RO Question 1

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	007/EK3.01	
Importance Rating	4.0	4.6

K/A: **Reactor Trip – Stabilization** - Knowledge of the **reasons** for the following **responses** as they apply to the reactor trip: Actions contained in EOP for reactor trip.

Proposed Question:

Following a Reactor Trip without a Safety Injection, the operators are required to "Check Total AFW Flow – Greater than 435 gpm."

The reason for performing this step is to:

A. Verify adequate secondary heat sink.

B. Prevent an uncontrolled RCS cooldown.

C. Verify all AFW pumps are in operation.

D. Ensure Steam Generator U-tubes are covered.

Proposed Answer: A

Explanation:

Answer A is correct. EOP Background for E-0.1 states that "verifying feed flow to the steam generators ensures a secondary heat sink for decay heat removal."

Answer B is incorrect. Uncontrolled RCS cooldown is determined by previous step that checks RCS temperature is stable or trending to 547˚F.

Answer C is incorrect. Only one motor driven pump is required to be in operation to meet the 435 gpm requirement.

Answer D is incorrect. Ensuring the Steam Generator U-tubes are covered is a function of level (>6%), SG level is not required for a heat sink, 435gpm will meet this requirement.

Technical Reference(s): EOP E-0.1 Step 2. Rev. 30;
E-0.1 Background Step 2, Page 13, Rev. 2;
LPE-0 Reactor Trip and SI Response, Page 35, Rev. 9

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7920 – Explain the basis of emergency procedure steps.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 2

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	008/AK1.01	
Importance Rating	3.2	3.7

K/A: **Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)** - Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Thermodynamics and flow characteristics of open or leaking valves.

Proposed Question:

The Unit is stable at 100% power when Pressurizer PORV 474 starts to leak. The following plant conditions exist:

- Pressurizer pressure = 2235 psig
- PRT pressure = 12 psig.
- PRT temperature = 140°F

If conditions remain constant, what will the Pressurizer PORV tail pipe temperature indicate?

A. 140°F.

B. 212°F.

C. 245°F.

D. Off scale high at 400°F.

Proposed Answer: C

Explanation:

Answer A is incorrect. Is the PRT temperature at present so if the isenthalpic process is ignored this would be a value to consider.

Answer B is incorrect. This is the temperature consistent with boiling in a non pressurized system.

Answer C is correct as this is the saturation temperature of the PRT at present.

Answer D is incorrect but would be the correct answer if the temperature element was on the upstream side of the PORV.

Technical Reference(s): Steam Tables
STG A-4A Pressurizer, Pressure & Level Control, Page 3-9, Rev. 13

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

Learning Objective: 40738 - Apply fundamental topics associated with the Pressurizer, Pressure & Level Control System.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8
55.43 ____

10 CFR Part 55 Content: Components, capacity, and functions of emergency systems.

Comments:

RO Question 3

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	009/EK.2.03	
Importance Rating	3.0	3.3

K/A: **Small Break LOCA** - Knowledge of the interrelations between the small break LOCA and the following: S/Gs

Proposed Question:

The principle difference in plant response of a Large Break LOCA (LBLOCA) versus a Small Break LOCA (SBLOCA) is only the:

A. SBLOCA clears the loop seal.

B. LB LOCA results in core uncovery.

C. SBLOCA results in peak clad temperatures greater than 1200°F.

D. SBLOCA needs the heat removal capacity from the Steam Generators.

Proposed Answer: D

Explanation:

Answer A is incorrect. A small break LOCA will clear the loop seal as well as a Large Break LOCA.

Answer B is incorrect. Small Break LOCAs can result in core uncovery as well as Large Break LOCAs.

Answer C is incorrect. Small Break LOCAs can result in Peak Clad temperatures exceeding 1200°F as well as Large Break LOCAs.

Answer D is correct. Per Background for E-1, secondary heat sink is not required for Large Break LOCAs, secondary heat sink is required for Small Break LOCAs.

Technical Reference(s): EOP E-1 Background Step 3, Page 52, Rev. 2.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7920 -Explain basis of emergency procedure step.

Question Source: Bank # INPO 22377
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 4

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	011/EA2.13	
Importance Rating	3.7	3.7

K/A: **Large Break LOCA** - Ability to determine and interpret the following as they apply to a Large Break LOCA: Difference between overcooling and LOCA indications.

Proposed Question:

Which of the following parameters is used to differentiate between a Large Break LOCA and a Faulted Steam Generator inside containment?

A. Steam Generator WR level.

B. Containment Radiation.

C. Containment Pressure.

D. Containment Temperature.

Proposed Answer: B

Explanation:

Answer A is incorrect. Steam Generator WR level is not used at determine if the SG is faulted or if the RCS is intact.

Answer B is correct. Containment Radiation will not increase for a Faulted Steam Generator, but will increase for an RCS LOCA.

Answer C is incorrect. Containment pressure will rise for both a Large Break LOCA and a Faulted Steam Generator.

Answer D is incorrect. Containment temperature will rise for both a Large Break LOCA and A Faulted Steam Generator.

Technical Reference(s): LPE-0, Reactor trip and SI Response, Pages 23 & 24, Rev. 9

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5441 – State the symptoms of Major Accidents.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 5

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	015/017/AK2.10	
Importance Rating	2.8	2.8

K/A: **Reactor Coolant Pump (RCP) Malfunctions** - Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions and the following: RCP indicators and controls.

Proposed Question:

Given the following:

- Unit 1 is at 100% power
- RCS Pressure is 2235 psig.

Which of the following RCP seal parameters indicates a FAILURE of the # 2 Seal?

	#1 Seal DP (psid)	#1 Seal Leakoff Flow (gpm)	RCDT Level Trend
A.	1500	2.0	Trending Up
B.	1500	0.5	No Change
C.	2200	2.0	No Change
D.	2200	0.5	Trending Up

Proposed Answer: D

Explanation:

Answer A is incorrect. Symptoms indications are for a number 1 seal failure based on #1 seal DP at 1500 psid.

Answer B is incorrect. Symptoms indications are for a number 1 seal failure based on #1 seal DP at 1500 psid.

Answer C is incorrect. Symptom indications are incorrect in that number 1 seal flow would be low and RCDT level would be increasing.

Answer D is correct. Symptoms are correct, would expect normal number 1 seal DP, Low number 1 seal leakoff flow and increase in RCDT level.

Technical Reference(s): OP AP-28, Reactor Coolant Pump Malfunction, Rev. 4
STG A-6, Reactor Coolant Pump, Pages 3-8 & 3-9, Rev. 14
LAR-1, RCP Failures, Page 14 of 30, Rev. 9

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6137 – Explain effects of RCP seal failures.

Question Source: Bank # P-1608
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 6 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	022/2.4.31	
Importance Rating	4.2	4.1

K/A: **Loss of Reactor Coolant Makeup** - Emergency Procedures/Plan: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question:

Assume the Reactor Makeup system is in AUTOMATIC and NO operator action.

If VCT Level Transmitter LT-112 fails HIGH to 100%, then divert valve LCV-112A will:

A. Modulate open to the LHUTs, and charging pump suction will shift to the RWST on VCT LOW level.

B. Modulate open to the RWST, and the VCT LOW level alarm will annunciate in the control room.

C. Trip open to the LHUTs, and the VCT HIGH level alarm will annunciate in the control room.

D. Trip open to the RWST, and automatic blender operation will maintain VCT level.

Proposed Answer: C

Explanation:

Answer A is incorrect. High failure of LT-112 causes LCV-112A to trip open, not modulate open, modulation is controlled by LT-114. Charging pump suction swap is a function or low level and requires both transmitters to reach 5% VCT level.

Answer B is incorrect. High failure of LT-112 causes LCV-112A to trip open, not modulate open, modulation is controlled by LT-114, also flow goes to LHUT and not RWST. VCT Low Level will function off of LT-114 when actual level reaches 5%.

Answer C is correct. High failure of LT-112 causes LCV-112A to trip open, when indicated level reaches 87%. High Level alarm is driven from LT-112 and not LT-114 and alarms at 87% indicated level.

Answer D is incorrect. High failure of LT-112 causes LCV-112A to trip open, when indicated level reaches 87%. Flow goes to LHUT and not RWST. Automatic Blender operation is driven from LT-112 and auto makeup is prevented if indicated level is greater than 24%.

Technical Reference(s): OP AP-19, Malfunction of Reactor Makeup Control System, Page 2 and Appendix A, Rev. 4.
LPA-19, Malfunction of Reactor Makeup Control System, Page 6, Rev. 8.
OIM Page B-1-4, Rev. 26

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40449 - Discuss abnormal conditions associated with the CVCS.

Question Source: Bank # A-0087
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 7

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	025/AA1.11	
Importance Rating	2.9	3.0

K/A: **Loss of Residual Heat Removal System (RHRS)** - Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: Reactor building sump level indicators.

Proposed Question:

A large break LOCA has occurred, and the crew has transitioned to EOP E-1.3, “Transfer to Cold Leg Recirculation.”

Which of following indications would require a transition to EOP ECA-1.1, “Loss of Emergency Coolant Recirculation”?

A. Containment Recirculation Sump levels are less than 92.0 ft on both channels.

B. Containment Structure Sumps levels are off scale low.

C. Refueling Water Storage Tank Level is less than 4% on all channels.

D. Reactor Cavity Sump level off scale low.

Proposed Answer: A

Explanation:

Answer A is correct. EOP E-1.3 “Transfer to Cold Leg Recirculation”, Step 6.d. RNO states that if sump level is less than 92.0 ft then a transition to ECA 1.1 is warranted.

Answer B is incorrect. Containment structure sumps are for normal plant operation and air supply to the level detector will be isolated, causing low level indication.

Answer C is incorrect. RWST level of less than 4% requires that all pumps taking suction must be secured, RHR suction is not from the RWST at this time. No transition to required.

Answer D is incorrect. Reactor Cavity sump level is expected to be low and the air to the level detector is isolated, causing low level indication.

Technical Reference(s): EOP E-1.3, “Transfer to Cold Leg Recirculation”, Step 6.d., page 4, Rev. 25.
EOP E-1.3 Background, Page 19, Rev. 2
LPE-1C, Recirculation Modes and LOCA Outside Containment, page 11, Rev. 9.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3549 – Give the symptoms and/or the indications for a loss of emergency coolant Recirculation.

Question Source: Bank #
Modified Bank # P-6251 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 8 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	026/AA2.04	
Importance Rating	2.5	2.9

K/A: **Loss of Component Cooling Water (CCW)** - Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW.

Proposed Question:

Component Cooling Water (CCW) temperature is TRENDING UP due to fouling of the In-service ASW/CCW Heat Exchanger.

The Alternate ASW/CCW train is not available.

Which of the following parameters would require a manual Reactor Trip per the Abnormal Procedures?

A. Reactor Coolant Pump Radial Bearing outlet temperature is 232°F.

B. Reactor Coolant Pump Number 1 seal return temperature is 205°F.

C. CCW pump and motor bearings temperature is 150°F.

D. CCW supply temperature is 80°F.

Proposed Answer: A

Explanation:

Answer A is correct. Per AP-28 Fold out page, if pump radial bearing temperature exceeds 225°F, a reactor Trip is required.

Answer B is incorrect. Per AP-28 fold out page, if pump number 1 seal temperature exceeds 235°F a reactor trip is required.

Answer C is incorrect. Per AR PK01-09,"CCW Pumps" CCW Pumps can be run up a temperature of 180°F, without affecting the lifetime of the component.

Answer D is incorrect. Per AP-11, Section A, temperature limit for CCW system is 120°F, as long as one pump and heat exchanger are in service, then heat loads are reduced to minimize temperature increase.

Technical Reference(s): LPA-11, Malfunction of CCW System, Page 11, Rev. 10
AR PK01-09, CCW Pumps, Pages 3, Rev. 14
OP AP-11, Malfunction of CCW System, Section D. Page 14, Section A, pages 2 & 3, Rev. 23.
OP AP-28 Reactor Coolant Pump Malfunction, section E, Page 25 and Fold out page, Rev. 4.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3466 – Discuss the effects and actions associated with a loss of CCW.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 9 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	27/AA2.04	
Importance Rating	3.7	4.3

K/A: **Pressurizer Pressure Control (PZR PCS) Malfunction** - Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: Tech-Spec limits for RCS pressure.

Proposed Question:

Unit 1 is at 100% power, with all systems aligned for automatic operation.

Which of the following Pressurizer Channel malfunctions would place Unit 1 closer to the Departure from Nucleate Boiling (DNB) Tech Spec?

Assume no operator actions have occurred.

A. PT-456 (Backup) fails LOW.

B. PT-457 (Control) fails HIGH.

C. LT-459 (Control) fails LOW.

D. LT-459 (Backup) fails HIGH.

Proposed Answer: B

Explanation:

Answer A is incorrect. PT-456 (B) failing low will not cause a plant transient.

Answer B is correct. PT-457 (C) failing high will cause a plant transient. Control channel failure will cause a PORV PCV-474 to open, which drops Pressurizer Pressure to 2185 psig, PORV interlock pressure which closes the open PORV. Pressure will not increase due to the master controller sensing high pressure and preventing the Pressurizer Backup Heaters from energizing. DNB TECH SPEC Pressure is 2175 psig.

Answer C is incorrect. LT-459 (C) failing Low will cause a loss of letdown and an increase in charging flow, this will cause an increase in Pressurizer pressure.

Answer D is incorrect. LT-459 (B) failing High will not cause a plant transient.

Technical Reference(s): LTAA-3, Instrument and Control Failure Analysis, Pages 23 – 25, Rev. 9.
LA-4A , Pressurizer, Pressure & Level Control, Pages 34 & 71, Rev. 10.
AR PK05-17, PZR Pressure Low, Page 2, Rev. 11
Tech Spec 3.4.1, Reactor Coolant System, RCS Pressure, Temperature and Flow Departure from Nucleate Boiling (DNB) Limits, Page 3.4.1, Rev. 8.
COLR 1, COLR for Diablo Canyon Unit 1 Cycle 15, Page 5, Rev. 0.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 66040 – Discuss significant Technical Specifications and Equipment Control Guidelines associated with the Pressurizer, Pressure & Level Control System.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ____

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 10

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	038/EK3.04	
Importance Rating	3.9	4.1

K/A: **Steam Generator Tube Rupture (SGTR)** - Knowledge of the **reasons** for the following **responses** as they apply to the SGTR: Automatic actions provided by each PRM.

Proposed Question:

A Steam Generator Tube Rupture has been diagnosed on Steam Generator 1-1.

Which of the following identifies an automatic action for the radiation monitors designed to sense this condition and the reason for this response?

	Automatic Action	Reason
A.	Isolates Steam Jet Air Ejector Discharge.	Prevent the spread of contamination to the environment.
B.	Isolates the effected Steam Generator 10% dump valve.	Prevents the spread of contamination to the environment
C.	Closes SG Blowdown Sample Valves - Inside Containment.	Prevents the spread of contamination to the secondary plant.
D.	Closes SG Blowdown Sample Valves - Outside Containment.	Prevents spread of contamination to the secondary plant.

Proposed Answer: D

Explanation:

Answer A is incorrect. RE-15 Stream Jet Air Ejector Exhaust has no automatic actions.

Answer B is incorrect. RE-72 Main Steam Line Radiation Monitor has no automatic actions.

Answer C is incorrect. RE-19 or RE-23 will not isolate SG Blowdown inside containment.

Answer D is correct. RE-19 or RE-23 will isolate SG Blowdown outside containment to limit the contamination to the secondary plant caused by a SG Tube Rupture.

Technical Reference(s): LG 4-A, Radiation Monitoring System, page 54, Rev. 4
LPE3 SGTR response, Page 20, Rev. 10
EOP E-3 Background, Page 67, Rev. 2.
EOP E-3, Steam Generator Tube Rupture, Step 3, Page 6, Rev. 28.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8469 - Analyze automatic features and interlocks associated with the Radiation Monitoring System.

Question Source: Bank # ______
Modified Bank # INPO - 30387 (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 11

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	054/AK3.01	
Importance Rating	4.1	4.4

K/A: **Loss of Main Feedwater (MFW)** - Knowledge of the **reasons** for the following **responses** as they apply to the Loss of Main Feedwater (MFW): Reactor and/or turbine trip, manual and automatic.

Proposed Question:

Unit 1 is at 100% power when both Main Feedwater Pumps trip.

Which of the following responses should be taken and why?

	Response	Reason
A.	Manually trip the Reactor	Prevent a mismatch between Reactor Power and Secondary Heat Sink.
B.	Verify the Reactor has tripped.	Trip of Both Main Feedwater Pumps should have caused an automatic Reactor Trip.
C.	Manually trip the Main Turbine	Prevent a mismatch between Reactor Power and Secondary Heat Sink.
D.	Verify the Main Turbine has tripped.	Trip of Both Main Feedwater Pumps should have caused an automatic Turbine Trip.

Proposed Answer: A

Explanation:

Answer A is correct. This is an immediate action step from AP-15 associated with power being greater than 80% power. Loss of Feedwater leads to loss of heat sink for the reactor.

Answer B is incorrect. There is not an automatic reactor trip associated with the loss of both main Feedwater pumps.

Answer C is incorrect. Loss of Feedwater leads to loss of heat sink for the reactor. Reactor is tripped and not the turbine to reduce heat input to the RCS.

Answer D is incorrect. There is not an automatic turbine trip associated with the loss of both main Feedwater pumps.

Technical Reference(s): LPA-15, Loss of Feedwater Flow, Page 7, Rev. 11
OP AP-15, Loss of Feedwater Flow, Section A, Page 3, Rev. 18.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 9693 - State the steps and transitions in procedures that are considered immediate actions.

Comment [RLS1]: 9693

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 5
55.43 ______

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 12 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	055/EA1.06	
Importance Rating	4.1	4.5

K/A: **Loss of Offsite and Onsite Power (Station Blackout**) - Ability to operate and/or monitor the following as they apply to a Station Blackout: Restoration of power with one ED/G.

Proposed Question:

Following a loss of all AC power, Diesel Generator 1-2 was restored, and is now supplying 4 kV Bus G.

The Site Emergency Coordinator has approved implementation of EOP ECA-0.3, Restore 4 kV Buses, Appendix X, “Crosstie of a Vital Bus” to crosstie 4 kV Bus G to 4 kV Bus F.

During performance of the appendix, the 4 kV vital bus auto transfer reset pushbuttons are depressed, which of the following relays are reset as a result of this action?

A. Transfer to Diesel and Shed Load only.

B. Transfer to Startup, Transfer to Diesel and Sequential Starting of Loads.

C. Transfer to Startup, Shed Load and Sequential Starting of Loads.

D. Transfer to Diesel and Transfer to Startup only.

Proposed Answer: B

Explanation:

Answer A is incorrect. Transfer to Diesel relay is reset, but the shed load relay is not reset by this action.

Answer B is correct. Transfer to Startup relays, Transfer to Diesel relays and Sequential Starting of Loads relays are all reset when the pushbutton is depressed.

Answer C is incorrect. Transfer to Startup relays and Sequential Starting of Loads relays are reset, the shed load relay is not reset by this action.

Answer D is incorrect. Transfer to Diesel relays and Transfer to Startup relays are not the only relays reset, Sequential Starting of Loads relays are reset also.

Technical Reference(s): EOP ECA-0.3, Restore 4 kV Buses, Appendix X, Page 28, Rev. 14.
LJ-15, Electrical Power Transfer, Page 13, Rev. 4.

Proposed references to be provided to applicants during examination: None

Learning Objective: 3827 - Explain emergency operation of AC distribution systems.

Question Source: Bank # A-0660
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 13 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	058/2.2.37	
Importance Rating	3.6	4.6

K/A: **Loss of DC Power** - Equipment Control: Ability to determine operability and/or availability of safety related equipment.

Proposed Question:

The following conditions exist on Unit 1 due to electrical faults:

- Vital DC Bus 1-2 is de-energized.
- 480V bus G is de-energized.

What effect, if any, will this event have on the capability of the Turbine Driven Auxiliary Feed Pump 1-1 to feed all four Steam Generators?

A. AFW Pump 1-1 will NOT be capable of an auto start and all associated LCVs will be deenergized.

B. AFW Pump 1-1 steam supply valves, FCV-37 and FCV-38 will fail CLOSED, preventing an auto start.

C. AFW Pump 1-1 will be capable of an auto start, but all associated LCVs will fail CLOSED.

D. NO effect, AFW Pump 1-1 and associated LCVs will still function normally.

Proposed Answer: A

Explanation:

Answer A is correct. Appendix B of AP-23, “Loss of Vital DC Bus”, discusses the equipment lost based on the loss of Vital Bus 1-2, FCV-95 will not open, DC power supply and TDAFP LCV’s will be open and deenergized.

Answer B is incorrect. Steam supply valves FCV-37 and FCV-38 will fail as is on loss of power, normally open valves.

Answer C is incorrect. TDAFP will not auto start, no DC power available to open FCV-95.

Answer D is incorrect. TDAFP and LCV will both be affected by this loss of DC power.

Technical Reference(s): LPA-23, Loss of Vital DC Bus, Page 14, Rev. 8.
OP AP-23, Loss of Vital DC Power, Appendix B, Page 9, Rev. 12.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7116 - Explain the consequences of loss of DC vital bus.

Question Source: Bank # B-0372
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 14 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	065/2.1.23	
Importance Rating	4.3	4.4

K/A: **Loss of Instrument Air** - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question:

Instrument air pressure is dropping and the crew is responding by using OP AP-9 “Loss of Instrument Air.”

During the performance of the procedure, the Main Feedwater Reg valves fail closed, resulting in a Reactor Trip.

Which of the following describes the proper procedure usage of EOP E-0 “Reactor Trip or Safety Injection” and OP AP-9, for this event?

A. AOP only will be used.

B. EOP only will be used.

C. EOP and AOP will be implemented in parallel.

D. AOP will be implemented and EOP will be referenced.

Proposed Answer: C

Explanation:

Answer A is incorrect. EOP will also be used in parallel.

Answer B is incorrect. AOP will also be used in parallel.

Answer C is correct. Per OP AP-9 step 1 (continuous action step) RNO states that the procedure should be implemented in parallel with EOP E-0.

Answer D is incorrect. EOP is not referenced; it is used in parallel with AOP.

Technical Reference(s): LPA-9, Loss of Instrument Air, Page 14, Rev. 9.
OP AP-9, " Loss of Instrument Air", Page 2, Rev. 23
OP1.DC10 'Conduct of Operations", Attachment 7.1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operation procedure to be used to mitigate an operational event.

Question Source: Bank # INPO 10143
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: DCPP NRC Exam 1/25/99

RO Question 15

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	E04/EA1.3	
Importance Rating	3.8	4.0

K/A: **LOCA Outside Containment** - Ability to operate and/or monitor the following as they apply to the LOCA Outside Containment: Desired operating results during abnormal and emergency situations.

Proposed Question:

A reactor trip and SI occurred from full power. The crew transitioned from E-0, Reactor Trip or Safety Injection to ECA-1.2, LOCA Outside Containment.

The crew closed 8809A, RHR to Cold Legs 1 and 2 in an effort to isolate the leak.

Per ECA-1.2, which of following parameters indicate that the leak has been isolated?

A. RCS Pressure trending up rapidly.

B. Pressurizer Level trending up rapidly.

C. RHR pressure trending up slowly.

D. RVLIS Level trending up slowly.

Proposed Answer: A

Explanation:

Answer A is correct. If closing 8809A isolated the RCS leak, RCS pressure would be rising rapidly

Answer B is incorrect. ECA-1.1 uses RCS pressure to verify the leak is isolated and not Pressurizer level.

Answer C is incorrect. ECA 1-1 uses RCS pressure to verify the leak is isolated and not RHR Pressure, also pressure increase is rapid and not slow when the leak is isolated.

Answer D is incorrect. ECA 1-1 uses RCS pressure to verify the leak is isolated and not RVLIS Level, also pressure increase is rapid and not slow when the leak is isolated.

Technical Reference(s): LPE-1C, Recirculation Modes and LOCA Outside Containment, Page 44, Rev. 9.
EOP ECA-1.2, LOCA Outside Containment, Pages 2 & 3, Rev. 6A.
EOP ECA-1.2 Background, Page 7, Rev. 2.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 42461 - Explain basis of emergency steps of ECA-1.2

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ____

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 16 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	E05/EK2.1	
Importance Rating	3.7	3.9

K/A: **Loss of Secondary Heat Sink** - Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question:

Unit 1 has experienced a Loss of Secondary Heat Sink and a Safety Injection.

The crew is performing the following step in EOP FR-H.1, “Loss of Secondary Heat Sink”;

- TRY To Establish Main Feedwater Flow To At Least One Steam Generator.

The crew has determined that the Main Feedwater Reg Valves are closed.

What actions must be taken in order to open the Main Feedwater Reg Valves?

A. Cycle the Reactor Trip Breakers only.

B. Reset SI signal, Cycle the Reactor Trip Breakers, Reset Feedwater Isolation.

C. Reset SI signal, Reset Phase A Isolation, Reset Feedwater Isolation.

D. Reset Feedwater Isolation only.

Proposed Answer: B

Explanation:

Answer A is incorrect. SI Signal must be reset; this will allow the cycling of the reactor trip breakers to clear the seal in feature.

Answer B is correct. Per EOP FR-H.1 and Reactor Protection STG; SI must be reset, Reactor Trip Breakers cycled closed and then open to break the seal in and Feedwater Isolation must be reset on both trains.

Answer C is incorrect. Trip Breakers must be cycled and Phase A is not required to be reset.

Answer D is incorrect. SI Must be reset and Reactor Trip Breakers must be cycled.

Technical Reference(s): STG B6A, Reactor Protection System, Pages 2.2-36 to 38, Rev. 15.
EOP FR-H.1, Response to Loss of Secondary Heat Sink, Page 6, Rev. 23.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37614 - Analyze automatic features and interlocks associated with the MFW system.

Question Source: Bank # A-0730
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 17

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	E11/EK1.2	
Importance Rating	3.6	4.1

K/A: **Loss of Emergency Coolant Recirculation** - Knowledge of the operational implications of the following concepts as they apply to the Loss of Emergency Coolant Recirculation: Normal, abnormal and emergency operating procedures associated with Loss of Emergency Coolant Recirculation

Proposed Question:

While responding to a LOCA, a transition to ECA-1.1, "Loss of Emergency Coolant Recirculation Capability," was performed due to a loss of emergency coolant recirculation.

Makeup is being added to the RWST, Containment Spray pumps have been reduced to the minimum required, and ECCS reduced to one train of SI flow.

What are these actions designed to do?

A. Prevent damage to vital equipment.

B. Permit the RCS to stabilize at an equilibrium condition.

C. Increase RCS Subcooling.

D. Delay the time to RWST depletion.

Proposed Answer: D

Explanation:

Answer A is incorrect. The major actions of ECA-1.1 do not include preventing damage to vital equipment.

Answer B is incorrect. The major actions of ECA-1.1 do not include the stabilization of RCS conditions.

Answer C is incorrect. RCS subcooling is not the reason for reducing flow, subcooling is actually reduced later in the procedure.

Answer D is correct. This is a major action from ECA-1.1 per the EOP Background Document. Makeup is added to the RWST to extend the time the SI pumps and containment spray pumps can take suction from the RWST. In addition, RWST outflow is minimized by stopping any unnecessary containment spray pumps and decreasing the SI pump flowrate.

Technical Reference(s): LPE-1C, Recirculation Modes and LOCA Outside Containment, Page 31, Rev. 9.
EOP ECA-1.1 Background, Page 4, Rev. 2.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 42460 -Explain basis of emergency procedure steps for ECA-1.1.

Question Source: Bank # B-0053
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ________

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 18

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	E12/EK1.3	
Importance Rating	3.4	3.7

K/A: **Uncontrolled Depressurization of all Steam Generators** - Knowledge of the operational implications of the following concepts as they apply to the Uncontrolled Depressurization of all Steam Generators: Annunciators and conditions indicating signals, and remedial actions associated with the Uncontrolled Depressurization of all Steam Generators

Proposed Question:

Unit 1 experienced a Reactor Trip and Safety Injection from 100%.

ECA-2.1, "Uncontrolled Depressurization of All Steam Generators," is currently in use.

While performing step 3, "Control Feedflow to Minimize RCS Cooldown," the following conditions are observed:

- All Steam Generator (SG) Narrow Range Levels are off scale low.

Based on the indications, what are the Auxiliary Feedwater (AFW) requirements to the faulted Steam Generators(SG) and why?

A. Stop all AFW flow to minimize the cooldown.

B. Maintain a minimum of 25 gpm AFW flow to one SG, to maintain a heat sink.

C. Maintain a minimum of 25 gpm AFW flow to each faulted SG, to prevent dryout.

D. Maintain Total AFW flow at greater than 435 gpm to maintain a heat sink.

Proposed Answer: C

Explanation:

Answer A is incorrect. AFW flow is not stopped, a cooldown is expected, 25 gpm AFW flow is established to keep the SG tubes wet.

Answer B is incorrect. Heat sink requirements will not be met, flow is throttled to 25 gpm to all SGs.

Answer C is correct. 25 gpm AFW flow is the minimum measurable flow to the SGs, maintaining minimum flow keeps the SG “wet.”

Answer D is incorrect. AFW flow is minimized in this instance, heat sink requirement will not be met.

Technical Reference(s): LPECA-2, Uncontrolled depressurization of ALL S/Gs. Page 12; Rev. 8.
EOP ECA-2.1, Uncontrolled Depressurization of ALL Steam Generators, Page 3, Rev. 22.
EOP ECA-2.1 Background Document, Pages 16 & 26; Rev. 2.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7920 - Explain the basis of emergency procedure steps.

Comment [JLB2]: obj 7920

Deleted: s in ECA-2.1

Question Source: Bank # P-36171
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 19

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	003/AK2.05	
Importance Rating	2.5	2.8

K/A: **Dropped Control Rod** - Knowledge of the interrelations between the Dropped Control Rod and the following: Control rod drive power supplies and logic circuits.

Proposed Question:

During recovery of a dropped rod in Control Bank A, Group 1, the ROD CONTROL URGENT FAILURE annunciator is received once the withdrawal of the affected rod commences.

Which of the following identifies the correct cause for the alarm?

A. Power Cabinet has a Regulation Failure.

B. Logic Cabinet has Slave Cycler Failure.

C. Power Cabinet has a Logic Error.

D. Logic Cabinet has a Multiplex Error.

Proposed Answer: A

Explanation:

Answer A is correct. Power Cabinet 2AC supplies power to Control Bank A, Group 2. A regulation failure in power cabinet of affected banks unaffected group due to open lift coil disconnects is expected during dropped rod recovery, due to the actual current to the coils not equal to the demanded current.

Answer B is incorrect. A slave cycle failure is a logic cabinet failure, a logic cabinet failure could cause an Urgent failure, but not during a dropped rod recovery.

Answer C is incorrect. Power Cabinet 2AC supplies power to Control Bank A, Group 2. A logic error is caused by a simultaneous zero current order to the stationary and movable coils. This is not the case during a dropped rod recovery.

Answer D is incorrect. Logic Cabinets do not have Multiplex errors, this error is associated with a Power Cabinet. Multiplex errors is when current is sensed with no demand is present.

Technical Reference(s): LPA-12, Control Rod Malfunction, Page 16, Rev. 11.
LA-3A, Rod Control Review, Pages 48 & 61, Rev. 9.
OP AP-12C, Dropped Control Rod, Page 6, Rev. 11.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 36990 – Describe controls, indications, and alarms associated with the Rod Control System.

Question Source: Bank # P-51615
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 20

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	024/AK2.03	
Importance Rating	2.6	2.5

K/A: **Emergency Boration** - Knowledge of the interrelations between the Emergency Boration and the following: Controllers and positioners.

Proposed Question:

Given the following Unit 1 plant conditions:

- Emergency Boration is required to restore adequate Shutdown Margin.
- The normal Makeup Controller is out of service for a software upgrade.

Which of the following is the next preferred “Alternate Boration” method, per OP AP-6 “Emergency Boration”?

A. Swap Charging Pump suction to the RWST via 8805A & B.

B. Locally OPEN CVCS-8471, Manual Emergency Borate Valve.

C. Open CVCS 8104, Emergency Borate Valve.

D. Open FCV-110A, Boric Acid Makeup FCV.

Proposed Answer: C

Explanation:

Answer A is incorrect. This method is the 3rd preferred method of emergency boration.

Answer B is incorrect. This method is the 4th preferred method of emergency boration.

Answer C is correct. This is the next preferred method, per AP-6 step 2.

Answer D is incorrect. This method would not supply Boric Acid flow unless additional valves were also opened in the flowpath.

Technical Reference(s): LPA-6, Emergency Boration, Page 8, Rev. 9.
OP AP-6, Emergency Boration, Step 2a, Page 4, Rev. 17.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 4149 -Describe the Emergency Boration process.

Question Source: Bank # P-35170
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 21 (Rev.1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	028/AA1.05	
Importance Rating	2.8	2.9

K/A: **Pressurizer (PZR) Level Control Malfunction** - Ability to operate and/or monitor the following as they apply to the Pressurizer Level Control Malfunctions: Initiation of excess letdown per the CVCS

Proposed Question:

Unit 1 is at 100% power with the following conditions:

- Pressurizer Level Control Channel has failed low.
- Letdown Isolation Valve associated with the control channel has failed in the closed position and can not be reopened.

Which of the following alignments would allow for continued operation at 100% power, while the valve is being repaired?

A. Isolate normal charging only.

B. Reduce Charging to Minimum and place Excess Letdown in service.

C. Reduce Charging to Minimum only.

D. Isolate normal charging and place Excess Letdown in service.

Proposed Answer: B

Explanation:

Answer A is incorrect. Isolation of normal charging will still supply water to the RCP seals. A portion of Seal Injection flow will go to the RCS and cause Pressurizer Level to increase slowly, eventually leading to a Reactor Trip from high Pressurizer level.

Answer B is correct. Reducing charging to Minimum (to seals only) and placing excess letdown in service will maintain a flow balance in the CVCS which will allow for continued operation at 100% power.

Answer C is incorrect. Reducing charging flow will slow the increase in Pressurizer level, but it alone will not be enough to prevent a Reactor Trip from high Pressurizer level.

Answer D is incorrect. Normal charging is reduced and not isolated. Excess letdown is placed in service.

Technical Reference(s): LPA-18, Loss of Letdown, Page 7, Rev. 8
OP AP-18, Letdown Line Failure, Page 2, Rev. 8.
AR PK04-21, Letdown Press/ Flo Temp, Page 3, Rev. 15.
AR PK05-21, PZR Level Hi/LO, Page 2, rev. 9.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5671 - Explain the impact on plant operation of a loss of CVCS letdown flow.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 22 (Rev.1)

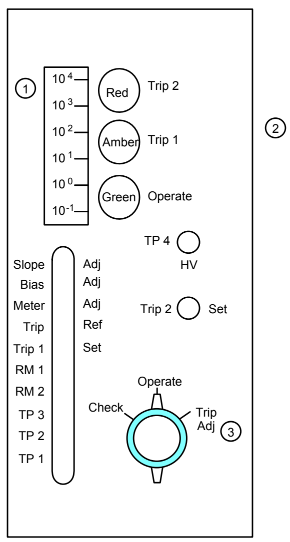
Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	036/AA2.01	
Importance Rating	3.2	3.9

K/A: **Fuel Handling Incidents** - Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: ARM system indications.

Proposed Question:

Refer to the figure of RM-58 Fuel Handling Building (FHB) Radiation Control Module for the question that follows:



Given the following indications:

- Red Light is ON for Trip 2
- Amber Light is ON for Trip 1
- Green Operate Light is ON

Based on the indications, what automatic actions should have occurred?

A. FHB Evacuation Alarm only.

B. FHB Evacuation Alarm and FHB ventilation swapped to Iodine Removal Mode.

C. Auxiliary Building and FHB Ventilation swapped to Iodine Removal Mode.

D. Auxiliary Building Ventilation swapped to Iodine removal mode only.

Proposed Answer: B

Explanation:

Answer A is incorrect. This condition would also cause the FHB ventilation system to swap to the Iodine removal mode.

Answer B is correct. Red light ON, indicates a HI Alarm set point has been exceeded and the FHB evacuation alarm should be sounding and Iodine removal ventilation should be in service.

Answer C is incorrect. Auxiliary Building Ventilation is not swapped to Iodine Removal mode based on RM-58 input signal, only FHB Ventilation is shifted to Iodine Mode.

Answer D is incorrect. Auxiliary Building Ventilation is not swapped to Iodine Removal mode based on RM-58 input signal.

Technical Reference(s): STG G4A, Radiation Monitoring, Pages 2-18-23, Rev. 9.
LPA-21, Irradiated Fuel Damage, Page 9, Rev. 8.
AR PK11-10, FHB High Radiation, RE 58 and 59, Rev. 11.
OP AP-21, Irradiated Fuel Damage, Pages 1 & 3, Rev. 10.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6573 – Explain the automatic actions that occur due to a fuel handling building radiation alarm.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 23

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	051/AA2.02	
Importance Rating	3.9	4.1

K/A: **Loss of Condenser Vacuum** - Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip.

Proposed Question:

Which of the following conditions would require an immediate Turbine Trip?

	MEGAWATTS	Condenser Pressure (IN HG ABS)
A.	150	6.5
B.	750	7.5
C.	1000	9.8
D.	1182	10.0

Proposed Answer: C

Explanation:

Answer A is incorrect. Megawatts and condenser pressure fall with in the acceptable operating region.

Answer B is incorrect. Megawatts and condenser pressure fall with in the acceptable operating region.

Answer C is correct. Megawatts and condenser pressure fall with in the trip the turbine immediately region.

Answer D is incorrect. Megawatts and condenser pressure fall with in the acceptable operating region.
.

Technical Reference(s): LPA-7, Degraded Condenser, Page 5, Rev. 9.
OP AP-7 Degraded Condenser, Attachment 6.2, Rev, 34.

Proposed references to be provided to applicants during examination: OP A-7 Att. 6.2

Learning Objective: 7968 - Given initial conditions and assumptions, determine if a turbine trip is required.

Question Source: Bank # INPO - 19212
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ________

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 24

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	061/AA1.01	
Importance Rating	3.6	3.6

K/A: **Area Radiation Monitoring (ARM) System Alarms** - Ability to operate and/or monitor the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Automatic actuation.

Proposed Question:

Which of the following Area Radiation Monitors (ARMs) has an automatic action (other than an alarm) when the alarm setpoint is reached?

A. RM-1, Control Room.

B. RM-2, Containment.

C. RM-25, Control Room Ventilation Intake.

D. RM-30, Containment High Range.

Proposed Answer: C

Explanation:

Answer A is incorrect. RM-1 Control Room area radiation monitor has no automatic actions.

Answer B is incorrect. RM-2 Containment area radiation monitor has no automatic actions.

Answer C is correct. A high alarm on RM-25 Control room ventilation intake radiation monitor will transfer Control Room vent to Mode 4.

Answer D is incorrect. RM-30 Containment High Range area radiation monitor has no automatic actions.

Technical Reference(s): LG-4A, Radiation Monitoring System, Page 58, Rev. 4.
LH-5, Control Room Ventilation System, Pages 36-38, Rev. 10.
AR PK15-06, Control Room Vent, Rev. 17.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8469 - Analyze automatic features and interlocks associated with the Radiation Monitoring System.

Question Source: Bank #
Modified Bank # INPO - 23405 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 25

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	E02/EK1.1	
Importance Rating	3.2	3.8

K/A: **SI Termination** - Knowledge of the operational implications of the following concepts as they apply to the SI Termination: Components, capacity, and function of emergency systems.

Proposed Question:

Unit 1 experienced a Steam Generator Tube Rupture (SGTR).

The crew is at the step in E-3 “Steam Generator Tube Rupture”, that checks to see if “ECCS Flow Should Be Terminated.”

Which of the following would be the effect on the plant, if Safety Injection Termination is delayed beyond the time assumed in the FSAR analysis for a SGTR?

A. Steam Generator overfill.

B. Loss of Subcooling Margin.

C. Loss of Secondary Heat Sink.

D. Reactor Vessel Head bubble formation.

Proposed Answer: A

Explanation:

Answer A is correct. If SI Flow is not terminated in a timely fashion, leakage into the steam generator will eventually fill the SG with water and potentially lift the SG Safety valves.

Answer B is incorrect. Loss of subcooling is not a concern at this step in the procedure.

Answer C is incorrect. Loss of Heat Sink will not occur, even if the ruptured generator is needed for cooldown, as level will be increasing.

Answer D is incorrect. Voiding may occur in the upper head region, but is not a significant safety concern; SI is terminated even if a void exists.

Technical Reference(s): LTAA-9, Steam Generator Tube Ruptures, Page 16, Rev. 8.
LPE-3, EOP E-3 SGTR Response, Page 25, Rev. 10.
EOP E-3, Steam Generator Tube Rupture, Page 23, Rev. 28A.
E-3 Background Document, Page 120, Rev. 2.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6855 -Explain the effect of SI termination and/or re-initiation on control of the plant.

Question Source: Bank # P-0796
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 8
55.43 ________

10 CFR Part 55 Content: Components, capacity, and functions of emergency systems.

Comments:

RO Question 26

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	E09/EK1.3	
Importance Rating	3.3	3.6

K/A: **Natural Circulation Operations** - Knowledge of the operational implications of the following concepts as they apply to the Natural Circulation Operations: Annunciators and conditions indicating signals, and remedial actions associated with the Natural Circulation Operations.

Proposed Question:

A natural circulation cooldown is in progress with the following indications.

- PK03-22 “CRDM FANS SUCT TEMP HI-LO PPC” is in alarm.
- No CRDM cooling fans are running.

The crew is at the step in E-0.2, "Natural Circulation Cooldown," that has them initiate an RCS Depressurization with the RCS Cooldown.

RCS pressure is at 1500 psig, what is (are) the approximate allowable temperature limit(s) that should be maintained?

A. Between 390°F and 600°F

B. Between 320°F and 390°F

C. Less than 600°F

D. Less than 320°F

Proposed Answer: B

Explanation:

Answer A is incorrect. These temperatures result in the temp/pressure being in the unacceptable range, higher limit would be correct if normal subcooled liquid curve is used.

Answer B is correct. These temperatures are in the acceptable region of the appendix T curve.

Answer C is incorrect. This temperature would be in the unacceptable range on the appendix T curve, but is correct if the normal subcooled liquid curve was used.

Answer D is incorrect. This temperature would be in the unacceptable range on the appendix T curve.

Technical Reference(s): LPE-0.2, Natural Circulation Cooldown, Page 10, Rev. 11.
AR PK03-22, CRDM FANS SUCT TEMP HI/LO PPC, Rev. 11.
EOP E-0.2, Natural Circulation Cooldown, Page 13 & Appendices C & T, Rev. 20.

Proposed references to be provided to applicants during examination: Appendices C & T

Learning Objective: 5854 - Explain the conditions that affect steam voids in the reactor vessel head.

Question Source: Bank # B-0599
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ________

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 27

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	E16/EK3.2	
Importance Rating	2.9	3.3

K/A: **High Containment Radiation** - Knowledge of the **reasons** for the following **responses** as they apply to the High Containment Radiation: Normal, abnormal and emergency operating procedures associated with High Containment Radiation

Proposed Question:

The following Unit 1 conditions exist:

- Large Break LOCA has occurred.
- Containment Radiation Levels are elevated.

The crew has entered EOP FR-Z.3 “Response to High Containment Radiation Level.”

What is the basis for the caution statement in the procedure, which warns against placing the Iodine Removal Units in service if a LOCA is in progress?

A. Their efficiency is drastically reduced during High Radiation conditions.

B. Starting the fans could result in high levels of Hydrogen in Containment.

C. Starting the fans will result in over current trips of the fan motors.

D. The carbon filters could catch on fire during accident conditions.

Proposed Answer: D

Explanation:
Answer A is incorrect. Concern is not related to efficiency of the fans and filters.

Answer B is incorrect. Concern is not related to the production of Hydrogen in containment.

Answer C is incorrect. Concern is not related to the potential for an over current trip of the motor.

Answer D is correct. The carbon filters could catch on fire in an accident due to absorption of water.

Technical Reference(s): LPE-ZI, Containment and Inventory FRGs, Page 17, Rev. 9.
EOP FR-Z.3, Response to High Containment Radiation Level, Page 2, Rev. 9
EOP FR-Z.3 Background

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7920 – Explain the basis of emergency procedure steps.

Question Source: Bank # ____
Modified Bank # R-1087 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 10
55.43 ____

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 28 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	003/2.1.27	
Importance Rating	3.9	4.0

K/A: **Reactor Coolant Pump System (RCPS)** - Conduct of Operations: Knowledge of system purpose and/or function.

Proposed Question:

A Loss of ALL offsite power occurs with the plant initially operating at 100%.

Immediately after the reactor trip when reactor power drops to about 8-9% (due to prompt drop after the trip), adequate core cooling exists because of:

A. continued RCP flow.

B. natural circulation.

C. ECCS flow.

D. power being sufficiently low.

Proposed Answer: A

Explanation:

Answer A is correct. The purpose of the RCP Flywheel is to increase the pump coast down time, to extend the period of forced flow following a pump trip.

Answer B is incorrect. Not enough time has elapsed for natural circulation flow to have been established.

Answer C is incorrect. ECCS flow will not have been established at this time.

Answer D is incorrect. Cooling is still required at this power level, AFW flow is insufficient to provide enough cooling at this power level.

Technical Reference(s): LA-6, Reactor Coolant Pump, Page 21, Rev. 10.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 68286 - State the purpose of the reactor coolant pumps system and equipment.

Question Source: Bank # S-0444
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 29 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	003/K5.03	
Importance Rating	3.1	3.5

K/A: **Reactor Coolant Pump System (RCPS)** - Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. in the shutdown loop.

Proposed Question:

Given the following conditions:

Unit 1 tripped from 29% power.

RCP 1-1 breaker tripped open when the buses swapped.

Which of the following describes the correct response and reason for the associated RCS Loop 1 Tavg indication?

A. Lower, due to reverse flow in the loop.

B. Lower, due to no steam flow.

C. Higher, due to increased steam flow.

D. Higher, due to no flow in the loop.

Proposed Answer: A

Explanation:

Answer A is correct. Tavg indication will lower because Thot will drop to approximately equal to Tcold in the loop, due to reverse flow in the loop.

Answer B is incorrect. Tavg indication will lower, do to reverse flow and not increased steam flow, steam flow will drop on the trip, not increase.

Answer C is incorrect. Tavg indication will not go up.

Answer D is incorrect. Tavg indication will not go up.

Technical Reference(s): STG A-6, Reactor Coolant Pump, Page 3-6, Rev. 14.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40521 - Describe the operation of the RCP.

Question Source: Bank # P-72907
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 5
55.43 ________

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 30

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	004/K6.04	
Importance Rating	2.8	3.1

K/A: **Chemical and Volume Control System (CVCS)** - Knowledge of the effect of a loss or malfunction on the following CVCS components: Pumps.

Proposed Question:

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

- ECCS CCP 1-1 running
- ECCS CCP 1-2 off
- Normal CCP 1-3 off
- Letdown in service at 75 gpm

Which of the following actions would occur if CCP 1-1 trips?

A. CCP 1-2 will auto start.

B. Letdown Isolation Valves (LCV-459/460) will close.

C. Letdown Orifice Valve(s) (CVCS-8149 A/B/C) will close.

D. Containment Letdown Isolation Valve (CVCS-8152) will close.

Proposed Answer: C

Explanation:

Answer A is incorrect. Charging pumps will only auto start for a safety injection signal, a second pump must be manually started to replace the tripped pump.

Answer B is incorrect. Letdown isolation valves will close automatically on loss of Pressurizer level as sensed by the control and backup channels for Pressurizer Level.

Answer C is correct. Letdown orifice valves will close if no charging pump is running to protect the letdown system.

Answer D is incorrect. Containment Letdown Isolation valve will close only on a Phase A or High HX room temperature.

Technical Reference(s): LB-1A, Chemical and Volume Control System, Page 12, Rev. 11
OIM Page B-1-2, Rev. 22.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5080- Analyze automatic features and interlocks associated with the CVCS.

Question Source: Bank # ____
Modified Bank # R-57798 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 7
55.43 ____

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 31

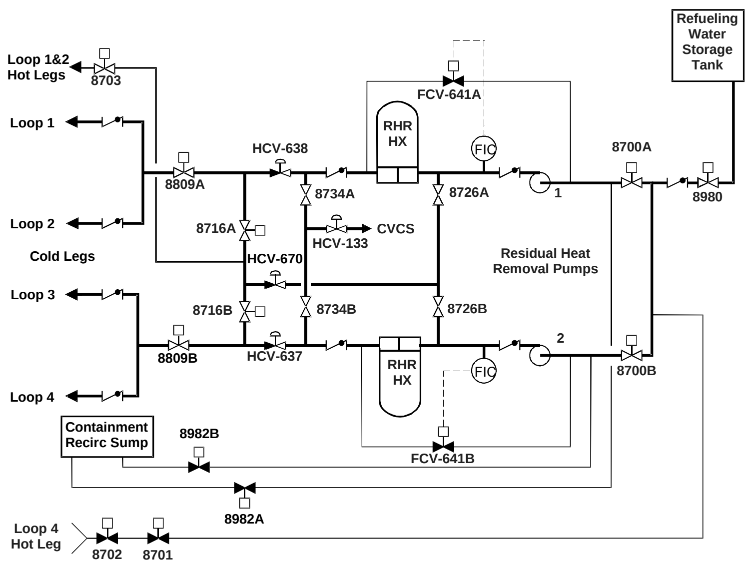
Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005/K4.05	
Importance Rating	2.5	2.9

K/A: **Residual Heat Removal System (RHRS)** - Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: Relation between RHR flowpath and refueling cavity.

Proposed Question:

Use the following drawing of the RHR System.



Which of the following describes the current RHR system use, based on the lineup depicted?

A. Filling the refueling cavity.

B. Cold Leg Recirculation.

C. RCS Cooldown.

D. ECCS Injection.

Proposed Answer: A

Explanation:

Answer A is correct. RHR-8980 is throttled open supplying water from the RWST to the refueling cavity via the RHR system.

Answer B is incorrect. Loop 4 hot leg valves 8701 an 8702 would not be open and valves 8982A and B would be open drawing water off the containment recirc sump.

Answer C is incorrect. RHR-8980 would not be open during normal cooldowns.

Answer D is incorrect. Loop 4 hot leg valves 8701 an 8702 would not be open and 8980 would be full open and HCV-637 and HCV-638 would be full open.

Technical Reference(s): STG B2, Residual Heat Removal System, Page 3-5, Rev. 16.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 35319 – Describe the operation of the RHR System.

Question Source: Bank # P-49446
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments: DCPP RO Exam 1995

RO Question 32 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005/A4.01	
Importance Rating	3.6	3.4

K/A: **Residual Heat Removal System (RHRS)** - Ability to manually operate and/or monitor in the control room: Controls and indication for RHR pumps.

Proposed Question:

Unit 1 is at 100% power.

During a surveillance test on RHR Pump 1-2, the following control switch indications are observed as the pump is started:

Red Light - OFF	Green Light - ON
White Light - ON	Blue Light - ON

What is the current status of RHR pump 1-2?

A. Running, with the auto/ manual selector switch in manual.

B. Tripped on bus differential.

C. Running, with the RWST low level trip signal defeated.

D. Tripped on overcurrent.

Proposed Answer: D

Explanation:

Answer A is incorrect. Pump will not be running. Running indication is based on red light being on. There is no selector switch for auto or manual on the RHR pumps on on ASW and CCW pumps.

Answer B is incorrect. Pump is tripped but not by bus differential, this will give you a blue light on the bus breaker and not the pump indication.

Answer C is incorrect. Pump will not be running. RWST low level defeat switch indication is not on the pump switch..

Answer D is correct. Pump is tripped and blue light indicates the trip is caused by an over current relay at the pump breaker.

Technical Reference(s): STG B2, Residual Heat Removal System, Page 2-6 & 7, Rev. 16.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 35318 -Describe controls, indications, and alarms associated with the RHR system.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ____

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 33

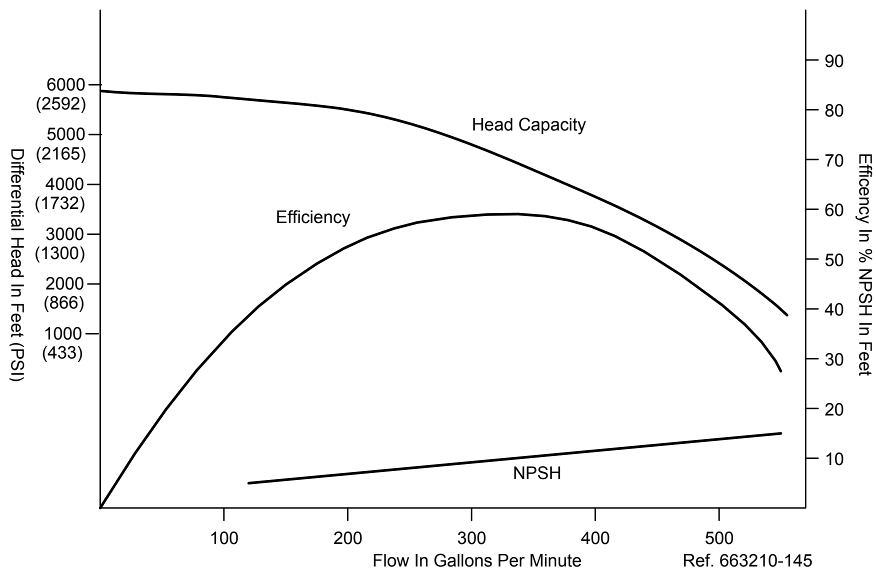
Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	006/K5.06	
Importance Rating	3.5	3.9

K/A: **Emergency Core Cooling System (ECCS)** - Knowledge of the operational implications of the following concepts as they apply to the ECCS: Relationship between ECCS flow and RCS pressure.

Proposed Question:

Use the following drawing to answer this question.



If RCS pressure were 1200 psig following a Safety Injection actuation, the expected flow rate from one CENTRIFUGAL CHARGING PUMP would be CLOSEST to:

A. 200 gpm.

B. 300 gpm.

C. 400 gpm.

D. 500 gpm.

Proposed Answer: D

Explanation:

Answer A is incorrect. This value would be obtained if the pump efficiency curve is used instead of the Head capacity curve.

Answer B is incorrect. This value would be obtained if normal RCS operating pressure were used instead of 1200 psig on the pump head capacity curve.

Answer C is incorrect. This value would be obtained if 1200 psig was used an applied to the pump efficiency curve.

Answer D is correct. This is the value that would be obtained using 1200 psig and the head capacity curve for the pump.

Technical Reference(s): LB-3, Emergency Core Cooling System, Page 13, Rev. 9
OIM, Page B-3-1a, Rev. 27.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6878- Describe ECCS components.

Question Source: Bank # A-0016
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 34 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	006/K4.16	
Importance Rating	3.2	3.5

K/A: **Emergency Core Cooling System (ECCS)** - Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Interlocks between RHR valves and RCS.

Proposed Question:

Given:

- RCS Wide Range pressure at 355 PSIG
- PZR Vapor space temperature instrument has failed to 700°F

How would the current plant conditions affect the ability to open RHR-8701 and 8702, Reactor Coolant Loop 4 Suction Isolations to RHR System?

A. RHR-8702 can NOT be opened. RHR-8701 is NOT affected.

B. RHR-8701 can NOT be opened. RHR-8702 is NOT affected.

C. Both RHR-8701 & 8702 can NOT be opened.

D. Both RHR-8701 & 8702 can be opened.

Proposed Answer: B

Explanation:

Answer A is incorrect. RHR 8702 can be opened and RHR 8701 cannot open due to PZR steam space temperature being greater than 475°F.

Answer B is correct. RHR 8701 cannot be opened due to PZR steam space being greater than 475°F and RHR 8702 is not affected and can open.

Answer C is incorrect. RHR 8701 cannot be opened due to PZR steam space being greater than 475°F and RHR 8702 is not affected and can open.

Answer D is incorrect. RHR 8701 cannot be opened due to PZR steam space being greater than 475°F and RHR 8702 is not affected and can open

Technical Reference(s): LB2, Residual Heat Removal System, Page 17, Rev. 13.
OIM, Page A-1-2, Rev. 26.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 35317 - Analyze automatic features and interlocks associated with the RHR System.

Question Source: Bank # A-0825
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 35

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	007/A1.02	
Importance Rating	2.7	2.9

K/A: **Pressurizer Relief Tank/Quench Tank System (PRTS)** - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank pressure

Proposed Question:

If a Pressurizer PORV fails open which of the following actions concerning the Pressurizer Relief Tank (PRT) will occur assuming NO operator action?

A. Relief valve will lift at 50 psig.

B. Rupture disks will rupture at 100 psig.

C. Primary water spray valve will automatically open at 3 psig.

D. Vent valve to the waste gas vent header will automatically open at 10 psig.

Proposed Answer: B

Explanation:

Answer A is incorrect. PRT does not use relief valves to protect against over pressure, rupture disks are used.

Answer B is correct. Rupture disks are designed to protect the tank and will blow at 100 psig.

Answer C is incorrect. Primary water spray valve does not open automatically based on pressure.

Answer D is incorrect. Vent valve to the waste gas vent header is blocked closed when PRT pressure reaches 10 psig.

Technical Reference(s): LA-4B, Pressurizer Relief Tank, Pages 7 & 19, Rev. 9.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 4950 - Describe the operation of the Pressurizer Relief Tank System.

Question Source: Bank # A-2173
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 5
55.43 ______

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 36

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	008/K3.01	
Importance Rating	3.4	3.5

K/A: **Component Cooling Water System (CCWS)** - Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS.

Proposed Question:

Component Cooling Water Return from the Non-Regenerative Heat Exchanger Temperature Control Valve, TCV-130 fails open due to a broken air line.

Assuming no action by the crew, which of the following describes the effect of this failure on the plant?

A. Letdown temperature goes down; the reduction in letdown temperature causes the letdown demineralizers to remove more boron, resulting in a minor dilution.

B. Letdown temperature goes up; the rise in letdown temperature causes the letdown demineralizers to remove less boron, resulting in a minor dilution.

C. Letdown temperature goes up; the rise in letdown temperature causes the letdown demineralizers to remove less boron, resulting in a minor boration.

D. Letdown temperature goes down; the reduction in letdown temperature causes the letdown demineralizers to remove more boron, resulting in a minor boration.

Proposed Answer: A

Explanation:

Answer A is correct. TCV-103 fails open on loss of air, provides more cooling flow which results in cooler letdown flow. Cooler letdown flow causes the demineralizers to remove more boron, which results in a minor dilution.

Answer B is incorrect. Temperature goes down as a result of the valve failing open, not closed. Demineralizers will remove more boron, not less.

Answer C is incorrect. Temperature goes down as a result of the valve failing open. Effect is not a boration.

Answer D is incorrect. No boration will occur if more boron is removed, this is a dilution.

Technical Reference(s): LB-1A, Chemical and Volume Control System, Page 24, rev. 11.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5668 - Apply fundamentals topics associated with the CVCS.

Question Source: Bank # INPO - 23455
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 37 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	010/K2.01	
Importance Rating	3.0	3.4

K/A: **Pressurizer Pressure Control System (PZR PCS)** - Knowledge of bus power supplies to the following: PZR heaters

Proposed Question:

From 100% power, electrical faults have caused a reactor trip and loss of offsite power.

If the Non-Vital 480V AC Buses CANNOT be re-energized, what can be done to restore power to the Pressurizer Heaters?

A. Groups 1 & 3 can be aligned Diesel Generators 1-3 and 1-2

B. Groups 2 & 4 can be aligned to Diesel Generators 1-3 and 1-1

C. Groups 2 & 3 can be aligned to Diesel Generators 1-2 and 1-1.

D. Groups 1 & 4 can be aligned to Diesel Generators 1-3 and 1-1.

Proposed Answer: C

Explanation:

Answer A is incorrect. Group 1 can not be aligned to a vital power source.

Answer B is incorrect. Group 4 can not be aligned to a vital power source.

Answer C is correct. Groups 2 and 3 can be aligned to a vital power source.

Answer D is incorrect. Groups 1 and 4 can not be aligned to a vital power source.

Technical Reference(s): LA-4A, Pressurizer, Pressure & Level Control, Page 13, Rev. 10.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 9990 - State the power supplies to Pressurizer, Pressure & Level Control System components.

Question Source: Bank #
Modified Bank # A-0404 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 38

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	012/K1.08	
Importance Rating	2.9	3.1

K/A: **Reactor Protection System** - Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following systems: MFW.

Proposed Question:

Which of the following provides a direct automatic trip of the Main Feed Water Pumps?

A. 2 out of 3 Low-Low Level in any SG.

B. Reactor Trip coincident with Low Tavg.

C. 2 out of 3 High-High Level in any SG.

D. Reactor Trip coincident with Low-Low Tavg.

Proposed Answer: C

Explanation:

Answer A is incorrect. Results in a reactor trip and automatic start of AFW pumps.

Answer B is incorrect. Results in a partial Feed Water Isolation, but does not trip the Main Feed Water Pumps.

Answer C is correct. SG High-High level (P-14) results in a trip of Main Feed Water Pumps along with a Feed Water Isolation.

Answer D is incorrect. P-12, Low-Low Tavg closes group one and two steam dump valves.

Technical Reference(s): OIM Page B-6-2, Protection Interlocks and Permissives, Rev. 26.
OIM Page B-6-12, Feedwater Isolation Signals, Rev. 27.
LB-6A, Reactor Protection System, Pages 39 & 41, Rev. 9.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37048 - Analyze automatic features and interlocks associated with the Reactor Protection System.

Question Source: Bank # INPO - 29401
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 39

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	013/K2.01	
Importance Rating	3.6	3.8

K/A: **Engineered Safety Features Actuation System (ESFAS**) - Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control.

Proposed Question:

Unit 1 is operating at steady-state, 100% power with all equipment operable and aligned for normal full power conditions.

What is the effect on the status and operation of engineered safeguards features (ESF) equipment if vital 120 VAC instrument bus 1-1 (PY-11) de-energizes?

A. Most SSPS channel 1 input bay relays for both trains will trip.

B. Most SSPS channel 1 input bay relays for train “A” only will trip.

C. None of the SSPS input bay relays on either train will trip.

D. Most SSPS channel 1 input bay relays for train “B” only will trip.

Proposed Answer: A

Explanation:

Answer A is correct. PY-11 maintains most of the input bay relays for both trains in the energized, non-trip condition.

Answer B is incorrect. Input relays on both trains will actuate and SSPS train A will lose slave relay power (the logic cards have redundant power).

Answer C is incorrect. Input relays on both trains will actuate, but the train A equipment will not auto start/actuate without slave relay power.

Answer D is incorrect. Input relays on both trains will actuate and SSPS train A will lose slave relay power (the logic cards have redundant power).

Technical Reference(s): LB-6B, Eagle 21 & Solid State Protection System. Page 10. Rev. 10.
OIM Page B-6-1b, SSPS Power Supplies, Rev. 19.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3291 - State the power supplies to Eagle 21 and Solid State Protection System components.

Question Source: Bank # DCPP/NRC-Dec. 07
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam DCPP- December 2007

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ____

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 40

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	013/K5.02	
Importance Rating	2.9	3.3

K/A: **Engineered Safety Features Actuation System (ESFAS)** - Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Safety system logic and reliability.

Proposed Question:

In the Solid State Protection System, Single Failure Criteria is achieved by:

A. Ensuring a minimum 2 of 3 logic when a single channel supplies both protection and control circuitry.

B. Using several protection channels with a minimum 2 of 3 logic for a trip.

C. Requiring any channel that is being tested to remain operable and thus capable of supplying trip logic on demand.

D. Tripping the reactor upon receiving a failure of a single protection channel.

Proposed Answer: B

Explanation:

Answer A is incorrect. Protection and Control circuits require 2 out of 4 logic for control of the plant.

Answer B is correct. Protection circuits require a minimum logic of 2 of 3 to trip the plant.

Answer C is incorrect. Channels are removed from service for testing and do not remain operable.

Answer D is incorrect. Reactor will not be tripped based on the failure of one protection channel.

Technical Reference(s): LB-6B, Eagle 21 and Solid State Protection System. Pages 6 & 7, Rev. 10.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 41312 - Explain significant Eagle-21/SSPS design features and the importance to nuclear safety.

Question Source: Bank # S-0165
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 41 (Rev. 1)

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	022/K4.04	
Importance Rating	2.8	3.1

K/A: **Containment Cooling System (CCS)** - Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Cooling of control rod drive motors.

Proposed Question:

Power is lost and subsequently regained to the Control Rod Drive Motor (CRDM) fans.

When the power is restored, the selected CRDM fans:

A. Must be manually restarted from their respective control switches.

B. Must be manually restarted by locally closing the motor circuit breakers.

C. Automatically restart in a time-delayed order.

D. Automatically restart when CRDM temperatures exceed 150˚F.

Proposed Answer: C

Explanation:

Answer A is incorrect. Selected fans will automatically restart, manual start is not required.

Answer B is incorrect. Fan are not controlled from the motor circuit breakers and fans will automatically restart.

Answer C is correct. Fans will restart in a predetermined order on a timer if power is lost and regained.

Answer D is incorrect. Fans are not controlled by temperature setpoints.

Technical Reference(s): LH-6, CRDM Cooling System, Page 10, Rev. 9.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3527 - Analyze automatic features and interlocks associated with the CRDM Cooling System.

Question Source: Bank # A-0521
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 42

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	022/A1.02	
Importance Rating	3.6	3.8

K/A: **Containment Cooling System (CCS)** - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure.

Proposed Question:

Unit 2 has experienced a DBA LOCA and Containment Spray has actuated.

Which of the following describes the impact of only ONE containment spray pump operating after ESF actuation?

A. Containment Recirculation Sump inventory will have an acid pH value.

B. 10 CFR 100 off-site exposure limits will be exceeded due to reduced iodine scrubbing capability.

C. Peak containment pressure limit may be exceeded unless at least two (2) Containment Fan Cooler Units (CFCUs) are running.

D. Peak containment pressure limit may be exceeded unless at least four (4) Containment Fan Cooler Units (CFCUs) are running.

Proposed Answer: C

Explanation:

Answer A is incorrect. Containment Spray additive tank will still empty into the spray system, sump pH will not be effected.

Answer B is incorrect. 10 CFR 100 exposure limits will not be exceeded with only one CS pump.

Answer C is correct. One train of Containment Spray and 2 of 5 CFCUs provide sufficient heat removal to maintain Containment pressure below its design value of 47 psig following a design basis LOCA or MSLB..

Answer D is incorrect. Only 2 CFCUs are required to meet the requirements.

Technical Reference(s): LI-2, Containment Spray System, Page 6, Rev.10.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40802 - Explain significant Containment Spray System design features and the importance to nuclear safety.

Question Source: Bank # P-49419
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge x
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5
55.43

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 43

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	026/K1.01	
Importance Rating	4.2	4.2

K/A: **Containment Spray System (CSS)** - Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: ECCS.

Proposed Question:

A Design Basis LOCA has occurred on Unit 1.

EOP E-1.3, Transfer to Cold Leg Recirculation," has just been completed.

Refueling Water Storage Tank Level is 2%.

Which of the following ECCS pumps will be supplying flow directly to the Containment Spray system?

A. Both Containment Spray Pumps.

B. ONE Containment Spray Pump only.

C. Both Residual Heat Removal Pumps.

D. ONE Residual Heat Removal Pump only.

Proposed Answer: D

Explanation:

Answer A is incorrect. Both Containment spray pumps are secured when RWST level reaches 4%..

Answer B is incorrect. Both Containment Spray pumps are secured, not just one when RWST Level reaches 4 %.

Answer C is incorrect. Only one Train of RHR flow is used for Containment Spray flow.

Answer D is correct. One train of RHR supplies core cooling and the other train is aligned to the containment spray system and suction of remaining ECCS pumps.

Technical Reference(s): STG B3, Emergency Core Cooling System, Pages 3-13 & 14, Rev. 16.
STG I-2, Containment Spray System, Page 3-10, Rev. 12.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6055 - Identify system interrelationships between the CSS and other plant systems.

Question Source:

Bank #		
Modified Bank #		(Note changes or attach parent)
New	X	

Question History: Last NRC Exam N/A

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	8
55.43	

10 CFR Part 55 Content: Components, capacity, and functions of emergency systems.

Comments:

RO Question 44 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	039/K1.05	
Importance Rating	2.5	2.6

K/A: **Main and Reheat Steam System (MRSS)** - Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: T/G.

Proposed Question:

During a secondary plant start-up on Unit 1, what is the effect on the MSR bypass valves (CV-1s) and supply valves (CV-2s) when the RAMP pushbutton, on the MSR HMI control screen, is depressed?

A. CV-1 valves open based on a 75˚F/hr heatup rate and after they have been open for 15 minutes, all CV-2 valves open at the same time.

B. CV-1 valves open based on feedback from hot reheat temperature;
CV-2 valves open as soon as CV-1 is full open.

C. CV-1 valves open, based on main turbine first stage impulse pressure signal;
CV-2 valves open fully in a staggered fashion.

D. CV-1 valves open based on a 75˚F/hr heatup rate and after they have been open for 30 minutes, CV-2 valves open fully in a staggered fashion.

Proposed Answer: D

Explanation:

Answer A is incorrect. CV-2 valves open in a staggered fashion once CV-1 valves have been open for 30 minutes.

Answer B is incorrect. CV-1 valves stroke open for a 75˚F heat up rate, CV-2 valves open in staged fashion.

Answer C is incorrect. 1st stage impulse pressure is not used and CV-2 do open in a staggered fashion.

Answer D is correct. After the CV-1 valves have been open for 30 minutes, CV-2 valves open fully in a staggered fashion. LP A CV-2s open first, 12 minutes later LP B CV-2s open, and then 12 minutes later LP C's CV-2s open.

Technical Reference(s): LC-5, Moisture Separator Reheaters, Pages 22 – 25, Rev. 9.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40878 – Describe MSR Components.

Question Source: Bank # ______
Modified Bank # A-0747 (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 4
55.43 ______

10 CFR Part 55 Content: Secondary Coolant and auxiliary systems that affect the facility.

Comments: Unit 1 Design Change

RO Question 45 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	059/A4.12	
Importance Rating	3.4	3.5

K/A: **Main Feedwater (MFW) System** - Ability to manually operate and/or monitor in the control room: Initiation of automatic feedwater isolation.

Proposed Question:

A Reactor Trip and Safety Injection have occurred on Unit 1.

What will be the position of the Main Feedwater valves following this transient?

A. Main Feedwater reg valves, bypass valves, and Feedwater isolation valves will all be CLOSED.

B. Main Feedwater reg valves and bypass valves will be CLOSED; the Feedwater isolation valves will be OPEN.

C. Main Feedwater reg valves will be OPEN, until the controller demand is reduced to "0"; the bypass valves will be CLOSED; the Feedwater isolation valves will be OPEN.

D. Main Feedwater reg valves and bypass valves, will be OPEN; the Feedwater isolation valves will be CLOSED.

Proposed Answer: A

Explanation:

Answer A is correct. Feedwater Isolation valves will all be closed; they close on Safety Injection signal.

Answer B is incorrect. Feedwater Isolation valves close on Safety Injection signal.

Answer C is incorrect. Main Feedwater Regulation valves will be closed, even if demand is high.

Answer D is incorrect. Main and Bypass valves will be closed.

Technical Reference(s): LC-8A, Main Feedwater System, Page 39, Rev. 11.
OIM, Feedwater Isolation Signals, Page B-6-12, Rev. 27.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37614 - Analyze automatic features and interlocks associated with the MFW system.

Question Source: Bank # P-0052
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ________
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 46

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	061/K3.02	
Importance Rating	4.2	4.4

K/A: **Auxiliary / Emergency Feedwater (AFW) System** - Knowledge of the effect that a loss or malfunction of the AFW System will have on the following: S/G.

Proposed Question:

Given the following conditions:

Unit 2 was at 100% power when a Reactor Trip occurred.

PT-434, AFW pump 2-3 discharge pressure transmitter has failed to zero.

What effect will this failure have on LCV-115 and LCV-113, AFW supply valves to Steam generators 2-3 and 2-4?

The LCVs will fail:

A. Closed, and must be taken to manual to restore control.

B. Open, and must be taken to manual to restore control.

C. Open, AFW pump 2-3 must be secured to stop flow.

D. As is, and must be locally operated.

Proposed Answer: A

Explanation:

Answer A is correct. This input is used as run-out protection. Discharge pressure transmitter failing to zero will cause the LCVs to go closed on low pump discharge pressure. Manual control is available from the controller.

Answer B is incorrect. Valves will not fail open, they will fail closed.

Answer C is incorrect. Valves will not fail open, they will fail closed. There is no requirement to stop the pumps.

Answer D is incorrect. Valves will not fail as is; this is true of the turbine driven AFW pump LCVs.

Technical Reference(s): LD-1, Auxiliary Feedwater System, Page 29 & 30, Rev11.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37635- Describe controls, indications, and alarms associated with the Auxiliary Feedwater System.

Question Source: Bank # ______
Modified Bank # A-0692 (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 47

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	061/A2.04	
Importance Rating	3.4	3.8

K/A: **Auxiliary / Emergency Feedwater (AFW) System** - Ability to (a) predict the impacts of the following malfunctions or operations on the AFW System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: pump failure or improper operation.

Proposed Question:

Unit 1 was at 100% power when a Reactor Trip occurred.

AFW pump 1-1 and 1-2 will not start.

What actions must be taken in order to supply AFW to the 1-1 and 1-2 Steam Generators from the 1-3 AFW pump?

A. Cut out the pump running interlock bypass for LCV-110 & LCV-111,
select auto for LCV-110 & LCV-111 and
locally open manual pump discharge cross-tie valves

B. Cut in the pump running interlock bypass for LCV-110 and LCV-111,
select manual control for LCV-110 and LCV-111 and
locally open manual pump discharge cross-tie valves

C. Cut out auto start from main feed pump trip for 1-2 AFW pump,
select manual control for LCV-110 & LCV-111 and
locally open manual pump cross-tie valves

D. Cut in the pump running interlock bypass for all four LCVs,
select auto control on all four LCVs, and
open pump discharge cross-ties from VB-3.

Proposed Answer: B

Explanation:

Answer A is incorrect. The interlock must be cut in for the LCVs and the LCVs need to be selected to manual.

Answer B is correct. The actions listed will align the -13 AFW pump to feed all 4 SGs.

Answer C is incorrect. The interlock must be cut in for the LCVs.

Answer D is incorrect. Must select manual control for the LCVs and discharge crossties are local and not on the control board.

Technical Reference(s): LD-1, Auxiliary Feedwater System, Page 31, Rev. 11.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37635 - Describe controls, indications, and alarms associated with the AFW system.

Question Source: Bank # S-1260
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ________

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 48

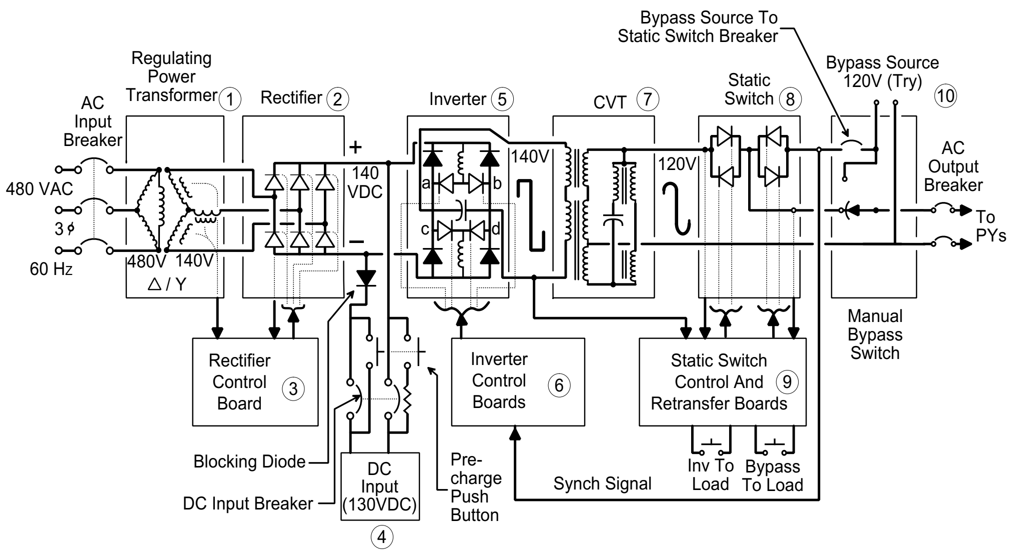
Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	062/A3.04	
Importance Rating	2.7	2.9

K/A: **A.C. Electrical Distribution System** - Ability to monitor automatic operation of the A.C. Distribution System, including: Operation of inverter (e.g., precharging synchronizing light, static transfer).

Proposed Question:

Use the following drawing of a Vital Instrument AC System UPS.



Which of the following conditions will initiate an auto transfer of a Vital Instrument AC UPS static switch to its backup power supply?

A. Inverter output failure.

B. Rectifier failure.

C. Opening the DC input breaker.

D. Opening the AC input breaker.

Proposed Answer: A

Explanation:

Answer A is correct. Static switch electronically transfers load from the Constant Voltage Transformer (CVT) output to the backup regulating transformer (TRY) input on loss of inverter output.

Answer B is incorrect. Rectifier failure will cause loss of ac input but, DC input is still available so no transfer is required.

Answer C is incorrect. Opening the DC input breaker will not cause the inverter to transfer to backup power, due to the AC source still being available.

Answer D is incorrect. Opening the AC input breaker will not cause the inverter to transfer to backup power, due to the DC source still being available.

Technical Reference(s): STG J-10, Instrument AC System, Pages 2.1 - 8 – 10, Rev. 14.
LJ-10, Instrument AC System, Page 31, Rev. 8.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3332 – Discuss abnormal conditions associated with the Instrument AC System.

Question Source: Bank # S-47507
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 49 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	063/A2.01	
Importance Rating	2.5	3.2

K/A: **D.C. Electrical Distribution System** - Ability to (a) predict the impacts of the following malfunctions or operations on the D.C. Electrical System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds.

Proposed Question:

Unit 1 is stable at 100% power when the control room operators receive multiple seemingly unrelated alarms.

The following indications are also observed:

- Loss of indicating lights to some secondary plant valves.
- Unwarranted repositioning of some secondary valves.
- Unexpected pump starts.
- Loss of secondary efficiency.

In response to the above indications, the control room crew should:

A. start the standby air compressor to recover instrument air pressure.

B. initiate action to restart the Plant Process Computer.

C. start the Nash Vacuum pump to restore Main Condenser vacuum.

D. initiate action to identify and isolate multiple DC grounds.

Proposed Answer: D

Explanation:

Answer A is incorrect. Low instrument air pressure will cause valve to change positions, but not affect indicating lights. Starting the standby air compressor will not correct the problem.

Answer B is incorrect. Loss of the PPC will affect indicated secondary efficiency but not actual efficiency. Restarting will not correct the problem.

Answer C is incorrect. Loss of main condenser vacuum will cause loss of efficiency but not cause valves to reposition. Starting the Nash Vacuum pump will not correct the problem.

Answer D is correct. All indications given are a result of multiple Grounds, isolation of grounds will correct the problem.

Technical Reference(s): AR PK20-22, 125V DC BUS 11,12,13 GROUND, Page 1, Rev. 8A.
OP J-9:V, Detection of a DC Ground, Pages 1-5, Rev. 9.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37793 - Describe controls, indications, and alarms associated with the DC Power System.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 50 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064/A2.05	
Importance Rating	3.1	3.2

K/A: **Emergency Diesel Generator (ED/G) System** - Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loading the ED/G.

Proposed Question:

Unit 1 is at 100% Power.

STP M-9A “Diesel Engine Generator Surveillance Test” is in progress on Diesel Generator 1-2.

The control room operator is preparing to parallel the Diesel Generator with its associated 4kV bus.

What is the **minimum** load required to be picked up on the Diesel Generator, after the output breaker is closed and the reason for it?

A. 500 kW, Prevent the Diesel from tripping on High Jacket Water Temperature.

B. 500 kW, Prevent the Diesel Generator Breaker from tripping on Reverse Power.

C. 1300 kW, Prevent the Diesel from tripping on High Jacket Water Temperature.

D. 1300 kW, Prevent the Diesel Generator Breaker from tripping on Reverse Power.

Proposed Answer: B

Explanation:

Answer A is incorrect. 500 kW is correct, but the load picked up is independent from the jacket water cooling system.

Answer B is correct. Precaution 10.9 from STP M-9A states that when paralleling a DG, pick up load (0.5 MW) as soon as possible after the beaker is closed. This will prevent the DG breaker from tripping on directional power (reverse power).

Answer C is incorrect. 1300 kW is incorrect; this is the kW that the diesel would be loaded to if it ran at less than 650 kW for greater than 1 hour. Load picked up is independent from the jacket water cooling system.

Answer D is incorrect. 1300 kW is incorrect; this is the kW that the diesel would be loaded to if it ran at less than 650 kW for greater than 1 hour. No trip will occur.

Technical Reference(s): LJ-6B, Diesel Generator System, Page 87, Rev. 12.
STG J-6B, Diesel Generator System, Page 3-1, Rev. 19.
STP M-9a, Diesel Engine Generator Routine Surveillance Test, Page 9, Rev. 76A.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6408 - Discuss significant precautions and limitations associated with the Diesel Generator System.

Question Source: Bank #
Modified Bank # P-49443 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5
55.43

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 51

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064/K6.07	
Importance Rating	2.7	2.9

K/A: **Emergency Diesel Generator (ED/G) System** - Knowledge of the effect of a loss or malfunction of the following will have on the ED/G System: Air receivers.

Proposed Question:

Unit 1 is at 100%.

A relief valve has failed open on Diesel Generator (DG) 1-1 Starting Air Receiver 1-1A.

The leakage exceeds the capacity of the starting air compressor.

Which of the following describes the affect of this failure on DG 1-1, if a start signal occurs prior to any operator action?

A. Start in the normal allowed time, from the 1-1B Starting Air Receiver via all four starting air solenoids.

B. Will start but, starting time will exceed the surveillance required response time.

C. Start in the normal allowed time via the 2 starting air solenoids associated with the 1-1B Starting Air Receiver.

D. Will not start because the starting air system will be depressurized.

Proposed Answer: C

Explanation:

Answer A is incorrect. Diesel will start in normal time, but the start will be with only 2 of the 4 starting motors.

Answer B is incorrect. Diesel will start, but will not exceed the surveillance time as the starting air systems are redundant.

Answer C is correct. Start will occur in normal time via 2 starting air solenoids associated with the 1-1B starting air system.

Answer D is incorrect. Air systems are separate but redundant, one system failing will not affect the others operation.

Technical Reference(s): LJ-6B, Diesel Generator System, Pages 54-60, Rev. 12.
Tech Spec 3.8.8 Bases, Page 39, Rev. 4.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37728 - Decribe Diesel Generator System components.

Question Source: Bank # INPO - 23185
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 52 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	073/A1.01	
Importance Rating	3.2	3.5

K/A: **Process Radiation Monitoring (PRM) System** - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM System controls including: Radiation levels.

Proposed Question:

Unit 1 is in MODE 6.

High Radiation is sensed on Containment Purge Radiation Monitor RE-44A during a Containment Purge.

What occurs automatically as a result of this high radiation condition?

A. Containment purge and pressure/vacuum relief pathways are isolated.

B. Charcoal filter upstream of Exhaust fans E-1 and E-2 are placed in service.

C. Charcoal filter upstream of Exhaust Fan E-3 is placed in service.

D. Containment Purge Supply Fan S-3 and Exhaust fan E-3 both trip.

Proposed Answer: A

Explanation:

Answer A is correct. High radiation on RM-44A or RM-44B will cause an isolation of the ventilation flow paths out of containment only, no effect on fans.

Answer B is incorrect. There are no charcoal filters that are placed in service due to this condition; charcoal filters are associated with auxiliary building ventilation systems.

Answer C is incorrect. There are no charcoal filters that are placed in service due to this condition; charcoal filters are associated with auxiliary building ventilation systems.

Answer D is incorrect. There are no automatic trip signals for the ventilation fans; they must be shutdown manually if the ventilation system is isolated.

Technical Reference(s): LG-4B Digital Radiation Monitoring System, Pages 34 – 37, Rev. 4.
LH-4 Containment Purge System, Page 18, Rev. 10.
OIM, Containment Vent Isolation (CVI) Functions, Page B-6-9a, Rev. 27.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3275 - Analyze automatic features and interlocks associated with the DRMS.

Question Source: Bank # S-3069
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 11
55.43 ________

10 CFR Part 55 Content: Purpose and operation or radiation monitoring systems, including alarms and survey equipment.

Comments:

RO Question 53

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	076/K3.07	
Importance Rating	3.7	3.9

K/A: **Service Water System (SWS)** - Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads.

Proposed Question:

Which of the following describes how cooling is ultimately provided to the Component Cooling Water (CCW) system following a total loss of the Auxiliary Salt Water (ASW) system?

A. Temporary vendor supplied pumps and piping transport water from the ocean to the ASW lines via connections in the ASW vacuum breaker vault.

B. Temporary vendor supplied pumps and piping transport water from the ocean directly to the CCW heat exchangers.

C. Long term cooling water pumps and hoses transport water from the East Reservoir directly to the CCW heat exchangers.

D. Long term cooling water pumps and hoses transport water from the East Reservoir to the ASW lines via connections in the ASW vacuum breaker vault.

Proposed Answer: A

Explanation:

Answer A is correct. Per AP-11, Appendix D, Instructions for loss of ultimate heat sink; once available, the source of cooling water for the CCW heat exchangers will be supplied by temporary portable industrial pumps taking a suction from the intake cove and pumping through temporary lines connected to the ASW vacuum breaker piping.

Answer B is incorrect. Piping is connected to the vacuum breaker piping and not directly to the CCW heat exchanger.

Answer C is incorrect. Long term cooling water pumps are reserved for use during a loss of Auxiliary Feed Water flow and piping is connected to the ASW vacuum breaker and not the CCW heat exchanger.

Answer D is incorrect. Long term cooling water pumps are reserved for use during a loss of Auxiliary Feed Water flow and are not used for these conditions.

Technical Reference(s): LPA-11, Malfunction of CCW System, Page 16, Rev. 10.
OP AP-10, Loss of Auxiliary Salt Water, Page 2, Rev. 9.
OP AP-11, Malfunction of Component Cooling Water System, Appendix D, Instructions for Loss of Ultimate Heat Sink. Pages 34 -37, Rev. 24.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5360 - Discuss abnormal conditions associated with the ASW system.

Question Source: Bank # A-0032
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 54

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078/A4.01	
Importance Rating		

K/A: **Instrument Air System (IAS)** - Ability to manually operate and/or monitor in the control room: Pressure gauges.

Proposed Question:

An Instrument Air leak is occurring on Unit 1.

The Reactor Operator notes that the Main Feedwater Reg Valves are slowly closing.

The Instrument Air Pressure header pressure gauge PI-380 should read approximately how much pressure based on the plant response.

A. 75 - 78 psig.

B. 86 - 89 psig.

C. 93 - 96 psig

D. 100 -103 psig

Proposed Answer: A

Explanation:

Answer A is correct. The main Feedwater Regulation valves are expected to begin closing at 75 psig.

Answer B is incorrect. This pressure is above where instrument air to containment valve FCV-584 will begin to close.

Answer C is incorrect. This is the pressure where the stand-by compressors will auto start and load.

Answer D is incorrect. This is pressure associated with the Lag compressor loading as pressure decreases.

Technical Reference(s): LK-1, Compressed Air System, Pages 8, 42 and 46, Rev. 9.
OP AP-9, Loss of Instrument Air, Page 2, Rev. 24.
OIM, Compressor Load/Unload Setpoints, Page K-1-2, Rev. 26.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7226 - Describe controls, indications, and alarms associated with the Compressed Air System.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 55

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	103/A3.01	
Importance Rating	3.9	4.2

K/A: **Containment System** - Ability to monitor automatic operation of the Containment System, including: Containment isolation.

Proposed Question:

A spurious Phase "A" Containment Isolation signal has occurred.

What is the effect on the Reactor Coolant Pumps (RCPs)?

A. CCW to the RCP thermal barrier will be lost; seal injection must be maintained.

B. RCP seal injection is isolated; so CCW flow to the thermal barrier must be verified.

C. RCP #1 seal leakoff will be isolated and flow will be directed to the PRT via a relief valve.

D. CCW cooling flow from the RCP oil coolers is isolated and must be restored within 5 minutes.

Proposed Answer: C

Explanation:

Answer A is incorrect. CCW will not be lost to the thermal barrier, this is caused by a Phase ‘B”.

Answer B is incorrect. RCP Seal Injection is NOT affected by any signal, but normal charging is isolated on SI signal.

Answer C is correct. The Phase A signal will cause #1 seal return isolation valves CV8112 and CV8100 to close. This will cause the 150# relief valve to the PRT to open maintaining adequate #1 seal flow.

Answer D is incorrect. CCW flow will not be lost the oil coolers, this is caused by a Phase “B”.

Technical Reference(s): STG B-6a, Reactor Protection System, Page 2.2-20, Rev. 15.
OIM Phase A, Page B-6-7, Rev. 27.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40928 - Analyze automatic features and interlocks associated with the Containment Structure.

Question Source: Bank # INPO - 27
Modified Bank # __________ (Note changes or attach parent)
New __________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 56 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	002/K3.03	
Importance Rating	4.2	4.6

K/A: **Reactor Coolant System (RCS)** - Knowledge of the effect that a loss or malfunction of the RCS will have on the following: Containment.

Proposed Question:

Which of the following provides for containment pressure control during a Loss of Cooling Accident once the Refueling Water Storage Tank is empty?

A. Continued operation of the Containment Fan Coolers only

B. Operation of the Containment Fan Coolers and aligning RHR to Containment Spray.

C. Operation of the Containment Fan Coolers and making up to the RWST for continued spray.

D. Continued operation of the Containment Fan Coolers and Containment pressure relief as needed.

Proposed Answer: B

Explanation:

Answer A is incorrect. Containment Fan Coolers used in conjunction with the RHR system.

Answer B is correct. Containment Fan Coolers used in conjunction with the RHR system

Answer C is incorrect. RWST makeup is not done, unless there is a loss of Recirculation, no additional inventory in containment is required.

Answer D is incorrect. Containment pressure relief in not used during a LOCA.

Technical Reference(s): LI-1, Containment Structure, Page 46, Rev. 11.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37591 - Describe system interrelationships between the Containment Structure and other plant systems.

Question Source:	Bank #	INPO - 10402	
	Modified Bank #	________	(Note changes or attach parent)
	New	________	

Question History: Last NRC Exam N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	______
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 57

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	011/2.4.4	
Importance Rating	4.5	4.7

K/A: **Pressurizer Level Control System (PZR LCS)** - Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

Unit 1 is at 2% power.

The following SEQUENTIAL events have just occurred:

- charging flow has trended down to zero.
- letdown isolated and heaters turned off.
- pressurizer level on LR-459 PZR Level Recorder trended up to 75%.

Pressurizer level control selector switch is in the LT-459/461 position.
Pressurizer pressure control selector switch is in the PT-457/456 position.

No operator actions have been taken.

Which procedure will the crew enter first?

A. AP-1 “Excessive Reactor Coolant System Leakage” for an RCS leak.

B. AP-5 “Malfunction of Eagle 21 Protection or Control Channel” for a failure of the Pressurizer level control channel.

C. AP-13 “Malfunction of Reactor Pressure Control System” for a failure of the Pressurizer Pressure Controller.

D. E-0, Reactor Trip or Safety Injection, due to Reactor Trip from High Pressurizer level.

Proposed Answer: B

Explanation:

Answer A is incorrect. Symptoms are not associated with an RCS leak, charging flow will increase, not decrease.

Answer B is correct. Reactor power is less than P-10 which blocks the Reactor Trip from High Pressurizer level at 70%, the Control channel has failed high and needs to be selected out, this is done in AP-5.

Answer C is incorrect. Failure is not a pressure control issue; AP-13 is used if the pressure controller fails.

Answer D is incorrect. This would be correct if the plant tripped, in this case the plant will not trip and so E-0 is not appropriate at this time.

Technical Reference(s): LA-4A, Pressurizer, Pressure & Level Control, Page 70, Rev. 10.
OIM, Pressurizer Level Control, Page A-4-2a, Rev. 22
OIM, Pressurizer Level Channel Failures, Page A-4-2b, Rev. 27.
OIM, Protection and Interlock Permissives, Page B-6-2, Rev. 26.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 4560 - Describe the operation of the Pzr, Pzr Pressure and Level Control System.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ______

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 58

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	014/K1.01	
Importance Rating	3.2	3.6

K/A: **Rod Position Indication System (RPIS)** - Knowledge of the physical connections and/or cause-effect relationships between the RPIS and the following systems: CRDS.

Proposed Question:

Which of the following describes the method utilized by the Digital Rod Position Indicating System (DRPI) to determine actual rod position?

A. Inductance coils sense the position of each control rod's drive shaft and relate this to position of the rod in the core.

B. The current signal to the Stationary Gripper coil for each rod subgroup is measured for time duration and converted to a digital indication.

C. The LVDT associated with each rod sub group measures rod drive shaft position by induced current signal which is averaged to give the digital counter position.

D. The IN and OUT position contactors for the Control Rod IN-HOLD-OUT switch provides the signal that moves the digital counter.

Proposed Answer: A

Explanation:

Answer A is correct. Inductance coils detect the position of the rod due to an impedance change as the control rod drive shaft passes through its center.

Answer B is incorrect. Current signal is not measured for each rod subgroup.

Answer C is incorrect. LVDTs are not used in this application and no averaging is performed.

Answer D is incorrect. IN/HOLD/OUT switch contactor does not supply a signal to the digital rod position system.

Technical Reference(s): LA-3B, Digital Rod Position Indication System, Page 8, Rev. 11.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40701 - Describe DRPI system components

Question Source: Bank # P-1275
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ________

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 59 (Rev.1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	017/K5.01	
Importance Rating	3.1	3.9

K/A: **In-Core Temperature Monitor (ITM) System** - Knowledge of the operational implications of the following concepts as they apply to the ITM System: Temperature at which cladding and fuel melt.

Proposed Question:

An Inadequate Core Cooling condition exists on Unit 1.

The onset of Fuel Cladding failure is expected to occur at which of the following temperatures?

A. 700˚F.

B. 1300˚F.

C. 1600˚F.

D. 2200˚F.

Proposed Answer: B

Explanation:

Answer A is incorrect. Fuel cladding damage occurs at 1300˚F, so no damage has occurred at this point.

Answer B is correct. This is the temperature where fuel cladding failure occurs.

Answer C is incorrect. This is the temperature where oxidation of the cladding occurs and Hydrogen is generated.

Answer D is incorrect. This is the temperature where the fuel begins to overheat.

Technical Reference(s): LMCDCDA, MCD – Core Damage Assessment, Page 7, Rev. 6.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3808 - Interpret indications of core damage.

Comment [JPSJ3]: Obj 3808

Question Source: Bank # ____
Modified Bank # P-6273 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 5
55.43 ____

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 60 (Rev.1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	027/K2.01	
Importance Rating	3.1	3.4

K/A: **Containment Iodine Removal System (CIRS)** - Knowledge of bus power supplies to the following: Fans.

Proposed Question:

Containment Iodine Removal Fan E-15 has been placed in service on Unit 1.

A total loss of offsite power has occurred that results in a Reactor Trip.

How will the loss of offsite power affect Fan E-15?

A. De-energized since 480v bus 12I has lost power.

B. Energized from 480v bus 12I and still in service.

C. Energized from 480v bus 1F and still in service.

D. De-energized due to the redundant breaker trip on the power transfer.

Proposed Answer: A

Explanation:

Answer A is correct. Iodine removal fan E-15 is powered from 480v bus 12I, this is a non-vital bus that will not have power on the transfer.

Answer B is incorrect. Power will not be available to bus 12I, a non vital bus, so the fan will not be energized.

Answer C is incorrect. E-15 is powered from a non-vital bus and not vital bus 1F. Fan will not be running.

Answer D is incorrect. There is no automatic trip associated with the redundant breaker for a power transfer.

Technical Reference(s): LH-3, Iodine Removal System, Page 7, Rev. 7.
STG H-3, Iodine Removal System, page 2-2, Rev. 8.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40819 - State the power supplies to Iodine Removal System components.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ____

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 61

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	028/K6.01	
Importance Rating	2.6	3.1

K/A: **Hydrogen Recombiner and Purge Control System (HRPS)** - Knowledge of the effect of a loss or malfunction of the following will have on the HRPS: Hydrogen recombiners.

Proposed Question:

Given the following conditions:

A Large Break LOCA has occurred.

Both Hydrogen Recombiners are in service.

If one of the operating Recombiners trips, which of the following describes the effect on the removal of Hydrogen from Containment?

A. Hydrogen concentration will remain below 4% with only one Recombiner in operation.

B. Hydrogen concentration will rise above 4% but remain below 13% with only one Recombiner in operation.

C. Hydrogen concentration will remain below 4% only if the Containment Purge System is placed in service in addition to the Recombiner.

D. Hydrogen concentration will remain below 4% only if Containment Spray is placed in service in addition to the Recombiner.

Proposed Answer: A

Explanation:

Answer A is correct. Either train will meet design function, without additional actions.

Answer B is incorrect. 4% is the limit. 13% was chosen as the approximate value for explosive mixture.

Answer C is incorrect. Purge system would not be placed in service as a result of a recombiner failure.

Answer D is incorrect. Spray will not be in service at the pressures that H2 recombiners operate at.

Technical Reference(s): STG H-9, Containment Hydrogen Recombiners, Page 1-4, Rev. 10.
ECG 23.4, Hydrogen Recombiners, Bases, Page 2, Rev. 1.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3864 - Describe the operation of the Containment Hydrogen Recombiner System.

Question Source: Bank # INPO - 23368
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 62 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	033/K4.01	
Importance Rating		

K/A: **Spent Fuel Pool Cooling System (SFPCS)** - Knowledge of Spent Fuel Pool Cooling System design feature(s) and/or interlock(s) which provide for the following: Maintenance of spent fuel level.

Proposed Question:

Which of the following design features of the Spent Fuel Pool (SFP) cooling system mitigates inadvertent draining of the SFP?

A. SFP pump high discharge flow trip.

B. SFP Automatic makeup.

C. Discharge piping anti-siphon hole.

D. SFP pump low level trip.

Proposed Answer: C

Explanation:

Answer A is incorrect. SFP pumps will not trip on high discharged flow.

Answer B is incorrect. There is no automatic makeup to the SFP based on low level.

Answer C is correct. The SFP cooling return line (Discharge Piping) has a 0.5” hole in the pipe to act as an anti-siphon feature to prevent draining the pool to the normal discharge location.

Answer D is incorrect. The is no automatic trip of the SFP pumps based on low level in the SFP.

Technical Reference(s): LB-7, Spent Fuel Pool Cooling System, Page 8, Rev. 11.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40508 - Describe SFP Cooling System components.

Question Source: Bank #
Modified Bank # INPO - 28903 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 63

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	035/A1.01	
Importance Rating	3.6	3.8

K/A: **Steam Generator System (S/GS)** - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the S/GS controls including: S/G wide and narrow range level during startup, shutdown, and normal operations.

Proposed Question:

Unit 1 is at 2% power.

Auxiliary Feedwater is in service and Main Feedwater is not available.

Assuming no operator action, which Steam Generator 1-1 Level channel failure will cause a P-14 “Steam Generator High-High Level Turbine Trip” actuation?

A. Narrow Range Channel LI-517 fails LOW.

B. Narrow Range Channel LI-518 fails HIGH.

C. Narrow Range Channel LI-519 fails LOW.

D. Wide Range Channel LI-501 fails HIGH.

Proposed Answer: C

Explanation:

Answer A is incorrect. Narrow range channel LI-517 is not used by the AFW system to control level in the Steam Generator.

Answer B is incorrect. Narrow range channel LI-518 is not used by the AFW system to control level in the Steam Generator, and would only bring in 1 P-14 Bistable.

Answer C is correct. Narrow range channel LI-519 is used to control SG level when AFW is in service. If the level channel fails low the AFW flow control system will provide maximum AFW flow and a P-14 signal will occur at 75% narrow range.

Answer D is incorrect. Wide range channel LI-501 is not used by the AFW system to control level in the Steam Generator.

Technical Reference(s): LA-5, Steam Generators, Page 22, Rev. 10.
LD-1, Auxiliary Feedwater System, Page 30, Rev. 11.
OIM Steam Generator Indications, Page C-8-5, Rev. 20.
OIM Protection Interlocks and Permissives, Page B-6-2, Rev. 26.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40559 - Describe controls, indications, and alarms associated with the Steam Generators.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ______

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 64 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	045/A3.04	
Importance Rating	3.4	3.6

K/A: **Main Turbine Generator (MT/G) System** - Ability to monitor automatic operation of the MT/G System, including: T/G trip.

Proposed Question:

The plant was operating at 100% power when condenser vacuum dropped below the turbine trip set point.

All plant systems functioned properly EXCEPT PCV-23, Auto Stop Oil EH Fluid Drain Valve (Interface valve), remains closed.

The turbine will:

A. Automatically trip due to denergizing the trip block solenoid valve.

B. NOT automatically trip, but manual trip is available from the control room.

C. Automatically trip by directly dumping EH emergency trip header fluid to drain via a solenoid actuated valve.

D. NOT automatically trip, but manual trip from the manual trip lever on the front standard is available.

Proposed Answer: C

Explanation:

Answer A is incorrect. The trip block solenoid valve is energized to trip the Turbine, not deenergized.

Answer B is incorrect. The Turbine will automatically trip via SV-40 on the EH header.

Answer C is correct. The Turbine will automatically trip via the EH Backup interface valve, SV-40.

Answer D is incorrect. The Turbine will automatically trip; manual trip will not be effective.

Technical Reference(s): LC-3B, Turbine Control Oil, Pages 18 - 22 & 36 - 43, Rev. 9
OIM Main Turbine Auto Stop Oil, Page C-3-3, Rev. 27.
OIM Turbine Trip Logic, Page C-3-5, Rev. 26.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 37644 - Analyze automatic features and interlocks associated with the Turbine Control Oil System.

Question Source: Bank # A-0131
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

10 CFR Part 55 Content: Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

Comments:

RO Question 65

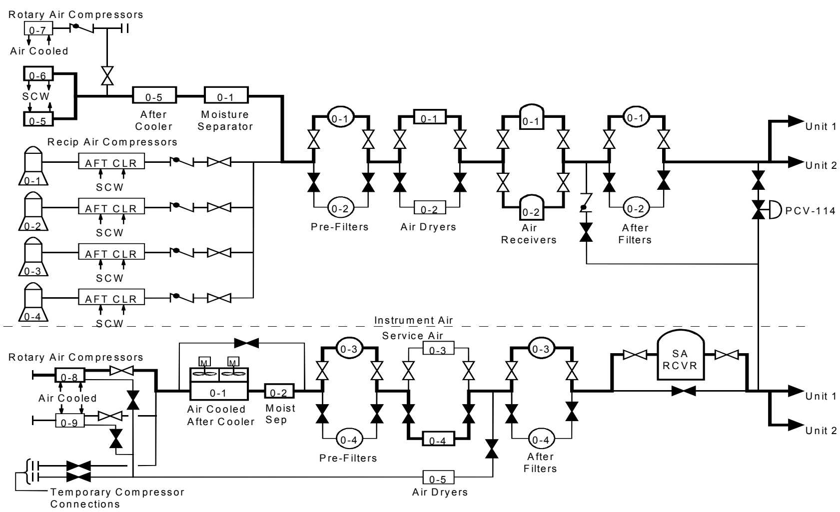
Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	079/A2.01	
Importance Rating		

K/A: **Station Air System (SAS)** - Ability to (a) predict the impacts of the following malfunctions or operations on the SAS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Cross-connection with IAS.

Proposed Question:

Using the following drawing of the Station Air System:



Operators are aligning the Service Air system to back feed into the Instrument Air System thru PCV-114.

What is the reason the operating procedure has the service air receiver isolated.

A. The higher pressure in the Service Air Receiver will back down the Instrument Air Compressors.

B. To prevent contamination of the Instrument Air system from the Service Air Receiver.

C. To increase service air flowrate.

D. The Service Air Dryers are not used in this lineup, which would cause excessive moisture to collect in the Service Air Receiver.

Proposed Answer: B

Explanation:

Answer A is incorrect. Service and Instrument air systems are normally at the same pressure so backing down of the instrument air compressors is not a concern.

Answer B is correct. To prevent contamination from the service air receiver back to the instrument air system.

Answer C is incorrect. Capacity of system is not related to having service air receiver available and will not affect back pressure.

Answer D is incorrect. Instrument air dryers remain in service during this line up.

Technical Reference(s): LK-1, Compressed Air System, Page 40, Rev. 9
STG K-1, Compressed Air System, Page 3-1, Rev. 12.
OP K-1:V Service Air System – Make Available and Place In Service, Page 3, Rev. 23.

Proposed references to be provided to applicants during examination: CAS-01

Learning Objective: 7232 - Describe the operation of the Compressed Air System.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5
55.43 ____

10 CFR Part 55 Content: Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure, and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

RO Question 66

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.1.2	
Importance Rating	4.1	4.4

K/A: **Generic** - Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question:

Unit 1 is at 100% Power.

The Control Operator notes that all channels of Pressurizer pressure have dropped below 1950 psig.

No automatic actions have occurred.

What action should the Control Operator take?

A. Inform the Shift Foreman and wait for direction.

B. Immediately inform the Shift Foreman and trip the reactor.

C. Immediately energize all Pressurizer Heaters and inform the Shift Foreman.

D. Channel check other indications to diagnose the cause of the pressure drop.

Proposed Answer: B

Explanation:

Answer A is incorrect. It would be inappropriate to wait for the shift foreman to make a decision and to wait for direction, since the setpoint for the trip has been exceeded.

Answer B is correct. OP1.DC10, Conduct of Operations states that licensed operators are expected to manually initiate engineered safety feature (ESF) actions, (Reactor trips and Safety Injections), when a plant parameter is approaching an automatic setpoint. In this case Pressurizer Pressure is at the reactor trip setpoint.

Answer C is incorrect. The reactor trip setpoint for the Pressurizer low pressure has been reached the appropriate action is to trip the reactor not to energize the Pressurizer heaters.

Answer D is incorrect. The reactor trip setpoint for the Pressurizer low pressure has been reached the appropriate action is to trip the reactor not channel check other indicators.

Technical Reference(s): LADM-1, Conduct of Operations, Page 11, Rev. 10.
OP1.DC10, Conduct of Operations, Page 15, Rev. 15.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 41662 - Describe the general duties and responsibilities of Control Room Operators.

- Expectations for initiating reactor trips and safety injections.

Question Source: Bank # P-7066
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ________

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 67 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.1.37	
Importance Rating	4.3	4.6

K/A: **Generic** - Conduct of Operations: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question:

Only licensed operators are permitted to operate which of the following controls per OP1.DC10, Conduct of Operations?

1. Rod Control
2. Makeup Control
3. Feedwater Control
4. Pressurizer Control
5. Turbine Control

A. 1 and 2 only.

B. 2, 3 and 5 only.

C. 3, 4 and 5 only.

D. 1, 2 and 5 only.

Proposed Answer: D

Explanation:

Answer A is incorrect. Feedwater controls are not defined as controls that can directly affect reactivity or power level of the reactor.

Answer B is incorrect. Feedwater controls are not defined as controls that can directly affect reactivity or power level of the reactor.

Answer C is incorrect. Pressurizer controls are not defined as controls that can directly affect reactivity or power level of the reactor.

Answer D is correct. Controls are specifically considered to be Control Rods, makeup controls and turbine controls, all three directly affect the reactivity of power level of the reactor.

Technical Reference(s): LADM10, Conduct of Operations, Page 8, Rev. 10.
OP1.DC10, Conduct of Operations, Page 5, Rev. 15.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 41659 - Describe the requirements for the operation of equipment which may effect reactivity, including:

- . Definition of reactor controls

Question Source: Bank # ______
Modified Bank # P-5232 (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 68

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.2.12	
Importance Rating	3.7	4.1

K/A: **Generic** - Equipment Control: Knowledge of surveillance procedures.

Proposed Question:

Which of the following surveillance tests would require an Operations Test Director to be designated?

A. M-9A, Diesel Engine Generator Routine Surveillance Test.

B. I-1B, Routine Daily Checks Required by Licenses.

C. R-19, Shutdown Margin Determination.

D. I-1A, Routine Shift Checks Required by Licenses.

Proposed Answer: A

Explanation:

Answer A is correct. This test requires operation of plant equipment along with data taking and should have a Test Director. Significant coordination is required with field and control room operators.

Answer B is incorrect. This test requires only simple data to be taken, no test Director is required.

Answer C is incorrect. This test requires only simple data to be taken, no test Director is required.

Answer D is incorrect. This test requires only simple data to be taken, no test Director is required.

Technical Reference(s): OP1.DC10, Conduct of Operations, Page 57, Rev. 15.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 9724 - Explain Operator responsibilities during maintenance, test and surveillance activities.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 10
55.43 ____

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 69 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.2.14	
Importance Rating	3.9	4.3

K/A: **Generic** - Equipment Control: Knowledge of the process for controlling equipment configuration or status.

Proposed Question:

Excess Letdown has been placed in service per the operating procedure OP B-1A:IV.

The control room operator placed "Pink Tags" on the Excess Letdown Isolation valves.

The use of the Pink Tag is:

A. Acceptable, only if a LBIE screen is performed prior to the end of the shift.

B. Unacceptable, unusual system lineups are tracked by using of the abnormal status board only.

C. Unacceptable, can only be used in conjunction with an approved clearance.

D. Acceptable, can be used to draw attention to components that are not normally in service, for short durations.

Proposed Answer: D

Explanation:

Answer A is incorrect. LBIE screen is not required for Pink Tags; they do not control status of equipment.

Answer B is incorrect. While the system lineup is required to be placed on the abnormal status board, nothing precludes placing a pink tank on the component as well.

Answer C is incorrect. Pink Tags are not used for clearances.

Answer D is correct. The Pink Tag process may be used to designate an unusual component status and to draw attention to off normal components.

Technical Reference(s): LADM-5, Misc. Operations Department Polices, Page 17, Rev. 10.
OP1.DC10, Conduct of Operations, Page 45, Rev. 15.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3654 - Identify and discuss Operations Department policy statements.

Comment [MRN4]: (3654)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 70

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.2.43	
Importance Rating	3.0	3.3

K/A: **Generic** - Equipment Control: Knowledge of the process used to track inoperable alarms.

Proposed Question:

A Main Control room annunciator alarm has been alarming and resetting at a rate greater than six times in an hour, and has become a distraction for the control room operators.

What is the proper classification of the alarm, and what actions are required to defeat the alarm per OP1.DC24, Control of Annunciator Problems?

A. Standing Annunciator, Control Operator will defeat the alarm at the Annunciator Maintenance Terminal in the cable spreading room.

B. Standing Annunciator, Maintenance will defeat the alarm at the Annunciator Maintenance Terminal in the cable spreading room.

C. Continually Alarming Annunciator, Control Operator will defeat the alarm at the Annunciator Maintenance Terminal in the cable spreading room.

D. Continually Alarming Annunciator, Maintenance will defeat the alarm at the Annunciator Maintenance Terminal in the cable spreading room.

Proposed Answer: D

Explanation:

Answer A is incorrect. Alarm is classified as a Continually Alarming Annunciator and the control operator will not defeat the alarm.

Answer B is incorrect. Alarm is classified as a Continually Alarm Annunciator.

Answer C is incorrect. Control Operator will not defeat the alarm.

Answer D is correct. Alarm is classified as a Continually Alarming Annunciator and Technical Maintenance will defeat the alarm in the cable spreading room.

Technical Reference(s): STG J12, Annunciator System, Page 2-30, Rev. 8.
OP1. DC24, Control of Annunciator Problems, Page 2, Rev. 7B.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7122 - Explain the operation of Alarm Computer system.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 10
55.43 ____

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 71

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.3.4	
Importance Rating	3.2	3.7

K/A: **Generic** – Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question:

Which of the following describes the 10CFR20 Limits and the Diablo Canyon **Administrative Limit** for radiation exposure for a calendar year?

	10CFR20 Limit	DCPP Admin Limit
A.	4500 mREM	2000 mREM
B.	4500 mREM	4000 mREM
C.	5000 mREM	2000 mREM
D.	5000 mREM	4500 MREM

Proposed Answer: D

Explanation:

Answer A is incorrect. 10CFR20 limit is 5000 mREM and not 4500 mREM, 4500 mREM is the Admin limit, DCPP Admin Guideline is 2000 mREM.

Answer B is incorrect. 10CFR20 limit is 5000 MREM and not 4500 mREM , 4500 mREM is the Admin limit, 4000 mREM is 90% of 4500 mREM.

Answer C is incorrect. 10CFR20 limit is 5000 mREM and the DCPP Admin Guideline is 2000 mREM.

Answer D is correct. 10CFR20 limit is 5000 mREM and the DCPP Admin limit is 4500 mREM.

Technical Reference(s): RP1.ID6, Personnel Dose Limits and Monitoring Requirements, Attachment 8.1, Rev. 8.
OIM Administrative Radiation Exposure Limits, Page S-1-1, Rev. 20.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: STATE the DCPP administrative exposure guidelines.

Question Source: Bank #
Modified Bank # INPO - 30401 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12
55.43

10 CFR Part 55 Content: Radiological safety principles and procedures.

Comments:

RO Question 72

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.3.13	
Importance Rating	3.4	3.8

K/A: **Generic** – Radiation Control: Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters.

Proposed Question:

An accessible area of the Auxiliary Building has the following readings:

- General Area Radiation level of 1250 mrem/hr.

Which of the following area postings should be displayed at the entrance to this area?

A. Locked High Radiation Area

B. High Radiation Area.

C. Radiation Area only.

D. Very High Radiation Area.

Proposed Answer: A

Explanation:

Answer A is correct. Locked Hi Radiation Area is an area accessible to personnel with radiation levels greater than 1000 mrem/hr.

Answer B is incorrect. A High Radiation area is an area accessible to personnel with radiation levels greater than 100 mrem/hr.

Answer C is incorrect. A Radiation area is an area accessible to personnel with radiation levels greater than 5 mrem/hr.

Answer D is incorrect. A very high Radiation Area is area accessible to personnel with radiation levels greater than 500 rads/hr.

Technical Reference(s): LPM-2, Portable Rad Inst and Rad Con Review. Page 6, Rev. 6.
RCP D-240, Radiological Postings, Pages 1-4, Rev. 18.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: DEFINE the following:
1. Radiological Controls Area
2. Radiation Area
3. High Radiation Area
4. Locked-High Radiation Area
5. Very High Radiation Area
6. Surface Contamination Area
7. Airborne Radioactivity Area

Comment [TRP5]: Changed High – High Radiation Area to Locked – High Radiation Area, and Hot Particle Zone as it is now a Red Zone on the CAR rating for Surface Contamination Areas. All this per Bop Clark – Radiation Protection.

Question Source: Bank #
Modified Bank # INPO - 20261 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12
55.43

10 CFR Part 55 Content: Radiological safety principles and procedures.

Comments:

RO Question 73

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.4.14	
Importance Rating	3.8	4.5

K/A: **Generic** - Emergency Procedures/Plan: Knowledge of general guidelines for EOP usage.

Proposed Question:

Per the Emergency Operating Procedure (EOP) rules of usage, when do the items in an EOP Foldout page apply?

A. At any step in the procedure, unless otherwise noted in the procedure.

B. Only when assigned to an individual during a procedure transition brief.

C. Only after any transition from EOP E-0, Reactor Trip or Safety Injection.

D. When a step in the procedure directs the Operator to the Foldout page.

Proposed Answer: A

Explanation:

Answer A is correct. Foldout pages are applicable at any step within the procedure, or procedure set that they appear.

Answer B is incorrect. Foldout pages for E-0 are not assigned to an individual at a briefing, but they do apply throughout the procedure.

Answer C is incorrect. Foldout pages for E-0 apply during it’s use, they are the responsibility of the SFM and WCSFM.

Answer D is incorrect. Steps in procedures do not direct the operator to foldout page items.

Technical Reference(s): LPERULE, EOP Rules of Usage, Page 8, Rev. 11.
OP1.DC10, Conduct of Operations, Page 56, Rev. 15.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5435 - State rules of usage for Emergency and Abnormal Operating Procedures as specified in the DCPP EOP User's Guide and OP1.DC10.

Question Source: Bank # P-5931
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 74 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.4.25	
Importance Rating	3.3	3.7

K/A: **Generic** - Emergency Procedures/Plan: Knowledge of fire protection procedures.

Proposed Question:

A fire has been reported in the Unit 2 250v DC battery room.

The fire brigade has been dispatched to the scene of the fire.

In addition to bringing the fire preplans, gurney and defibrillator to a fire scene, which of the following functions does the designated OPS Responder also perform per CP M-6 "Fire"?

1. Takes a direct role in the fire fighting efforts.
2. Performs emergency notifications to offsite personnel.
3. Communicates with the control room on status of fire fighting efforts.
4. Notifies the Control Room when any power block door is propped open.

A. 2 and 3 only.

B. 1 and 4 only.

C. 1 and 2 only.

D. 3 and 4 only.

Proposed Answer: D

Explanation:

Answer A is incorrect. OPS Responder does not perform emergency notifications.

Answer B is incorrect. OPS Responder does not take a active role in fire fighting efforts.

Answer C is incorrect. Ops Responder does not take an active role in fighting the fire nor does he perform emergency notifications.

Answer D is correct. OPS Responder communicates with the control room on status of fire fighting efforts and notifies the control room when any power block door is propped open.

Technical Reference(s): CP M-6 Fire, Attachment 7.2, Operations responder Checklist, Page 1, Rev. 31.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: Explain the duties and responsibilities of the Operations Responder during a medical, hazardous material, or fire emergency, including information that needs to be relayed between Control Room personnel and the Fire Brigade Leader.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

RO Question 75

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	3	
Group #	N/A	
K/A #	G 2.4.37	
Importance Rating	3.0	4.1

K/A: **Generic** - Emergency Procedures/Plan: Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question:

Shift Manager has declared an UNUSUAL EVENT based on a leak in containment of greater than 25 gpm.

Which individual has the primary responsibility for the emergency response effort?

A. Site Emergency Coordinator.

B. Interim Site Emergency Coordinator.

C. Recovery Manager.

D. Operations Manager.

Proposed Answer: B

Explanation:

Answer A is incorrect. Site Emergency Coordinator position is staffed at the ALERT and higher emergency classifications, this position is not staffed on an Unusual Event.

Answer B is correct. ISEC position is staffed by the Shift Manager at the UE action level and above.

Answer C is incorrect. Recovery Manager position is staffed at the ALERT and higher emergency classification levels, this position is not staffed on an Unusual Event.

Answer D is incorrect. Operation Manager will be notified of the Unusual Event, but does not assume command and control of the event.

Technical Reference(s): LEP-2, Emergency Plan Procedures, Page 47, Rev. 10.
EP G-2, Interim Emergency Response Organization, Page 1

and Attachment 6.1, Rev. 31.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3705 - As describes in EP G-2, state the responsibilities of the following:

- Interim Site Emergency Coordinator (ISEC)
- Interim Emergency Operations Coordinator (IEOC)
- Control Room Communicator #1
- Control Room Communicator #2
- Emergency Evaluation Coordinator (EEC)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 10
55.43 ____

10 CFR Part 55 Content: Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

SRO Question 1

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #		009/EA2.14
Importance Rating		4.4

K/A: **Small Break LOCA** - Ability to determine and interpret the following as they apply to a small break LOCA: Actions to be taken if PTS limits are violated.

Proposed Question:

While operating at 100% power, a LOCA occurred.

Safety Injection was actuated, and E-0 "Reactor Trip or Safety Injection" was entered.

Transition criteria have been met for E-1 “Loss of Reactor or Secondary Coolant” based on the Reactor Coolant System not being intact.

The following plant parameters exist:

- Secondary radiation monitors show NO upward trends, NO alarms, and NO upward spikes
- Containment pressure is 23 PSIG and TRENDING UP.
- All RCS Cold Leg Temps are 230°F and TRENDING DOWN.
- RCS Pressure is 1000 PSIG and TRENDING DOWN.
- All S/G pressures are 400 PSIG and STABLE.

Based on the above parameters the Shift Foreman should:

A. GO TO FR-P.1, “Response to Imminent Pressurized Thermal Shock Condition."

B. Transition to E-1, "Loss of Reactor or Secondary Coolant."

C. GO TO FR-P.2, "Response to Anticipated Pressurized Thermal Shock Condition."

D. GO TO, FR-Z.1, “Response to High Containment Pressure."

Proposed Answer: A

Explanation:

Answer A is correct. Based on All T-Colds being to the right of limit A and being less than 240°F it is appropriate that a transition to FR-P.1 be performed, This is a magenta path.

Answer B is incorrect. A PTS concern (magenta path) has a higher priority than E-1, a transition to FR-P.1 is required.

Answer C is incorrect. All cold leg temperatures are below 240°F, if temperatures were above 240°F and also above 270°F then a transition to FR-P.2 would be appropriate.

Answer D is incorrect. Current containment pressure leads to a Magenta path for containment, but the PTS concern is a higher priority.

Technical Reference(s): LPE-FR, Functional Restoration Guidelines, pages 5,6 & 7, Rev. 8.
EOP F-0, Critical Safety Function Status Trees, Rev. 14

Proposed references to be provided to applicants during examination: EOP F-0

Learning Objective: 38107 - Apply the Rules of Usage in EOPs for the CSFSTs and FRGs, including:

- the six status trees
- the priority of use of the status trees
- the priority of use of the color of each CSF
- when to monitor and/or implement the CSFSTs and FRGs

Question Source: Bank # B-0188
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 2

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #		011/2.4.4
Importance Rating		4.7

K/A: **Large Break LOCA** - Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

A Large Break LOCA has occurred, resulting in a reactor trip and safety injection.

EOP E-1, "Loss of Reactor or Secondary Coolant" is in progress.

The following conditions currently exist:

- RCS Pressure is 40 PSIG and stable.
- RHR system flow is at 0 gpm.
- RWST Level is 32% and TRENDING DOWN SLOWLY.
- Containment Recirculation Sump level is 94 feet and TRENDING UP SLOWLY.

Based on the above parameters the Shift Foreman should:

A. Continue in E-1, "Loss of Reactor or Secondary Coolant."

B. GO TO E-1.3, "Transfer to Cold Leg Recirculation."

C. GO TO FR-Z.2, "Response to Containment Flooding."

D. GO TO ECA-1.2, "LOCA Outside Containment."

Proposed Answer: B

Explanation:

Answer A is incorrect. E-1 foldout page directs the operator to transition to E-1.3 when RWST level reaches 33%.

Answer B is correct. E-1 Foldout page item is met for transition to E-1.3 based on the RWST level being below 33%, RHR flow is zero because the RHR pumps tripped off automatically on low RWST level of 33%. 92 ft is the expected level for the Containment Recirculation Sump following a DBA LOCA.

Answer C is incorrect. Containment flooding is only a concern if Recirc sump level is greater than 95.75 ft..

Answer D is incorrect. LOCA outside containment has not occurred; level in containment and the RWST are at the expected values. E-1.3 will check for this condition.

Technical Reference(s): LPE-1A, Loss of Coolant Response, Page 25, Rev. 10.
EOP E-1, Loss of Reactor or Secondary Coolant, Step 13, page 14 and Foldout Page, Rev. 24.
EOP E-1.3, Transfer to Cold Leg Recirculation, Step 6, Page 4, Rev. 25.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event.

Question Source: Bank # ____
Modified Bank # B-0101 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 3 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		026/2.2.36
Importance Rating		4.2

K/A: **Loss of Component Cooling Water (CCW)** - Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question:

Unit 1 is at Rated Thermal Power. At 0400 this morning (June 13), the Unit 1 CCW Pump 1-1 was declared inoperable after being tagged out for planned maintenance.

At 0700 this morning, the Unit 1 CCW Pump 1-2 was declared inoperable due to a failed relay in the pump's electrical supply breaker.

At 1000 this morning, the Unit 1 CCW Pump 1-1 was restored to service.

Based on this sequence of events, Technical Specification 3.7.7 (attached) requires Unit 1 to be in Mode 3 no later then _________ on June 16. Assume there are no technical specification extensions in effect.

A. 0400

B. 1000

C. 1300

D. 1600

Proposed Answer: B

Explanation:

Answer A is incorrect. This is the requirement to be in compliance per action statement 3.7.7.A but does not include the 6 hours provided by action statement 3.7.7.B.

Answer B is correct. When the first pump is taken out of service, the 72 hour TS clock starts per LCO 3.7.7.A. When the second pump is declared inoperable a second TS clock starts per 3.0.3 and runs in parallel with the first TS clock. When pump 1-1 is restored to service, the TS clock associated with 3.0.3 stops and the clock associated with 3.7.7.A continues to run. Since one CCW loop remains unavailable, the clock associated with action 3.7.7.A expires at 0400 (distractor A). Action 3.7.7.B is entered requiring the unit to be in Mode 3 6 hours later per 3.7.7.B. Thus, the unit must be in Mode 3 no later than 1000.

Answer C is incorrect. This answer is based on the TS clock starting at 0700 when the second pump is declared INOP. This is incorrect since the one pump out of service TS clock had already started at 0400.

Answer D is incorrect. This answer assumes the TS clock for 3.7.7.A starts when one of the inoperable pumps is returned to service leaving one pump inoperable. The single pump inoperable TS clock (3.7.7.A) started at 0400 and continues to run since there was no time when both pumps were operable to reset this clock. Therefore, this answer is incorrect.

Technical Reference(s): LF-2, Component Cooling Water, Page 36, Rev. 11.
STG F2, Component Cooling Water System, Page 4-1, Rev. 15.
Tech Specs 3.7.7, Vital Component Cooling Water (CCW) System, Pages15 & 16, Rev. 9.
Tech Spec 3.7.7, Vital Component Cooling Water (CCW) Basis, Pages 36 to 40, Rev. 4c.
Tech Specs Example 1.3-2.
OIM Component Cooling Water, Page F-2-1, Rev. 27.

Proposed references to be provided to applicants during examination: T.S. 3.7.7 & TS section 1.3 for completion times

Learning Objective: 66054 - Discuss significant Technical Specifications and Equipment Control Guidelines associated with the CCW System.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2

10 CFR Part 55 Content: Facility operating limitations in the technical specifications and their bases.

Comments:

SRO Question 4

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #		058/AA2.03
Importance Rating		3.9

K/A: **Loss of DC Power** - Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on to operate and monitor plant systems.

Proposed Question:

Unit 1 is at 100% power.

The following indications are received in the main control room:

- Feedwater Flow has dropped to zero on all Steam generators
- 4 kV bus G Aux, startup and Diesel Generator breaker have no indicating lights lit.
- Main Annunciator system has received numerous seemingly unrelated alarms.
- 4 kV Bus G pump control switch indicating lights are out.
- 4 kV Bus G bus potential white light is lit.

Which of the following procedures will provide the guidance on how to recover from this event?

A. OP AP-4, “Loss of Vital or Non-Vital Instrument AC.”

B. OP AP-15, “Loss of Feedwater Flow”

C. OP AP-23, “Loss of Vital DC Bus”

D. OP AP-27, “Loss of Vital 4 kV and/or 480V Bus”

Proposed Answer: C

Explanation:

Answer A is incorrect. Multiple seemingly unrelated annunciator alarms will occur, but breaker control is DC power related and not AC power related.

Answer B is incorrect. Loss of Feedwater flow is caused by the Feedwater solenoids de-energizing and failing closed do to loss of DC Power.

Answer C is correct. Loss of DC will affect the Feedwater solenoids; failing closed the Feedwater regulating valves, also supplies DC power for breaker control, which is the best indication of a failed DC bus.

Answer D is incorrect. AP-27 is used to recover from a loss of vital AC bus, Vital bus is lost do to loss of DC power.

Technical Reference(s): LPA-23, Loss of Vital DC Bus, Page 6, Rev. 8.
OP AP-23, Loss of Vital DC Bus, Appendix B, Pages 9 -13, Rev. 12.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 5

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		065/AA2.05
Importance Rating		4.1

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

Proposed Question:

Unit 1 is at 100% power.

A seismic event occurs, resulting in the following plant indications:

- Instrument Air Pressure is LOWERING rapidly.
- All Plant Instrument Air Compressors are running and loaded.
- All Steam Generator Narrow Range Levels are at 30% and LOWERING rapidly.
- Pressurizer Level is 55% and STABLE.
- Pressurizer Pressure is 2235 and STABLE.

Assume other plant parameters have not changed.

Which of the following procedures and actions should the Shift Foreman use to respond to this event?

A. OP AP-9, “Loss of Instrument Air”; and direct that the reactor be tripped.

B. OP AP-9, “Loss of Instrument Air”; and commence a rapid plant shutdown.

C. OP AP-25, Rapid Load Reduction or Shutdown” and commence a rapid plant shutdown.

D. OP AP-15, Loss of Feedwater Flow” and direct that the reactor be tripped.

Proposed Answer: A

Explanation:

Answer A is correct. Continuous action in step 1 of AP-9 verifies control of the plant. If the plant is not under control, “SG Water Levels” then the reactor is verified to be tripped.

Answer B is incorrect. Continuous action in step 1 of AP-9 verifies control of the plant. If the plant is not under control, it is not ramped down, it is tripped.

Answer C is incorrect. AP-25 is not appropriate for this situation; the plant is tripped when control of SG level is lost due to failure of Main Feedwater Regulation valves.

Answer D is incorrect. AP-15 is not appropriate for this situation; it is used for loss of One Main Feedwater pump at power and not loss of feed flow caused by the main Feedwater regulation valves failing closed.

Technical Reference(s): LPA-9, Loss of Instrument Air, Page 10, Rev. 9.
OP AP-9, Loss of Instrument Air, Page 2, Rev. 24.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operation procedure to be used to mitigate an operational event.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 6

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		1
K/A #		E11/2.1.7
Importance Rating		4.7

K/A: **Loss of Emergency Coolant Recirculation** - Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question:

Crew is performing E-1.3, "Transfer to Cold Leg Recirculation."

FCV-8982B, RHR Pump 2 Suction from Containment Recirculation Sump, is found frozen in the CLOSED position and can NOT be opened locally.

If RHR pump 1-1 is unavailable because of a Bus differential trip, the Shift Foreman should…

A. Remain in E-1.3 until one Recirculation path can be established.

B. Go to ECA-1.1, "Loss of Emergency Coolant Recirculation."

C. Implement ECA-0.3, “Restore 4 kV Buses” to restore power to RHR Pump 1-1.

D. Maintain RHR pump 1-2 suction from the RWST and commence filling the RWST while continuing in E-1.3.

Proposed Answer: B

Explanation:

Answer A is incorrect. Remaining in E-1.3 will not provide any actions to reestablish core cooling, which was secured when the RHR pump was tripped at 33% level in the RWST..

Answer B is correct. Per E-1.3 Step 8f RNO, if no RHR pumps can be started (1-1 has no power and 1-2 has no suction flow path) then the crew is directed to ECA-1.1.

Answer C is incorrect. Restoration of power to the buses will not be able to be accomplished with a Bus differential and is not an option in E-1.3..

Answer D is incorrect. RHR pump suction from the RWST is only possible if the RWST low level cut out switches are cut out in the cable spreading room, this is not an option in E-1.3.

Technical Reference(s): LPE-1C Recirculation Modes and LOCA Outside Containment, Page 27, Rev. 9.
EOP E-1.3, Transfer to Cold Leg Recirculation, Page 7, Rev. 25.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3552 - Given initial conditions, assumptions, and symptoms, determine the correct Emergency Operating Procedure to be used to mitigate an operational event.

Question Source: Bank # B-0184
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 7 (Rev. 1)

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		1
Group #		2
K/A #		33/AA2.04
Importance Rating		3.6

K/A: **Loss of Intermediate Range Nuclear Instrumentation** - Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Satisfactory overlap between source-range, intermediate-range and power-range instrumentation.

Proposed Question:

Given the following conditions:

- Plant startup is in progress.
- Intermediate range channel N-35 is reading 7 X 10^{-10} amps.
- Intermediate range channel N-36 is reading 7 X 10^{-10} amps.

Channel N-36 suddenly fails off-scale high.

Based on the above parameters the Shift Foreman should:

A. Continue plant startup and comply with TS 3.3.1 for one intermediate range neutron flux channel inoperable.

B. Continue the plant startup and enter OP AP-5, “Malfunction of Eagle 21 Protection on Control Channel”, due to the intermediate range channel failure.

C. Enter EOP E-0, “Reactor Trip or Safety Injection”, due to exceeding the source range high positive rate trip set point.

D. Enter EOP E-0, “Reactor Trip or Safety Injection”, due to exceeding the intermediate range high flux trip setpoint.

Proposed Answer: D

Explanation:

Answer A is incorrect. Plant will trip based intermediate range hi flux, tech spec action is not required to be entered.

Answer B is incorrect. Plant will trip based on intermediate range hi flux, OP AP-5 will not be entered, no actions in AP-5 for intermediate range, power range only.

Answer C is incorrect. Source range hi flux trip will be blocked at this power level based on overlap of source range and intermediate range channels. Reactor will trip but, not for this reason.

Answer D is correct. Reactor will trip because intermediate range hi flux trip is not blocked at this power level. E-0 will be entered due to the trip.

Technical Reference(s): OIM Page B-4-1, Rev. 27.
LB-4, Excore Nuclear Instrumentation System, Page 58, Rev. 10.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 5992 - Discuss abnormal conditions associated with the Excore Nuclear Instrumentation System.

Question Source: Bank # ______
Modified Bank # A-0123 (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 8

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #		036/2.1.23
Importance Rating		4.4

K/A: **Fuel Handling Incidents** - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question:

Unit 1 is in a refueling outage.

A fuel assembly is being lowered from the manipulator crane into the upender in containment.

A HIGH RADIATION alarm, PK11-21 is received.

- Containment Area Radiation Monitor RE-2, is reading 57 mR/hr.

The refueling crew notes gas bubbles are being released from the assembly.

To place the fuel assembly in a "Safe Position," the fuel assembly should be:

A. Moved to the RCCA change fixture carriage.

B. Returned to the reactor vessel and lowered to a safe location.

C. Lowered in to the upender and the upender frame should be lowered.

D. Lowered in the refueling mast, gripper engaged, to the reactor cavity floor.

Proposed Answer: C

Explanation:

Answer A is incorrect. Placing the assembly in the RCCA fixture will not allow the fuel to be lowered to a safe location; this is used to change out RCCA and is located away from the upender and transfer cart.

Answer B is incorrect. Moving the assembly to the vessel could cause more damage, since the assembly is in the transfer system, the best location is the transfer system.

Answer C is correct. Fuel assembly should be lowered into the upender and the upender frame should be lowered to place the fuel in the lowest position possible in the refueling cavity.

Answer D is incorrect. This location is not specified in OPAP-21 as a safe location.

Technical Reference(s): LPA-21, Irradiated Fuel Damage, Page 8, Rev. 8.
OP AP-21, Irradiated Fuel Damage, Page 2, Rev. 10.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 6488 - State the safe locations for fuel assemblies for fuel handling incidents in the fuel building or containment.

Question Source: Bank # B-0350
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ________
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 9

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #		037/AA2.05
Importance Rating		3.3

K/A: **Steam Generator (S/G) Tube Leak** - Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Past history of leakage with current problem.

Proposed Question:

Unit 1 is operating at 100%.

Previous shift Chemistry reports showed total SG tube leakage was of 45 gpd.

Latest sample was taken at 1200 hours indicates 160 gpd leakage in SG 1-1.

Air Ejector Off-Gas monitors, RM15 and RM15R count rates have TRENDED UP.

The leak rate trended up in a linear fashion.

Which of the following action(s) is required?

A. Manually initiate a Safety Injection and GO TO EOP E-0, “Reactor Trip or Safety Injection.

B. Use OP L-4, "Normal Operation at Power," to be in Hot Standby by 1200 tomorrow.

C. Use OP L-4, "Normal Operation at Power," to be in Hot Standby by 1800 today.

D. Manually initiate a Reactor Trip and Implement AP-3, “Steam Generator Tube Failure”

Proposed Answer: C

Explanation:

Answer A is incorrect. Safety injection is not required for this situation; leak is well within the capacity of the charging and makeup systems.

Answer B is incorrect. Action is from Action Level 2, which is met, but the leak being larger than 150 gpd meets the Action Level 3b criterion which is more restrictive.

Answer C is correct. Action Level 3b criterion is met, which requires the plant to be in Mode 3 within 6 hours and Mode 5 within 36 hours.

Answer D is incorrect. Manual Reactor trip is not required for this situation, control of plant is not jeopardized based on this leak size, a normal plant shutdown should occur to limit stress on the steam generators.

Technical Reference(s): LPA-3, S/G Tube Leakage and Tube Failure, Page 7, Rev. 11.
OP 0-4, Primary to Secondary Steam Generator Tube Leakage Detection, Page 10, Rev. 18.

Proposed references to be provided to applicants during examination: OP 0-4

Learning Objective: 3794 - Given initial conditions, assumptions, and symptoms, predict the operational implications for any size SG tube leak.

Comment [R3H06]: Upgraded objective to include 5862

Question Source: Bank # B-0096
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam DCPP Dec. 2007

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: 2 Distracters have been replaced.

SRO Question 10

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #		E02/2.2.44
Importance Rating		4.4

K/A: **SI Termination** - Equipment Control: Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question:

While operating at 100% power, a PZR spray valve is stuck in the full OPEN position.

A reactor trip and SI occur due to low PZR pressure.

The transient is terminated when RCPs 1-1 and 1-2 are stopped.

Which emergency procedure will be implemented after the actions of E-0, "Reactor Trip or Safety Injection," are completed?

A. E-1.1, "SI Termination"

B. E-0.1, "Reactor Trip Response"

C. E-0.2, "Natural Circulation Cooldown"

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

Proposed Answer: A

Explanation:

Answer A is correct. When checks for “RCS intact” are performed none of the conditions to transfer to E-1 will be present, E-0 will then check for adequate subcooling and then terminate SI flow, this will be followed by a transition to E1.1 to complete the termination of SI.

Answer B is incorrect A Safety Injection has occurred, transition will not be made to E-0.1.

Answer C is incorrect. 2 Reactor Coolant Pumps have been secured but the other 2 are running so a natural circulation cooldown is not required.

Answer D is incorrect. E-1 would not be entered, there is no transition required based on the condition of the RCS.

Technical Reference(s): LPE-0, Reactor Trip and SI Response, Page 26, Rev. 9.
EOP E-0, Reactor Trip or Safety Injection, Page13, Rev. 32.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 7685 - State the criteria for SI termination.

Question Source: Bank # B-0197
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ________
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 11

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		004/A2.18
Importance Rating		3.1

K/A: **Chemical and Volume Control System (CVCS)** - 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: High VCT level.

Proposed Question:

Unit 1 is at 100% power.

The BOPCO reports that LT-112, Volume Control Tank (VCT) Level indicator has failed HIGH.

The Control Operator reports that PPC Point L0112A, indicates VCT level at 5% and TRENDING DOWN.

Which of the following procedures provides the appropriate mitigation strategy to recover from this event?

A. OP AP-5, “Malfunction of Eagle 21 Protection or Control Channel”, to select an alternate VCT Level control channel.

B. OP AP-19, “Malfunction of Reactor Makeup Control System”, to ensure automatic transfer of Charging Pump suction to the Refueling Water Storage Tank has occurred.

C. OP AP-19, “Malfunction of Reactor Makeup Control System”, to manually transfer suction of the running Charging Pumps to the Refueling Water Storage Tank.

D. OP AP-5, “Malfunction of Eagle 21 Protection or Control Channel”, to ensure automatic transfer of Charging Pump suction to the Refueling Water Storage Tank has occurred.

Proposed Answer: C

Explanation:

Answer A is incorrect. VCT Level transmitters are not part of the plant control scheme that is included in AP-5.

Answer B is incorrect. Automatic swap to the RWST will not occur with LT-112 failed High, so no automatic transfer will occur.

Answer C is correct. Charging pump suction must be manually transferred to the RWST because LT-112 has failed HI. AP-19 provides the steps to realign and vent the pumps if suction is lost.

Answer D is incorrect. AP-5 does not contain steps associated with LT-112 and verification of automatic actions associated with VCT level transmitters.

Technical Reference(s): LPA-19, Malfunction of Reactor Makeup Control System, Page 6, Rev. 8.
OP AP-19, Malfunction of Reactor Makeup Control System, Page 2, Rev. 4.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 40449 - Discuss abnormal conditions associated with the CVCS.

Question Source:
- Bank # ______
- Modified Bank # ______ (Note changes or attach parent)
- New X

Question History: Last NRC Exam N/A

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 12

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		1
K/A #		005/2.4.4
Importance Rating		4.7

K/A: **Residual Heat Removal System (RHRS)** - Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

Unit 1 is in Mode 6 with the following conditions:

- Residual Heat Removal (RHR) system is service with vessel level near mid-loop.
- RHR pumps were shut down due to signs of cavitation.
- Current RVRLIS level is 106' and 6" with level slowly TRENDING DOWN.

Which procedure describes the correct method used to raise the level of the Reactor Coolant System for the given conditions?

A. OP AP-24, "Shutdown LOCA", Aligns and starts one Safety Injection Pump.

B. OP AP SD-2, "Loss of RCS Inventory", Gravity fills the RCS from the RWST.

C. OP AP-16, Malfunction of the RHR System", Gravity fills the RCS from the RWST.

D. OP AP SD-5. Loss of Residual Heat Removal", Restarts the RHR pumps.

Proposed Answer: B

Explanation:

Answer A is incorrect. OP AP-24 is not used in this MODE of operation; it is designed for modes 3 and 4 only.

Answer B is correct. OP AP SD-2 provides several methods to refill the vessel, one method is to gravity fill the RCS from the RWST via 8980, vessel must be refilled prior to starting pumps again.

Answer C is incorrect. OP AP-16 in not used in the MODE of operation, it is designed for MODE 4 only.

Answer D is incorrect. OP AP SD-5 would not be used until level is restored in the vessel, pumps were shutdown due to low level.

Technical Reference(s): LPA-SD, Abnormal Shutdown Procedures, Pages 25 & 28, Rev. 7.
OP AP SD-2, Loss of RCS Inventory, Page 5, Rev. 16.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3477 - Given an abnormal condition, summarize the major actions of the abnormal operating procedure to mitigate an event in progress.

Question Source: Bank # ____
Modified Bank # B-0348 (Note changes or attach parent)
New ____

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 13

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		062/2.4.8
Importance Rating		4.5

K/A: **A.C. Electrical Distribution System** - Emergency Procedures/Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question:

The following conditions exist:

- A grid disturbance has resulted in a reactor trip with loss of offsite power.
- All 3 Diesel Generators started and energized their respective buses.
- Power has been restored to the 230kV switchyard.

The operators have just reached the step of EOP E-0.1, "Reactor Trip Response," to "Check All AC Buses ENERGIZED BY OFFSITE POWER."

Which one of the following procedures should be used to restore power to the non-vital buses?

A. ECA-0.0, “Loss of All Vital AC Power.”

B. EOP ECA-0.3, “Restore 4 kV Buses.”

C. OP AP-26, “Loss of Offsite Power.”

D. OP AP-27, “Loss of Vital 4kV and /or 480V Bus.”

Proposed Answer: C

Explanation:

Answer A is incorrect. ECA-0.0 is entered when all vital buses are de-energized; in this case, All Vital Buses are still energized from the Diesel Generators.

Answer B is incorrect. ECA-0.3 entered or referenced based on vital bus availability, in ths situation, all vital busses are energized. If ECA-0.3 is entered, it would be exited at step one as all vital buses are energized.

Answer C is correct. E-0 is entered on the trip and since all 4 kV vital buses are energized OP AP-26 is implemented to restore the non-vital 4 kV buses.

Answer D is incorrect. OP AP-27 is used to restore power to an individual 4 kV vital bus, when operating.

Technical Reference(s): LPE0, Reactor trip and Safety Injection Response, Page 39, Rev. 9.
LPA-26, Loss of Offsite Power, Page 5, Rev. 5.
EOP E-0.1, Reactor Trip Response, Page 13, Rev. 30.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event.

Comment [R1BT7]: Obj7920

Question Source: Bank #
Modified Bank # P-51673 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 14

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		063/A2.02
Importance Rating		3.1

K/A: **D.C. Electrical Distribution System** - Ability to (a) predict the impacts of the following malfunctions or operations on the D.C. Electrical System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of ventilation during battery charging.

Proposed Question:

Unit 1 is at 100% power.

Maintenance Services requests to work on both Auxiliary Building Switchgear Ventilation system supply and Exhaust fans (S-27 and E-27).

What are the operational concerns and the response by the SFM?

A. Elevated H_2 (Hydrogen) concentration in the Vital Battery Rooms; Implement compensatory measures for blocked open doors per ECG 80.1, Doors required for HELB, HVAC, ECCS Function or Flood Protection.

B. Reduced Battery Capacity, Implement compensatory measures for blocked open doors per ECG 80.1, Doors required for HELB, HVAC, ECCS Function or Flood Protection.

C. Reduced Battery Capacity, Declare the batteries inoperable per Tech Spec 3.8.4, DC Sources - Operating.

D. Elevated H_2 (Hydrogen) concentration in the Vital Battery Rooms; Declare the batteries inoperable per tech Spec 3.8.4, DC Sources - Operating.

Proposed Answer: A

Explanation:

Answer A is correct. H_2 concentration will increase in the battery rooms without sufficient ventilation, compensatory action would be to open doors to provide ventilation from other systems. Blocked open doors are controlled by ECG 80.1.

Answer B is incorrect. Capacity of the battery will not be diminished, ECG is correct.

Answer C is incorrect. Capacity of the battery will not be diminished, TS is incorrect.

Answer D is incorrect. H_2 concentration will increase in the battery rooms without sufficient ventilation; Batteries do not have to be declared inoperable until operational limits are exceeded, there are no limits for high temperature.

Technical Reference(s): LH-10 Miscellaneous Building Ventilation System, Pages 11 & 22, Rev. 4.
ECG 80.1, Doors required for HELB, HVAC, ECCS Function, or Flood Protection, Rev. 5.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 65936 - Discuss significant Technical Specifications and Equipment Control Guidelines associated with the Miscellaneous Building Ventilation System.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2

10 CFR Part 55 Content: Facility operating limitations in the technical specifications and their bases.

Comments:

SRO Question 15 (Rev. 1)

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		1
K/A #		103/A2.05
Importance Rating		3.9

K/A: **Containment System** - Ability to (a) predict the impacts of the following malfunctions or operations on the Containment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry.

Proposed Question:

Units 1 & 2 are at 100% power.

A 0.04 g magnitude earthquake is confirmed by ground motion felt in the control room.

Both units remained at 100% power.

Besides CP M-4 “Earthquake”, which of the following procedures and actions should the Shift Manager use to respond to this event?

A. EOP E-0, “Reactor Trip or Safety Injection”, direct BOTH units to perform a Reactor Trip.

B. OP L-4, “Normal Operation at Power”, to place BOTH units in MODE 3 within 72 hours.

C. OP AP-25, “Rapid Load Reduction or Shutdown”, reduce power on BOTH units to less than P-10 within 2 hours.

D. OP AP-31, “Rapid Containment Entry”, to inspect containment fire zones within 2 hours.

Proposed Answer: D

Explanation:

Answer A is incorrect. Seismic setpoint reactor trip setpoint is 0.3 g, so threshold has not been reached. Reactor Trip is not required.

Answer B is incorrect. There is no requirement to reduce load to MODE3 based on this size of an earthquake.

Answer C is incorrect. There is no requirement to reduce load to MODE3 based on this size of an earthquake.

Answer D is correct. Per CP M-4, for earthquakes greater than 0.02g the fire zones listed in ECG 18.3 (Containment is included) must be inspected within 2 hours of the event. OP AP-31 is used to gain access to containment in a rapid fashion.

Technical Reference(s): CP M-4, Earthquakes, Page 6, Rev. 23.
ECG 18.3, Fire Detection Instrumentation, Page 2, Rev. 9.
LPE-7, Miscellaneous Emergency Procedures, Page 17, rev. 13.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3295 - Describe the major actions to take for earthquakes:
- Over 0.02 g

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 16

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		2
K/A #		002/A2.02
Importance Rating		4.4

K/A: **Reactor Coolant System (RCS)** - Ability to (a) predict the impacts of the following malfunctions or operations on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of coolant pressure.

Proposed Question:

Unit 1 is currently at 100% power

The following indications are observed:

- Pressurizer level is 51% and slowly TRENDING DOWN.
- Pressurizer Pressure is 2150 psig and slowly TRENDING DOWN.
- All Pressurizer Backup Heaters are ON.
- Charging flow is 125 gpm.
- Containment Pressure is 0.5 psig and slowly TRENDING UP.

Which of the following procedure flow paths would be expected, based on the above indications?

A. OP AP-1, “Excessive Reactor Coolant Leakage”; EOP-E.0, Reactor Trip or Safety Injection and EOP E-1, “Loss of Reactor or Secondary Coolant.”

B. OP AP-1, “Excessive Reactor Coolant Leakage”; EOP-E-0, Reactor Trip or Safety Injection and EOP E-0.1, “Reactor Trip Response.”

C. OP AP-3, “Steam Generator Tube Failure”; EOP-E.0, Reactor Trip or Safety Injection and EOP E-3, “Steam Generator Tube Rupture.”

D. OP AP-3, “Steam Generator Tube Failure”; EOP-E.0, Reactor Trip or Safety Injection and EOP E-0.1, “Reactor Trip Response.”

Proposed Answer: A

Explanation:

Answer A is correct. OP AP-1 is entered and pressure is less than 2210 psig, then a Safety Injection is required, since SI was actuated, the next procedure transition would be E-1 to determine the long term recovery plan.

Answer B is incorrect. Since Safety Injection is actuated, E-0.1 will not be entered.

Answer C is incorrect. Even though RCS pressure is low and charging flow is high, the fact that containment pressure is trending up means that the leak is from the RCS and not the Steam Generator Tubes.

Answer D is incorrect. Even though RCS pressure is low and charging flow is high, the fact that containment pressure is trending up means that the leak is from the RCS and not the Steam Generator Tubes.

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Technical Reference(s): LPA-1, Excessive Reactor Coolant System Leakage, Pages 13 & 17, Rev. 8.
OP AP-1, Excessive Reactor Coolant System Leakage, Page 3, Rev. 18.
EOP E-0, Reactor Trip or Safety Injection, Pages 4 & 10, Rev. 32.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3538 - List the leakage criteria which would require manual initiation of a safety injection.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 17

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #		011/2.4.45
Importance Rating		4.3

K/A: **Pressurizer Level Control System (PZR LCS)** - Emergency Procedures/Plan: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question:

Unit 1 is operating at 100% power.

The following control room alarms are received:

- PK 05-01 RCP NO. 11 – Seal injection Flow Lo
- PK 05-02 RCP NO. 12 – Seal Injection Flow Lo
- PK 05-03 RCP NO. 13 – Seal Injection Flow Lo
- PK 05-04 RCP NO. 14 – Seal Injection Flow Lo

The following control room indications are observed:

- Pressurizer level - 55% - Steady
- Charging Flow – 0 gpm
- Letdown Flow – 0 gpm
- RCP Seal Injection Flows at 0 gpm on all pumps.

Based on the above indications, which of the following procedures provides the appropriate recovery actions for this event?

A. OP AP-5, "Malfunction of Eagle 21 Protection or Control Channel."

B. OP AP-17, "Loss of Charging."

C. OP AP-18, "Letdown Line Failure"

D. OP AP-28, "Reactor Coolant Pump Malfunction."

Proposed Answer: B

Explanation:

Answer A is incorrect. This is not an instrument failure, AP-5 would not be appropriate.

Answer B is correct. AP-17 symptoms include the loss of charging and seal injection flow, along with the loss of letdown due to the loss of the running charging pump interlock.

Answer C is incorrect. Charging and seal injection flow would not go to 0 gpm on a letdown line failure.

Answer D is incorrect. Reactor Coolant pump operation is not jeopardized as long as CCW is supplied to the thermal barrier heat exchanger.

Technical Reference(s): LAP-17, Loss of Charging, Page 5, Rev. 9.
OP AP-17, Loss of Charging, Page 2, Rev. 28.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3478 - Given initial conditions, assumptions, and symptoms, determine the correct abnormal operating procedure to be used to mitigate an operational event.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 18

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		2
K/A #		056/A2.04
Importance Rating		2.8

K/A: **Condensate System** - Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of condensate pumps.

Proposed Question:

Unit 1 is operating at 60% Power with both Main Feedwater Pumps in service.

Condensate Pump 1-3 is cleared for seal replacement.

The following alarm is received in the main control room:

- PK 10-06 CNDs & CNDS BSTR PPS – OC Trip

The following control room indications are observed:

- Blue light on Control Switch for Condensate Pump Set 1-2
- Main Feedwater Pump suction header pressure is 250 psig and TRENDING DOWN slowly.

Which of the following procedure(s) provides the appropriate mitigation strategy to recover from this event?

A. OP AP-2, “Full Load Rejection”, directs tripping one Main Feedwater pump to boost main feed water suction header pressure.

B. OP AP-2, “Full Load Rejection”, directs a reduction in turbine load to boost main feed water pump suction header pressure.

C. OP AP-15, “Loss of Feedwater Flow”, directs tripping one Main Feedwater pump to boost main feed water suction header pressure.

D. OP AP-15,”Loss of Feedwater Flow”, directs a reduction in turbine load to boost main feed water pump suction header pressure.

Proposed Answer: D

Explanation:

Answer A is incorrect. AP-2 is used for full load rejections, no load rejection will occur for this failure. Action to trip one Feedwater pump is not appropriate for this failure.

Answer B is incorrect. AP-2 is used for full load rejections, no load rejection will occur for this failure. Action to reduce turbine load is not included in AP-2 because it is assumed to have occurred.

Answer C is incorrect. AP-15 is the correct procedure, but tripping of a running MFW pump is not an action contained in the procedure.

Answer D is correct. Per AP-15 if Main Feedwater suction pressure is less than 260 psig and greater than 190 psig, main turbine load is reduced to increase suction header pressure.

Technical Reference(s): LPA-15, Loss of Feedwater Flow, Page 13, Rev. 11.
OP AP-15, Loss of Feedwater Flow, Section D, Page 11, Rev. 18.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 3477 - Given an abnormal condition, summarize the major actions of the abnormal operating procedure to mitigate an event in progress.

Comment [RLS8]: 3477

Question Source: Bank # ______
Modified Bank # P-54490 (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 19

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		N/A
K/A #		G 2.1.37
Importance Rating		4.6

K/A: **Generic** - Conduct of Operations: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question:

Which of the following activities has the potential to affect reactivity and therefore requires the shift foreman conduct a reactivity briefing?

1. Adjusting Steam Generator Blowdown flows.
2. Testing of RHR Pump 1-1 in MODE 1.
3. Swapping Service Cooling Water heat exchangers.
4. Spent Fuel Pool Water Addition.
5. Letdown Heat Exchanger CCW Cooling adjustment.

A. 1, 4 and 5 only.

B. 1, 2 and 4 only.

C. 3, 4 and 5 only.

D. 1, 2, 3, 4 and 5.

Proposed Answer: A

Explanation:

Answer A is correct. All items are listed in OP1.ID3, Reactivity Management Program as potentially affecting reactivity and requiring a reactivity briefing.

Answer B is incorrect. Running RHR pump in MODE 1 will not change reactivity, no water will pump forward.

Answer C is incorrect. Swapping Service Cooling water heat exchangers will not affect reactivity.

Answer D is incorrect. Running RHR Pump 1-1 in MODE 1 or swapping Service Cooling Water heat exchangers will not affect reactivity.

Technical Reference(s): LADM-9, Reactivity Management, Pages 6 & 7, Rev. 5.
OP1. ID3, Reactivity Management Program, Page 5, Rev. 1A.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 67250 - State reactivity management requirements and expectations.

- Requirements for reactivity briefs

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 6

10 CFR Part 55 Content: Procedures, and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Comments:

SRO Question 20

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		N/A
K/A #		G 2.1.40
Importance Rating		3.9

K/A: **Generic** - Conduct of Operations: Knowledge of refueling administrative requirements.

Proposed Question:

Unit 1 Reactor Core Reload is in progress.

The control operator reports the following parameters:

	RCS temp	Boron	N31	N32
- Prior to fuel motion	62°F	2570 PPM	13 CPS	15 CPS
- Latest reading	70°F	2550 PPM	22 CPS	0 CPS

Based on the indications; Core loading should be:

A. Allowed to continue.

B. Suspended, because of the source range response.

C. Suspended, because of the temperature change.

D. Suspended, because of the boron change.

Proposed Answer: B

Explanation:

Answer A is incorrect. Fuel movement must be suspended due to the failure of N32, Both source ranges are required.

Answer B is correct. Fuel movement must be suspended, one source range has failed (N32), and both source ranges are required by tech specs and OP B8DS2.

Answer C is incorrect. Temperature change is 8°, change of 20°F is required before suspension of core reload is required.

Answer D is incorrect. Boron change is 20 PPM, change of 50 PPM is required before suspension of core reload is required.

Technical Reference(s): LB-8, Fuel Handling & Equipment, Page 63, Rev. 12.
OP B-8DS2, Core Loading, Pages 4 & 7, Rev. 39.
Tech Specs 3.9.3, Nuclear Instrumentation, Page 3.9-2, Rev. 3.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 36965 - Discuss significant precautions and limitations associated with the Fuel Handling System.

Question Source: Bank #
Modified Bank # S-47409 (Note changes or attach parent)
New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 7

10 CFR Part 55 Content: Fuel handling facilities and procedures.

Comments:

SRO Question 21 (Rev. 1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		3
Group #		N/A
K/A #		G 2.2.6
Importance Rating		3.6

K/A: **Generic** - Equipment Control: Knowledge of the process for making changes to procedures.

Proposed Question:

Implementation of an “Interim“ On the Spot Change (OTSC) which effects the operational status of plant equipment, but does not constitute a change of intent to a procedure, must be authorized by __________ and must get final approval within ____ days.

A. Plant Staff Review Committee (PSRC), 14 days

B. Plant Management staff with an SRO, 14 days

C. Operations Director, 30 days

D. Site Procedure group, 30 days

Proposed Answer: B

Explanation:

Answer A is incorrect. PSRC approval is not required for OTSCs. 14 days is correct.

Answer B is correct. Interim review by an SRO is required and final approval must occur within 14 days.

Answer C is incorrect. Operations Director is not required to review OTSCs and the final approval must be done in 14 days.

Answer D is incorrect. Site Procedure group is not required to review OTSC, and final approval must be done in 14 days.

Technical Reference(s): LADM-7, Procedure Use and Change Requirements, Page 11, Rev. 5.
AD1.ID2, Procedure Process Control, Pages 21 & 22 and Attachment 2, Rev. 27.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 9735 - Explain procedure for making on the spot changes (OTSC).

Question Source: Bank # XB-0201
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 3

10 CFR Part 55 Content: Facility license procedures required to obtain authority for design and operating changes in the facility.

Comments:

SRO Question 22

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		N/A
K/A #		G 2.2.25
Importance Rating		4.2

K/A: **Generic** - Equipment Control: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question:

Which of the following is the Technical Specifications BASES, associated with the 10% Atmospheric Dump Valves?

A. Reduce Steam Generator Pressure in response to a Loss of Heat Sink.

B. Reduce Steam Generator Pressure to prevent Steam Generator Overpressurization.

C. Conduct a cooldown to establish adequate subcooling in response to a Large Break LOCA.

D. Conduct a cooldown to establish adequate subcooling in response to a Steam Generator Tube Rupture.

Proposed Answer: D

Explanation:

Answer A is incorrect. ADVs are used for Loss of Heat Sink, if the condenser is not available, but the Tech Spec Bases is for a Steam Generator Tube Leak.

Answer B is incorrect. ADVs are used to reduce Steam Generator Pressure, but the Tech Spec Bases is for a Steam Generator Tube Leak.

Answer C is incorrect. One ADV is required from each of four steam generators to ensure that at least two ADV lines are available to conduct a unit cooldown following an SGTR and not a Large Break LOCA.

Answer D is correct. One ADV is required from each of four steam generators to ensure that at least two ADV lines are available to conduct a unit cooldown following an SGTR.

Technical Reference(s): LC-2B, Steam Dump System, Pages 6 & 55, Rev. 10.
Tech Specs Bases, ADVs, 3.7.4, Page 20, Rev. 4.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 9694G - Discuss 3.7 Technical Specification bases.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 2

10 CFR Part 55 Content: Facility operating limitations in the technical specifications and their bases.

Comments:

SRO Question 23 (Rev.1)

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		3
Group #		N/A
K/A #		G 2.3.6
Importance Rating		3.8

K/A: **Generic** - Radiation Control: Ability to approve release permits.

Proposed Question:

Which of the following authorizes and approves all liquid Radwaste discharge permits?

A. Chemistry Foreman and Shift Foreman.

B. Rad Waste Foreman and Shift Foreman.

C. Chemistry Foreman and Shift Manager.

D. Rad Waste Foreman and Shift Manager

Proposed Answer: A

Explanation:

Answer A is correct. The SFM signs the discharge checklist; both the Chemistry Foreman and Shift Foreman sign the discharge permit.

Answer B is incorrect. The Rad Waste foreman will do the discharge, but does not authorize the discharge.

Answer C is incorrect. Shift Manager does not approve discharge permits.

Answer D is incorrect. The Rad Waste foreman will do the discharge, but does not authorize the discharge and the Shift Manager does not approve discharge permits.

Technical Reference(s): LG-1, Liquid Radwaste System, Page 28, Rev. 8.
OP G-1:II, Liquid Radwaste System – Discharge of Liquid Radwaste, Pages 2 & 6, Rev. 35.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 8454 - Discuss significant precautions and limitations associated with the Liquid Radwaste System.

Question Source: Bank # P-68285
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 4

10 CFR Part 55 Content: Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

SRO Question 24

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		3
Group #		N/A
K/A #		G 2.4.23
Importance Rating		4.4

K/A: **Generic** - Emergency Procedures/Plan: Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

Proposed Question:

The control room operators have entered EOP FR-H.1 “Response to Loss of Secondary Heat Sink,” due to red path on the Heat Sink Critical Safety Function Status Tree.

The Emergency Evaluation Coordinator (EEC) identifies a red path on the “Integrity” Critical Safety Function Status Tree.

The Shift Foreman should:

NOTE - FR-P.1 - Response to Imminent Pressurized Thermal Shock Condition

A. Continue with FR-H.1, Fuel/Cladding Integrity is a higher priority than RCS Integrity.

B. GO TO FR-P.1, RCS Integrity is a higher priority than Fuel/Clad Integrity.

C. Implement FR-P.1 while continuing in FR-H.1; to minimize cooldown caused by FR-H.1 actions.

D. Immediately return to Step 1 of FR-H.1, to reassess secondary conditions.

Proposed Answer: A

Explanation:

Answer A is correct. FR H.1 is a higher priority than FR-P.1, higher priority CSFSTs are always continued unless a higher priority challenge is identified. Fuel and Cladding Integrity are a higher priority for protection against radiation releases.

Answer B is incorrect. FR P.1 is not a higher priority FRG, it is lower. FR-P.1 actions will be addressed when FR-H.1 actions are completed.

Answer C is incorrect. FR-P.1 actions are not done in parallel with FR-H.1 actions; this is not allowed per rules of usage.

Answer D is incorrect. Returning to step 1 is not required when FR-P.1 conditions are met.

Technical Reference(s): LPE-FR, Functional Restoration Guidelines, Page 5, Rev. 8.
EOP F-0, Critical Safety Function Status Trees, Page 2 & 3, Rev. 14.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 38107 - Apply the Rules of Usage in EOPs for the CSFSTs and FRGs, including:

- the priority of use of the six CSFSTs

Question Source: Bank # ______
Modified Bank # INPO - 20219 (Note changes or attach parent)
New ______

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Question 25 (Rev. 1)

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		N/A
K/A #		G 2.4.32
Importance Rating		4.0

K/A: **Generic** - Emergency Procedures/Plan: Knowledge of operator response to loss of all annunciators.

Proposed Question:

Unit 1 is stable at 100% power.

A loss of the main annunciator trains A and B has occurred with the following indications:

- All annunciator windows are OFF.
- Alarm printer has failed.
- RONAN CRT has failed.

The correct response to the above indications is to:

A. Trip the Reactor and enter E-0, “Reactor Trip or Safety Injection” to place the unit in MODE 3.

B. Enter OP AP-25, ‘Rapid Load Reduction” and commence a controlled plant shutdown to MODE 3.

C. Enter AR PK15-22, “Main Annunciator System Trouble” to establish surveillance requirements and classify the event.

D. Enter OP AP-27, “Loss of Vital 4kV or 480v bus” and transfer the annunciator system to its backup power supply.

Proposed Answer: C

Explanation:

Answer A is incorrect. Step 5.2 of annunciator response procedure directs the operator to avoid any load changes.

Answer B is incorrect. Step 5.2 of annunciator response procedure directs the operator to avoid any load changes.

Answer C is correct. AR PK 15-22 directs the operator to avoid any load changes, establish continuous surveillance of all control room parameters and classify the event.

Answer D is incorrect. OP AP-27 does not have any actions associated with the loss of the main annunciator system and its backup power supply.

Technical Reference(s): STG J-12, Annunciator System, Pages 3-5 & 3-6, Rev. 8.
AR PK15-22, Main Annunciator System Trouble, Page 5, Rev. 13.

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 71128 – Explain the operation of the Annunciator system.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

10 CFR Part 55 Content: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: