria
- ALL Charging pumps are operating to maintain Pressurizer level within the desired band
- 11B and 12A RCPs are operating
- A plant cooldown is in progress to reach SDC cooling initiation
- RCS Pressure is being lowered to maintain RCS subcooling low in the band
- The STA reports present RCS pressure trend will challenge continued RCP operation

Which **ONE** of the following is occurring and what is the required action to maintain RCPs operating?

A. Cooldown rate is too excessive;
Adjust the ADVs, to reduce the cooldown rate, which will raise subcooling.

B. Aux Spray is in use;
Secure Aux Spray by reopening charging header isolations and shut the Aux Spray isolation.

C. Aux Feedwater feed rate is excessive;
Reduce feed rate to S/Gs to lower cooldown rate and stabilize RCS pressure.

D. Aux Spray is in use;
Secure all but one charging pump to reduce RCS depressurization.

Answer: B

Answer Explanation:

A. **Incorrect** - Cooldown rate is not too excessive as Pzr level is being maintained with all charging pumps running. Shutting ADVs allows RCS to heatup resulting in RCS subcooling becoming even smaller and further challenge continued RCP operation.

B. **Correct** - This is why RCS subcooling is lowering as RCS pressure is lowered. Reopening charging header stops and shutting Aux Spray isolation will stop subcooling from continuing to lower and maintain RCP operation.

C. **Incorrect** - Lowering AFW feed rate will cause RCS to heatup as this is primary method of heat removal since HPSI pumps are secured. This will further challenge subcooling limit for continued RCP operation.

D. **Incorrect** - This is why RCS pressure is lowering. Securing 2 of 3 charging pumps will slow depressurization however subcooling will continue to be lowered and pressurizer level will begin to lower as HPSI pumps are secured per throttling criteria. Information provided in question stem states all 3 charging pumps operating are maintaining Pzr level with cooldown in progress.

2012 NRC SRO EXAM MASTER KEY

Question 91 (Q97057)			
Topic:	Actions required to recover Pzr level in EOP-5		
Tier/Group:	2/1		
K/A Info:	004 - CVCS • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.17 - Low PZR pressure		
SRO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given RCS parameters, identify the appropriate response for Loss of Coolant Accident (LOCA) per EOP-5.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	EOP-5, Loss of Coolant Accident Step J		
Comments:	None		

92. 006 - ECCS (A2.05)

Unit -1 is at 100% power when the following control room alarm annunciates:

EAST ECCS PP RM LVL HI

- The ABO looking through the Component Cooling Room access hatch observes a significant amount of water in the room and continuing to rise
- The CRO observes 11 Refueling Water Tank (RWT) level is 462 inches and lowering

Which **ONE** of the following groups represents ALL affected components and what directions should be provided to the crew?

A. 12 and 13 HPSI, 12 LPSI, and 12 Containment Spray Pump;
Shut the RWT outlet MOV on "B" train ECCS header, place 13 HPSI, 12 LPSI, and 12 Containment Spray Pumps in Pull To Lock.

B. 11 HPSI, 11 LPSI, and 11 Containment Spray Pump;
Shut the RWT outlet MOV on "A" train ECCS header, place 11 HPSI, 11 LPSI, and 11 Containment Spray Pumps in Pull To Lock.

C. 11 and 12 HPSI pumps, 11 LPSI, and 11 Containment Spray Pump;
Shut the RWT outlet MOV on "A" train ECCS header, place 11 HPSI, 11 LPSI, and 11 Containment Spray Pumps in Pull To Lock.

D. 13 HPSI, 12 LPSI, and 12 Containment Spray Pumps;
Shut the RWT outlet MOV on "B" train ECCS header, place 13 HPSI, 12 LPSI, and 12 Containment Spray Pumps in Pull To Lock.

Answer: C

Answer Explanation:

A. **Incorrect** – 12 HPSI Pump is located in the affected room for the alarm provided but the other components are located in the West ECCS room. Actions are for the "B" train but the leak is on the "A" train header.

B. **Incorrect** – Components provided are located in room with leak occurring, however, 12 HPSI pump is also affected. Actions provided are correct to address the leak as OI-3A requires 12 HPSI Pp handswitch in Pull to Lock.

C. **Correct** – All components provided are located in room with leak occurring. Actions provided are correct to address the leak as OI-3A has 12 HPSI Pp handswitch always in Pull to Lock.

D. **Incorrect** – These components are located in the West ECCS pump room and are not affected. These are actions to isolate the "B" train ECCS components. The leak is on "A" train ECCS header.

2012 NRC SRO EXAM MASTER KEY

Question 92 Info (Q97058)			
Topic:	ECCS header rupture		
Tier/Group:	2/1		
K/A Info:	006 - ECCS • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.11 - Rupture of ECCS header.		
SRO Importance:	4.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	Alarm Response Manual 1C10 window J-14 OI-3A, Safety Injection and Containment Spray		
Comments:	None		

93. 059 - Main Feedwater (2.4.4)

Unit-2 has just completed a startup following a refueling outage. Given the following events and conditions:

- Reactor power is 4%
- 22 SGFP is out of service for emergent repairs

21 SGFP is operating on Main Steam when the following occurs:

- 21 SGFP speed lowers from 3300 rpm to 1200 RPM
- 21 and 22 S/G levels lower to minus (-) 20 inches and continue slowly lowering

Which **ONE** of the following statements (1) correctly describes your direction to the operators to restore S/G water levels and (2) when a reactor trip would be ordered?

A. (1) Reduce power to less than 1%, initiate AFW flow to S/Gs and allow S/G levels to slowly recover while maintaining T_{COLD} within 2°F of program;
(2) Trip the reactor if S/G levels are approaching minus (-) 40 inches

B. (1) Immediately align the Auxiliary Steam supply and slowly restore S/G water levels. Withdraw CEAs to maintain T_{COLD} above 515 °F;
(2) Trip the reactor if T_{COLD} lowers to 515 °F.

C. (1) Immediately align the Auxiliary Steam supply and maximize feedwater flow to restore S/G water levels;
(2) Trip the reactor if S/G levels are approaching minus (-) 40 inches.

D. (1) Reduce power to less than 1% and maximize AFW flow to S/Gs to restore S/G levels. Withdraw CEAs to maintain T_{COLD} above 515 °F;
(2) Trip the reactor if T_{COLD} lowers to 515 °F.

Answer: A

Answer Explanation:

A. **Correct** - This is the correct sequence of actions required by AOP-3G which would be implemented based on conditions listed in question stem.

B. **Incorrect** - Per OI-12A, this is a controlled evolution and will take several minutes between each adjustment of the Aux Steam Supply valve. Doing this would allow S/G levels to continue lowering and reach trip criteria. Withdrawing CEAs is not one of the methods provided to control RCS temperature but it will raise reactor power and may cause a plant trip on high power. MTC is very low at BOL and the effects of withdrawing CEAs will be to raise power substantially while raising T_{COLD} relatively slowly. If the examinees are not familiar with the 1995 LER for S/G overfeed event, these are the actions that were taken during that event complicating crew response resulting in an automatic reactor trip.

C. **Incorrect** - Promptly shifting back to the auxiliary steam supply will overspeed the SGFP and overfeed the S/G causing T_{COLD} to lower. Per OI-12A, this is a controlled evolution and will take several minutes between each adjustment of the Aux Steam Supply valve. Partially correct as AOP-3G requires tripping the reactor if SG level approaches -40 inches

D. **Incorrect** - Withdrawing CEAs is not one of the methods provided to **control** RCS temperature but it will raise reactor power and may cause a plant trip on high power. 515 °F is the minimum temperature for critical operations.

Question 93 Info (Q49870)			
Topic:	Shifting SGFP steam supplies at low power		
Tier/Group:	2/1		
K/A Info:	059 - Main Feedwater 2.4 - Emergency Procedures / Plan • 2.4.4 - Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.		
SRO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	2006 SRO (08/06)		
Exam Bank History:	LOI-2010 1C03 Exam (08/11)		
Technical references:	AOP-3G, Main Feedwater Malfunctions, page 23, 30, 31, and 33		
Comments:	Updated to reflect current procedure actions		

94. 063 - DC Electrical Distribution (2.2.42)

Given the following conditions on Unit-1 at 100%:

- The following alarms are received on control panel 1C34:
  - Window U-13: 11, 12 125V DC BUS U/V
  - Window U-15: 11, 12, 23, 24 125V BATT CHGR FAILURE
- DC Bus 11 voltage indication on panel 1C24 is 122 VDC and lowering slowly

Which **ONE** of the following describes (1) the failure that has occurred, and (2) the operability of DC Bus 11 in accordance with Technical Specifications?

A. (1) 11 and 24 Battery Chargers have failed;
(2) DC Bus operability will be restored when bus voltage is restored to > 125 VDC by BOTH Battery Chargers being restored to the bus.

B. (1) 12 and 23 Battery Chargers have failed;
(2) DC Bus operability will be restored when bus voltage is restored to > 125 VDC by BOTH Battery Chargers being restored to the bus.

C. (1) 11 and 23 Battery Chargers have failed;
(2) DC Bus operability will be restored when bus voltage is restored to > 125 VDC by EITHER Battery Charger.

D. (1) 12 and 24 Battery Chargers have failed;
(2) DC Bus operability will be restored when bus voltage is restored to > 125 VDC by EITHER Battery Charger.

Answer: C

Answer Explanation:

A. **Incorrect** – Only battery charger 11 normally supplies the Bus. Operability, per T.S. 3.8.4, requires a single battery charger on the Bus

B. **Incorrect** - 12 Charger does not supply DC Bus 11. Operability, per T.S. 3.8.4, requires a single battery charger on the Bus

C. **Correct** - Listed battery chargers are those that normally supply the Bus. T.S. 3.8.4 will be met when either battery charger is restored to the DC bus and voltage is > 125 VDC.

D. **Incorrect** – Neither battery charger supplies DC Bus 11. DC Bus operability will not be restored given battery chargers 12 and 24.

Question 94 Info (Q54964)			
Topic:	Battery Chargers inoperability on DC bus		
Tier/Group:	2/1		
K/A Info:	063 – DC Electrical Distribution • 2.2 - Equipment Control • 2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given plant conditions, determine if 125 VDC busses are operable per appropriate tech specs.		
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2006 Audit Remediation (11/08)		
Technical references:	Tech Spec 3.8.4 – DC Sources, Operating 1C34-ALM, HVAC Systems Control windows U-13 and U-15		
Comments:	None		

95. 029 - Containment Purge (2.2.37)

Given the following:

- Core Alts are in progress
- The Containment Purge system is in operation
- RI-5316A (Containment Area Monitor) exhibited erratic operation and the ESFAS Sensor Channel ZD CRS Sensor module was pulled to comply with Tech Specs

Currently, which **ONE** of the following explains the effect of the Out Of Service Containment Area Monitor on (1) CRS/Containment Purge operation, and (2) fuel handling?

A. (1) CRS actuation logic is reduced to 1 out of 3 logic and Containment Purge may remain in operation;
(2) Fuel handling may continue.

B. (1) ALL CRS sensor channels must be operable, therefore, immediately secure Containment Purge;
(2) Immediately suspend fuel handling within containment.

C. (1) CRS actuation is reduced to 2 out of 3 logic and Containment Purge may remain in operation;
(2) Fuel handling may continue.

D. (1) CRS actuation requires a 2 out of 4 logic, therefore, immediately secure Containment Purge;
(2) Immediately suspend fuel handling within containment.

Answer: A

Answer Explanation:

A. **Correct** - Since channel removed from service (i.e. tripped), CRS requires 1 of remaining 3 channels to trip and actuate to secure Containment Purge. Fuel handling may continue in this case.

B. **Incorrect** - This is true, however, the tech spec actions allow continued operation of Containment Purge; second part would only occur if unable to place channel in trip within 4 hours.

C. **Incorrect** - Examinee may forget fact that first RMS channel is tripped requiring only one more channel to trip to actuate CRS and secure Containment Purge.

D. **Incorrect** - The effect of the OOS sensor has provided one of the required 2 out of 4 trip logic to actuate CRS. Containment Purge remains in operation and fuel handling continues within containment.

2012 NRC SRO EXAM MASTER KEY

Question 95 Info (Q97059)			
Topic:	RMS channel OOS for Containment Radiation Signal		
Tier/Group:	2/2		
K/A Info:	029 - Containment Purge • 2.2 - Equipment Control • 2.2.37 - Ability to determine operability and/or availability of safety related equipment.		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	None		
Technical references:	Tech Spec 3.3.7 – Containment Radiation Signal		
Comments:	Modified from Q50710		

96. 041 - Steam Dump/Turbine Bypass Control (A2.03)

Unit-1 was operating at 100% power when a loss of instrument air occurred. Given the following events and conditions:

- The operators enter AOP-7D, Loss of Instrument Air
- Instrument air pressure is 50 PSIG and lowering at a rapid and continuous rate

Which statement correctly describes (1) the effect on the plant (2) direction provided to the crew and (3) required action(s) to control RCS temperature?

A. (1) The TBVs will not quick-open below 40 PSIG;
(2) Trip the reactor at 40 PSIG and lowering;
(3) Operate the TBVs in manual to control RCS temperature

B. (1) The FRVs will fail as-is at 40 PSIG;
(2) Trip the reactor at 40 PSIG and lowering;
(3) Operate the Steam Driven Auxiliary Feedwater Train to control S/G/ level

C. (1) The TBVs will not quick-open below 50 PSIG;
(2) Trip the reactor at 50 PSIG and lowering;
(3) Operate the ADVs, in manual, as required to control RCS temperature

D. (1) The FRVs will fail-as-is at 50 PSIG;
(2) Trip the reactor at 50 PSIG and lowering;
(3) Operate the Motor Driven Auxiliary Feedwater Train to control S/G/ level

Answer: C

Answer Explanation:

A. **Incorrect** - The TBV's will not quick open below 40 PSIG and AOP-7D specifies a reactor trip at 50 PSIG I/A Header pressure and the TBVs are unavailable

B. **Incorrect** - AOP-7D specifies a reactor trip at 50 PSIG I/A Header pressure

C. **Correct** – AOP-7D initial actions are to start the Saltwater Air Compressors (SWACs) which provide air to the ADVs. The 50 PSIG trip value was chosen to enable FRVs and TBVs post-trip response. The TBVs are able to quick open fully at 50 PSIG. The FRVs ramp shut, removing the immediate need to trip the SGFPs due to overfeeding effects on the RCS and provide opportunity to maintain normal heat removal methods as long as possible.

D. **Incorrect** - The FRVs fails as-is at 40 PSIG

Question 96 Info (Q50779)			
Topic:	Loss of I/A effects on the TBVs		
Tier/Group:	2/2		
K/A Info:	041 - Steam Dump/Turbine Bypass Control • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: • A2.03 - Loss of IAS		
SRO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine the Operator actions for a loss of Instrument Air in the following situations: Modes 1 and 2		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	None		
Technical references:	AOP-7D, Loss of Instrument Air EOP-0, Post-Trip Immediate Actions		
Comments:	None		

97. 2.4 - Emergency Procedures/Plan (2.4.1)

Given the following conditions on Unit 1:

- Reactor Startup in progress
- Reactor power is 10%
- The following alarm is received in the control room:
  - CNDSR EXH HOOD TEMP HI VAC LO
- The crew enters AOP-7G, Loss of Condenser Vacuum
- All Condenser Air Removal Units are verified running
- Condenser vacuum indicates 23.5 inches Hg and is lowering Rapidly

Which **ONE** of the following actions should be directed?

A. Trip the reactor and implement EOP-0, Post Trip Immediate Actions.

B. Insert CEAs to reduce reactor power to less than 1%.

C. Trip the Turbine and implement EOP-0, Post Trip Immediate Actions.

D. Initiate RCS boration to reduce reactor power to less than 1%.

Answer: A

Answer Explanation:

A. **Correct** - Per AOP-7G, Loss of Condenser Vacuum, requirements, Condenser vacuum has reached the low vacuum trip setpoint of 23.5 inches Hg, requiring a reactor trip and implementation of EOP-0, Post-Trip Immediate Actions

B. **Incorrect** - This step from the AOP is applicable for an initial power level of < 5%.

C. **Incorrect** – Question stem does not indicate the Main Turbine is paralleled to the grid or being warmed up. If vacuum reaches 22.5 inches Hg, the turbine trip automatically and the reactor will not trip automatically as the Loss of Load trip is disabled < 14% power.

D. **Incorrect** - This step from the AOP is applicable for an initial power level of < 5%.

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Question 97 Info (Q54991)			
Topic:	Actions for a loss of Condenser Vacuum		
Tier/Group:	Generic		
K/A Info:	2.4 - Emergency Procedures/Plan • 2.4.1 - Knowledge of EOP entry conditions and immediate action steps.		
SRO Importance:	4.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given a loss of condenser vacuum and/or plant conditions and parameters, determine the correct operator response(s).		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2006 Audit Exam		
Technical references:	AOP-7G, Loss of Condenser Vacuum		
Comments:	None		

98. 2.3 - Radiation Control (2.3.11)

Unit-2 is at 90% power. Given the following events and conditions:

- RCS activity is at normal values
- A 30 GPD tube leak develops in 22 S/G

Which **ONE** of the following statements correctly describes the response of (1) 2-RIC-5422A (22 MAIN STM N-16 RAD MON) and 2-RIC-5422 (22 MAIN STM EFFL RAD MON), and (2) Required action?

A. (1) 2-RIC-5422A and 2-RIC-5422 show no increase;
(2) Current leak rate does not meet any AOP entry criteria, continue to monitor.

B. (1) 2-RIC-5422A shows observable increase and 2-RIC-5422 shows no increase;
(2) Implement AOP-2A, Excess RCS Leakage.

C. (1) 2-RIC-5422A and 2-RIC-5422 show observable increase;
(2) Place the unit in Hot Standby within 6 hours and Cold Shutdown within 30 hours.

D. (1) 2-RIC-5422A and 2-RIC-5422 show observable increase;
(2) Implement AOP-10, Abnormal Secondary Chemistry Conditions.

Answer: D

Answer Explanation:

A. **Incorrect** - Above 50% power, 2-RIC-5422A (N-16 gamma monitor) and 2-RIC-5422 (Main Steam Effluent rad monitor) will be in service and see an increase. Each are able to detect a 5 GPD tube leak at normal operating temperature. 5 GPD through any one S/G is criteria for entering AOP-10.

B. **Incorrect** - Above 50% power, 2-RIC-5422A (N-16 gamma monitor) and 2-RIC-5422 (Main Steam Effluent rad monitor) will be in service and see an increase. Each are able to detect a 5 GPD tube leak at normal operating temperature. At this point leak rate is not exceeding any Tech Spec limits so placing plant in Hot Standby and subsequently Cold Shutdown is not warranted.

C. **Incorrect** - With power level above 50%, 2-RIC-5422A (N-16 gamma monitor) and 2-RIC-5422 (Main Steam Effluent rad monitor) will be in service and see an increase for this RCS leak. Each can detect a 5 GPD tube leak at normal operating temperature. Entry into AOP-2A is required when S/G leakage reaches 50 GPD through any one S/G.

D. **Correct** - Both these monitors see an observable increase based on this 30 GPD tube leak and power level above 50%. 5 GPD through any one S/G is criteria for entering AOP-10 and continuing to monitor per Att. 2.

Question 98 Info (Q91996)			
Topic:	Main Steam Line RMS response based on Rx Power		
Tier/Group:	3		
K/A Info:	2.3 – Radiation Control • 2.3.11 – Ability to control radiation releases		
SRO Importance:	4.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	Identify the Radiation Monitors that have a control interface with another system and State their control functions.		
10 CFR Part 55 Content:	55.43(b)(4)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	Remediation LOI 2010 Panel Comp (01/12)		
Technical references:	AOP-10, Abnormal Secondary Chemistry Conditions and bases		
Comments:	None		

99. 003 – Reactor Coolant Pump System (A2.01)

Given the following 22B RCP seal parameters at 100% power:

• Middle seal pressure	2000 PSIA
• Upper seal pressure	130 PSIA
• VCT pressure	40 PSIA
• Controlled Bleedoff pressure	52 PSIA
• Lower seal Temperature	195°F
• Controlled Bleedoff flow	2.7 GPM

(1) Which of the following describes the impact on plant operation and

(2) What direction will you provide the crew?

A. (1) Increased monitoring of 22B RCP seal parameters;
(2) Direct the OWC to immediately contact the system engineer to provide evaluation of continued operability.

B. (1) Commence an expeditious plant shutdown;
(2) Cooldown the RCS to less than 350° F, then secure 22B RCP.

C. (1) Immediately trip the reactor, verify reactivity control safety function;
(2) Secure 22B RCP based on Controlled Bleedoff flow exceeding 2.6 GPM.

D. (1) Two RCP seals have failed, continued operation requires GS-SO permission;
(2) Direct the OWC to immediately contact the system engineer to provide evaluation of continued operability.

Answer: B

Answer Explanation:

A. **Incorrect** – Per the alarm manual these are the actions to take based on one seal failed. Parameters given indicate two seals have failed. Controlled Bleedoff flow higher than normal confirms the upper seal has failed per criteria in OI-1A with less than 300 PSID across a seal stage.

B. **Correct** – Per the alarm manual this is the action to take based on two seals failed. RCP would be secured during plant cooldown when RCS temperature is below 350° F (per OP-5 this is when the first two RCPs are secured). Controlled Bleedoff flow higher than normal confirms the upper seal has failed per criteria in OI-1A with less than 300 PSID across a seal stage.

C. **Incorrect** – Parameters given have not exceeded any reactor trip criteria. Controlled Bleedoff flow higher than normal confirms the upper seal has failed per criteria in OI-1A with less than 300 PSID across a seal stage.

D. **Incorrect** – Two seals are failed which requires RCP be shutdown per OI-1A and the alarm manual. Controlled Bleedoff flow higher than normal confirms the upper seal has failed per criteria in OI-1A with less than 300 PSID across a seal stage.

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Question 99 Info (Q14387)			
Topic:	RCP seal failure actions		
Tier/Group:	2/1		
K/A Info:	003 – Reactor Coolant Pump System A2 - 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: A2.01 - Problems with RCP seals, especially rates of seal leak-off		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine the actions required for single or multiple RCP seal failures.		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2008 1C06 & Reactor Reg (04/09)		
Technical references:	1C06 - ALM, RCS Control Alarm Manual OI-1A, Reactor Coolant System and Pump Operations		
Comments:	None		

100. 2.2 - Equipment Control (2.2.36)

Using Provided Reference(s):

Unit 1 and Unit 2 were operating at 100% power. Given the following events and conditions:

- Maintenance requested to take the 1A Diesel Generator (DG) out of service for surveillance.
- OI-49 (Operability Verification) was performed on Unit 1 ZB train equipment.
- All other DGs and offsite power sources were verified to be operable.

Which **ONE** of the following statements correctly and completely describes the impact of this maintenance on the status of 11 HPSI Pump?

A. 11 HPSI pump is considered operable while the 1A DG is out of service regardless of the status of the remaining HPSI pumps.

B. 11 HPSI pump is considered NOT operable while the 1A DG is out of service regardless of the status of the remaining HPSI pumps.

C. 11 HPSI pump is considered operable while the 1A DG is out of service unless both the 12 and 13 HPSI pumps are declared to be inoperable.

D. 11 HPSI pump is considered operable while the 1A DG is out of service unless the 13 HPSI pump is declared to be inoperable.

Answer: D

Answer Explanation:

A. **Incorrect** - If examinee is unfamiliar with how to apply the requirements of LCO 3.8.1 action B.3 this may be selected. The 11 HPSI pump would be inoperable only if 13 HPSI pump became inoperable for these conditions.

B. **Incorrect** - If examinee is unfamiliar with how to apply the requirements of LCO 3.8.1 action B.3 this may be selected. The 11 HPSI pump would be inoperable only if 13 HPSI pump became inoperable for these conditions.

C. **Incorrect** - The 11 HPSI pump would be inoperable only if 13 HPSI pump became inoperable for these conditions. 12 HPSI pump is NOT qualified as a HPSI pump in the safety analysis because it is mechanically aligned to the 11 loop but electrically aligned to 14 4KV bus.

D. **Correct** - This is the correct interpretation of LCO 3.8.1 action B.3.

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Question 100 Info (Q50790)			
Topic:	HPSI Pp operability with DG out of service		
Tier/Group:	3		
K/A Info:	2.2 - Equipment Control • 2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.		
SRO Importance:	4.2		
Proposed references to be provided to applicant:	T.S 3.8.1		
Learning Objective:	Given a Mode of operation and a set of equipment conditions, identify applicable Technical Specifications (TS) Conditions and Technical Requirement Manual (TRM) Non-Conformances.		
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2006 (08/06)		
Exam Bank History:	LOI-2006 Recovery Exam (10/08)		
Technical references:	Tech Spec 3.8.1 Action B.3 OI-49, Operability Verification page 18		
Comments:	None		

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1. 007 - Reactor Trip-Stabilization-Recovery (EK1.04)

Shortly after a reactor trip, when reactor power indicates 10^{-3} %, a stable negative SUR is attained. Reactor power will decrease to 10^{-4}% in approximately ______________ seconds.

A. 90

B. 180

C. 360

D. 540

Answer: B

Answer Explanation:

A. **Incorrect** - Following a Rx trip, the SUR is -1/3 DPM and it will take approximately 3 minutes (180 seconds) not 90 seconds to lower power to 10E-4%.

B. **Correct** - Following a Rx trip, the SUR is -1/3 DPM and it will take approximately 3 minutes (180 seconds).

C. **Incorrect** - Following a Rx trip, the SUR is -1/3 DPM and it will take approximately 3 minutes (180 seconds) not 6 minutes (360 seconds) to lower power to 10E-4%.

D. **Incorrect** - Following a Rx trip, the SUR is -1/3 DPM and it will take approximately 3 minutes (180 seconds) not 9 minutes (540 seconds) to lower power to 10E-4%.

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Question 1 (QS50408)			
Topic:	Which RPS response is correct for a reactor trip?		
Tier/Group:	1/1		
K/A Info:	EPE - 007 Reactor Trip • EK1 - Knowledge of the operational implications of the following concepts as they apply to the reactor trip: • EK1.04 - Decrease in reactor power following reactor trip (prompt drop and subsequent decay)		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-58-1-01		
10 CFR Part 55 Content:	55.41(b)(8)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	LOR 11-6D Biennial written exam (12/11)		
Technical references:	EOP-0 Technical Bases page 12		
Comments:	None		

2. 027 - Pressurizer Pressure Control System Malfunction (AK2.03)

Given the following:

- Unit-1 is at 100% power
- RCS Pressure Control is in AUTO
- Pressurizer Backup Heaters are in AUTO
- RCS Pressure is 2250 PSIA

What is the IMMEDIATE plant response if the selected Pressurizer Pressure controller setpoint fails to 2500 PSIA?

A. Spray valve controller goes to minimum output, proportional heaters output goes to maximum, and all backup heaters energize.

B. Spray valve controller goes to minimum output, proportional heaters output goes to maximum, and all backup heaters remain off.

C. Spray valve controller goes to maximum output, proportional heaters output goes to maximum, and all backup heaters deenergize.

D. Spray valve controller goes to minimum output, proportional heaters output goes to minimum, and all backup heaters remain off.

Answer: B

Answer Explanation:

A. Incorrect - Spray valves remain closed and Backup Heaters remain off until actual pressure lowers to 2200 PSIA. Proportional Heaters go to maximum. Spray will collapse the Pressurizer bubble causing Pressurizer level to rise.

B. Correct - The Pressurizer Spray valves would remain closed, Proportional Heaters energize to maximum to raise PZR pressure to setpoint, and Backup Heaters remain off until actual pressure lowers to 2200 PSIA.

C. Incorrect - The Pressurizer Spray valves remain closed and the Backup Heaters remain off until actual pressure lowers to 2200 PSIA.

D. Incorrect - The Pressurizer Spray valves remain closed and the Proportional Heaters would go to maximum output.

Question 2 (Q97000)			
Topic:	Plant response to a change in the Pzr pressure controller setpoint.		
Tier/Group:	1/1		
K/A Info:	027 - Pressurizer Pressure Control System (PZR PCS) Malfunction: • AK2 - Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: • AK2.03 - Controllers and positioners		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-064A2-1		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/ Analysis	
Last NRC Exam used on:	N/A		
Exam Bank History:	LOR11-6B Biennial Written Exam (11/11)		
Technical references:	System Description - 064D, RCS Instrumentation; ALM-1C06, RCS Control		
Comments:	Modified version of Q92862		

3. 059 - Main Feedwater (A2.03)

Given the following conditions on Unit 1:

- Reactor power is 100%.
- The following annunciator window alarms are received in the sequence listed:
  - 1C03, C-28, 11 SGFP DISCH PRESS HI
  - 1C03, C-38, 11 SG FW CONTR CH LVL
  - 1C03, C-39, 12 SG FW CONTR CH LVL
  - 1C03, C-44, 11 SGFPT SPD CONTR SYS TROUBLE

The CRO observes the following at the control panel:

- 11 SGFPT speed is lowering
- 11 SGFP discharge pressure is 1352 PSIG and lowering
- 11 and 12 SG levels are (+) 32 inches and slowly rising
- Main Feed Reg Valves are responding as expected

Which **ONE** of the following describes the status of the Feedwater system; and the action required for the plant conditions?

A. 11 SGFPT discharge pressure has ONLY exceeded the setpoint for SGFPT setback (Runback);
Trip the reactor, trip 11 SGFP, and perform EOP-0, Post-Trip Immediate Actions.

B. 11 SGFPT discharge pressure has exceeded the setpoint for SGFPT setback (Runback) AND SGFPT trip;
Trip 11 SGFP and reduce SG levels using the guidance in AOP-3G, Malfunction of Main Feedwater System.

C. 11 SGFPT discharge pressure has ONLY exceeded the setpoint for SGFPT setback (Runback);
Operate 11 SGFP in manual to reduce speed and restore SG levels per AOP-3G, Malfunction of Main Feedwater System.

D. 11 SGFPT discharge pressure has exceeded the setpoint for SGFPT setback (Runback) AND SGFPT trip;
Trip the reactor, trip 11 SGFP, and perform EOP-0, Post-Trip Immediate Actions.

Answer: C

Answer Explanation:

A. **Incorrect** - . S/G level trip setpoint (+50 inches) not yet reached but discharge pressure above 1350 PSIG initiates the setback circuit. S/G level at +32 inches will actuate a S/G level control channel alarm. Not necessary to trip reactor until attempt made to control 11 SGFP speed manually which, if successful, will restore S/G levels.

B. **Incorrect** - SGFPT trip setpoint not reached (1450 PSIG); Tripping SGFP at 100% power results in being unable to control S/G levels. A rapid downpower would be necessary to continue operating at power but being successful to control S/G levels would most likely cause an automatic reactor trip.

C. **Correct** – Alarm response manual validates that automatic runback signal will initiate whenever pressure exceeds 1350 psig and automatically start to lower SGFP speed. Since MFRVs are closing to compensate for high levels, it is necessary for operator to take manual control and adjust speed to restore S/G levels.

D. **Incorrect** – SGFPT Setback is initiated but SGFPT trip not reached. Tripping SGFP at 100% power with setback initiated would result in a more rapid drop in S/G levels causing a reactor trip before attempts made to control SGFP speed in manual and restore S/G levels. EOP-0 not required, level is not above trip setpoint (+50 inches)

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Question 3 (Q74572)			
Topic:	Main Feedwater		
Tier/Group:	2/1		
K/A Info:	059 - Main Feedwater • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.03 - Overfeeding event		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the actions taken for a SGFP speed controller failure.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2006 Audit Remediation Exam (11/08)		
Technical references:	AOP-3G, Main Feedwater Malfunctions; ALM-1C03, Condensate and Feedwater Control		
Comments:	None		

4. 003 - Reactor Coolant Pump (A4.04)

Unit-1 is operating at 100% power. The following RCP parameters are being monitored:

	11B RCP	12A RCP
VCT pressure	40 PSIG	40 PSIG
Upper seal	870 PSIA	400 PSIA
Middle seal	1750 PSIA	1325 PSIA
Lower cavity seal temperature	120 °F	124 °F
Bleedoff Flow	2.2 GPM	0.0 GPM
Controlled Bleedoff Temperature	122 °F	145 °F

Which **ONE** of the following statements correctly describes the condition of the RCP seals?

A. 11B RCP lower seal degraded; 12A RCP upper seal degraded with vapor seal failed.

B. 11B RCP middle seal degraded; 12A RCP upper and vapor seal failed.

C. 11B RCP seals are normal; 12A RCP middle seal degraded.

D. 11B RCP lower seal failed; 12A RCP vapor seal failed.

Answer: A

Answer Explanation:

A. **Correct** – Based on 11B RCP middle and upper seal pressures the lower seal is degraded. Based on 12A RCP middle seal pressure higher than normal the upper seal is degraded with vapor seal failed based on controlled bleedoff flow.

B. **Incorrect** – 11B middle seal is higher as it is breaking down ½ of remaining pressure drop; 12A RCP upper seal has not completely failed yet as lower and middle seal are reducing RCS pressure by ½ the current value. Per OI-1A, there is > 300 PSID across upper seal so it is reducing pressure but not by ⅓ the value.

C. **Incorrect** – Normal pressures are ⅓ the value (i.e. 1500/750); 12A RCP middle seal pressure is adjusting due to degraded upper seal. Also since 12A RCP controlled bleedoff flow is 0.0, the vapor seal has failed.

D. **Incorrect** – The 11B lower seal is degraded not failed. The 12A RCP upper seal is degraded and the vapor seal has failed.

Question 4 (Q97001)			
Topic:	11B RCP seal status		
Tier/Group:	2/1		
K/A Info:	003 Reactor Coolant Pump System (RCPS) • A4 - Ability to manually operate and/or monitor in the Control Room: • A4.04 - RCP seal differential pressure instrumentation		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given a set of RCP seal indications, determine the status of the seal(s).		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on NRC exam		
Exam Bank History:	None		
Technical references:	OI-1A, Reactor Coolant System And Pump Operations		
Comments:	Modified **Q28840** to add 2nd RCP seal conditions.		

5. 003 - Reactor Coolant Pump (K6.14)

Given the following conditions on Unit-1 in MODE 5:

- RCS Temperature is 190 °F with a plant heatup in progress per OP-1
- PZR level being maintained at 150 inches
- Shutdown Cooling (SDC) has just been secured
- RCS Pressure is being maintained at 290 PSIA IAW OP-1
- 11A and 12B RCPs Oil Lift pumps have been started and operated for at least one minute
- 11A RCP was started five (5) minutes ago
- Just prior to starting 12B RCP, the "OIL LIFT PP PRESS LO" annunciator alarms and will not clear
- 12B RCP Upper and lower oil reservoir levels checked using plant computer indicate normal values

Which **ONE** of the following is appropriate action based on current conditions?

A. Raise RCS pressure to allow single RCP operation using applicable pump operating curve per OI for RCS and Pump operations.

B. Start 12B RCP, after 30 seconds ensure oil lift pump stops automatically and check clear the "OIL LIFT PP PRESS LO" alarm.

C. Stop 12B Oil lift pump, start 12A RCP oil lift pump and operate for at least one minute, then start 12A RCP.

D. Secure 11A RCP, lower RCS pressure, and reinitiate SDC operation IAW OP-1.

Answer: D

Answer Justification:

A. **Incorrect** – This condition is not allowed per OP-1 to operate a single RCP to commence initial plant heatup.

B. **Incorrect** – 12B RCP will not start as oil lift pressure is interlocked with the RCP starting circuit.

C. **Incorrect** – Per OP-1, when selecting a pair of RCPs to start, the second pump must be started within 5 minutes of first one started per Caution prior to step in OP-1. 12A RCP oil lift pump must operate for one minute before starting RCP and this exceeds time limit between RCP starts of OP-1.

D. **Correct** – Per OP-1, it states if a pair of RCPs cannot be started, then reinitiate SDC. One of first steps is to lower RCS pressure before opening SDC Header return isolations.

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Question 5 (Q97002)			
Topic:	12B RCP status		
Tier/Group:	2/1		
K/A Info:	003 Reactor Coolant Pump System (RCPS) • K6 - Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: • K6.14 - Starting requirements		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine which set of RCPs are the preferred set for initial starting and identify the initial RCP starting criteria.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New Question		
Exam Bank History:	None		
Technical references:	OP-1, Plant Heatup from Cold Shutdown OI-1A, Reactor Coolant System And Pump Operations		
Comments:	None		

6. 009 - Small Break LOCA (EK3.24)

Given the following plant conditions on Unit-2:

- A small break LOCA has occurred
- EOP-0 actions have been completed
- The appropriate Optimal Recovery Procedure has been implemented
- Containment pressure peaked at 3.1 PSIG and is slowly lowering
- Both S/Gs levels at (-) 70 inches and rising slowly
- RCS T_{COLD} is 520 °F and lowering
- Pressurizer (Pzr) level is 140 inches and rising rapidly
- Aux Spray is initiated and PZR pressure is 1100 PSIA and lowering
- CET subcooling is 45 °F
- RVLMS lights 1 and 2 are illuminated

Which **ONE** of the following are the appropriate actions per conditions stated?

A. Reduce charging flow to a single pump then secure aux spray to stabilize RCS pressure.

B. Secure both HPSI pumps simultaneously and adjust cooldown to stabilize RCS temperature.

C. Slow the cooldown rate and secure Auxiliary Spray to maintain subcooling in the specified band.

D. Reduce HPSI flow by throttling HPSI header valves or stopping HPSI Pumps one at a time to maintain Pzr level in the specified band.

Answer: D

Answer Justifications:

A. **Incorrect** - This is allowed but is only done **after** HPSI flow has been secured. Reducing Aux Spray will stop lowering RCS pressure, however, the biggest rise in PZR level is attributed to HPSI flow into RCS.

B. **Incorrect** - Securing both HPSI pumps together would stop rise in PZR level immediately. Step for HPSI throttling/termination specifically states when conditions met to stop HPSI Pumps one at a time or throttle HPSI header valves. Stabilizing RCS temperature would only stop RCS depressurization, however, charging flow would still be injecting into RCS causing PZR level to continue rising.

C. **Incorrect** - Subcooling is not being jeopardized at the current value or with the current trends. Examinee must know the subcooling limits for the given condition and perform an analysis to eliminate this distracter. These actions may affect HPSI flow but are not the EOP-5 actions.

D. **Correct** - ALL conditions are met to throttle/terminate HPSI flow which is the most correct action to take for plant conditions.

Question 6 (Q97003)			
Topic:	Actions to control PZR level		
Tier/Group:	1/1		
K/A Info:	009 Small Break LOCA / 3 EK3 - Knowledge of the reasons for the following responses as they apply to the small break LOCA: • EK3.24 - ECCS throttling or termination criteria		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given plant conditions, determine actions to take for ECCS throttling/termination criteria.		
10 CFR Part 55 Content:	55.41(b)(5)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/ Analysis	
Last NRC Exam used on:	New Question		
Exam Bank History:	None		
Technical references:	EOP-5, Loss of Coolant Accident, and Technical Bases		
Comments:	None		

7. 011 - Large Break LOCA (EA2.13)

Considering an ESDE and a LOCA, both which cause containment pressure to peak at 30 PSIG, which **ONE** of the following conditions can be used to differentiate between the accidents?

A. RCS subcooling conditions may be at saturation during the LOCA.

B. Total hydrogen generation is greater during the ESDE.

C. S/Gs are a major contributor to heat removal during the LOCA.

D. RCS inventory loss is greater during the ESDE.

Answer: A

Answer Explanation:

A. **Correct** - Due to loss of inventory from the RCS, subcooling may reach saturation during a LOCA. During an ESDE, subcooling is increased as RCS cools down from faulted S/G, no inventory is lost.

B. **Incorrect** – Hydrogen is generated during the ESDE and the LOCA. During the LOCA the amount of hydrogen produced depends on the duration of core uncovery and the maximum core temperature reached. During the ESDE hydrogen is produced, however, no RCS fluid is released into the containment to add to hydrogen being produced.

C. **Incorrect** – During a large break LOCA the RCS and S/Gs are uncoupled. During a small break LOCA the S/Gs become a significant contributor to RCS heat removal.

D. **Incorrect** – Although in both cases PZR level goes to zero, NO inventory is lost during an ESDE although RCS response from panel indication makes it appear as if inventory has been lost in both cases. The RCS shrinks from the uncontrolled cooldown making it appear that RCS inventory is being lost. Actual inventory is lost during any LOCA.

Question 7 (Q97004)			
Topic:	Actions to control PZR level		
Tier/Group:	1/1		
K/A Info:	011 Large Break LOCA / 3 • EA2 - Ability to determine or interpret the following as they apply to a Large Break LOCA: • EA2.13 - Difference between overcooling and LOCA indications		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Compare the following plant parameters response to differentiate between the design basis accidents, ESDE and a LOCA, occurring:		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	New Question		
Exam Bank History:	None		
Technical references:	EOP-5, Loss of Coolant Accident and Technical Bases		
Comments:	None		

8. 008 - Pressurizer Vapor Space Accident (AK2.02)

Given the following conditions:

- Unit-1 is performing a plant cooldown
- PORVs are in Variable MPT Enable
- An RCS overpressure condition occurred
- The cause of the high pressure condition is corrected

Which **ONE** of the following provides the complete operator response to this condition, if any?

A. No action required, the PORVs will close automatically.

B. Place PORV Override handswitches in override to close.

C. Shut both PORV block valves, when PORVs reseat reopen the block valves.

D. Place PORV Override handswitches in override to close; return to auto when PORVs are shut.

Answer: D

Answer Explanation:

A. **Incorrect** – Plausible as during normal operation these valves will reclose. When in single or variable MPT enable must place in override to close and when PORVs are shut return to auto.

B. **Incorrect** - When in single or variable MPT enable must place in "override to close" to shut the PORVs but must be returned to AUTO to restore MPT overpressure protection.

C. **Incorrect** – Plausible as PORVs will reclose when not in LTOP conditions once block valves have been shut. Per alarm manual, this action is only required if a PORV fails to close or has opened due to a failed transmitter (each PT operates only one PORV for MPT).

D. **Correct** - When in single or variable MPT enable must place in override to close but to restore overpressure protection must be returned to AUTO.

Question 8 (Q97061)			
Topic:	Actions to control PZR level		
Tier/Group:	1/1		
K/A Info:	008 Pressurizer Vapor Space Accident / 3 • AK2 - Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: • AK2.01 - Valves		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given PORV HS positions, RCS temperature and RCS pressure, determine whether PORVs are enabled or disabled for MPT.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New		
Exam Bank History:	None		
Technical references:	1C06-ALM, RCS Control window E-21		
Comments:	None		

9. 029 - ATWS (2.4.45)

Given a Reactor trip on High Pressurizer Pressure:

Which **ONE** of the following specifically identifies that the Diverse Scram System (DSS) has actuated to automatically trip the reactor?

A. Window D-05: "Prot Ch Trip"

B. Window D-46: "MG Set No Output"

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

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

Answer: B

Answer Explanation:

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

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

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

D. **Incorrect** - This alarm, by itself, would not identify a DSS trip. It can occur as the result of any one of the following conditions:

- RPS generated trip
- Manual Rx trip
- DSS generated Rx trip
- A single faulted UV relay.

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Question 9 (Q97005)			
Topic:	Determining when DSS has actuated		
Tier/Group:	1/1		
K/A Info:	029 - Anticipated Transient Without Scram • 2.4 - Emergency Procedures / Plan • 2.4.45 - Ability to prioritize and interpret the significance of each annunciator / alarm.		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Identify the cause and effect of the following alarms on Control Element Drive System (CEDS) …		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI 2010 Panel Comp remediation (01/12)		
Technical references:	1C05-ALM, Reactivity Control Alarm Manual OI-34, Engineered Safety Features Actuation System, Appendix O		
Comments:	Modified from Q36552		

10. 022 - Loss of Reactor Coolant Makeup (AK3.03)

During a Unit-2 reactor startup with power at 6%, a transient occurs resulting in a rapid rise in RCS T_{AVE}. The following conditions exist at control panel 2C07:

- Window F-01: "Regen Hx Out Temp Hi" alarms and Regen HX L/D Out Temp, 2-TIC-221, indicates 471° F
- Window F-09: "L/D Press" alarms

Five minutes later "Regen Hx Out Temp Hi" alarm is clear and 2-TIC-221 indicates 395°F and lowering

Which **ONE** of the following actions is required to restore a Letdown flowpath?

A. Place 2-CVC-515-CV & 2-CVC-516-CV handswitches in CLOSE, then OPEN.

B. Place 2-CVC-515-CV & 2-CVC-516-CV handswitches in CLOSE; momentarily depress the handswitch for 2-CVC-515-CV; place 2-CVC-515-CV & 2-CVC-516-CV handswitches in OPEN.

C. Place 2-CVC-515-CV & 2-CVC-516-CV handswitches in CLOSE; momentarily depress the handswitch for 2-CVC-516-CV; place 2-CVC-515-CV & 2-CVC-516-CV handswitches in OPEN.

D. Depress the handswitches for 2-CVC-515-CV and 2-CVC-516-CV, to OPEN.

Answer: B

Answer Explanation:

A. **Incorrect** – Upon on loss of letdown, procedure direction (OI-2A) is provided to secure charging preventing thermal shock to charging nozzles. This action does not reset the high temperature closing circuit associated with 2-CVC-515-CV.

B. **Correct** - Per OI-2A section 6.7.C (Restore from Excess Flow Check Valve Actuation) these actions are required to establish a L/D flowpath.

C. **Incorrect** - Reset of the high temperature closing circuit is only associated with 2-CVC-515-CV. 2-CVC-516 handswitch has a reset function only for a loss of control power which did not occur in the stem statement.

D. **Incorrect** - These actions do not reset the high temperature closing circuit associated with 2-CVC-515-CV.

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Question 10 (Q97006)			
Topic:	Restoring letdown flow		
Tier/Group:	1/1		
K/A Info:	022 - Loss of Reactor Coolant Makeup • AK3 - Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: • AK3.03 - Performance of lineup to establish excess letdown after determining need		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Describe how letdown flow is restored following closure of the Excess Flow Check Valve.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New		
Exam Bank History:	No previous use		
Technical references:	2C07-ALM, CVCS Alarm Manual OI-2A, Chemical & Volume Control System		
Comments:	None		

11. 055 - Station Blackout (EA1.01)

Following a plant transient resulting in a loss of ALL AC power and reactor trip, you are directed to verify Natural Circulation.

Which **ONE** of the following would be observed when comparing Core Exit Thermocouple (CET) response to RCS temperature trends?

A. CET temperatures are approximately T_{AVE} but trend with T_{HOT}.

B. CET temperatures are always slightly lower than T_{HOT} but trend with T_{HOT}.

C. CET temperatures trend consistent with T_{COLD} which is constant or lowering.

D. CET temperatures trend consistent with T_{HOT} which is constant or lowering.

Answer: D

Answer Explanation:

A. **Incorrect** – Per EOP-7 Basis step K, CET temperatures trend consistent with T_{HOT}. Hot leg RTD temperature should be consistent with core exit thermocouples. Adequate natural circulation flow ensures core exit thermocouple temperatures will be approximately equal to the hot leg RTDs temperature within the bounds of the instruments' inaccuracies. Generally speaking, the CET temperatures will be somewhat higher than T_{HOT}.

B. **Incorrect** - Per EOP-7 Basis step K, CET temperatures trend consistent with T_{HOT}. Hot leg RTD temperature should be consistent with core exit thermocouples. Adequate natural circulation flow ensures core exit thermocouple temperatures will be approximately equal to the hot leg RTDs temperature within the bounds of the instruments' inaccuracies. Generally speaking, the CET temperatures will be somewhat higher than T_{HOT}.

C. **Incorrect** - Per EOP-7 Basis step K, CET temperatures trend consistent with T_{HOT}. Hot leg RTD temperature should be consistent with core exit thermocouples. Adequate natural circulation flow ensures core exit thermocouple temperatures will be approximately equal to the hot leg RTDs temperature within the bounds of the instruments' inaccuracies. Generally speaking, the CET temperatures will be somewhat higher than T_{HOT}.

D. **Correct** - Per EOP-7 Basis step K, CET temperatures trend consistent with T_{HOT}. Hot leg RTD temperature should be consistent with core exit thermocouples. Generally speaking, the CET temperatures will be somewhat higher than T_{HOT}.

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Question 11 (Q97007)			
Topic:	In-core Thermocouple temperatures trend		
Tier/Group:	1/1		
K/A Info:	055 Station Blackout / 6 • EA1 - Ability to operate and monitor the following as they apply to a SBO: • EA1.01 - In-core thermocouple temperatures		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the plant parameters used to verify natural circulation is occurring or being maintained.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/ Analysis	
Last NRC Exam used on:	New Question		
Exam Bank History:	None		
Technical references:	EOP-7 and EOP-7 Technical Basis		
Comments:	None		

12. 065 - Loss of Instrument Air (AA1.04)

Unit 1 was operating at 100% power when a reduction in instrument air header pressure occurred. Given the following events and conditions:

- Instrument Air header pressure lowered to 87 PSIG
- Plant Air header pressure lowered to 84 PSIG

Which of the following actions should have occurred?

(1) The standby instrument air compressor started

(2) CNTMT IA SUPPLY CV, 1-IA-2085-CV, shuts

(3) Plant air header automatic isolation valve (PA-2059-CV) closed

(4) Plant air to instrument air cross connect valve (PA-2061-CV) opened

A. Actions 1 , 2, and 3 only

B. Actions 1, 3, and 4 only

C. Actions 1 and 4 only

D. Actions 1, 2, 3, and 4.

Answer: B

Answer Explanation:

A. **Incorrect** – Actions 1 and 3 have occurred but Action 2 has not. Action 1 occurred at 93 PSIG IA pressure, Action 2 occurs at 75 PSIG IA pressure, and Action 3 occurred at 85 PSIG PA pressure.

B. **Correct** – Action 1 occurred at 93 PSIG IA pressure, Action 3 occurred at 85 PSIG PA pressure, and Action 4 occurred at 88 PSIG IA pressure.

C. **Incorrect** - Action 1 occurred at 93 PSIG IA pressure and Action 4 occurred at 88 PSIG IA pressure, however, Action 3 also occurs.

D. **Incorrect** – Action 2 has not occurred based on IA pressure value. Actions 1, 3, and 4 have occurred based on IA and PA pressure.

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Question 12 (Q97008)			
Topic:	Lowering IA pressure effects		
Tier/Group:	1/1		
K/A Info:	065 - Loss of Instrument Air / 8 • AA1- Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: • AA1.04 – Emergency Air Compressor		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given lowering instrument air conditions, determine the actions needed and why.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	LOI-2008 RO (06/08)		
Exam Bank History:	None		
Technical references:	AOP-7D-1, Loss of Instrument Air Page 5		
Comments:	Modified Q50747 - added IA pressure value to each distractor along with basis behind action.		

13. 056 - Loss of Offsite Power (2.2.3)

Given the following conditions:

- Both Units have tripped from 100% power due to loss of offsite power
- Both 4KV ESF buses on each unit are reenergized from dedicated DG
- 13 and 23 AFW Pumps are operating to recover S/G levels with the following flow rates observed in EOP-0:
  - 11 S/G is 290 GPM
  - 12 S/G is 280 GPM
  - 21 S/G is 270 GPM
  - 22 S/G is 260 GPM

Which **ONE** of the following represents the status of the Total AFW ***and*** individual unit AFW Flow limits and expected action?

A. Total AFW flow is below maximum allowed,
Unit -1 and 2 AFW flows are below maximum allowed;

Monitor S/G levels and adjust AFW flows to maintain within band.

B. Total AFW flow is below maximum allowed,
Unit-1 AFW flow is below maximum allowed,
Unit-2 AFW flow limit is 300 GPM;

Reduce Unit-2 AFW flows to 150 GPM per S/G.

C. Total AFW flow is above maximum allowed,
Unit-1 and Unit-2 AFW flow limits are 300 GPM;

Reduce Unit-1 and Unit-2 AFW flows to 150 GPM per S/G.

D. Total AFW flow is above maximum allowed,
Unit-1 AFW flow is limited to 300 GPM,
Unit-2 AFW flow is below maximum allowed;

Reduce Unit-1 AFW flows to 150 GPM per S/G.

Answer: B

Answer Explanation:

A. **Incorrect** – Total Flow and U-1 AFW are correct; U-2 AFW flow is limited to 300 GPM as 23 AFW Pp is being powered from 2B DG.

B. **Correct** – Status of each flow limit is correct; Reducing U-2 AFW flow to 150 GPM to each S/G ensures that the 2B DG is not overloaded.

C. **Incorrect** – Only U-2 is exceeding AFW flow limits, only need to reduce flow to U-2 S/Gs to 150 GPM to ensure the 2B DG is not overloaded.

D. **Incorrect** – U-2 AFW flow is limited to 300 GPM as 23 AFW Pp is being powered from 2B DG, reducing flow to 150 GPM per S/G ensures the 2B DG is not overloaded.

Question 13 (Q97009)			
Topic:	AFW flow limits during LOOP on each unit		
Tier/Group:	1/1		
K/A Info:	056 Loss of Off-site Power / 6 • 2.2 - Equipment Control • 2.2.3 - Knowledge of the design, procedural, and operational differences between units.		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given plant conditions, determine if 13(23) AFW flow limits are being met.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New Question		
Exam Bank History:	None		
Technical references:	OI-32A-1 & 2, Auxiliary Feedwater System EOP-2-1 & 2, Loss of Offsite Power/Loss of Forced Circulation		
Comments:	None		

14. 040 - Steam Line Rupture (AK1.03)

Given the following conditions on Unit 1:

- Reactor has tripped.
- The crew has transitioned to EOP-4, Excess Steam demand Event, due to 12 S/G pressure lowering uncontrollably.
- 12 ADV is shut and control has been transferred to 1C43.
- The following conditions are indicated:
  - 11 S/G pressure is 700 PSIA
  - 12 S/G pressure is 200 PSIA
  - RCS pressure is 1350 PSIA
  - Core Exit Thermocouple temperatures are 445°F
  - RCS Loop 12 T_{COLD} is 410°F

Which **ONE** of the following provides the needed response, and reason for the response, as the faulted S/G continues to blowdown?

A. Reduce pressure in 11 S/G to 500 PSIA;
Maintains heat removal to limit the possibility of a PTS transient.

B. Reduce pressure in 11 S/G to 500 PSIA;
Prevents a steam generator tube rupture once 12 S/G is dry.

C. Reduce pressure in 11 S/G to 350 PSIA;
Maintains heat removal to limit voids forming in the unaffected S/G.

D. Reduce pressure in 11 S/G to 350 PSIA;
Minimizes ΔP between S/Gs to limit the possibility of a PTS transient.

Answer: A

Answer Explanation:

A. **Correct** - This pressure is within 25°F of CETs and does not add to the RCS cooldown rate as S/G saturation temperature is established above the CET temperatures and limits the possibility of a PTS transient following an excessive cooldown of the RCS.

B. **Incorrect** – This pressure is within 25 °F of CETs and does not add to the RCS cooldown rate as S/G temperature is established above the CET temperatures but the reason is wrong.

C. **Incorrect** – Lowering 11 S/G pressure to this value (432°F) **adds** to the RCS cooldown from faulted S/G although it is within 25°F of CETs. Actions of EOP-4 are concerned about RCS voids forming during this event but this is not the reason for lowering unaffected S/G pressure.

D. **Incorrect** - Lowering 11 S/G pressure to this value (432°F) **adds** to RCS cooldown from faulted S/G although it is within 25 °F of CETs. Lowering unaffected S/G pressure is not to limit the ΔP between the S/Gs. Actions are to maintain within 25°F of CETs to restrict RCS heatup following blowdown to limit possibility of PTS transient.

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Question 14 (Q91846)			
Topic:	In-core Thermocouple temperatures trend		
Tier/Group:	1/1		
K/A Info:	040 Steam Line Rupture - Excessive Heat Transfer / 4 • AK1 - Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: • AK1.03 - RCS shrink and consequent depressurization		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given conditions and/or parameters associated with an ESDE, determine the appropriate operator actions.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of previous use		
Exam Bank History:	LOI-2008 Audit (11/08)		
Technical references:	EOP-4 and EOP-4 Technical Bases		
Comments:	None		

15. 025 - Loss of RHR System (AK2.03)

Given the following plant conditions:

- Unit-2 is in Mode 5
- Shutdown Cooling (SDC) is in operation
- S/G Nozzle Dams are installed
- 22 SDC HX has significant tube leakage requiring emergent repairs

Which **ONE** of the following actions is required per the applicable Tech Spec?

A. Initiate action to restore the inoperable SDC loop to operable status within 1 hour.

B. Restore the inoperable SDC loop to operable status within 1 hour or align spent fuel pool cooling to supplement shutdown cooling.

C. Initiate action to restore the inoperable SDC loop to operable status immediately or initiate action to restore the SGs to operable status.

D. Initiate action to restore the inoperable SDC loop to operable status immediately.

Answer: D

Answer Explanation:

A. **Incorrect** - Examinee may expect an hour is allowed to initiate action to return loop to operable status. Based on conditions, S/Gs are unavailable as nozzle dams installed, therefore, RCS loops not filled and TS LCO 3.4.8 is applicable.

B. **Incorrect** - Examinee may expect an hour is allowed to initiate action to return loop to operable status; however, no provision is made for use of SFP cooling to supplement SDC. Based on conditions, S/Gs are unavailable as nozzle dams installed, therefore, RCS loops not filled and TS LCO 3.4.8 is applicable.

C. **Incorrect** - Examinee may know immediate action is required to return loop to operable status; however, no provision is made in the LCO for returning the S/Gs to operable status as the RCS loop(s) are not filled.

D. **Correct** – Per LCO 3.4.8 Action A is the required operator response.

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Question 15 Info (Q20373)			
Topic:	Knowledge of T.S. LCOs of 1 hour or less		
Tier/Group:	1/1		
K/A Info:	025 Loss of RHR System / 4 • AK2 - Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: • AK2.03 - Service water or closed cooling water pumps		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of previous use		
Exam Bank History:	None		
Technical references:	Tech Spec 3.4.8		
Comments:	Enhanced question by adding bullets for conditions and ensured each distractor had same wording		

16. 058 - Loss of DC Power (AA1.01)

1Y01 has been shifted to the Inverter Back-up bus, 1Y11. Subsequently, a loss of transformer P-13000-1 occurs.

Which **ONE** of the following describes the effect this loss has on plant system(s) response? (Assume a normal electrical distribution system line-up)

A. 1Y11 remains energized from P-13000-2;
1A DG reenergizes 11 4KV Bus;
1Y01 remains energized.

B. 1Y11 remains energized from P-13000-2;
11 4KV Bus remains energized;
1Y01 remains energized.

C. 1Y11 is deenergized on loss of P-13000-1;
1A DG reenergizes 11 4KV Bus;
1Y01 is reenergized.

D. 1Y11 is deenergized on loss of P-13000-1;
1A DG cannot reenergize 11 4KV Bus;
1Y01 remains deenergized.

Answer: A

Answer Explanation:

A. **Correct** - All statements are correct as to effect on plant and expected response to restore power to 4KV Bus 11.

B. **Incorrect** - First and third parts are correct, however, remaining part is wrong as 4KV Bus 11 is initially deenergized on loss of P-13000-1 and reenergized from 1A DG.

C. **Incorrect** - First and third statements are wrong since 1Y11 is energized from MCC-104 which gets power ultimately from P-13000-2 that has not been affected by this loss and 1Y01 was not deenergized.

D. **Incorrect** - All of these are wrong as previously stated.

Question 16 (Q51215)			
Topic:	Loss Backup Inverter Bus for DC Power		
Tier/Group:	1/1		
K/A Info:	058 Loss of DC Power / 6 • AA1 - Ability to operate and / or monitor the following as they apply to the Loss of DC Power: • AA1.01 - Cross-tie of the affected dc bus with the alternate supply		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the power supplies to the following: a. 125 VDC Battery Chargers b. 120 VAC Static Inverters		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of previous use		
Exam Bank History:	LOI-2010 Panel Comp (05/11)		
Technical references:	OI-26B, 120 Volt Vital AC and Computer AC 1C18-ALM, 13KV & 4KV Essential Feeder Bkrs Control Board Alarm Manual window M-07 AOP-7I-1, Loss Of 4kv, 480 Volt Or 208/120 Volt Instrument Bus Power OI-27D-2, Station Power 480 Volt System Breaker Lineup Att. 1B		
Comments:	None		

17.026 - Loss of Component Cooling Water (AK3.04)

Given the following conditions:

- Unit-1 is in Mode 3
- RCS temperature is 532 °F
- RCS pressure is 2250 PSIA
- ALL RCPs are running
- The following control room annunciators have alarmed:
  - "CCW Flow LO" and "CCW Temp HI" for each RCP
  - "CC Head Tk Lvl" with level at 30 inches and lowering
  - "Waste Proc Panel 1C63" with observed level rising on Miscellaneous Waste Receiver Tank

Which **ONE** of the following is the required action by the control room?

A. Implement AOP-7C, Loss of Component Cooling Water, secure ALL RCPs and implement EOP-2, Loss of Offsite Power/Loss Of Forced Circulation.

B. Implement EOP-0, Post-Trip Immediate Actions, perform reactivity control safety function, and then secure ALL RCPs.

C. Implement AOP-7C, Loss of Component Cooling Water, secure ALL RCPs, and concurrently implement AOP-3E, LOSS OF ALL RCP FLOW.

D. Implement AOP-2A, Excess RCS Leakage, secure ALL RCPs, and shut CC Containment Supply and Return Valves.

Answer: C

Answer Explanation:

A. **Incorrect** - Implementing AOP-7C and securing ALL RCPs is correct, however, AOP-7C does not direct implementing EOP-2 based on current mode.

B. **Incorrect** - Stem statement informs examinee unit is in MODE 3 at 532 °F. Implementing EOP-0 does not address other actions needed for loss of CCW. AOP-7C and 3E actions are more specific.

C. **Correct** – Securing ALL RCPs per AOP-7C and implementing AOP-3E are required actions to take.

D. **Incorrect** - Stopping RCPs and isolating CC to containment are needed actions, however, AOP-2A does not apply here. RCS fluid leaks into CC system from letdown or the RCPs would cause the head tank level to rise NOT lower. Securing ALL RCPs per AOP-7C and implementing AOP-3E are required actions to take.

Question 17 (Q40710)			
Topic:	Effect on the CCW flow header of a loss of CCW		
Tier/Group:	1/1		
K/A Info:	026 - Loss of Component Cooling Water • AK3 - Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: • AK3.04 - Effect on the CCW flow header of a loss of CCW		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given a loss of CC system, diagnose the event and take appropriate actions.		
10 CFR Part 55 Content:	55.41(b)(5)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of previous use		
Exam Bank History:	LOR 11-5E Session 5 weekly quiz (10/11)		
Technical references:	AOP-7C, Loss of CCW		
Comments:	Enhanced stem statement conditions and added distractor explanations.		

18. 038 - Steam Generator Tube Rupture (EA2.10)

Given the following conditions on Unit-1:

- A S/G tube leak developed and the crew performed the required AOP actions
- The Reactor was tripped per the AOP trip criteria and EOP-0 entered
- During EOP-0 actions 1Y10 deenergized due to an electrical fault

The Crew transitioned to the appropriate Optimal Recovery Procedure. The following conditions exist:

- RCS pressure is stable at 800 PSIA and approximately equal to ruptured S/G
- SIAS and SGIS were blocked during RCS depressurization and cooldown
- RCS cooldown continuing to initiate shutdown cooling
- The Ruptured S/G is isolated. Level is being maintained between 0 and +50 inches
- A Pressurizer bubble exists and Pressurizer level is being maintained between 101 and 180 inches

Which **ONE** of the following represents the status of charging and letdown flow paths for the EOP in use?

A. Charging flowpath is ONLY thru Aux Spray valve;
Letdown flowpath was isolated for inventory control but is available.

B. Charging flowpath is ONLY thru Loop Charging valves;
Letdown flow has been restored.

C. Charging flowpath is thru LOOP CHG valves **or** Aux Spray valve;
Letdown flowpath was isolated for inventory control and remains unavailable.

D. Charging flowpath is through the Aux HPSI Header.
Letdown flowpath remains unavailable due to a loss of Instrument Air.

Answer: C

Answer Explanation:

A. **Incorrect** - Wrong, both Loop CHG valves or Aux Spray valve are able to maintain charging flow path; Letdown was isolated in AOP-2A and remains unavailable as 1-CVC-515-CV fails shut due to loss of 1Y10.

B. **Incorrect** - Wrong, both Loop CHG valves or Aux Spray valve are able to maintain charging flow path; Prior to EOP-6 entry, letdown was isolated per AOP-2A or EOP-0 actions and remains unavailable as 1-CVC-515-CV fails shut on loss of 1Y10.

C. **Correct** - Both paths are available to maintain charging flow and letdown was isolated and remains unavailable as 1-CVC-515-CV fails shut on loss of 1Y10.

D. **Incorrect** - EOP-6 does not provide guidance for Charging via the Aux HPSI Header and 1-CVC-515-CV is closed due to the loss of 1Y10, not a loss of I/A.

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Question 18 (Q97010)			
Topic:	Charging and Letdown flow paths during a SGTR		
Tier/Group:	1/1		
K/A Info:	038 - Steam Generator Tube Rupture • EA2 - Ability to determine or interpret the following as they apply to a SGTR: • EA2.10 - Flow path for charging and letdown flows		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New Question		
Exam Bank History:	None		
Technical references:	EOP-6, Steam Generator Tube Rupture		
Comments:	None		

19. 077 - Generator Voltage & Grid Disturbances (2.1.23)

Given the following conditions:

- Both units are at 100% power
- Both units generator frequency begin to slowly oscillate between 59.8 HZ and 60.3 HZ
- Unit-1 generator terminal voltage begins to slowly oscillate between 24 KV and 26 KV

What is the required action?

A. Trip both units and implement EOP-0.

B. Check both Main Generator Voltage Regulators in AUTO.

C. Place Unit-1 Voltage Regulator in MANUAL to stabilize grid voltage.

D. Commence a rapid downpower on both units to maintain grid stability.

Answer: B

Answer Explanation:

A. **Incorrect** – Although grid disturbances are occurring, the frequency and voltage oscillations have not reached trip criteria on either unit.

B. **Correct** – Since conditions indicate grid instabilities, it is desirable to check voltage regulators in AUTO to counter small voltage swings. If large swing occurs, each regulator shifts to MANUAL, and if VOLTS/HZ is actuated each main generator will trip to protect itself.

C. **Incorrect** – AOP-7M desires maintaining regulators in AUTO to maintain grid stability. Shifting Unit-1 to MANUAL challenges grid stability. The voltage oscillations with no alarms indicate the voltage regulator is performing its function.

D. **Incorrect** – The AOP only directs a load reduction if partial grid losses have occurred and it is needed to lower frequency. Per stem statement this has not occurred.

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Question 19 (Q38808)			
Topic:	Major Grid Malfunctions		
Tier/Group:	1/1		
K/A Info:	077 - Generator Voltage and Electric Grid Disturbances • 2.1 - Conduct of Operations • 2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.		
RO Importance:	4.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given electrical grid disturbances, evaluate for entry conditions of AOP-7M and if met take the appropriate actions.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of previous use		
Exam Bank History:	LOR 11-3R weekly remediation exam (08/11)		
Technical references:	AOP-7M, Major Grid Disturbances and Technical Bases document		
Comments:	Added answer explanations for distractors. Changed one distractor to be enhance difficulty		

20. CE/E06 Loss of Main Feedwater (EK1.2)

Unit-1 has tripped from 100% power. The following conditions exist:

- A momentary loss of Control Room lighting occurred on **BOTH** Units
- 1C03 annunciator window C-25, "SGFP(s) Suct Press LO" in alarm
- 1C04 annunciator window W-03, "MOTOR Sys NO Flow" in alarm
- 1C04 annunciator window W-04, "TURB Sys NO Flow" in alarm
- Diesel Generators have automatically started with output breakers closed
- Both MSIVs were shut per EOP-0 Alternate Actions

Assuming **ALL** EOP-0 actions are complete, which procedure would be implemented for plant conditions?

A. EOP-8, Functional Recovery Procedure

B. EOP-4, Excess Steam Demand Event

C. EOP-3, Loss of ALL Feedwater

D. EOP-2, Loss of Offsite Power / Loss of Forced Circulation

Answer: C

Answer Explanation:

A. **Incorrect** – Although a LOOP has occurred with a Loss of All Feedwater EOP-8 is not the appropriate EOP to implement. EOP-3 addresses a LOOP within it actions.

B. **Incorrect** – MSIVs being shut are required manual actions due to LOOP as an alternate action for Turbine Trip to secure MSR lineup. Since non-vital power is unavailable, the MSIVs must be shut. No other conditions indicate an ESDE is in progress.

C. **Correct** – This is Optimal EOP to implement. Bullets 2 thru 4 conditions are indicative that a loss of all feedwater has occurred with a LOOP event as well. EOP-3 addresses a loss of offsite power in the major actions to allow crew to restore a source of feedwater. Motor system no flow alarm indicates issue with 13 AFW pump lost or system lineup (power is available to 4KV buses 11 and 14). Turbine system no flow alarm indicates issue with 11 and 12 AFW Pumps or system lineup.

D. **Incorrect** – Although a LOOP has occurred as evidenced by bullets 1, 2, 5 and 6 implementing EOP-2 does NOT address actions needed to restore feedwater to S/Gs which EOP-3 specifically provides.

Question 20 (Q97011)			
Topic:	Loss of ALL Feedwater		
Tier/Group:	1/1		
K/A Info:	CE/E06 Loss of Feedwater / 4 • EK1. Knowledge of the operational implications of the following concepts as they apply to the (Loss of Feedwater) • EK1.2 - Normal, abnormal and emergency operating procedures associated with (Loss of Feedwater)		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given various plant conditions and EOP-0 actions complete, implement the appropriate EOP.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New Question		
Exam Bank History:	None		
Technical references:	EOP-3, Loss of ALL Feedwater		
Comments:	None		

21. 062 - Loss of Nuclear Svc Water (AA2.03)

Unit-2 is at 100% power. STP O-8B-2 is in progress with the 2B DG paralleled to the 24 4KV bus and has been at full load for 30 minutes.

A transient occurs resulting in a "21 SRW HDR PRESS LO" and "U-2 4KV ESF MOTOR OVERLOAD" alarms. 21 SRW header pressure indicates 30 PSIG and steady.

The following temperatures exist:

- Main Turbine Thrust Bearing Metal is 145°F
- Main Turbine Journal Bearing Metal is 180°F
- Generator Hydrogen temperature is 46°C

Which **ONE** of the following actions should be taken first?

A. Shutdown the 2B DG.

B. Immediately trip the reactor and implement EOP-0.

C. Commence a power reduction per OP-3.

D. Start 23 SRW pump after verifying it is aligned to 21 SRW header.

Answer: D

Answer Explanation:

A. **Incorrect** - The 2B DG is unaffected by the transient as it is cooled by 22 SRW header.

B. **Incorrect** - No trip criteria have been exceeded.

C. **Incorrect** - A load reduction would not be required yet based on the given parameters. SRW loads in the Turbine Building are cross connected and although the temperatures on equipment would rise, an immediate load reduction would not be required. Coordination with system operator is to reduce MVARs to zero to minimize heating on main generator.

D. **Correct** - Indications are provided that the 21 SRW pump has tripped due to an electrical issue. AOP-7B directs that the swing SRW pump be mechanically aligned and started on the affected SRW header.

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Question 21 (Q39867)			
Topic:	SRW leak and isolation		
Tier/Group:	1/1		
K/A Info:	062 Loss of Nuclear Svc Water / 4 AA2 - Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: • AA2.03 - The valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given ANY of the following alarms, determine the cause and corrective actions required to clear the alarm(s): • 21(22) SRW HEAD TK LVL		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOR-09E Biennial Written exam (12/09)		
Technical references:	AOP-7B, Loss of Service Water 2C13-ALM, SRW And Misc Station Services Alarm Manual		
Comments:	None		

22. 001- Continuous Rod Withdrawal (AA2.04)

An End-of-Cycle (EOC) reactor start-up on Unit-1 is in progress 4 days after a forced outage shutdown. The following conditions exist:

- Boron equalization is in progress
- Critical data has been recorded at 1 x 10E-4% power
- The RO withdraws Reg. Group 4 CEAs to establish a sustained positive SUR of 0.8 DPM to raise power to the Point of Adding Heat (POAH).
- Shortly after CEA withdrawal is terminated the following are observed:
  - 1CO5 annunciator window D-15: "Power Lvl Rate Hi Ch Pre-Trip" alarms
  - CEA outward motion is observed with CEDS in "OFF"
  - S/G pressures are 910 PSIA and slowly rising

Which **ONE** of the following statements describes the required response?

A. Trip the Reactor and implement EOP-0, Post-Trip Immediate Actions per AOP-1B, CEA Malfunctions.

B. Place TBVs in MANUAL and lower output signal and insert CEAs or BORATE the RCS to lower SUR to zero to stabilize power.

C. Insert CEAs using Manual Sequential to lower SUR below 1.0 DPM as an excessive CEA withdrawal event has occurred.

D. Commence fast boration to the RCS to raise boron to 2300 PPM, trip the reactor, and implement EOP-0.

Answer: A

Answer Explanations:

A. **Correct** - This is the required action of AOP-1B since the CEDS control system was in OFF based on 2nd bullet of stem statement and it is malfunctioning. It is apparent that an uncontrolled CEA withdrawal is occurring.

B. **Incorrect** - Examinee notes that S/G pressures rising means T_{COLD} is rising and an RCS cooldown is NOT occurring. This is action directed from alarm manual condition #3.

C. **Incorrect** - If examinee believes an excessive withdrawal means CEAs are continuing to move OUT. Since SUR continued to rise after original SUR established, using CEDS to insert CEAs will most likely be unsuccessful. Once again this is action from alarm manual condition #1.

D. **Incorrect** - This is the action taken when the Reactor has gone critical below ZPDIL which is not the case. Since reactor is already critical actions of AOP-1A are to borate as needed and/or insert CEAs to control power if examinee assumes a boron dilution event is occurring.

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Question 22 (Q97013)			
Topic:	Uncontrolled CEA Withdrawal Event		
Tier/Group:	1/2		
K/A Info:	001 - Continuous Rod Withdrawal • AA2 - Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: • AA2.04 - Reactor power and its trend		
RO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given a CEA Malfunction the examinee will be able to identify, understand the basis and take appropriate actions per plant operating procedures to mitigate the event.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2006 Trip / Setpoint Criteria (09/08)		
Technical references:	AOP-1B, CEA Malfunctions 1C05-ALM, Reactivity Control Alarm Manual windows D-15		
Comments:	Modified version of Q42232, enhanced stem statement and strengthened distractors and explanations to reflect D-15 window response.		

23. 067 - Plant Fire On-site (AK3.04)

Given that AOP-9A is implemented:

Which **ONE** of the following statements explains why the Fairbanks Morse Diesel Generators are shutdown?

A. To prevent overloading the Diesel Generators as equipment starts because the Shutdown Sequencers may be inoperable.

B. To ensure fuel is conserved for continued extended operation of the 1A and 0C DGs.

C. To prevent engine damage due to the essential trips being bypassed with an active UV signal.

D. The 0C DG is aligned to power a Unit-1 and Unit-2 safety related 4KV bus simultaneously.

Answer: C

Answer Explanation:

A. **Incorrect** - The ESFAS sequencers are powered from 120VAC vital busses that are provided power from the DC busses.

B. **Incorrect** - Although tripping these DGs will conserve fuel oil they normally use, the 1A DG is not aligned as a power source, ONLY the 0C DG is aligned to 11 and 24 4KV busses of each unit to provide power and it has its own fuel supply.

C. **Correct** - This is per AOP-9A Rev. 12 page 3 bases document.

D. **Incorrect** - This is a true statement in and of itself. It is not, however, the reason for securing the Fairbanks Morse Engines. Plausibility lies in the fact that it could make sense to run just one DG rather than two, given the higher load capacity of the SACM engine.

Question 23 (Q97014)			
Topic:	AOP-9A bases for actions		
Tier/Group:	1/2		
K/A Info:	067 - Plant Fire On-site • AK3 - Knowledge of the reasons for the following responses as they apply to the Plant Fire on Site: • AK3.04 - Actions contained in EOP for plant fire on site		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given AOP-9A and the Technical Bases, list the actions performed by each watchstander and determine the bases for those actions.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2004 RO (04/04)		
Exam Bank History:	No record of previous use		
Technical references:	AOP-9A and Technical Bases document		
Comments:	Add to bank		

24. 061 - ARM System Alarms (AA1.01)

Which **ONE** of the following occurs on a loss of power to the Control Room ventilation RMS (0-RI-5350) monitor?

A. Control Room kitchen exhaust fan STOPS with gravity damper SHUT; BOTH post-LOCI filter fans START.

B. Control Room outside air supply and common exhaust dampers SHUT; BOTH post-LOCI filter fans START.

C. Operating Control Room HVAC outside air supply damper SHUTs; Selected post-LOCI filter fan STARTS.

D. Operating Control Room HVAC dampers shift to recirculation mode; Selected post-LOCI filter fan STARTS.

Answer: A

Answer Explanation:

A. **Correct** - Since Control Room ventilation normal lineup is in a recirculation mode, only actions stated occur automatically.

B. **Incorrect** - Control Room outside air dampers are already shut as recirculation mode is the normal ventilation lineup.

C. **Incorrect** - All actions listed are wrong.

D. **Incorrect** – System is already in recirculation mode per OI-22F. Loss of power causes both Post-LOCI filter fans to start not just one fan to start.

Question 24 (Q94793)			
Topic:	Loss of power to Control Room Ventilation RMS		
Tier/Group:	1/2		
K/A Info:	061 - ARM System Alarms • AA1 - Ability to operate and / or monitor the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: • AA1.01 - Automatic actuation		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	DESCRIBE the design features that provide for the following during operation of the CR Ventilation and Chilled Water System: • 100% recirculation during high radiation conditions (or loss of power to RMS)		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2010 1C22/34 exam (09/11)		
Technical references:	OI-35, Radiation Monitoring System OI-22F, Control Rm and Cable Spreading Rm Vent		
Comments:	Modified from Q24745		

25. 003 Dropped Control Rod (AK1.16)

During a Unit-2 initial startup after refueling, the following exist:

- Power was stabilized at 30% for NI CAL
- CEA withdrawal recommences to raise power when a Regulating Group 4 CEA drops fully to the bottom

Which **ONE** of the following actions is the initial response required?

A. Commence realignment of the dropped CEA and borate the RCS as needed to keep power constant.

B. Adjust turbine load to maintain T_{COLD} on program and stop reactor power from continuing to lower.

C. Withdraw remaining CEAs in steps as needed to maintain T_{COLD} on program and stabilize reactor power.

D. Initiate boration to counter effects of T_{COLD} lowering causing reactor power to rise above level prior to CEA drop.

Answer: B

Answer Explanations:

A. **Incorrect** - This action is not the initial response per AOP-1B. This is only done after plant is stabilized by adjusting turbine load, TBVs or ADVs, or initiating boration.

B. **Correct** - At BOL a positive MTC exists. A dropped CEA adds negative reactivity causing T_{COLD} to lower which with a positive MTC will add more negative reactivity causing T_{COLD} to continue to lower. Lowering turbine load will stabilize T_{COLD} and reactor power.

C. **Incorrect** - CEAs will add positive reactivity to compensate for T_{COLD} continuing to lower with positive MTC. However, CEAs shall **NOT** be used to control T_{COLD} per caution of AOP-1B.

D. **Incorrect** - If examinee doesn't recognize positive MTC exists (adds negative reactivity and causes T_{COLD} to continue to lower) may assume power will rise due to drop in T_{COLD} and initiate boration to prevent reactor power from exceeding power level prior to CEA drop.

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Question 25 (Q97015)			
Topic:	Dropped CEA actions at power and BOL		
Tier/Group:	1/2		
K/A Info:	003 Dropped Control Rod / 1 AK1 - Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: • AK1.16 - MTC		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	AOP-1B, CEA Malfunctions		
Comments:	Modified from Q37647		

26. 005 - Residual Heat Removal System (K5.03)

Given an RCS fill evolution in progress during Shutdown Cooling Ops:

(1) Which **ONE** of the following criteria defines adequate mixing and (2) Indicates a boron dilution event may be in progress?

A. (1) At least 1500 GPM flow thru the core **AND** at least 500 GPM flow through at least one S/G;
(2) Unexpected slow rise in SDC temperature.

B. (1) At least 1500 GPM flow thru the core **AND** at least 500 GPM flow through both S/Gs;
(2) Audible countrate, in the Control Room, increases.

C. (1) At least 3000 GPM flow thru the core **AND** at least 500 GPM flow through at least one S/G;
(2) An unexpected rise in RCS level.

D. (1) At least 3000 GPM flow thru the core **AND** at least 500 GPM flow through both S/Gs;
(2) An unexpected rise in countrate on Nuclear Instrumentation.

Answer: D

Answer Explanation:

A. **Incorrect** - Both parts are wrong ... 1500 GPM is the minimum SDC flow surveillance requirement for Mode 6 (logged in CRO logs), the 500 GPM flow is requirement per S/G. SDC temperature would not rise during any boron dilution event when shutdown. If anything it will lower as fill water is added and operator needs to adjust to maintain temperature.

B. **Incorrect** - 1500 GPM is the minimum SDC flow surveillance requirement for Mode 6 (logged in CRO logs), the 500 GPM flow is requirement per S/G.; popper noise becoming more frequent indicates counts are rising on NIs which could be expected during inadvertent dilution event.

C. **Incorrect** - Flow thru core is right but flow must be thru both S/Gs not just one. Rise in RCS level compared to change in RWT level may indicate another source of water is entering RCS causing an inadvertent dilution event.

D. **Correct** - Flow thru core and S/Gs meets criteria; NI counts rising is an indication that a boron dilution event may be occurring.

Question 26 (Q97016)			
Topic:	Reactivity effects of SDC fill water into RCS		
Tier/Group:	2/1		
K/A Info:	005 Residual Heat Removal System (RHRS) K5 - Knowledge of the operational implications of the following concepts as they apply the RHRS: • K5.03 - Reactivity effects of RHR fill water		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	Identify RCS dilution limitation including requirements for adequate mixing.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	AOP-1A, Inadvertent Dilution Event OP-7, Shutdown Operations		
Comments:	Modified from Q25960		

27. 69 - Loss of Containment Integrity (AA1.03)

Given the following condition on Unit-2 in Mode 4:

- The SDC header is placed in recirculation through the SIT recirculation leak-off isolation valves, 2-SI-463 and 2-SI-455, flow path return to the RWT

Which **ONE** of the following procedural controls ensures Containment Integrity is reestablished?

A. An approved Contingency Plan is activated to re-establish Containment Integrity when the evolution has been completed.

B. Containment Closure Deviation Sheets are used to ensure Containment Integrity is re-established.

C. A dedicated operator is stationed in continuous communication with the Control Room to restore valves to locked shut condition.

D. A Component Manipulation Form (CMF) is completed and closed out when the lineup is secured.

Answer: C

Answer Explanation:

A. **Incorrect** - Contingency plans are used in lower modes to address plant situations where maintenance or equipment situations challenge the MEEL or for High Risk evolutions.

B. **Incorrect** - Containment Closure Deviation tracking sheets are only used in Modes 5 and 6.

C. **Correct** - These are manual valves administratively controlled per NO-1-205 and per T.S.3.6.3. Per OI-3B there is a step to open these valves then shut and lock them by stationing a dedicated operator in continuous communication when this path is in use.

D. **Incorrect** - This activity is covered by an Operating Instruction. A CMF is not required.

Question 27 (Q20794)			
Topic:	Containment Isolation valves		
Tier/Group:	2/1		
K/A Info:	069 Loss of Containment Integrity / 5 AA1. Ability to operate and / or monitor the following as they apply to the Loss of Containment Integrity: • AA1.03 - Fluid systems penetrating containment		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the actions established to take per NO-1-205 when operating administratively controlled containment isolation valves during Recirc of SDC to RWT.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2002 RO (07/02)		
Exam Bank History:	LOI-2006 RO/SRO Audit Remediation Exam (05/08)		
Technical references:	OI-3B, Section 6.2 Recirculation of SDC header NO-1-114, Containment Closure		
Comments:	None		

28. 076 - High Reactor Coolant Activity (AK2.01)

Which of the following statements describes the relationship of RCS activity to the Process Radiation Monitor, 1(2)-RI-202, installed in the Letdown line sample path?

A. ONLY RCS gross activity is monitored.

B. ONLY activity associated with a specific isotope related to fuel failure events is monitored.

C. Gross activity and activity associated with a specific isotope related to fuel failure events are monitored.

D. Gross activity and activity associated with a specific isotope related to fuel failure events are monitored continuously, during all accident conditions.

Answer: C

Answer Explanation:

A. **Incorrect** - The PRM detects both gross activity and activity associated with a specific isotope.

B. **Incorrect** - The PRM detects both gross activity and activity associated with a specific isotope.

C. **Correct** - This is the purpose of the monitor in the letdown system. If both are increasing it identifies that a fuel failure event is occurring rather than just a crud burst.

D. **Incorrect** - The PRM detects both gross activity and activity associated with a specific isotope. However, the PRM is isolated when SIAS is actuated and can no longer be relied upon.

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Question 28 (Q97017)			
Topic:	Process Rad Monitor relationship to RCS activity		
Tier/Group:	2/1		
K/A Info:	076 High Reactor Coolant Activity /9 • AK2. Knowledge of the interrelations between the High Reactor Coolant Activity and the following: • AK2.01 - Process radiation monitors		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Identify the purpose of the Process Radiation Monitor.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory/ Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2004 RO (02/04)		
Exam Bank History:	None		
Technical references:	System Description (SD) 041 - CVCS		
Comments:	Modified from Q14577		

29. CE/A16 - Excess RCS Leakage (2.1.20)

Unit-2 is in Mode 5 with the following conditions:

- The plant has been shut down for 3 days
- RCS temperature is 130° F
- RCS is at atmospheric pressure with the Pzr manway installed
- Pressurizer level is 120" and lowering at approximately 20 inches per minute
- The appropriate AOP has been entered

Which **ONE** of the following actions will restore RCS level in accordance with the AOP?

A. Start all available Charging pumps.

B. Open a LPSI Pump normal suction valve.

C. Start a HPSI pump and throttle flow to less than 210 GPM.

D. Start a HPSI pump and maintain RCS pressure less than 260 PSIA.

Answer: D

Answer Explanation:

A. **Incorrect** - Starting all available Charging Pumps is the first step specified in AOP-3B. However, information provided, in the stem of the question, indicates Charging Pumps alone will not restore RCS level at this leak rate requiring that a HPSI Pump be started per Att. 7, Filling the RCS.

B. **Incorrect** - While opening the LPSI Pp Normal Suction may supply makeup water to the RCS (if the RWT Outlet MOV is open), AOP-3B does not direct this action. This action requires local operator action outside of the control room.

C. **Incorrect** - Per AOP-3B Attachment 7, flow into the RCS is limited to less than 210 GPM unless a leak exists. Indications of a leak are provided in the stem of the question.

A. **Correct** - Per AOP-3B Attachment 7, Filling the RCS: When RCS temperature is less than 365° F AND the RCS vent opening is less than 2.6 square inches, flow into the RCS is limited to less than 210 GPM unless a leak exists. If a leak exists, flow may exceed 210 GPM as long as pressure is maintained less than 380 PSIA (or 260 PSIA if the SDC Header Return Isolation valves, 1-SI-651-MOV and 1-SI-652-MOV, are open).

Question 29 Info (Q24883)			
Topic:	RCS leakage into CCW and cannot be isolated		
Tier/Group:	1/2		
K/A Info:	CE/A16 - Excess RCS Leakage • 2.1 - Conduct of Operations • 2.1.20 - Ability to interpret and execute procedure steps.		
RO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given various AOPs and bases documents with a set of plant conditions, navigate the procedures correctly to mitigate the effects of various malfunctions		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/ Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	No record of use on any exam		
Technical references:	AOP-3B, Abnormal Shutdown Cooling Conditions		
Comments:	None		

30. 059 - Accidental Liquid Rad Waste Release (AK2.01)

The Miscellaneous Waste Monitor Tank (MWMT) is being discharged per an approved release permit when the LIQUID WASTE DISCH RMS monitor, 0-RIC-2201, alarms high. Upon investigation, the Control Room observes the LIQUID WASTE DISCH CVs, 0-MWS-2201-CV and 0-MWS-2202-CV, have not shut automatically.

Which **ONE** of the following is the expected operator response?

A. Verify valves 0-MWS-2201-CV and 0-MWS-2202-CV shut.

B. Stop the MWMT pump being used to discharge the MWMT.

C. Ensure valves 0-MWS-103 and 0-MWS-105 are shut to isolate the Unit-2 SG Blowdown overboard discharge path.

D. Continue discharge of MWMT using the procedure for 0-RE-2201 not available and energized.

Answer: A

Answer Explanation:

A. **Correct** - Per 1C22-ALM, RMS Alarm Manual, this is the appropriate action. Verify means to make it happen if it hasn't. In this case, placing the handswitches for 0-MWS-2201-CV and 0-MWS-2202-CV in close would cause the valves to shut, terminating the accidental liquid waste release.

B. **Incorrect** - This action is specified by AOP-6B, Accidental Liquid Waste Release, if the discharge CVs fail to shut when handswitches placed to shut position.

C. **Incorrect** –This action is specified by AOP-6B, Accidental Liquid Waste Release, if the discharge CVs fail to shut when handswitches placed to shut position.

D. **Incorrect** – This action per Alarm Manual due to an RMS failure. Question Stem stated due to high alarm.

Question 30 (Q97019)			
Topic:	Liquid Waste Monitor, 0-RIC-2201, automatic actions		
Tier/Group:	1/2		
K/A Info:	059 Accidental Liquid RadWaste Release / 9 AK2 - Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: • AK2.01 - Radioactive-liquid monitors		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine the automatic actions upon a high alarm on 0-RIC-2201.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	AOP-6B, Accidental Liquid Waste Release 1C22 Alarm Response Manual Window D32		
Comments:	Modified from original Q74951		

31. CE/E09 Functional Recovery (EK1.2)

Which one of the following conditions would require the implementation of EOP-8, Functional Recovery procedure?

A. Reactivity Control safety function cannot be met in EOP-0 due to no power available to CEA indications.

B. A loss of offsite power results in a reactor trip, and the EOP-0 flowchart recommends EOP-6 implementation.

C. The EOP-0 flowchart recommends implementing both EOP-3 and EOP-7 and single event diagnosis is not possible.

D. EOP-4 is implemented but the Final Safety Function Acceptance Criteria is not being met.

Answer: C

Answer Explanation:

A. **Incorrect** - The EOP-0 Diagnostic flowchart would recommend considering EOP-7, Station Blackout in this case.

B. **Incorrect** - The EOP-0 Diagnostic flowchart would recommend considering EOP-2 and EOP-6, Steam Generator Tube Rupture. EOP-6 is written to address a LOOP coincident with a SGTR.

C. **Correct** - These are the conditions needed to enter EOP-8.

D. **Incorrect** - Final acceptance criteria not being met is incorrect. EOP-8 would be implemented if the Intermediate Safety Function Status Check(s) is/are not met.

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Question 31 (Q25083)			
Topic:	EOP-8 entry conditions		
Tier/Group:	1/2		
K/A Info:	CE/E09 - Functional Recovery • EK1- Knowledge of the operational implications of the following concepts as they apply to the (Functional Recovery) • EK1.2 - Normal, abnormal and emergency operating procedures associated with (Functional Recovery)		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine conditions when EOP-8 may be entered		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	2010 RO Recertification Test		
Exam Bank History:	LOI-2006 SRO practice (03/08)		
Technical references:	EOP-8, Functional Recovery Procedure		
Comments:	None		

32. 004 - Chemical and Volume Control (K6.13)

RCS boration is in progress when a loss of instrument air occurs.

Which **ONE** of the following modes of boration would require the RO to select an alternate path due to loss of instrument air?

A. BA pumps to charging pump suction (fast boration).

B. Gravity feed to charging pump suction.

C. RWT to charging pump suction.

D. Borate Makeup mode to VCT.

Answer: D

Answer Explanation:

A. **Incorrect** – Only an MOV is used for this flowpath. Loss of IA has no effect on this valve.

B. **Incorrect** – Once again this flowpath is aligned using only MOVs, NO CVs, so no effect from loss of IA.

C. **Incorrect** – This flowpath also uses only an MOV therefore, loss of IA has no effect.

D. **Correct** – Boric Acid Flow control valve, 1-FIC-210Y, fails closed on loss of IA and 1-CVC-512-CV also fails closed.

Question 32 (Q20598)			
Topic:	Loss of IA effects on when borating to VCT		
Tier/Group:	2/1		
K/A Info:	004 - Chemical and Volume Control • K6 - Knowledge of the effect of a loss or malfunction on the following CVCS components: • K6.13 - Purpose and function of the boration/dilution batch controller		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Explain how CVCS responds to the following conditions: Loss of Instrument Air		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/ Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2002 RO (07/02)		
Exam Bank History:	LOR 11-6A Biennial Written exam		
Technical references:	OI-2B, CVCS Boration, Dilution, and Makeup Operations page 70 AOP-1A, Inadvertant Boron Dilution Attachment 1 pages 2, 3, & 4		
Comments:	None		

33. 005 - Residual Heat Removal System (A4.02)

Unit-1 is in Mode 6 with the RCS drained to the 37.5 Foot elevation.

Which **ONE** of the following prerequisites must be established prior to shifting LPSI pumps in this condition?

A. Raise RCS level to at least 38 feet and Reduce SDC flow to 800 GPM.

B. Adjust 1-FIC-306 and 1-HIC-3657 to maintain SDC flow at 1500 GPM.

C. Reduce SDC flow to 800 GPM using 1-FIC-306 and 1-HIC-3657.

D. Verify the designated LPSI header MOVs are throttled to limit SDC flow to 1700 GPM if a loss of power occurs to 1-SI-306-CV.

Answer: C

Answer Explanation:

A. **Incorrect** - Raising the RCS level to 38' would place the plant in a condition where the idle LPSI Pp could be started and the previously running LPSI Pp could be stopped without SDC flow limitations.

B. **Incorrect** - This prerequisite is associated with SDC flowrate limitations prior to draining the RCS to below the 37.6 ft. elevation.

C. **Correct** - OI-3B states: PLACE the SDC FLOW CONTR, 1-FIC-306 in MANUAL AND REDUCE SDC Flow to approximately 800 GPM by adjusting the SDC FLOW CONTR, 1-FIC-306 and SDC TEMP CONTR, 1-HIC-3657.

D. **Incorrect** - 1700 GPM is the limit, established in OP-7, when the UGS is installed, to prevent damage to the ICI thimbles.

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Question 33 (Q20727)			
Topic:	Determine the prerequisites for shifting LPSI pumps.		
Tier/Group:	2/1		
K/A Info:	005 - Residual Heat Removal System (RHRS) • A4 - Ability to manually operate and/or monitor in the control room: • A4.02 - Heat exchanger bypass flow control		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/ Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2010 1C08, 09, & 10 mid-term		
Exam Bank History:	None		
Technical references:	OI-3B, Shutdown Cooling OP-7, Shutdown Operations		
Comments:	None		

34. 006 - Emergency Core Cooling (K5.07)

When initiating SDC, OP-5 specifies the RCS cooldown should be stopped and new baseline temperatures obtained after SDC is initiated.

Which **ONE** of the following is the reason for obtaining new baseline temperature data?

A. The indicated temperature difference between the SDC HX outlets and the hot legs prevents accurate cooldown determination when SDC is initiated.

B. The indicated temperature difference between the hot and cold legs prevents accurate cooldown determination when SDC is initiated.

C. The indicated temperature difference between the hot legs prevents accurate cooldown determination when SDC is initiated.

D. The indicated temperature difference between the CETs and TR-351 prevents accurate cooldown determination when SDC is initiated.

Answer: D

Answer Explanation:

A. **Incorrect** - This is accurate once flow is directed thru the SDC HXs and this is the return temperature to the RCS.

B. **Incorrect** - The temperature difference between hot and cold legs is accurate once natural circulation has been established after the RCPs are secured and prior to SDC initiation.

C. **Incorrect** – Although SDC suction is only from one hot leg, there is no temperature difference observed between hot legs.

D. **Correct** –OP-5 states: “Due to the indicated temperature differential between the CETs and 1-TR-351, accurate cooldown determination is not available when SDC is initiated. When initiating SDC, the cooldown should be stopped and new baseline temperatures obtained after SDC is initiated.

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Question 34 (Q97022)			
Topic:	Why new set of baseline data is obtained prior to reinitiating SDC		
Tier/Group:	2/1		
K/A Info:	006 Emergency Core Cooling K5 - Knowledge of the operational implications of the following concepts as they apply to ECCS: • K5.07 - Expected temperature levels in various locations of the RCS due to various plant conditions		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Initiate SDC during a plant cooldown upon securing RCPs		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	OP-5, Section 6.3 step G note, page 41		
Comments:	None		

35. 007 - Pressurizer Relief/Quench Tank (K1.03)

A loss of load transient resulted in a plant trip with PORVs lifting. Which of the following would indicate that the quench tank rupture disk has failed?

A. “CNTMT NORMAL SUMP LVL HI” alarm annunciates.

B. “QUENCH TK •TEMP•LVL•PRESS” alarm annunciates.

C. Reactor Coolant System pressure lowers more rapidly.

D. “11 RCDT •PRESS HI•LVL” alarm annunciates

Answer: A

Answer Explanation:

A. **Correct** - The sump alarm with the quench tank pressure rapidly lowering is indication that the rupture disk has failed as quench tank overflows to the normal sump.

B. **Incorrect** - Although quench tank pressure has been lost, level and temperature remain high keeping window in alarm.

C. **Incorrect** - The small range of backpressure associated with an intact or open quench tank has little effect of PORV relief capacity.

D. **Incorrect** - The Quench Tank is connected to the RC Drain Tank through a normally closed Quench Tank Drain Valve. RCDT parameters will be unaffected by an open PORV.

Question 35 (Q20628)			
Topic:	Quench Tank rupture disk failing		
Tier/Group:	2/1		
K/A Info:	007 - Pressurizer Relief Tank/Quench Tank System • K1 - Knowledge of the physical connections and/or cause/effect relationships between the PRTS and the following systems: • K1.03 - RCS		
RO Importance:	3.0		
Proposed references to be provided to applicant:	None		
Learning Objective:	Identify indications that a quench tank rupture disk has failed.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/ Fundamental	☒ Comprehension/ Analysis	
Last NRC Exam used on:	LOI-2002 RO (07/02)		
Exam Bank History:	LOI-2008 1C06 RRS (04/09)		
Technical references:	1C06 –ALM, RCS Control Alarm Manual, 1C10-ALM, ESFAS 13 Alarm Manual 1C33-ALM, Waste Processing System Alarm Manual		
Comments:	None		

36. 007 - Pressurizer Relief/Quench Tank (A4.01)

Given containment pressure on Unit 2 has reached 5.0 PSIG during an event, which one of the following valves requires manual action to close if open?

A. IA CONTAINMENT ISOLATION, 2-IA-2080-MOV

B. RCS SAMPLE ISOL valve, 2-PS-5464-CV

C. DW CNTMT ISOL valve, 2-DW-5460-CV

D. SRW SUPP TO 22 BD HX, 2-SRW-1640-CV

Answer: C

A. **Incorrect** - IA CONTAINMENT ISOLATION, 2-IA-2080-MOV automatically closes on receipt of a CIS (Containment Pressure greater than 2.8 PSIG).

B. **Incorrect** - RCS SAMPLE ISOL valve, 2-PS-5464-CV automatically closes on receipt of a SIAS (Containment Pressure greater than 2.8 PSIG or RCS pressure less than 1725 PSIA).

C. **Correct** - This valve receives no automatic ESFAS signal to close. It is an administratively controlled valve. CIS verification checklist (EOP Att. 4 Page 1) directs shutting this valve if open.

D. **Incorrect** - SRW SUPP TO 22 BD HX, 2-SRW-1640-CV automatically closes on receipt of a CSAS (Containment Pressure greater than 4,25 PSIG).

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Question 36 (Q97023)			
Topic:	Restore quench tank temperature		
Tier/Group:	2/1		
K/A Info:	007 Pressurizer Relief Tank/Quench Tank System • A4 - Ability to manually operate and/or monitor in the control room: • A4.01 - PRT spray supply valve		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	EOP Attachments 2, 3, and 4		
Comments:	None		

37. 008 - Component Cooling Water System (K4.09)

A loss of offsite power has occurred causing the Unit-2 reactor to trip. Given the following:

- Prior to the trip, 21 Component Cooling pump was running
- RCS pressure is 1700 PSIA and slowly lowering
- 480V bus 24A is deenergized
- No operator actions have been performed for the Vital Auxiliaries safety function

Which combination of annunciator windows, in alarm, indicate that 23 Component Cooling pump is operating?

A. "ACTUATION SYS SIAS TRIP" and "CCW PPS • SIAS BLOCKED • AUTO START"

B. "CC PPS DISCH PRESS LO" and "ACTUATION SIGNAL BLOCKED"

C. "U-2 4KV ESF MOTOR OVERLOAD" and "SEQUENCER INITIATED"

D. "U-2 480V ESF U/V TRIP" and "23 CC PP BKR L/U IMPR"

Answer: A

Answer Explanation:

A. **Correct** - PZR pressure at 1700 PSIA generates a SIAS signal sent to ALL 3 CCW Pump starting circuits and 480V bus 24A deenergized provides a U/V condition to actuate the second alarm. This bus also supplies power to 22 CCW Pump which is unavailable, therefore, 23 CCW Pump will start (since normally aligned to 480V bus 24B) after one second upon receipt of SIAS signal when 22 CCW Pump fails to start.

B. **Incorrect** - The first alarm indicates NO CCW Pumps are running and second alarm occurs upon a LOOP and SIAS indicating all LOCI sequencer steps have not been completed for train A and/or B (this alarm clears when all steps timeout once power is restored).

C. **Incorrect** - The Second alarm occurs for conditions stated in stem statement. First alarm is wrong as CCW Pumps are 480V loads not 4KV loads.

D. **Incorrect** - The first alarm will occur when 480V bus 24A is deenergized. The second alarm means annunciates when 23 CCW Pp where it deviates from the standard lineup of one breaker racked in with its associated disconnect shut.

Question 37 (Q97024)			
Topic:	The "standby" feature for CCW pumps		
Tier/Group:	2/1		
K/A Info:	008 Component Cooling Water System • K4 - Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: • K4.09 - The "standby" feature for the CCW pumps		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given plant conditions, determine the status of "standby" CCW pump		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2008 RO Audit (05/10)		
Technical references:	2C08-ALM, ESFAS 21 Alarm Manual 2C13-ALM, SRW and Misc Station Services Alarm Manual 2C17-ALM, 4KV & 480V Normal FDR BKR Alarm Manual 1C19-ALM, 13KV & 4KV Essential Feeder Bkrs Control Board Alarm Manual		
Comments:	Modified from Q92262		

38. 010 - Pressurizer Pressure Control (K2.01)

A Loss of Offsite Power has occurred with the 1B Diesel Generator failing to start. Assuming no electrical buses are tied, which of the following is correct?

A. Pressurizer backup heater banks 1 and 3 are available from 1C43 only.

B. Pressurizer backup heater bank 1 is available from 1C43 only.

C. Pressurizer backup heater banks 1 and 3 are available from 1C06 and 1C43.

D. Pressurizer backup heater bank 3 is available from 1C43 only.

Answer: B

Answer Explanation:

A. **Incorrect** - Pressurizer Backup Heater banks 1 & 3 are powered from 480V Busses 11B and 14B respectively. The stem states the 1B DG failed to start which means Pressurizer Backup Heater bank 3 is NOT available.

B. **Correct** - Pressurizer Backup Heater banks 1 & 3 are powered from 480V Busses 11B and 14B respectively. The stem states the 1B DG failed to start which means only Pressurizer Backup Heater bank 1 is available. Because 1Y10 de-energizes, as a result of the 1B DG Start Failure, all Pressurizer Heaters receive a signal to turn off. Operation of the Pressurizer Heater(s) under these conditions requires transferring control to 1C43 via a local keyswitch.

C. **Incorrect** - Pressurizer Backup Heater banks 1 & 3 are powered from 480V Busses 11B and 14B respectively. The stem states the 1B DG failed to start which means Pressurizer Backup Heater bank 3 is NOT available. Because 1Y10 de-energizes, as a result of the 1B DG Start Failure, all Pressurizer Heaters receive a signal to turn off. Operation of the Pressurizer Heater(s) under these conditions requires transferring control to 1C43 via a local keyswitch.

D. **Incorrect** - Pressurizer Backup Heater banks 1 & 3 are powered from 480V Busses 11B and 14B respectively. The stem states the 1B DG failed to start which means Pressurizer Backup Heater bank 3 is NOT available.

Question 38 (Q14487)			
Topic:	EDG power to the pressurizer heaters		
Tier/Group:	2/1		
K/A Info:	010 - Pressurizer Pressure Control • K2 - Knowledge of bus power supplies to the following: • K2.04 – Pzr Heaters		
RO Importance:	3.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/ Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2010 ESFAS Panel Comp (9/11)		
Exam Bank History:	None		
Technical references:	AOP-7I, Loss Of 4kv, 480 Volt Or 208/120 Volt Instrument Bus Power		
Comments:	None		

39. 012 = Reactor Protection (K3.01)

Unit-1 is at 100% power when a malfunction occurs in RPS Channel B causing the Power Trip Test Interlock (**PTTI**) to actuate.

Which **ONE** of the following describes the effect on the Control Rod Drive System?

A. 1 of 2 required pre-trips from VOPT or APD for CEA Motion Inhibit is met.

B. 1 of 4 required pre-trips from APD or HI-SUR for CEA Withdrawal Prohibit is met.

C. 1 of 2 required pre-trips from VOPT or TM/LP for CEA Withdrawal Prohibit is met.

D. 1 of 4 required pre-trips for HI SUR or TM/LP for CEA Motion Inhibit is met.

Answer: C

Answer Explanation:

A. **Incorrect** - These trips occur in RPS from **PTTI**; however, they do NOT provide an input to CMI circuit for CEAs.

B. **Incorrect** - ONLY APD is tripped due to **PTTI**; however, APD does not provide an input into CWP circuit to prevent outward movement of CEAs.

C. **Correct** - BOTH of these trips occur from **PTTI** and each provides 1 of 2 required pre-trips to CWP circuit to prevent outward movement of CEAs.

D. **Incorrect** - ONLY TM/LP is tripped on **PTTI**; however, it does not provide an input to CMI circuit for CEAs.

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Question 39 Info (Q97026)			
Topic:	Effect of PTTI occurring to CEAs		
Tier/Group:	2/1		
K/A Info:	012 Reactor Protection • K3 - Knowledge of the effect that a loss or malfunction of the RPS will have on the following: • K3.01 - CRDS		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine the effect on the Control Rod Drive System when the Power Trip Test Interlock occurs.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/ Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	1C05-ALM, Reactivity Control Alarm Panel SD-058, Reactor Protective System Description		
Comments:	None		

40. 012 - Reactor Protection (2.4.31)

During a Unit -1 power escalation IAW OP-3, the annunciator window “HI POWER TRIP RESET DEMAND” alarm is received on 1C05.

(1) What condition has caused the alarm to actuate?

(2) What are the consequences of taking **NO** actions for this alarm?

A. (1) Actual Reactor Power is 8.4% away from the Reactor Trip setpoint.
(2) A “POWER LVL HI CH PRE-TRIP” alarm will be received if Reactor Power is allowed to rise an additional 2.6%.

B. (1) Actual Reactor Power is 5.8% away from the Reactor Trip setpoint.
(2) A “POWER LVL HI CH PRE-TRIP” alarm will be received if Reactor Power is allowed to rise an additional 1.5%.

C. (1) Actual Reactor Power is 2.6% away from the Reactor Trip setpoint.
(2) A Reactor Trip will occur if Reactor Power is allowed to rise an additional 2.6%.

D. (1) Actual Reactor Power is 1.5% from the Reactor Trip setpoint.
(2) A Reactor Trip will occur if Reactor Power is allowed to rise an additional 1.5%.

Answer: C

Answer Explanation:

A. **Incorrect** - Examinee may not understand the Variable Overpower Trip (VOPT) setpoint and how it is measured with respect to current reactor power. 8.4% is the margin gained between reset and the new trip setpoint. A 2.6% rise from the last reset will not cause an alarm. See explanation for correct answer.

B. **Incorrect** - Power has to rise approximately 5.8% from the last VOPT reset for the “HI POWER TRIP RESET DEMAND” alarm to annunciate. A power rise of 2.6% will not give this alarm.

C. **Correct** - When the VOPT reset pushbutton is depressed the high power trip setpoint is increased to a power level that is approximately 8.4% higher than current power, As power continues to rise the “HI POWER TRIP RESET DEMAND” will annunciate at approximately 2.6% away from the trip setpoint with the pre-trip occurring approximately 1.5% away from the trip setpoint. If the VOPT setpoint is not reset, the reactor will trip.

D. **Incorrect** – The High power pre-trip alarm annunciates at approximately 1.5% away from the trip setpoint. 1.5% away from the trip setpoint. If the VOPT setpoint is not reset, the reactor will trip.

2012 NRC RO EXAM MASTER KEY

Question 40 (Q75291)			
Topic:	Significance of RPS alarm and NO action taken		
Tier/Group:	2/1		
K/A Info:	012 - Reactor Protection • 2.4 - Emergency Procedures / Plan • 2.4.31 - Knowledge of annunciator alarms, indications, or response procedures.		
RO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	Identify the source of the VOPT Reset Demand alarm, and determine the effect on plant if **NO** action taken.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/ Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2010 RPS, AOP-7H, Pwr Dist. Tech Specs (05/11)		
Technical references:	Alarm Manual 1C05 Alarm window D-12		
Comments:	None		

41. 013 - Engineered Safety Features Actuation System (K5.02)

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

Time 0

11 S/G Pressure is 860 PSIA
12 S/G Pressure is 860 PSIA
11 S/G Level is 0 inches
12 S/G Level is 0 inches

Time +1 Min

11 S/G Pressure is 856 PSIA
12 S/G Pressure is 740 PSIA
11 S/G Level is minus (-) 120 inches
12 S/G Level is minus (-) 175 inches

Time +2 Min 30 seconds

11 SG Pressure is 800 PSIA
12 SG Pressure is 740 PSIA
11 SG Level is minus (-) 100 inches
12 SG Level is minus (-) 180 inches

Assuming **NO** operator actions, what is the current status of Auxiliary Feed Water?

A. AFW is supplying 11 S/G ONLY

B. AFW is supplying 12 S/G ONLY

C. AFW is supplying neither S/G

D. AFW is supplying 11 & 12 S/G

Answer: D

Answer Explanation:

A. **Incorrect** - AFAS has initiated based on 12 S/G levels below -170 inches for > 20 seconds and AFAS block did occur to 12 S/G initially but has cleared since block valves remain in AUTO and reopen when condition no longer exists. Thus AFW is supplying BOTH S/Gs.

B. **Incorrect** - AFAS has initiated and AFW is being supplied to both S/Gs.

C. **Incorrect** - Conditions for generating an AFAS have lasted for 30 seconds and AFW is being supplied to both S/Gs.

D. **Correct** - Based on timeline, AFAS has initiated and AFW is supplying both S/Gs.

2012 NRC RO EXAM MASTER KEY

Question 41 (Q92255)			
Topic:	AFAS / AFAS Block with NO operator action		
Tier/Group:	2/1		
K/A Info:	013 Engineered Safety Features Actuation • K5 - Knowledge of the operational implications of the following concepts as they apply to the ESFAS: • K5.02 - Safety system logic and reliability		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	Explain the initiating plant conditions and predict the AFAS response actions for the following: • AFAS Start • AFAS Block		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2006 RO (06/08)		
Exam Bank History:	LOI-2008 SRO Audit (05/10)		
Technical references:	1C03-ALM, Condensate and Feedwater Control Alarm Manual EOP-0, Post Trip Immediate Actions		
Comments:	None		

42. 013 - Engineered Safety Features Actuation System (A3.01)

ESFAS channel ZD sensor module for SIAS CP (Containment Pressure) is erratic. The sensor channel has been bypassed for troubleshooting. During troubleshooting I&C technicians remove the SIAS CP module from the sensor cabinet.

Which **ONE** of the following is the effect on ESFAS when this occurs?

A. The SIAS CP is no longer bypassed and each logic cabinet receives a trip input signal.

B. The SIAS CP is no longer bypassed and ONLY logic cabinet A receives a trip input signal.

C. The SIAS CP trip remains bypassed preventing a trip input signal from being sent to each logic cabinet.

D. The SIAS CP is no longer bypassed and ONLY logic cabinet B receives a trip input signal.

Answer: A

Answer Explanation:

A. **Correct** – The bypass key only works as long as sensor module keeps continuity of circuit. Since module withdrawn, power path is broken and trip input signal is sent to each logic cabinet for SIAS CP.

B. **Incorrect** - First part is true, however, each logic cabinet receives a SIAS CP trip input signal.

C. **Incorrect** - Even though bypass key is still installed, the power to circuit was removed thus allowing a SIAS CP trip input signal sent to each logic cabinet.

D. **Incorrect** - First part is true, however, each logic cabinet receives a SIAS CP trip input signal.

2012 NRC RO EXAM MASTER KEY

Question 42 (Q97027)			
Topic:	ESFAS Sensor Module Maintenance Bypass Circuit		
Tier/Group:	2/1		
K/A Info:	013 - Engineered Safety Features Actuation (ESFAS) • A3 - Ability to monitor automatic operation of the ESFAS including: • A3.01 - Input channels and logic		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the operation of ESFAS that includes: • Sensor module maintenance bypass channel circuit		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	OI-34, ESFAS Fig. 1, Sensor Maintenance Bypass Circuit		
Comments:	None		

43. 022 - Containment Cooling (A1.01)

Given the following conditions on Unit-1 at 100% power:

- Containment Cooling System is in a normal lineup with ALL Containment Air Coolers (CACs) available
- 11, 12, and 13 CAC Fans operating in FAST speed
- 11 CAC Emergency SRW Outlet valve is open

An event occurs resulting in a Reactor Trip with the following conditions:

- All equipment functions as designed upon the trip
- RCS pressure is 1910 PSIA and lowering
- Containment pressure 0.7 PSIG and rising
- Containment humidity for Dome and Rx Cavity are respectively 38% and 52% and both rising
- Containment temperature is 110 °F and rising

Which **ONE** of the following describes the required operation of the Containment Air Coolers for Containment Environment Safety Function in EOP-0?

A. Start 14 CAC in FAST and ensure open ALL CAC Emergency SRW Outlet valves.

B. Start 14 CAC in FAST with ALL CAC Normal SRW Outlet valves open.

C. Open the Emergency SRW Outlet valves on 12 and 13 CACs.

D. No additional manipulation of the CACs is required.

Answer: A

Answer Explanation:

A. **Correct** - Since containment pressure is degrading, alternate actions of EOP-0 require that ALL CACs be started and the Emergency SRW Outlet valves opened.

B. **Incorrect** - First part is required action, however, the Emergency SRW Outlet valves are opened to assist in lowering pressure and temperature.

C. **Incorrect** - EOP-0 specifically states ensure open ALL CAC Emergency SRW Outlet valves for containment pressure > 0.7 psig or containment temperature > 120 °F.

D. **Incorrect** - Taking no actions does not meet expectations of EOP-0 based on parameter trends.

Question 43 (Q74583)			
Topic:	Containment Air Cooler operation in EOP-0		
Tier/Group:	2/1		
K/A Info:	022 - Containment Cooling • A1 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: • A1.01 - Containment temperature		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the purpose of each of the safety function boxed steps of EOP-0.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2008 RO Audit (11/08)		
Technical references:	EOP-0 Containment Environment Safety Function		
Comments:	None		

44. 026 - Containment Spray (A3.02)

Following a Unit-2 plant trip from 100% power, a LOCA has occurred. CIS and SIAS have actuated.

Which condition represents the response of the Component Cooling (CC) system valves and equipment to the current ESFAS signals? Consider **ONLY** the Component Cooling side of the system.

A. Each CC HX outlet valve opens, the CC containment isolation valves shut, and all CCW pumps start.

B. Each CC HX outlet valve opens, each SDC HX inlet valve opens, and 21 and 22 CC pumps start.

C. Each SDC HX outlet valve opens, the CC containment isolation valves shut, and all CC pumps start.

D. Each SDC HX outlet valve opens, the CC containment isolation valves shut, and 21 and 22 CC pumps start.

Answer: D

Answer Explanation:

A. **Incorrect** - ONLY the CC containment isolation valves response is correct. ALL CC pumps receive a SIAS start signal but 13 (23) pump only starts if 12or 22 CC Pp does not start within one second after receiving a SIAS.

B. **Incorrect** - The CC HX outlet valves do NOT receive any ESFAS signal but remaining response is correct.

C. **Incorrect** - First two responses are correct, and third response as stated in A above does not occur.

D. **Correct** - All 3 responses are expected actions of CC upon a SIAS and CIS.

2012 NRC RO EXAM MASTER KEY

Question 44 (Q20370)			
Topic:	CC system response to SIAS and CIS		
Tier/Group:	2/1		
K/A Info:	026 - Containment Spray • A3 - Ability to monitor automatic operation of the CSS, including: • A3.02 - Verification that cooling water is supplied to the containment spray heat exchanger		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine the response on CCW system when a SIAS, CIS, and CSAS occur		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2008 RO Audit (11/08)		
Technical references:	EOP Attachment 2 page 4 of 5 EOP Attachment 4 page 1 of 2 EOP Attachment 6		
Comments:	None		

45. 026 - Containment Spray (2.1.28)

Given a total loss of Component Cooling, which of the following actions ensures core cooling is maintained during a LOCA with Recirculation Actuation Signal (RAS) in progress?

A. Secure one HPSI pump, align one Containment Spray pump for injection, and throttle HPSI flow to minimum allowed per EOP attachment.

B. Align one Containment Spray Pump for injection and stop ALL running HPSI pumps.

C. Stop ALL running HPSI pumps, start a LPSI pump using RAS override for injection and THEN throttle LPSI flow to minimum allowed per EOP attachment.

D. Align BOTH Containment Spray Pumps for injection and stop ALL running HPSI pumps.

Answer: B

Answer Explanation:

A. **Incorrect** - Securing one HPSI pump with NO CC does not protect remaining HPSI pump from overheating. The Containment Spray pump can operate without CCW flow as ECCS pump room air coolers provide SW cooling into room. Flow may be throttled through LPSI header valves.

B. **Correct** - These are required actions when CCW flow cannot be restored during RAS per EOP-5 Block Step S.1.f.2

C. **Incorrect** - Use of a LPSI pump during RAS is not allowed as the large flow initially may lead to clogging the sump screens with debris resulting in loss of NPSH for other pumps taking suction from the sump.

D. **Incorrect** - Aligning both spray pumps for safety injection would lower flow to cool the containment possibly preventing a lowering of containment temperature and pressure.

2012 NRC RO EXAM MASTER KEY

Question 45 (Q97028)			
Topic:	Containment Spray Pp purpose during LOCA		
Tier/Group:	2/1		
K/A Info:	026 Containment Spray • 2.1 - Conduct of Operations • 2.1.28 - Knowledge of the purpose and function of major system components and controls.		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given EOP-5 implemented, verify RAS actions.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New		
Exam Bank History:	None		
Technical references:	EOP-5 Step S.1.f.2 and Technical Bases document		
Comments:	None		

46. 039 - Main and Reheat System (K3.06)

Unit-1 was operating at 100% power when a loss of 1Y10 occurs and the Main Turbine trips due to loss of vacuum.

How should the ADVs/TBVs respond *immediately* upon the reactor trip?

A. ADVs ramp open, TBVs ramp open

B. ADVs quick open, TBVs ramp open

C. ADVs ramp open, TBVs remain shut

D. ADVs quick open, TBVs remain shut

Answer: D

Answer Explanation:

A. **Incorrect** - A loss of vacuum tripping the main turbine also makes TBVs inoperable. ADVs initially quick open upon trip from 100% power.

B. **Incorrect** - ADVs quick open, however, as stated in A TBVs are inoperable due to loss of vacuum.

C. **Incorrect** - First part is wrong and TBVs remain shut upon the reactor trip.

D. **Correct** - This is response to trip at 100% power with a loss of vacuum that trips the main turbine.

2012 NRC RO EXAM MASTER KEY

Question 46 (Q24658)			
Topic:	ADV/TBV response on loss of vacuum and 1Y10		
Tier/Group:	2/1		
K/A Info:	039 - Main and Reheat Steam (MRSS) • K3 - Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: • K3.06 - SDS		
RO Importance:	2.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	Evaluate ADV/TBV operation upon Loss of 1Y10 and loss of vacuum causing a reactor trip.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2006 RO/SRO Audit Remediation (05/08)		
Technical references:	AOP-7G-1, Loss of Vacuum		
Comments:	None		

47. 061- Auxiliary/Emergency Feedwater (A2.07)

Given the following:

- A loss of Service Water resulted in a Unit -1 trip and loss of the Instrument Air compressors 30 minutes ago.
- 13 AFW Pump is unavailable

(1) How is the AFW system affected and;

(2) What operator actions are required to maintain Steam Generator levels?

A. (1) The operating AFW pump trips on overspeed;
(2) Adjust the local speed adjust knob to minimum, reset the overspeed trip device, raise AFW Pump discharge pressure to 100 PSI above S/G pressure.

B. (1) The operating AFW pump speed will rise to the maximum governor setting;
(2) Adjust the local speed adjust knob to maintain AFW Pump discharge pressure 100 PSI greater than S/G pressure.

C. (1) The operating AFW pump speed will lower to the minimum governor setting;
(2) Adjust the AFW Pump Speed Controller, at 1C04, to obtain the desired AFW flow rate.

D. (1) S/G levels rise due to the flow control valves failing open;
(2) Align the Liquid N2 System to supply S/G FLOW CONTR valves via the AFW System Air Accumulators.

Answer: B

Answer Explanation:

A. **Incorrect** - The AFW Pump(s) run up to max speed, they do not trip. Actions taken would be correct if AFW Pump(s) did trip.

B. **Correct** - Effect of loss of I/A is as noted and AOP-7D provides direction to perform actions to locally control AFW Pump speed

C. **Incorrect** - AFW pump speed goes to maximum due to the loss of I/A. The AFW Pump Speed Controller at 1C04 has no effect on AFW Pp speed due to the loss of I/A. Examinee may think AFW Pp speed control (governor) is supplied by the AFW air accumulators that provide a source of air to other AFW components in an extended loss of Instrument Air situation.

D. **Incorrect** - The AFW Flow Control CVs will not fail open due to being supplied air via the AFW air accumulators (good for a minimum of 2 hours). S/G level would be controlled by maintaining AFW Pp speed 100 PSI above S/G pressure. Controlling FCVs thru use of liquid N_2 is directed by EOP-7, Station Blackout which assumes the AFW air accumulators have been depleted.

2012 NRC RO EXAM MASTER KEY

Question 47 (Q14608)			
Topic:	Effects to Unit-2 AFW valves on loss of Instrument Air		
Tier/Group:	2/1		
K/A Info:	061 - Auxiliary Feedwater • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.07 - Air or MOV failure		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given a loss of any 125 VDC Vital Bus, evaluate the effect on each unit and required actions.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	AOP-7D-2, Loss of Instrument Air and bases pages 7 and 12		
Comments:	None		

48. 062 - AC Electrical Distribution (K3.01)

Following a reactor trip, which **ONE** of the following bus losses would require operator actions to maintain the Core and RCS heat removal safety function per EOP-0?

A. 2Y09

B. MCC-107

C. 13B 480V Bus

D. 12 4KV Bus

Answer: B

Answer Explanation:

A. **Incorrect** - Loss of 2Y09 major effect would be ALL Low Pressure Feedwater Heater High Level Dumps fail open and challenge MFW operation when operating at power. Since the reactor has tripped these high level dumps receive a signal to open on the trip and loss of 2Y09 during EOP-0 has little affect on MFW thus will not challenge the Core and RCS heat removal safety function.

B. **Correct** - MCC-107 lost results in tripping off ALL Unit-1 Circ Water Pumps. This leads to a loss of vacuum and a trip of the SGFPs and loss of Turbine Bypass Valves. Initiation of AFW will be the alternate action necessary in EOP-0 for Core and RCS Heat Removal and ADVs will be used to control RCS temperature.

C. **Incorrect** - 13B 480V bus loss will result in the loss of MCC-116. This will result in potential loss of 2 Condensate pumps. A single Condensate pump will be able to support MFW requirements and AFW will not be needed in EOP-0.

D. **Incorrect** - The Main Feed system continues to operate, just at a reduced capacity as only 1 Condensate pump and 2 Condensate Booster pumps have been lost.

2012 NRC RO EXAM MASTER KEY

Question 48 Info (Q51239)			
Topic:	AC Electrical Distribution – Bus loss causing LONHR		
Tier/Group:	2/1		
K/A Info:	062 - AC Electrical Distribution • K3 - Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: • K3.01 - Major system loads		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given an electrical bus malfunction, diagnose the event and take appropriate actions per AOP-7I.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOR 11-6B Biennial Exam (11/11)		
Technical references:	AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power EOP-0, Post Trip Immediate Actions		
Comments:	None		

49. 063 - DC Electrical Distribution (A1.01)

Given the following conditions on Unit-1:

- A Station Blackout is in progress.
- EOP-7, Station Blackout, has been implemented.

Which **ONE** of the following describes why the Plant Computer Inverter, 1Y05A, is deenergized?

A. Removes a large DC load from 11 DC Bus allowing the bus to meet the 4 hour discharge rate.

B. Removes a large DC load from 12 DC Bus allowing the bus to meet the 4 hour discharge rate.

C. Removes a large DC load from 12 DC Bus allowing the bus to meet the 2 hour discharge rate.

D. Removes a large DC load from 11 DC Bus allowing the bus to meet the 2 hour discharge rate.

Answer: B

Answer Explanation:

A. **Incorrect** - 12 DC Bus has minimal load on it during normal operation. With SBO occurring, the load does not change. 1Y05A is not powered from 11 125V DC Bus.

B. **Correct** - Per EOP-7 Step J basis. Removing this load was identified during PRA that would allow the bus to be maintained for 4 hours just on battery.

C. **Incorrect** - Once again 12 DC Bus has minimal load on it during normal operation. Calculations performed verify that during a SBO each battery can carry required loads for at least one hour and most likely 4 hours.

D. **Incorrect** - Per UFSAR each station battery is designed to last at least 2 hours, however, EOP-7 states that removing this load will allow 12 DC Bus to meet a 4 hour discharge rate. 1Y05A is powered from 12 125V DC Bus.

2012 NRC RO EXAM MASTER KEY

Question 49 Info (Q74577)			
Topic:	Shedding Computer Inverter load during SBO		
Tier/Group:	2/1		
K/A Info:	063 - DC Electrical Distribution • A1 - Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: • A1.01 - Battery capacity as it is affected by discharge rate		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	STATE the electrical performance and design attributes of the 125 VDC, and 120 VAC Vital Busses.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2006 RO Audit (11/08)		
Technical references:	EOP-7, Station Blackout and Technical Bases		
Comments:	None		

50. 064 - Emergency Diesel Generator (K1.04)

Given a Loss of Offsite Power to both units, the following conditions exist:

- 1A Diesel Generator is out of service for maintenance
- 2B Diesel Generator did not load due to a faulted 4KV bus

Which **ONE** of the following statements is correct?

A. 11 DC bus is being supplied ONLY by 11 battery charger.

B. 21 DC bus is being supplied ONLY by 21 battery charger.

C. 12 DC bus is being supplied by 24 battery charger.

D. 22 DC bus is being supplied by 22 battery charger.

Answer: D

Answer Explanation:

A. **Incorrect** - 11 Bus will receive power from 23 battery charger. 11 Battery Charger is not available due to the unavailability of the 1A DG.

B. **Incorrect** - 21 battery charger is powered from 24A 480V Bus, which remains deenergized as the 2B DG did not load.

C. **Incorrect** - 24 battery charger is powered from 24B 480V Bus which remains deenergized as the 2B DG did not load.

D. **Correct** - 22 battery charger is powered from 21B 480V Bus which is reenergized from 2A Diesel Generator.

2012 NRC RO EXAM MASTER KEY

Question 50 Info (Q97030)			
Topic:	Emergency DG and DC busses		
Tier/Group:	2/1		
K/A Info:	064 - Emergency Diesel Generator • K1 - Knowledge of the physical connections and/or cause/effect relationships between the ED/G system and the following systems: • K1.04 - DC distribution system		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the purpose of each of the safety function boxed steps of EOP-0.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	LOI-2004 RO		
Exam Bank History:	None		
Technical references:	AOP-7I-1 & 2, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power		
Comments:	Never put into bank		

51. 073 - Process Radiation Monitoring (K4.01)

Given the following conditions:

- 1-RIC-4095 operating as a substitute for 1-RIC-4014 per OI-8A
- S/G Blowdown is discharging to Unit-1 Circ Water
- The Blowdown Recovery HI-TEMP DUMP, 1-BD-4088-CV, is shut
- Annunciator window "UNIT 1 S/G B/D RECOVERY" has just alarmed at 1C22H due to HIGH alarm setpoint exceeded

Which **ONE** of the following reflects the response of the SG Blowdown system? (Assume ***NO*** operator action)

	B/D REC DISCH TO COND, 1-BD-4096-CV	B/D REC DISCH TO CW, 1-BD-4015-CV	B/D REC DISCH TO MWS, 1-BD-4097-CV	11(12) SG BOT B/D CNTMT ISOLs, 1-BD-4011-CV 1-BD-4013-CV
A.	Shut	Shut	Open	Shut
B.	Open	Open	Shut	Open
C.	Shut	Shut	Open	Open
D.	Open	Shut	Shut	Shut

Answer: C

Answer Explanation:

A. **Incorrect** - First 3 responses are correct based on alarm actions and system lineup. SG Bottom BD valves must be manually closed when 1-RIC-4095 is substituting for 1-RIC-4014 per OI-8A Note for Precaution 5.0E.

B. **Incorrect** - The RMS still provides a close signal to control circuit for each valve preventing operator from opening valves unless placed in RAD TRIP Override.

C. **Correct** - This is the correct response of system valves with the given system lineup. The operator must manually shut the SG Bottom BD valves since automatic actions to close occur only from RIC-4014 which is OOS.

D. **Incorrect** - Only BD recovery discharge CW response is correct. All others are wrong. BD does not transfer to Condenser from Circ Water upon a high RMS condition.

Question 51 (Q97031)			
Topic:	S/G Blowdown response upon RMS alarm		
Tier/Group:	2/1		
K/A Info:	073 Process Radiation Monitoring • K4 - Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: • K4.01 - Release termination when radiation exceeds setpoint		
RO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine the response to S/G Blowdown system valves upon 1-RIC-4095 high alarm		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/ Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2010 1C22/1C34 exam (09/11)		
Technical references:	OI-8A, S/G Blowdown System		
Comments:	Modified from Q24653 by adding response of S/G Blowdown valves on RIC-4095 high alarm		

52. 076 - Service Water (K2.01)

Which **ONE** of the following is the normal bus power alignment for 13 (23) SRW pumps?

A. 13 Pump - 14 Bus; 23 Pump - 24 Bus.

B. 13 Pump - 14 Bus; 23 Pump - 21 Bus

C. 13 Pump - 11 Bus; 23 Pump - 21 Bus

D. 13 Pump - 11 Bus; 23 Pump - 24 Bus

Answer: A

Answer Explanation:

A. **Correct** - These are the normal power alignments of the SRW pumps per OI-27C. Both of these pumps are capable of being aligned to either 4KV safety related bus when required by using disconnects.

B. **Incorrect** - Pump breaker power alignment is wrong for 23 SRW pump per OI-27C. Both of these pumps are capable of being aligned to either 4KV safety related bus when required by using disconnects.

C. **Incorrect** – Bus alignments for 13 and 23 are to 14 and 24 4KV busses respectively. Both of these pumps are capable of being aligned to either 4KV safety related bus when required by using disconnects.

D. **Incorrect** - 13 SRW Pump breaker normal power alignment is wrong per OI-27C. Both of these pumps are capable of being aligned to either 4KV safety related bus when required by using disconnects.

Question 52 Info (Q97032)			
Topic:	Service Water Pump power supplies		
Tier/Group:	2/1		
K/A Info:	076 - Service Water System (SWS) • K2 - Knowledge of bus power supplies to the following: • K2.01 - Service water		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the power supply alignment of SRW pumps for each unit.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI2006 1C02 Exam (10/07)		
Technical references:	OI-27C, 4.16 KV SYSTEM		
Comments:	Modified from Q20452; removed reference to headers		

53. 004 - CVCS (2.4.49)

Given the following conditions:

- Unit-2 is in Mode 1 at 100% Reactor Power,
- An electrical perturbation occurs
- The CEAPDS monitor has deenergized as a result of the electrical perturbation

What is (1) the minimum bus lost and (2) the immediate stabilizing actions expected to be performed?

A. (1) 1Y01, (2) Verify the Pzr Lvl Ch Sel and Pzr Htr Lo Lvl Cutoff Sel switches in the "Y" Position, and reset the Proportional Htrs by placing the H/Ss to off and back to auto.

B. (1) 2Y01, (2) Verify the Pzr Press Ch Sel, RRS Ch Sel, Pzr Lvl Ch Sel and Pzr Htr Lo Lvl Cutoff Sel switches are in the "Y" Position.

C. (1) 2Y09, (2) Fast borate to reduce reactor power and promptly reduce Turbine load to restore T_{COLD} to program.

D. (1) 2Y10, (2) Align Chg Pp suction to the VCT, Reduce Turbine load to restore T_{COLD} to program and place two Chg Pps in PTL.

Answer: D

Answer Explanation:

A. **Incorrect** - Conditions given in the stem indicate a loss of 2Y10 as a minimum. These actions apply to the effects of a loss of 1Y01 on Unit-2.

B. **Incorrect** - Conditions given in the stem indicate a loss of 2Y10 as a minimum. These actions apply to a loss of 2Y01, they are not appropriate for a loss of 2Y10.

C. **Incorrect** - A loss of 2Y09 does require the immediate actions stated in this distracter because the Feedwater Heater HLDCVs fails open causing a reactor power excursion. However, the CEAPDS monitor is NOT deenergized on a loss of 2Y09.

D. **Correct** - Loss of CEAPDS indicates 2Y10, as a minimum, is deenergized. The "Immediate Actions" plaque states the Charging Pump suction shifts to the RWT with all Charging Pumps running and directs opening the VCT outlet MOV, shutting the RWT outlet to the Charging Pump suction, adjusting turbine load to maintain T_{COLD} on program and placing two Charging Pumps in PTL.

Question 53 Info (Q51200)			
Topic:	Loss of 2Y10 immediate actions		
Tier/Group:	2/1		
K/A Info:	004 - CVCS 2.4 - Emergency Procedures / Plan • 2.4.49 - Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.		
RO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Mentally develop a methodology for diagnosing electrical malfunctions in the Control Room by using key control board indications		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any NRC exam		
Exam Bank History:	LOI-2006 Panel Comp		
Technical references:			
Comments:	None		

54. 078 - Instrument Air (K4.03)

Unit-2 is in Mode 3 when an RCS depressurization event occurs causing Pressurizer pressure to lower to 1700 PSIA.

Which **ONE** of the following occurs based on this event?

A. The Saltwater Air Compressors (SWACs) start and will continue providing air to operate the TBVs.

B. IA Containment isolation, 2-IA-2080-MOV, shuts isolating control system air to containment components.

C. Instrument Air compressors trip on high Aftercooler or Intercooler temperature; Plant Air compressor trips on high discharge or first stage temperature.

D. The B/U IA HDR PCV TO U-2, 2-IA-6301-PCV, will open to supply the IA header from the IA Storage Tanks.

Answer: C

Answer Explanation:

A. **Incorrect** - SWACs do start on SIAS but do NOT provide air to TBVs.

B. **Incorrect** – Stated conditions do not support actuation of CIS which closes this valve. Instrument Air to containment will be supplied by the Unit -1 Plant Air Compressor once the Unit-2 Plant Air Compressor trips.

C. **Correct** – Stated conditions support actuation of SIAS which isolates SRW to turbine building and eventually these compressors trip on high temperature conditions listed.

D. **Incorrect** - 11 Plant Air Compressor will be supplying the U-2 Instrument Air header via the cross-connect valves. Instrument Air header pressure would not lower to the setpoint for opening 2-IA-6301-PCV (85 PSIG).

Question 54 (Q97033)			
Topic:	Loss of SRW to Compressed Air system due to SIAS		
Tier/Group:	2/1		
K/A Info:	078 Instrument Air K4 - Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: • K4.03 - Securing of SAS upon loss of cooling water		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Evaluate the long-term effect of a SIAS on the compressed air system.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2008 RO Audit (11/08)		
Technical references:	Alarm Response Manual 2C13		
Comments:	Modified from Q20286		

55. 001- Control Rod Drive (K4.23)

Under which **ONE** of the following conditions will a stop motion signal be supplied to the group programmer modules? (UCS/LCS = Upper/Lower Computer Stop)

A. ONLY during Manual Group mode withdrawal when highest CEA in group reaches UCS at 130.5 inches.

B. ONLY during Manual Sequential mode insertion when lowest CEA in group reaches LCS at 10 inches.

C. During Manual Sequential or Manual Group mode withdrawal when lowest CEA in group reaches UCS at 135.0 inches.

D. During Manual Sequential or Manual Group mode insertion when highest CEA in group reaches LCS at 6 inches.

Answer: D

Answer Explanation:

A. **Incorrect** – Outward motion is terminated when the lowest (not highest) CEA, in the group, reaches 130.5 inches if selected to manual sequential <u>**or**</u> manual group mode.

B. **Incorrect** - Inward motion is terminated when the highest (not lowest) CEA, in the group, reaches 6 inches (vice 10 inches) if selected to manual sequential <u>**or**</u> manual group mode.

C. **Incorrect** - Outward motion is terminated when the lowest CEA, in the group, reaches 130.5 inches (not 135.0 inches) if either mode is selected. This is Upper Electrical Limit for reed switch indication.

D. **Correct** - Inward motion is terminated when the highest CEA, in the group, reaches 6 inches if either mode is selected.

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Question 55 (Q97034)			
Topic:	Rod Motion Inhibit		
Tier/Group:	2/2		
K/A Info:	001 – Control Rod Drive • K4 - Knowledge of CRDS design feature(s) and/or interlock(s) which provide for the following: • K4.23 - Rod motion inhibit		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:	During withdrawal or insertion, determine condition to stop CEA group motion when in manual sequential or manual group mode.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2010 1C07, AFW and AFAS exam (04/11)		
Technical references:	OI-42, CEDM System Operation OP-2, Plant Startup From Hot Standby To Minimum Load		
Comments:	Modified Q25785 to add variation of Manual Sequential and/or Manual Group to each distractor.		

56. 008 - Component Cooling Water System (A2.03)

The following conditions exist on Unit-1:

- 100% power with core burnup of 11,000 MWD/MTU
- 1-HIC-5206, 11 CC Hx Saltwater Flow Controller, output signal drifts from 8% to 12%

(1) Which **ONE** of the following is the plant response and (2) What action is required?

A. (1) Component Cooling HX outlet temperature rises causing RCS boron to lower and reactor power to rise;
(2) Place Letdown Hx Temp. Controller, 1-TIC-223, in MANUAL to maintain letdown temperature constant.

B. (1) Letdown HX outlet temperature rises causing RCS boron to rise and reactor power to lower;
(2) PLACE IX BYPASS, 1-CVC-520-CV, to BYPASS to stop the power reduction.

C. (1) Letdown HX outlet temperature lowers causing RCS boron to lower and reactor power to rise;
(2) PLACE IX BYPASS, 1-CVC-520-CV, to BYPASS to stop the power rise.

D. (1) Component Cooling HX outlet temperature lowers causing RCP seal pressure perturbations.
(2) Restore saltwater flow controller output signal to previous setting.

Answer: C

Answer Explanation:

A. **Incorrect** – CC Hx outlet temperature lowers which causes RCS boron to be lowered and raise reactor power. Appropriate action is to bypass the IXs to stabilize reactor power.

B. **Incorrect** – L/D outlet temperature lowers not raises and reactor power would rise not lower; Bypassing IXs will immediately terminate the reactivity addition.

C. **Correct** – This is the expected response to RCS Boron and power; bypassing IXs will immediately terminate the positive reactivity addition.

D. **Incorrect** – First part is correct. RCP seals will be affected due to increased flow thru CC Hx, however, the boron effect to the RCS is the immediate concern.

Question 56 (Q97035)			
Topic:	Temperature affects on CVCS IX resin		
Tier/Group:	2/1		
K/A Info:	008 - Component Cooling Water System • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations • A2.03 - High/low CCW temperature		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	EOP-0 Containment Environment Safety Function		
Comments:	None		

57. 002 - Reactor Coolant (K1.06)

PZR level is 10 inches below setpoint. If all systems are in **AUTO**, what should letdown flow be?

A. 0 GPM

B. 24 GPM

C. 30 GPM

D. 36 GPM.

Answer: C

Answer Explanation:

A. **Incorrect** - The Letdown Stop Valves would have to be shut for this value. Information in the stem does not support this conclusion.

B. **Incorrect** - The HIC has a flow limiter which prevents the letdown valves from closing below 30 gpm. Examinee may subtract RCP Bleedoff flow from minimum L/D flow to reach a total of 24 GPM.

C. **Correct** - The HIC has a flow limiter which prevents the letdown valves from closing below 30 gpm.

D. **Incorrect** - The HIC has a flow limiter which prevents the letdown valves from closing below 30 GPM. Examinee may add RCP Bleedoff flow to minimum L/D flow to reach a total of 36 GPM.

Question 57 (Q14530)			
Topic:	Interrelationship between RCS and CVCS		
Tier/Group:	2/2		
K/A Info:	002 - Reactor Coolant • K1 - Knowledge of the physical connections and/or cause-effect relationships between the RCS and the following systems: • K1.06 - CVCS		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Design of charging flow path to provide relief protection for REGEN HX.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	SD-41 CVCS		
Comments:	None		

58. 015 - Nuclear Instrumentation (A3.02)

Given the following:

- Reactor power is being raised from 50 to 100%
- T_{COLD} is on program
- The "Nuclear ΔT Power Ch Deviation"alarm is received.

Which **ONE** of the following actions is required to clear this alarm for the current power level?

A. Balance turbine load with reactor power.

B. Calibrate the Ex-core NI Channels.

C. Null the NI Pots to the Delta-T Pots.

D. Adjust the T_{COLD} Calibrate Pot.

Answer: B

Answer Explanation:

A. **Incorrect** - The stem statement identifies that T_{COLD} is on program meaning reactor power and turbine load are balanced for the current power.

B. **Correct** - The conditions specified in the stem of the question indicate the need, per the Alarm Manual, for calibration of the Excore NI Channels in accordance with OI-30, Nuclear Instrumentation.

C. **Incorrect** - Nulling NI Pots to ΔT Pots can only be performed when reactor power is < 30% per OI-30, Nuclear Instrumentation.

D. **Incorrect** - The T_{COLD} Calibrate Pot is not adjusted by Operations personnel.

2012 NRC RO EXAM MASTER KEY

Question 58 (Q20147)			
Topic:	NI alarm response		
Tier/Group:	2/2		
K/A Info:	015 Nuclear Instrumentation • A3 - Ability to monitor automatic operation of the NIS, including: • A3.02 - Annunciator and alarm signals		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No NRC Exam use		
Exam Bank History:	LOI-2010 RPS (05/11)		
Technical references:	OI-30, Nuclear Instrumentation 1C05-ALM, Reactivity Control Alarm Manual		
Comments:	None		

59. 017 - In-core Temperature Monitor (K6.01)

Which **ONE** of the following indicates a Core Exit Thermocouple (CET) input, to the Post Accident Monitoring System, has been bypassed?

A. A blue backlight and a "B" adjacent to the parameter.

B. Parameter will indicate with a magenta backlight.

C. A green "S" adjacent to the parameter.

D. Parameter will indicate with a "??".

Answer: A

Answer Explanation:

A. **Correct** - Per OI-1I, Post Accident Monitoring System, a blue backlight and a "B" adjacent to the parameter indicate a bypassed parameter.

B. **Incorrect** - Per OI-1I, Post Accident Monitoring System, failed parameters will indicate with a magenta backlight.

C. **Incorrect** - Per OI-1I, Post Accident Monitoring System, a green "S" indicates a substituted RVLMS Probe.

D. **Incorrect** – Per OI-1I, Post Accident Monitoring System, "??" indicates a parameter that is not valid due to insufficient data to support the parameter.

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Question 59 (Q97062)			
Topic:	PAMS operation with CETs bypassed		
Tier/Group:	2/2		
K/A Info:	017 In-Core Temperature Monitor • K6 - Knowledge of the effect of a loss or malfunction of the following ITM system components: • K6.01 - Sensors and detectors		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Determine how a bypassed CET is indicated by PAMS.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	OI-1I, Post Accident Monitoring System LOI-114-1-2, Post Accident Monitoring System (slide 35)		
Comments:	None		

60. 027 - Containment Iodine Removal (A4.01)

Given the following:

- Unit-1 reactor tripped due to a LOCA
- Containment pressure has reached 3.0 PSIG

Which **ONE** of the following describes Containment Iodine Removal Unit operation for existing plant conditions?

A. CIS starts ONLY 11 and 12 Iodine Removal Units.

B. CSAS starts ALL Iodine Removal Units.

C. SIAS starts ALL Iodine Removal Units.

D. CRS starts ONLY 11 and 12 Iodine Removal Units.

Answer: C

Answer Explanation:

A. **Incorrect** - ALL IRUs start on SIAS, not CIS. Both SIAS and CIS actuate at a Containment pressure of 2.8 PSIG.

B. **Incorrect** - All IRUs start on SIAS not CSAS. SIAS actuates at a Containment pressure of 2.8 PSIG. CSAS actuates at a Containment pressure of 4.25 PSIG. Stated conditions indicate a CSAS would not be actuated.

C. **Correct** - All IRUs start on SIAS. SIAS actuates at a Containment pressure of 2.8 PSIG.

D. **Incorrect** - All IRUs start on SIAS. CRS actuates based on high radiation as indicated on Containment Area Radiation Monitors. These monitors are only for refueling purposes and are disabled during normal power operation. CRS and starting Iodine Removal Units seem a logical fit if the examinee is unsure of the correct answer.

Question 60 (Q20669)			
Topic:	Containment IRU controls		
Tier/Group:	2/2		
K/A Info:	027 Containment Iodine Removal • A4 - Ability to manually operate and/or monitor in the control room: • A4.01 - CIRS controls		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the purpose of each of the safety function boxed steps of EOP-0.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2002 1C08, 09, and 10 (05/03)		
Technical references:	EOP Attachment 2		
Comments:	None		

61. 028 - Hydrogen Recombiner and Purge Control (K2.01)

Unit-2 has tripped from 100% due to a LOCA and loss of offsite power. The following conditions exist:

- The 0C DG was out of service prior to the trip
- The 2B DG had a start failure upon the loss of offsite power
- The Crew has implemented EOP-5

The CRS has directed the RO to perform the following action per EOP-5:

"**IF** hydrogen concentration can **NOT** be determined, **THEN** start the Hydrogen Recombiners per OI-41A, HYDROGEN RECOMBINERS."

Which Hydrogen Recombiner(s) have power available?

A. 21 and 22 Hydrogen Recombiners by tying MCCs.

B. 21 Hydrogen Recombiner.

C. 21 and 22 Hydrogen Recombiners.

D. 22 Hydrogen Recombiner.

Answer: B

Answer Explanation:

A. **Incorrect** - Examinee may believe these loads receive power from MCCs rather than 480V load centers. Tying MCCs together would be an action directed per EOP-5 if a single 4KV bus is lost. Hydrogen recombiners are **NOT** powered from MCC-204 or 214.

B. **Correct** - Hydrogen Recombiner 21 is only one available and powered from 480V bus 21B.

C. **Incorrect** - Examinee may recognize power supplies are correct, however, 22 is unavailable as 2B DG failed to start and reenergize 4KV Bus 24. Hydrogen Recombiner 22 is powered from 480V Bus 24B.

D. **Incorrect** - Hydrogen Recombiner 22 is unavailable as 2B DG has not repowered 4KV bus 24.

2012 NRC RO EXAM MASTER KEY

Question 61 (Q97037)			
Topic:	Hydrogen Recombiner Power Supplies		
Tier/Group:	2/2		
K/A Info:	028 Hydrogen Recombiner and Purge Control • K2 - Knowledge of bus power supplies to the following: • K2.01 - Hydrogen recombiners		
RO Importance:	2.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the power supplies to the hydrogen recombiners.		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2002 1C08, 09,10 Misc Remediation (06/03)		
Technical references:	AOP-7I, Section VIII, page 42 and Section XXVII, page 164		
Comments:	Modified from Q20688		

62. 045 - Main Turbine Generator (K5.17)

Unit-1 is recovering from a plant trip after extended full power operation (400 days).

- Reactor power is 30% and holding for NI Calibration
- No CEA motion or boration/dilution operations are in progress
- TBV Controller, 1-PIC-4056, is in auto and the setpoint is set at 900 PSIA
- Turbine Bypass Valve, 1-MS-3944-CV, has failed open

Which **ONE** of the following sets of actions is taken to stabilize the plant?

A. Insert CEAs, as necessary, to return Reactor power to the required value; Maintain turbine load constant and isolate the TBV to restore T_{COLD} to program.

B. Withdraw CEAs, as necessary, to maintain Reactor power; Maintain turbine load constant and isolate the TBV to restore T_{COLD} to program.

C. Insert CEAs, as necessary, to return Reactor power to the required value; Lower turbine load to restore T_{COLD} to program.

D. Withdraw CEAs, as necessary, to maintain Reactor power; Lower turbine load to restore T_{COLD} to program.

Answer: C

Answer Explanation:

A. **Incorrect** - Per AOP-7K, Overcooling Event in Mode One or Two, CEAs should be inserted, as necessary, to maintain reactor power constant (later in the core cycle) and the overcooling event is compensated for by adjusting turbine load.

B. **Incorrect** - Per AOP-7K, Overcooling Event in Mode One or Two, CEAs should be inserted, as necessary, to maintain reactor power constant (later in the core cycle) and the overcooling event is compensated for by adjusting turbine load.

C. **Correct** - Per AOP-7K, Overcooling Event in Mode One or Two, CEAs should be inserted, as necessary, to maintain reactor power constant (later in the core cycle) and the overcooling event is compensated for by adjusting turbine load.

D. **Incorrect** - Per AOP-7K, Overcooling Event in Mode One or Two, CEAs should be inserted, as necessary, to maintain reactor power constant (later in the core cycle) and the overcooling event is compensated for by adjusting turbine load.

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Question 62 (Q97038)			
Topic:	Main Turbine Generator and MTC relationship		
Tier/Group:	2/2		
K/A Info:	045 Main Turbine Generator K5 - Knowledge of the operational implications of the following concepts as the apply to the MT/B System: • K5.17 - Relationship between moderator temperature coefficient and boron concentration in RCS as T/G load increases		
RO Importance:	2.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Exam Bank History:	None		
Technical references:	AOP-7K, Overcooling Event in Mode 1 or Two		
Comments:	Modified version of Q92905.		

63. 011 - Pressurizer Level Control System (A1.01)

The selected Pressurizer Level control channel process variable fails low at 100% power.

Which **ONE** of the following describes the plant response? (Assume **NO** operator action is taken)

A. All heaters deenergize, letdown goes to minimum, standby charging pumps start, actual Pzr level / pressure rises and the reactor trips on High Pzr pressure.

B. All heaters energize, letdown goes to maximum, only the selected charging pump runs, actual Pzr level / pressure lowers and the reactor trips on TM/LP.

C. All heaters energize, letdown goes to minimum, actual Pzr level / pressure rise and the reactor trips on High Pressurizer Pressure.

D. All heaters deenergize; actual Pzr level / pressure lowers; the reactor trips on TM/LP.

Answer: A

Answer Explanation:

A. **Correct** - With the level controller failing low, PLCS would respond to an indicated level lower than set point. Letdown valves would throttle back to raise level to setpoint. All charging pumps would start on level deviation. All heaters would deenergize based on pressurizer level being less than 101". Pressurizer bubble would be compressed and RCS pressure will rise until RPS high pressure trip setpoint is reached.

B. **Incorrect** - heaters will not energize due to failed detector indicating less than 101 inches, letdown does not go to maximum, all charging pumps start heaters will deenergize, but pressure level rises.

C. **Incorrect** - heaters will not energize due to failed detector indicating less than 101 inches

D. **Incorrect** - heaters will deenergize, but pressure level rises.

2012 NRC RO EXAM MASTER KEY

Question 63 (Q14491)			
Topic:	Pressurizer Level Control Channel failure		
Tier/Group:	2/2		
K/A Info:	011 - Pressurizer Level Control System • A1 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR LCS controls including: • A1.01 – PZR level and pressure		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the operating range of the Containment Hi-Range Radiation Monitors and automatic actions occurring upon alarm setpoint exceeded.		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	None		
Exam Bank History:	LOI-2006 RO Remediation Audit (11/08)		
Technical references:	OI-35, Radiation Monitoring System; ARM 1(2) C10 annunciator window J-04.		
Comments:	Modified Q74600		

64. 075 - Circulating Water (2.1.32)

Which **ONE** of the following describes the reason why a Circulating Water Pump (CWP) handswitch is returned to AUTO and not held in the START position until starting current lowers to running current?

A. Holding the handswitch in START prevents the motor protective relay circuit from arming and immediately reopens the breaker.

B. Holding the handswitch in START prevents the motor protective relay circuit from arming and the only protection is an overcurrent trip.

C. Holding the handswitch in START prevents the starting current from dissipating and causes the motor to trip on overcurrent.

D. Holding the handswitch in START prevents the charging spring motor from recharging to allow closing breaker upon subsequent starts.

Answer: B

Answer Explanation:

A. **Incorrect** - First part of statement is true, however, breaker does not trip open immediately.

B. **Correct** - Per OI-14A, Caution on page 14, prior to starting any CWP this is stated.

C. **Incorrect** - Starting current will dissipate if handswitch held in START, it does not remain once pump is started. If held in start, only motor overcurrent protection is active to trip breaker open.

D. **Incorrect** - This does not prevent charging spring motor from recharging. Once breaker is closed the charging spring motor resets for next closing operation.

2012 NRC RO EXAM MASTER KEY

Question 64 (Q97040)			
Topic:	Circulating Water		
Tier/Group:	2/2		
K/A Info:	075 - Circulating Water • 2.1 - Conduct of Operations • 2.1.32 - Ability to explain and apply system limits and precautions.		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	Apply all system limits (cautions and notes) and precautions when starting or stopping a Circulating Water Pump.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	New question		
Exam Bank History:	None		
Technical references:	OI-14A, Circulating Water System, Section 6.1.B page 14.		
Comments:	None		

65. 035 - Steam Generator (K3.01)

A reactor trip has occurred from full power on Unit-2. The following conditions exist:

- Reactivity Control is complete.
- Pressurizer level has stabilized at 120".
- No automatic ESFAS actuations have occurred.
- RCS pressure is 1710 PSIA and slowly decreasing.
- Both SG levels are -150" and decreasing.
- SG pressures are 785 PSIA
- T_{COLD} is 516 ˚F and lowering

Which **ONE** of the following sets of operator actions is required?

A. Manually initiate SIAS, trip 2 RCPs, and shut the MSIVs.

B. Manually initiate SIAS, SGIS, and trip all RCPs.

C. Manually initiate SIAS, CIS, and AFAS.

D. Block SIAS, throttle AFW flow, and shut the MSIVs.

Answer: A

Answer Explanation:

A. **Correct** - SIAS should have been initiated by 1725 PSIA, per EOP-0, 2 RCPs are tripped after verifying SIAS.

B. **Incorrect** - RCS pressure is high enough to support 2 RCPs running per Attachment 1 and SGIS is not required to initiate above a S/G pressure 685 of PSIA.

C. **Incorrect** - AFAS setpoints are not challenged and there is no information to support initiating CIS.

D. **Incorrect** - SIAS should not be blocked in EOP-0; although, not stated, it is it is inferred that the conditions are shortly after the trip. If in an Optimal Recovery procedure, there are steps to block SIAS prior to actuation. Also, with S/G levels dropping throttling AFW should not be accomplished at this point.

2012 NRC RO EXAM MASTER KEY

Question 65 (Q25069)			
Topic:	Operator actions for SLB		
Tier/Group:	2/2		
K/A Info:	035 - Steam Generator • K3 - Knowledge of the effect that a loss or malfunction of the S/Gs will have on the following: • K3.01 - RCS		
RO Importance:	4.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any NRC exam		
Exam Bank History:	2010 LOR Session 2 quiz		
Technical references:	EOP-0, Post Trip Immediate Actions EOP-4, Excess Steam Demand Event		
Comments:	None		

66. Conduct of Operations (G2.1.3)

Which **ONE** of the following is required if relief for a brief period is necessary when performing the duties of a "Dedicated Operator" in the Control Room assigned by the Shift Manager (SM)?

A. Any licensed operator on watch in Control Room may relieve following a verbal brief by the "Dedicated Operator" on status of the evolution in progress and any special conditions that may require attention or action during "Dedicated Operators" absence.

B. A license candidate on training watch who attended the pre-job brief may relieve with SM permission after being verbally briefed on any special conditions that may require attention or action during the "Dedicated Operators" absence.

C. The Dedicated SRO who attended the pre-job brief for evolution in progress, received a verbal brief by the "Dedicated Operator" on status of evolution in progress, and requires no "hands-on" operations during the "Dedicated Operators" absence.

D. Relieving individual attended the pre-job brief and has permission from the SM/CRS to relieve the "Dedicated Operator", received a verbal brief on the status of the evolution in progress and any special conditions that may require attention or action during absence, and have no concurrent duties.

Answer: D

Answer Explanation:

A. **Incorrect** - One of the requirements is that the relieving individual must have NO concurrent duties when relieving the Dedicated Operator for a brief period.

B. **Incorrect** - A trainee may never assume the role of "Dedicated Operator" and is not allowed to manipulate controls on boards independently; second part of statement is right as permission is needed by relieving individual from SM/CRS and verbal brief on any special conditions that may require attention or action during absence is part of requirement.

C. **Incorrect** - As stated before Dedicated SRO may not have any concurrent duties and may be required to perform "hands-on" manipulations as needed during "Dedicated Operators" brief absence.

D. **Correct** - This is what is required per NO-1-200 page 28 Section 5.2.B.3

Question 66 (Q97041)			
Topic:	Short term relief of Dedicated Operator		
Tier/Group:	3		
K/A Info:	2.1 - Conduct of Operations • 2.1.3 - Knowledge of shift or short-term relief turnover practices.		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the purpose of each of the safety function boxed steps of EOP-0.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	N/A		
Technical references:	NO-1-200 Page 28 Section 5.2.B.3		
Comments:	None		

67. 2.1 - Conduct of Operations (2.1.23)

The plant tripped from 100% power due to a LOCA. EOP-0 actions were taken and the crew transitioned to EOP-5, Loss of Coolant Accident.

The following conditions exist 2 hours after entry into EOP-5:

- SIAS, CIS, and CSAS were verified in EOP-0
- RAS has actuated and been verified
- Containment pressure is 3.0 PSIG and slowly lowering
- RCS pressure is 360 PSIA and slowly lowering
- RCS subcooling is 0°F
- ALL RVLMS lights are OUT
- Containment Wide Range Level indicates 50 inches and steady
- Both Containment Spray Pumps are running normally
- HPSI flow is throttled (and balanced) to the minimum allowed per EOP Att. 10, HPSI Flow
- 11 and 13 HPSI Pump current and flow are fluctuating

Per EOP-5, which of the following actions must be taken to stabilize HPSI flow?

A. Secure both Containment Spray Pumps.

B. Throttle HPSI injection flow.

C. Secure ONLY one Containment Spray Pump.

D. Secure one HPSI Pump and readjust HPSI flow to minimum allowed.

Answer: A

Answer Explanation:

A. **Correct** - Since HPSI flow is at the minimum, EOP-5 Step S.1.j.2 states secure BOTH spray pumps and THEN check for acceptable HPSI pump performance.

B. **Incorrect** - Throttling HPSI flow even more is NOT allowed as it is at the minimum required cooling flow for time since LOCA.

C. **Incorrect** - Securing only one pump does provide relief for HPSI pumps; however, the sump level is adequate and NOT the cause of cavitation. It is the sump screens becoming clogged.

D. **Incorrect** - Securing a HPSI pump would significantly reduce the flow to the vessel and since spray pumps are still operating, it would be more appropriate to secure both of these pumps before stopping a HPSI pump.

2012 NRC RO EXAM MASTER KEY

Question 67 (Q97043)			
Topic:	Conduct of Operations		
Tier/Group:	3		
K/A Info:	2.1 - Conduct of Operations • 2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.		
RO Importance:	4.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given a LOCA in progress, evaluate plant conditions and perform the required action to prevent HPSI pump cavitation		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	NEW		
Exam Bank History:	None		
Technical references:	EOP-5 Block Step S.1.j.2		
Comments:	Adapted from Millstone 2, 2008 NRC RO exam		

68. 2.1 - Conduct of Operations (2.1.43)

Which **ONE** of the following explains the reason for the difference between the required shutdown boron concentration for Mode 3/4 and Mode 5?

A. Less positive reactivity is inserted during a cooldown in Mode 3 or 4 than Mode 5 at EOC.

B. More positive reactivity is inserted during a cooldown in Mode 3 or 4 than Mode 5 at EOC.

C. Less negative reactivity is inserted during a cooldown in Mode 3 or 4 than Mode 5 at EOC.

D. More negative reactivity is inserted during a cooldown in Mode 3 or 4 than Mode 5 at EOC.

Answer: B

Answer Explanation:

A. **Incorrect** - On a cooldown, **more** NOT less positive reactivity is added in Mode 3 or 4 than Mode 5 at EOC due to cooldown from a steam line break which is most restrictive accident to challenge SDM. Mode 5 is below 200°F and no cooldown from a steam accident would occur.

B. **Correct** - On a cooldown, more positive reactivity is added in Mode 3 or 4 than Mode 5 at EOC due to cooldown from a steam line break which is most restrictive accident to challenge SDM. Mode 5 is below 200 °F and no cooldown from a steam accident would occur.

C. **Incorrect** - The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line *outside* containment, initiated at the end of core life. Positive NOT negative reactivity is added at EOC.

D. **Incorrect** - The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line *outside* containment, initiated at the end of core life. Positive NOT negative reactivity is added at EOC.

2012 NRC RO EXAM MASTER KEY

Question 68 (Q14303)			
Topic:	Generic – Conduct of Operations		
Tier/Group:	3		
K/A Info:	2.1 – Conduct of Operations • 2.1.43 - Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOI-2010 1C05 (02/11)		
Technical references:	NEOPs 13 (23) and Tech Spec Bases 3.1.1		
Comments:	None		

69. 2.2 - Equipment Control (2.2.4)

Per EOP-0, which **ONE** of the following sets of actions is performed if any Unit-1 MSR 2nd Stage Source MOV or Unit-2 MSR 2nd Stage Control valve fails to shut after the immediate actions have been performed? Assume NO loss of power has occurred.

A. For Unit-1: shut BOTH MSIVs;
For Unit-2: shut BOTH MSIVs

B. For Unit-1: place the MSR 2nd Stg Stm Source MOVs handswitch, 1-HS-4025 in the closed position;
For Unit-2: depress the RESET button on the MSR control panel.

C. For Unit-1, close the MSR 2nd Stage High Load MOVs and verify the MSR 2nd Stage Bypass Control valve panel loaders in manual with panel loader output at zero;
For Unit-2, shut the Main Steam Supply to the MSR 2nd Stage isolation valve.

D. For Unit-1, shut the appropriate Main Steam Supply to MSR 2nd Stage manual isolation valve;
For Unit-2, verify the MSR 2nd Stage bypass control valve panel loaders in manual with panel loader output at zero.

Answer: C

Answer Explanation:

A. **Incorrect** - These actions are performed for loss of power conditions, turbine speed not lowering, MTSV fails to close(U-1) and TV fails to close (U-2)

B. **Incorrect** – These are the immediate actions for each unit which have been performed as stated in the stem.

C. **Correct** - Per EOP-0, these are the correct actions to do per alternate actions for turbine trip.

D. **Incorrect** - This is action for Unit-2 not Unit-1; these are actions for Unit-1 not Unit 2. Both of these actions are a part of alternate actions response.

2012 NRC RO EXAM MASTER KEY

Question 69 (Q97045)			
Topic:	Generic 2.2 – Equipment Control		
Tier/Group:	3		
K/A Info:	2.2 – Equipment Control • 2.2.4 - (multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation and procedural actions between units at a facility.		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall how a Unit 1 and Unit 2 turbine trip are verified.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	NEW		
Exam Bank History:	None		
Technical references:	EOP-0 Unit 1 and Unit 2 Ensure Turbine Trip step B.3		
Comments:	None		

70. 2.2 - Equipment Control (2.2.42)

Unit-1 is in Mode 1 and the latest leakage reports are:

- 8.3 GPM - Pressurizer safety valve leakage
- 1.8 GPM - leakage past check valves from the RCS to the SI system
- 0.2 GPM - 12 Steam Generator primary-to-secondary leakage
- 10.9 GPM - total leakage

Which **ONE** of the following pairs of Technical Specification RCS leakage limits is exceeded?

A. Primary to Secondary leakage and Identified leakage.

B. Primary to Secondary leakage and Pressure Boundary leakage.

C. Identified leakage and Unidentified leakage.

D. Pressure Boundary leakage and Identified leakage.

Answer: A

Answer Explanation:

A. **Correct** - 12 S/G Primary to secondary leakage (0.2 GPM x 60 x 24 = 288 GPD) exceeds the T.S. limit of 100 GPD. Identified leakage is 10.3 GPM which is greater than the T.S. limit of 10 GPM.

B. **Incorrect** - 12 S/G Primary to secondary leakage (0.2 GPM x 60 x 24 = 288 GPD) exceeds the T.S. limit of 100 GPD; however no pressure boundary leakage exists. Tech Specs define Pressure Boundary leakage as “LEAKAGE (except primary to secondary LEAKAGE) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall”.

C. **Incorrect** - Identified leakage of 10.3 GPM is greater than the T.S. limit of 10 GPM. Total leakage of 10.9 GPM minus Identified leakage of 10.3 GPM = 0.6 GPM unidentified leakage which does not exceed the T.S. limit of 1 GPM unidentified leakage.

D. **Incorrect** - Tech Specs define Pressure Boundary leakage as “LEAKAGE (except primary to secondary LEAKAGE) through a non-isolable fault in an RCS component body, pipe wall, or vessel wall”. No Pressure Boundary leakage exists. Identified leakage of 10.3 GPM is greater than the T.S. limit of 10 GPM.

2012 NRC RO EXAM MASTER KEY

Question 70 (Q97044)			
Topic:	Equipment Control – Tech Spec entry conditions		
Tier/Group:	3		
K/A Info:	2.2 - Equipment Control • 2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	Given RCS leakage values, determine the leakage limits exceeded per tech spec LCO 3.4.13		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory/Fundamental	☒ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	LOR 11-6C Biennial Written exam (11/11)		
Technical references:	Unit-1, Tech Spec 3.4.13 and leakage definitions		
Comments:	Modified from Q92906		

71. 2.3 - Radiation Control (2.3.4)

Per CCNPP procedures, which **ONE** of the following would be the **first** threshold TEDE dose limit requiring an extension and required approval?

A. TEDE annual dose limit to exceed 1250, but not greater than 4,000 millirem/yr; Your Department Manager and GS

B. TEDE annual dose limit to exceed 2000, but not greater than 3,000 millirem/yr; GS-RP, your department Manager and GS.

C. TEDE annual dose limit to exceed 3,000, but not greater than 4,000 millirem/yr; GS-RP, your department Manager and GS;

D. TEDE annual dose limit to exceed 4,000, but not greater than 5,000 millirem/yr; GS-RP, your department Manager and GS; PGM, and VP-CCNPP

Answer: B

Answer Explanation:

A. **Incorrect** - This value is still below the first threshold of 2,000 mRem/yr to requiring an extension and approval.

B. **Correct** - Per Table 2 of RP-1-100, the first dose extension and approval is required when exceeding 2,000 mRem/yr

C. **Incorrect** - This would be the next threshold requiring an extension per Table 2; also approval of PGM is required. However, this includes dose from ALL sources (this applies for contractors and permanent personnel who worked at other nuclear sites).

D. **Incorrect** - This is the next threshold per Table 2 and it requires approval from VP-CCNPP in addition to the approvals to exceed 3,000 mRem/yr without exceeding the federal limit of 5,000 millirem/yr.

Question 71 (Q97045)			
Topic:	Radiation Control - Exposure Limits		
Tier/Group:	3		
K/A Info:	2.3 - Radiation Control • 2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions.		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	State whose approval is required to exceed CCNPP administrative dose limits.		
10 CFR Part 55 Content:	55.41(b)(12)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use on any exam		
Exam Bank History:	None		
Technical references:	RP-1-100, Radiation Protection Table 2		
Comments:	None		

72. 2.2 - Equipment Control (2.2.43)

In accordance with CNG-OP-1.01-2003, Alarm Response and Control, if one or more inputs to a multiple input alarm is out of service, the alarm will be designated with a …

A. Black Dot

B. Blue Dot

C. Red Dot

D. Yellow Dot

Answer: D

Answer Explanation:

A. **Incorrect** - Per CNG-OP-1.01-2003, Alarm Response and Control, a Black dot placed on an annunciator window is used to signify one of the following:

- A maintenance activity in the station that causes an alarm on a repeated basis.
- For identification of a locked in alarm that is caused by a current station configuration due to maintenance in the field or an Operations' lineup.
- For placement on alarm windows of nuisance alarms with the approval of the Control Room Senior Reactor Operator.

B. **Incorrect** - Per CNG-OP-1.01-2003, Alarm Response and Control, a Blue dot placed on an annunciator window is used to signify the associated annunciator window has been taken out of service.

C. **Incorrect** - Per CNG-OP-1.01-2003, Alarm Response and Control, a Red dot placed on an annunciator window is used to signify the associated component or annunciator window is part of a tagout.

D. **Correct** - Per CNG-OP-1.01-2003, Alarm Response and Control, a Yellow dot placed on an annunciator window is used to signify that one or more inputs to a multiple input annunciator are out of service.

Question 72 (Q55022)			
Topic:	What color dot indicates an input to a multiple input annunciator window is OOS		
Tier/Group:	3		
K/A Info:	2.2 - Equipment Control • 2.2.43 - Knowledge of the process used to track inoperable alarms.		
RO Importance:	3.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous use		
Exam Bank History:	LOI-2006 Audit Exam		
Technical references:	CNG-OP-1.01-2003, Alarm Response and Control		
Comments:	None		

73. 2.3 - Radiation Control (2.3.12)

As a licensed operator you have assumed the watch as the ABO. You are signed in on RWP-2, Operations Activities, including Fuel Shuffle, and Non-High radiation areas.

An emergency situation requires you to enter a locked high radiation area. No EAL classification thresholds have been met.

Which **ONE** of the following choices describes the requirements to gain access to the area?

A. Sign in under an Emergency Work Permit (EWP) and obtain RP coverage.

B. Enter the area under your current Radiological Work Permit (RWP) without RP coverage.

C. Obtain RP coverage and enter the area under your current RWP.

D. Sign in under the applicable EWP, RP coverage is not required if another operator is available.

Answer: C

Answer Explanation:

A. **Incorrect** – Emergency Work Permits are only used when EAL of Alert or higher is declared. They are used for plant equipment, lifesaving, and protecting large populations. EWPs are not used under routine operations.

B. **Incorrect** - RWP has the following contingency:
*EMERGENCY CONTINGENCY: In the event of an emergency, responders may enter any areas using this activity. Continuous RP coverage is required. Following closure of the emergency, responders may not enter the RCA without approval of RP Supervision.

C. **Correct** - RWP has the following contingency:
*EMERGENCY CONTINGENCY: In the event of an emergency, responders may enter any areas using this activity. Continuous RP coverage is required. Following closure of the emergency, responders may not enter the RCA without approval of RP Supervision.

D. **Incorrect** - Emergency Work Permits are only used when EAL of Alert or higher is declared. They are used for plant equipment, lifesaving, and protecting large populations. EWPs are not used under routine operations.

2012 NRC RO EXAM MASTER KEY

Question 73 (Q74533)			
Topic:	Generic 2.3 – Radiation Control		
Tier/Group:	3		
K/A Info:	2.3 - Radiation Control • 2.3.12 - Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	Apply the requirements of RP-1-100 for Locked High Radiation Access.		
10 CFR Part 55 Content:	55.41(b)(12)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No record of use		
Exam Bank History:	LOI-2008 Admin Comp (06/10)		
Technical references:	RWP-2		
Comments:	None		

74. 2.4 - Emergency Procedures / Plan (2.4.14)

Upon entry into an emergency operating procedure it becomes necessary to perform actions that are not contained within the controlling technical procedure and that are not parallel actions.

Which **ONE** of the following describes the minimum approval required to deviate from the emergency operating procedure?

A. At least 2 Senior Reactor Operators.

B. The Shift Manager AND the Control Room Supervisor.

C. The Shift Technical Advisor.

D. The Shift Manager.

Answer: D

Answer Explanation:

A. **Incorrect** - NO-1-201, Calvert Cliffs Operating Manual, specifies "Deviations shall be approved by the SM or the CRS in the absence of the SM".

B. **Incorrect** - NO-1-201, Calvert Cliffs Operating Manual, specifies "Deviations shall be approved by the SM or the CRS in the absence of the SM".

C. **Incorrect** - NO-1-201, Calvert Cliffs Operating Manual, specifies "Deviations shall be approved by the SM or the CRS in the absence of the SM" but not the STA.

D. **Correct** - NO-1-201, Calvert Cliffs Operating Manual, specifies "Deviations shall be approved by the SM or the CRS in the absence of the SM". "For deviations approved by the CRS, the CRS shall inform the SM as soon as practical".

2012 NRC RO EXAM MASTER KEY

Question 74 (Q45188)			
Topic:	2.4 – Emergency Procedures		
Tier/Group:	3		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.14 - Knowledge of general guidelines for EOP usage.		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	Apply the requirements of NO-1-201, Calvert Cliffs Operating Manual, for deviation from an approved procedure.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	No previous NRC Exam use		
Exam Bank History:	LOI-2010 Panel Comp (06/11)		
Technical references:	NO-1-201, Calvert Cliffs Operating Manual		
Comments:	None		

75. 2.4 - Emergency Procedures / Plan (2.4.29)

You are attending LOR training with your Ops crew when an Alert is declared by the Operating Crew.

The Shift Manager (SM) makes an announcement over the plant page system, directing all ERO members to report to their designated assembly areas.

At which **ONE** of the following locations should you assemble?

A. Assemble in the South Service Building Cafeteria.

B. Assemble in the Control Room behind the electrical panels.

C. Assemble outside the GS-Ops Training Office on 2nd floor of OTF.

D. Assemble in the pre-designated area in the OTF/NOF first floor hallway.

Answer: A

Answer Explanation:

A. **Correct** - Per ERPIP-317, this is where Operators in training will assemble, for an Alert declaration or higher, for accountability and assignment of tasks when directed by Control Room.

B. **Incorrect** - This is where "On-Shift" Operators would assemble, for an Alert declaration or higher, if not involved in actions to address emergency event in progress.

C. **Incorrect** - This is where Ops Training personnel assemble, for an Alert declaration or higher, if they do not have an assigned position in the ERO.

D. **Incorrect** - This would only be appropriate for non ERO personnel who have a regular work location within the protected area. Operators are considered part of the ERO when on site.

2012 NRC RO EXAM MASTER KEY

Question 75 (Q97048)			
Topic:	Training Crew Assembly Area for ERPIP declaration		
Tier/Group:	3		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.29 - Knowledge of the emergency plan		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	Recall the purpose of each of the safety function boxed steps of EOP-0.		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory/Fundamental	☐ Comprehension/Analysis	
Last NRC Exam used on:	Millstone 2, 2008 RO exam		
Exam Bank History:	No record of use on any exam		
Technical references:	ERPIP-317, Operations Team (OSC)		
Comments:	Question stem modified from Millstone 2, 2008 RO exam to reflect Calvert Cliffs emergency plan		