EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

1 **ID: Q20176** **Points: 1.00**

Which **ONE** of the following sequences occurs to open the Reactor Trip Circuit Breakers for an automatic reactor trip (consider only the components stated)?

A. Trip unit relays *deenergize*;
Matrix relays *deenergize*;
Shunt trip coils *energize*.

B. Trip unit relays *energize;*
Matrix relays *energize*;
Shunt trip coils *deenergize*.

C. Trip unit relays *deenergize;*
Matrix relays *deenergize*;
UV trip coils *energize*.

D. Trip unit relays *energize;*
Matrix relays *energize*;
UV trip coils *deenergize*.

Answer: A

Answer Explanation:

A. Correct – This is the only sequence provided that will result in a reactor trip.

B. Incorrect – Trip Unit and Matrix Relays deenergize to trip. Shunt Trip Coils energize to trip.

C. Incorrect - The UV Trip Coils must deenergize to cause a reactor trip.

D. Incorrect - The Trip Unit relays and Matrix relays deenergize to cause a trip.

Question 1 Info			
Topic:	Which RPS response is correct for a reactor trip?		
Tier/Group:	1/1		
K/A Info:	EPE - 007 Reactor Trip • EK2 Knowledge of the interrelations between a reactor trip and the following: • EK2.02 - Breakers, relays and disconnects		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-58-1-01		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - LOI 2008 RPS, AOP-7H, Power Distribution T.S. Exam (June, 2009)		
Technical references:	System Description 058, Reactor Protective System		
Comments:	None		

2 **ID: Q92131** **Points: 1.00**

Which **ONE** of the following conditions specifically requires notification of Site personnel via plant-wide announcement, In accordance with CNG-OP-101-2001, Communications and Briefings?

A. A Containment entry is made when the reactor is critical.

B. EOP-6, Steam Generator Tube Rupture, is implemented.

C. Regulating CEA withdrawal is commenced for reactor startup.

D. Tech Spec LCO 3.6.1 is entered for Containment inoperability.

Answer: B

Answer Explanation:

A. Incorrect - Containment entry is **not** specifically called out for announcement to the Site by CNG-OP-1.01-2001, Communications and Briefings.

B. Correct - Implementation of an EOP is specifically called out for announcement to the Site by CNG-OP-101-2001, Communications and Briefings.

C. Incorrect – Commencing withdrawal of Regulating CEAs is **not** specifically called out for announcement to the Site by CNG-OP-101-2001, Communications and Briefings.

D. Incorrect - Entry into a T.S. LCO, with a completion time of 1 hour is **not** specifically called out for announcement to the Site by CNG-OP-1.01-2001, Communications and Briefings. This condition is plausible because it requires prompt notification of site management personnel via the Pager system in accordance with CNG-OP-1.01-2001, Communications and Briefings.

Question 2 Info			
Topic:	Plant page announcements during AOP / EOP conditions.		
Tier/Group:	1/1		
K/A Info:	038 - Steam Generator Tube Rupture (SGTR) • 2.1.14 - Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• NO-1-200, Conduct of Operations • CNG-OP-101-2001, Communications and Briefings		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

3 **ID: Q25950** **Points: 1.00**

Which **ONE** of the following is the **minimum** allowable RCS flow during dilution operations per the Technical Requirements Manual?

A. 3000 GPM

B. 1700 GPM

C. 1500 GPM

D. 1000 GPM

Answer: A

Answer Explanation:

A. Correct - TRM 15.1.1 specifies Reactor Coolant System (RCS) flow rate shall be ≥ 3,000 GPM. APPLICABILITY Modes 1, 2, 3, 4, 5, and 6, whenever a reduction in RCS boron concentration is being made from a source whose boron concentration is less than the present Shutdown Margin requirements (Refueling Boron for Mode 6) per COLR.

B. Incorrect - Per OP-7; Maximum SDC Flow is 1700 GPM when the Reactor is defueled and the UGS is installed. This will prevent damage to the ICI Thimbles.

C. Incorrect - The bases for T.S. SR 3.9.4.1 states; the flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability, and to prevent thermal and boron stratification in the core.

D. Incorrect - Per OP-7; When entering reduced inventory two LPSI header stops are shut and the remaining two LPSI loop header stops are throttled to limit flow to a maximum of 1000 GPM per loop.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 3 Info			
Topic:	Required flow for dilution		
Tier/Group:	2/1		
K/A Info:	005 - Residual Heat Removal System (RHRS) • K5 Knowledge of the operational implications of the following concepts as they apply the RHRS: • K5.09 - Dilution and boration considerations		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-203-5-3-009		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - 2002		
Technical references:	Technical Requirements Manual, T.N.C. 15.1.1		
Comments:	None		

4 **ID: Q92611** **Points: 1.00**

Reactor power is currently stable at 88%. Current Burnup is 10,000 MWD/MTU.

A malfunction occurs, causing the Letdown HX CCW Temperature Control valve to close.

Which of the choices below correctly identifies the initial response of T_{COLD} and reactor power to this failure? Assume no operator action.

A. T_{COLD} lowers, reactor power rises.

B. T_{COLD} rises, reactor power rises.

C. T_{COLD} lowers, reactor power lowers.

D. T_{COLD} rises, reactor power lowers.

Answer: C

Answer Explanation:

A. Incorrect – TIC-223, failing to 100%, causes CCW flow to be secured to the L/D Hx. As L/D temperature raises the CVCS Ion Exchangers slough boron. The rise in Boron concentration will cause reactor power to lower.

B. Incorrect - TIC-223, failing to 100%, causes CCW flow to be secured to the L/D Hx. As L/D temperature raises the CVCS Ion Exchangers slough boron. The rise in Boron concentration will cause reactor power to lower with a resultant lowering of T_{COLD}.

C. Correct - TIC-223, failing to 100%, causes CCW flow to be secured to the L/D Hx. As L/D temperature raises the CVCS Ion Exchangers slough boron. The rise in Boron concentration will cause Reactor power to lower, lowering T_{COLD}.

D. Incorrect - TIC-223, failing to 100%, causes CCW flow to be secured to the L/D Hx. As L/D temperature raises the CVCS Ion Exchangers slough boron. The rise in Boron concentration will cause reactor power to lower, lowering T_{COLD}.

Question 4 Info			
Topic:	Explain the effects of increasing/decreasing Letdown temperature		
Tier/Group:	2/1		
K/A Info:	004 - Chemical and Volume Control System • K3 Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: • K3.06 - RCS temperature and pressure		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 Nuclear Instrumentation Exam (May, 2009)		
Technical references:	OI-16, Component Cooling System, Precaution "C"		
Comments:	Adaptation of Bank question “Q14535”		

5 **ID: Q28840** **Points: 1.00**

Match each of the RCP seal pressure conditions in column A to the RCP seal status in column B. RCS pressure is 2250 psia.

Column "A" - Seal Parameters

	Middle Seal Pressure	Upper Seal Press	VCT Press
Case 1	1100	50	45
Case 2	1400	700	48
Case 3	1050	1050	50
Case 4	2250	1100	51

Column "B" – Seal status

1 Normal

2 Lower Seal Failed

3 Middle Seal Failed

4 Upper Seal Failed

A. 3,2,4,1

B. 4,1,3,2

C. 1,4,3,2

D. 3,1,2,4

Answer: B

Answer Explanation:

A. Incorrect – See explanation for correct answer.

B. Correct - Parameters indicate:

- Case 1 – Middle Seal failure with expected RCP Seal pressure breakdown on remaining seals
- Case 2 – Normal RCP Seal pressure breakdown
- Case 3 – Upper Seal failure with expected RCP Seal pressure breakdown on remaining seals
- Case 4 – Lower Seal failure with expected RCP Seal pressure breakdown on remaining seals.

C. Incorrect - See explanation for correct answer.

D. Incorrect - See explanation for correct answer.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 5 Info			
Topic:	11B RCP seal status		
Tier/Group:	2/1		
K/A Info:	003 Reactor Coolant Pump System (RCPS) • K1 Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: • K1.03 RCP seal system		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(3)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 Diesel Generators Exam (May, 2009)		
Technical references:	OI-1A, Reactor Coolant System And Pump Operations		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

6 **ID: Q28439** **Points: 1.00**

A caution within EOP-3, Loss of All Feedwater, states that Once Through Core Cooling (OTCC) must be initiated before CETs reach or exceed 560 °F.

What is the basis for this temperature limit?

A. Ensures the RCS is maintained subcooled throughout OTCC.

B. Ensures the inventory in the core will not be displaced into the Pressurizer.

C. Ensure RCS core flow is sufficient to lower core temperature.

D. Ensures RCS pressure remains high enough to prevent HPSI Pump damage.

Answer: C

Answer Explanation:

A. Incorrect - The RCS will be in a saturated condition due to the PORVs being opened

B. Incorrect - The RCS will be in a saturated condition due to the PORVs being opened. Water will be displaced into the low pressure area (the Pressurizer).

C. Correct - Per the EOP-3 Basis Doc, If OTCC initiated above this value the HPSI pump flow may be insufficient for core cooling flow.

D. Incorrect - Runout of the HPSI pumps is not probable (DBA). Would also be prevented by complying with procedure direction to verify HPSI flow PER EOP ATTACHMENT(10), HIGH PRESSURE SAFETY INJECTION FLOW.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 6 Info			
Topic:	Basis for initiating OTCC prior to 560 °F		
Tier/Group:	1/1		
K/A Info:	CE/E06 - Loss of Feedwater • EK3 - Knowledge of the reasons for the following responses as they apply to the (Loss of Feedwater) • EK3.2 - Normal, abnormal and emergency operating procedures associated with (Loss of Feedwater).		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	SRO-201-3-1-14		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – 2006		
Technical references:	EOP-3, Loss of All Feedwater		
Comments:	None		

7 **ID: Q28803** **Points: 1.00**

A Charging header leak could be positively identified by which **ONE** of the following?

A. Lowering Pressurizer level with minimum letdown flow and one charging pump operating.

B. Charging header pressure greater than RCS pressure with two charging pumps operating.

C. Charging header flow equals letdown flow with one charging pump operating and VCT level lowering.

D. RCS pressure greater than charging header pressure with one charging pump operating.

Answer: D

Answer Explanation:

A. Incorrect - This would be true for any leak greater than about 12 GPM but does not distinguish a charging header leak.

B. Incorrect - A charging header leak can be disguised with 2 CHG pumps running.

C. Incorrect - Is true for any small leak and would not distinguish a leak on the charging header.

D. Correct - per AOP-2A, a leak on the Charging header which exceeds the capacity of the charging pumps can be identified by Charging header pressure indicating less than RCS pressure. Identification of the leak may be missed if more than one charging pump is running.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 7 Info			
Topic:	Charging header leak identification		
Tier/Group:	1/1		
K/A Info:	022 Loss of Reactor Coolant Makeup • AA2. Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: • AA2.01 Whether charging line leak exists		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-107-1-3-50		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 AOP / EOP Exam (April 2010)		
Technical references:	AOP-2A, Excessive Reactor Coolant Leakage		
Comments:	None		

8 **ID: Q42247** **Points: 1.00**

Unit-1 was operating at 100% power, End-of Life (EOL), when 11A and 11B RCPs tripped.

Assuming all equipment responds as designed, and NO operator action has been performed, which of the following best describes the heat removal parameters 5 minutes after 11A and 11B RCP's have completely stopped?

A. 12 S/G steam flow greater than 11 S/G steam flow;
12 S/G pressure greater than 11 S/G pressure;
11 S/G feed flow greater than 11 S/G steam flow

B. 12 S/G steam flow is equal to 11 S/G steam flow;
12 S/G pressure is equal to 11 S/G pressure;
11 S/G feed flow less than 11 S/G steam flow

C. 12 S/G steam flow greater than 11 S/G steam flow;
12 S/G pressure greater than 11 S/G pressure;
11 S/G feed flow less than 11 S/G steam flow

D. 12 S/G steam flow less than 11 S/G steam flow;
12 S/G pressure is equal to 11 S/G pressure;
11 S/G feed flow greater than 11 S/G steam flow

Answer: A

Answer Explanation:

A. Correct - RPS will trip the unit when the first RCP is tripped. Once both RCP's stop rotating, the flow through 11 S/G will reverse and be less than 12 Loop. This will cause 12 S/G pressure to be higher and flow from it to be higher. Digital feed will be unaffected by the trip and will position both FRBVs to the same output.

B. Incorrect - The flow through 11 S/G will reverse causing 11 S/G temperature to be lower than 12 S/G temperature. This will cause 11 S/G pressure to be lower and flow from it to be lower.

C. Incorrect - Digital feed will be unaffected by the trip and will position both FRBVs to the same output.

D. Incorrect - Digital feed will be unaffected by the trip and will position both FRBVs to the same output.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 8 Info			
Topic:	Loss of a pair of RCPs		
Tier/Group:	1/1		
K/A Info:	015/017 - Reactor Coolant Pump (RCP) Malfunctions • AK1. Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): • AK1.04 Basic steady state thermodynamic relationship between RCS loops and S/Gs resulting from unbalanced RCS flow		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - 2004		
Technical references:	EOP-2, Loss of Offsite Power/Loss of Forced Circulation		
Comments:	None		

9 **ID: Q39488** **Points: 1.00**

Unit-1 is operating at 100% Reactor Power when an electrical perturbation occurs causing the Backup and 2nd Backup Charging Pumps to start.

(1) What Bus is lost and;
(2) Which of the following describes a necessary action, per the response procedure, for the bus that was lost?

A. 1Y09;
Promptly reduce Turbine load.

B. 1Y10;
Adjust Turbine load to maintain T_{COLD} on program.

C. MCC-104R;
Fast Borate to reduce reactor power.

D. MCC-114R;
Align Charging Pump suction to the VCT.

Answer: B

Answer Explanation:

A. Incorrect – Symptom provided is indicative of a loss of 1Y10. A loss of 1Y10 results in Charging Pump suction shifting to the RWT with resultant boration of the RCS. Stabilizing actions are to secure boration and adjust Turbine load to maintain T_{COLD} on program.

B. Correct – Symptom provided is indicative of a loss of 1Y10. A loss of 1Y10 results in Charging Pump suction shifting to the RWT with resultant boration of the RCS. Stabilizing actions are to secure boration and adjust Turbine load to maintain T_{COLD} on program.

C. Incorrect – Symptom provided is indicative of a loss of 1Y10 which is powered from MCC-104. 1Y10 would be the "minimum" Bus lost.

D. Incorrect – Symptom provided is indicative of a loss of 1Y10 which is powered from MCC-104. 1Y10 would be the "minimum" Bus lost.

Question 9 Info			
Topic:	Loss of 2Y10		
Tier/Group:	1/1		
K/A Info:	057 - Loss of Vital AC Electrical Instrument Bus • AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: • AK3.01 - Actions contained in EOP for loss of vital ac electrical instrument bus		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	AOP-7I-02		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Comprehensive Exam (Sept, 2008)		
Technical references:	AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power		
Comments:	None		

10 **ID: Q40530** **Points: 1.00**

The Unit was operating at power when a steam line ruptured. The following conditions exist:

- SIAS actuated.
- RCS pressure is 1000 PSIA and lowering.
- RCS temperature is 460 °F and lowering.

What is the major concern associated with RCS repressurization during this event?

A. HPSI Pump operation at shutoff head

B. S/G tube sheet differential pressure

C. Pressurizer PORV actuation

D. Reactor vessel thermal stresses

Answer: D

Answer Explanation:

A. Incorrect - HPSI Pumps would be injecting a total of approximately 750 GPM at the stated RCS pressure. HPSI Pumps are nowhere near running at shutoff head.

B. Incorrect - S/G tubes/tubesheet are designed to withstand full RCS pressure on the primary side with atmospheric pressure on the secondary side.

C. Incorrect - The unaffected S/G must be used to stabilize RCS temperature to prevent RCS inventory expansion which could cause the Pressurizer to go solid and induce conditions susceptible to Pressurized Thermal Shock.

D. Correct - The unaffected S/G must be used to stabilize RCS temperature to prevent RCS inventory expansion which could cause the Pressurizer to go solid and induce conditions susceptible to Pressurized Thermal Shock.

Question 10 Info			
Topic:	PTS		
Tier/Group:	1/1		
K/A Info:	040 - Steam Line Rupture • AK1. Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: • AK1.04 - Nil ductility temperature		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-201-4-S-06		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - 2006		
Technical references:	EOP-4, Excess Steam Demand Event		
Comments:	None		

11 **ID: Q26572** **Points: 1.00**

Unit-2 is operating at 100% power when a Loss of Offsite Power occurs. 21 and 22 AFW Pumps are unavailable. #23 AFW pump is started to establish Auxiliary Feedwater flow to 21 and 22 S/Gs with the following flow values:

- 2-FIC-4525A, 21 SG FLOW CONTR, indicates 270 GPM
- 2-FIC-4535A, 22 SG FLOW CONTR, indicates 280 GPM

Based on these parameters which **ONE** of the following is the correct operator response and basis for the response?

A. Maintain flow values; No operational limits have been exceeded.

B. Reduce AFW flow to prevent AFW Pump cavitation.

C. Reduce AFW flow to protect the DG from overloading.

D. Reduce AFW flow to prevent runout of the AFW Pump.

Answer: C

Answer Explanation:

A. Incorrect – 23 AFW Pump flow is limited to 300 GPM total flow when powered by the DG. Plausible because this would be true on Unit-1.

B. Incorrect – No information is supplied or implied to indicate the common suction flow limit of 1200 GPM is being exceeded.

C. Correct – 23 AFW Pump is being powered from the 2B DG. EOP-2, Loss of Offsite Power/Loss of Forced Circulation, Step IV.G.2.2 has a caution stating: "23 AFW PP flow limit is 300 GPM when power is supplied by a DG; otherwise the flow limit is 575 GPM".

D. Incorrect – 23 AFW Pump is being powered from the 2B DG. EOP-2, Loss of Offsite Power/Loss of Forced Circulation, Step IV.G.2.2 has a caution stating: "23 AFW PP flow limit is 300 GPM when power is supplied by a DG; otherwise the flow limit is 575 GPM".

Question 11 Info	
Topic:	23 AFW Pump Ops during a LOOP
Tier/Group:	1/1
K/A Info:	056 - Loss of Offsite Power • AK3 - Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: • AK3.02 - Actions contained in EOP for loss of offsite power.
RO Importance:	4.4
Proposed references to be provided to applicant:	None
Learning Objective:	SRO-201-2-1-13
10 CFR Part 55 Content:	55.41(b)(10)
Question source:	☒ Bank ☐ Modified ☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis
Last NRC Exam used on:	N/A
Exam Bank History:	None
Technical references:	EOP-2, Loss of Offsite Power/Loss of Forced Circulation.
Comments:	None

12 **ID: Q92132** **Points: 1.00**

Given the following:

- Unit-1 is at 100% power
- RCS Pressure Control is in AUTO
- RCS Pressure is 2250 PSIA

What is the **IMMEDIATE** plant response if the selected Pressurizer Pressure controller setpoint fails to 1500 PSIA?

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

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

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

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

Answer: D

Answer Explanation:

A. Incorrect - Proportional Heaters go to minimum. Plausible because spray will collapse the Pressurizer bubble causing Pressurizer level to rise. This could trigger Pressurizer Heater operation on insurge.

B. Incorrect - The Pressurizer Spray valves would open.

C. Incorrect - The Pressurizer Spray valves would open and the Proportional Heaters would go to minimum.

D. Correct - The Pressurizer Spray valves would open and the Proportional Heaters would go to minimum.

Question 12 Info			
Topic:	Plant response to a change in the Pzr pressure controller setpoint.		
Tier/Group:	1/1		
K/A Info:	027 - Pressurizer Pressure Control System (PZR PCS) Malfunction: • AK2 - Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: • AK2.03 - Controllers and positioners		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-064A2-1		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• System Description - 064D, RCS Instrumentation; • ALM-1C06, RCS Control		
Comments:	Modified version of Q14490		

13 **ID: Q92750** **Points: 1.00**

Unit-1 was conducting a plant startup with the following events and conditions:

- Annunciator "LOSS OF LOAD CH TRIP BYP" is in alarm
- The turbine has just been paralleled to the grid when condenser vacuum begins to degrade
- AOP-7G, Loss of Condenser Vacuum, has been implemented
- Condenser vacuum suddenly dropped to 22 inches Hg and stabilized at that value

Which one of the following statements describes the expected system response and/or required operator actions?

A. The turbine will trip automatically; the operators will trip the reactor; heat removal will be on the ADVs; SGFPs will continue to operate.

B. The reactor and turbine will be manually tripped; heat removal will be on the TBVs; SGFPs will continue to operate.

C. The turbine will trip automatically; the operators will trip the reactor; heat removal will be on the ADVs; SGFPs will trip.

D. The turbine will trip automatically; the operators will trip the reactor; heat removal will be on the TBVs; SGFPs will continue to operate.

Answer: A

Answer Explanation:

A. Correct - These actions are specified, for the given conditions, in AOP-7G.

B. Incorrect - The turbine will trip automatically at & the TBVs will be shut due to Condenser vacuum being less than 22.5"

C. Incorrect - The SGFPs will remain in operation (trip stpt = 20")

D. Incorrect - Heat removal will be via the ADVs, the TBVs will be shut due to Condenser vacuum being less than 22.5".

Question 13 Info			
Topic:	AOP-7G operator response(s).		
Tier/Group:	1/2		
K/A Info:	051 - Loss of Condenser Vacuum • AA2. Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: • AA2.02 Conditions requiring reactor and/or turbine trip		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	AOP-7G, Loss of Condenser Vacuum		
Comments:	Modified version of Q50782		

14 **ID: Q93030** **Points: 1.00**

A loss of Shutdown Cooling occurred on Unit-1. Heat removal has been restored using 11 LPSI Pump and 11 Shutdown Cooling Heat Exchanger (SDCHX).

Which **ONE** of the following choices correctly identifies instruments that must be used to ensure Heat Exchanger limits are not exceeded in accordance with AOP-3B, Abnormal Shutdown Cooling Conditions?

A. TR-351, SDC Temperatures **AND** FIC-306, SDC Flow Controller.

B. **ONLY** TR-351, SDC Temperatures.

C. TI-303X, 11 SDCHX Outlet Temperature **AND** FIC-306, SDC Flow Controller.

D. **ONLY** TI-303X, 11 SDCHX Outlet Temperature

Answer: D

Answer Explanation:

A. Incorrect – TR-351, SDC Temperatures, provides indication of temperatures to/from the RCS OI-3B, Shutdown Cooling, contains a caution specifying use of TI-303X, 11 SDCHX Outlet Temperature, to ensure the 14°F/min heatup rate limitation for the SDC HX is not exceeded. FIC-306, SDC Flow Controller, provides indication of total LPSI flow to the core and is not indicative the temperature change occurring in the SDCHX.

B. Incorrect – OI-3B, Shutdown Cooling, contains a caution specifying use of TI-303X, 11 SDCHX Outlet Temperature, to ensure the 14°F/min heatup rate limitation for the SDC HX is not exceeded.

C. Incorrect - OI-3B, Shutdown Cooling, contains a caution specifying use of TI-303X, 11 SDCHX Outlet Temperature, to ensure the 14°F/min heatup rate limitation for the SDC HX is not exceeded. FIC-306, SDC Flow Controller, provides indication of total LPSI flow to the core and is not indicative the temperature change occurring in the SDCHX.

D. Correct - OI-3B, Shutdown Cooling, contains a caution specifying use of TI-303X, 11 SDCHX Outlet Temperature, to ensure the 14°F/min heatup rate limitation for the SDC HX is not exceeded.

Question 14 Info			
Topic:	Monitoring RCS Cooldown on restoration of SDC		
Tier/Group:	1/1		
K/A Info:	025 Loss of Residual Heat Removal System (RHRS) • AA1. Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: • AA1.08 RHR cooler inlet and outlet temperature indicators		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-052-4-2 (slide 78)		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• AOP-3B, Abnormal Shutdown Cooling Conditions; • OI-3B, Shutdown Cooling		
Comments:	None		

15 **ID: Q24750** **Points: 1.00**

Which **ONE** of the following must be operable to ensure the Containment Purge System will be automatically secured should a fuel handling incident occur inside the Containment?

A. Containment High Range Monitors (RE-5317 A/B)

B. Main Vent Gaseous Monitor (RE-5415)

C. Containment Area Radiation Monitors (RE-5316 A thru D)

D. Wide Range Noble Gas Monitor (RIC-5415)

Answer: C

Answer Explanation:

A. Incorrect - The Containment High Range monitor has no connection to the Containment Purge System.

B. Incorrect - Main Vent Gaseous Monitor provides no automatic functions.

C. Correct - Per OI-36, Containment Purge System: **IF** moving irradiated fuel assemblies within the containment, **THEN** all four channels of Containment Area Radiation Monitors RI-5316A, B, C, and D are operable on the unit to be purged. (Tech Spec 3.3.7).

D. Incorrect - WRNGM provides no automatic functions.

Question 15 Info			
Topic:	Which instrument ensures that Cntmt Purge will be secured on a Fuel Handling Incident?		
Tier/Group:	1/2		
K/A Info:	036 - Fuel Handling Incidents • AA1. Ability to operate and / or monitor the following as they apply to the Fuel Handling Incidents: • AA1.01 Reactor building containment purge ventilation system		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-134-1-5-36		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 Panel Comprehensive Exam (October, 2009)		
Technical references:	Tech Spec 3.3.7, Containment Radiation Signal		
Comments:	None		

16 **ID: Q20295** **Points: 1.00**

Given a loss of Instrument Air on Unit-1 at 100% power.

Which **ONE** of the following alarms is expected to be received 10 PSIG **BELOW** the pressure at which a reactor trip is required?

A. BACK-UP IA INITIATED

B. FRV PNEUMATIC PRESS LO

C. INSTR AIR SYS MALFUNCTION

D. CNTMT IA ISOL 1-IA-2085-CV CLOSED

Answer: B

Answer Explanation:

A. Incorrect - This alarm indicates 1-PCV-6301 has opened as a result of I/A header pressure less than 85 PSIG (87 - 83 PSIG)

B. Correct - Alarms at approx. 40 PSIG. AOP-7D (LOSS OF INSTRUMENT AIR) directs tripping the reactor at 50 PSIG IA pressure.

C. Incorrect - Alarms at approx. 90 PSIG IA pressure.

D. Incorrect - Alarms at Approx. 75 PSIG IA pressure.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 16 Info			
Topic:	Loss of Instrument Air effects		
Tier/Group:	1/1		
K/A Info:	065 - Loss of Instrument Air • 2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-202-7-S-01-1		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	• 1C03-ALM, Window C-40 "FRV PNEUMATIC PRESS LO" • AOP-7D, Loss of Instrument Air		
Comments:	None		

17 **ID: Q92610** **Points: 1.00**

You are the CRO on Unit-1 when a plant trip occurs. While addressing the Vital Auxiliary safety function, you cannot verify CC flow to the RCPs. You attempt to stop 11A RCP by opening the normal feeder breaker from 1C06 but the breaker does not open.

Which **ONE** of the following actions will stop 11A RCP?

A. Open 252-1201 (RCP Bus Unit-1 feeder breaker), from 1C19.

B. Open the Alternate feeder breaker for 11A RCP, on 1C06.

C. Have the OSO open the RCP Bus Unit-1 feeder breaker in the Unit-2 Metalclad.

D. Open 252-2202 (RCP Bus Unit-1 feeder breaker), from 1C20.

Answer: A

Answer Explanation:

A. Correct - Opening breaker 252-1201 deenergizes the Unit-1 RCP Bus, securing all four RCPs, and all RCPs are being secured anyway.

B. Incorrect - If the normal feeder breaker is closed, the alternate feeder breaker would already be open.

C. Incorrect – Manual operation of the RCP Feeder versus remote operation is not preferred due to the industrial safety concerns with manual operation of a 13KV Breaker. Additionally, the RCP breaker needing to be opened is in the U-1 Metalclad, not the U-2 Metalclad.

D. Incorrect - Breaker 252-2202 is a normally open breaker.

Question 17 Info			
Topic:	RCP trip		
Tier/Group:	2/1		
K/A Info:	003 Reactor Coolant Pump System • 2.1.30 - Ability to locate and operate components, including local controls.		
RO Importance:	4.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	None		

18 **ID: Q50764** **Points: 1.00**

Both Units were operating at 100% power when a rupture occurred on the Unit-1 Instrument Air header. Given the following events and conditions:

- Plant air pressure dropped to 80 PSIG.
- The air leak was isolated by manual operator action
- Instrument air pressure increased to normal operating pressure

Which **ONE** of the choices below correctly describes the automatic response, if any, of 1-PA-2059-CV (PA HDR ISOL VLV) to the following:

(1) Lowering Instrument Air header pressure on the rupture and;
(2) Rising Instrument Air header pressure after the leak is isolated?

A. (1) The valve will open.
(2) No automatic response.

B. (1) The valve will open.
(2) The valve will close.

C. (1) The valve will close.
(2) No automatic response.

D. (1) The valve will close.
(2) The valve will open.

Answer: C

Answer Explanation:

A. Incorrect - Per AOP-7D, 1-PA-2059-CV automatically isolates P/A to P/A and must be manually opened.

B. Incorrect - Per AOP-7D, 1-PA-2059-CV automatically isolates P/A to P/A and must be manually opened.

C. Correct - Per AOP-7D, 1-PA-2059-CV automatically isolates P/A to P/A and must be manually opened.

D. Incorrect - Per AOP-7D, 1-PA-2059-CV automatically isolates P/A to P/A and must be manually opened.

Question 18 Info			
Topic:	Identify the design features that provide a backup for the instrument air system during a partial or		
Tier/Group:	2/2		
K/A Info:	079 - Station Air System (SAS) • K1 Knowledge of the physical connections and/or cause effect relationships between the SAS and the following systems: • K1.01 IAS.		
RO Importance:	3.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(4)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - 2006		
Technical references:	AOP-7D, Loss of Instrument Air		
Comments:	None		

19 **ID: Q50736** **Points: 1.00**

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

0200 LOCA occurred inside the Containment
0203 Containment pressure peaked at 20 PSIG
0240 Containment pressure dropped below 4 PSIG
0245 RWT level reached 0.75 feet but RAS failed to actuate

- Containment pressure is 3.5 PSIG and slowly lowering
- Containment sump level is 40 inches and rising
- CSAS has NOT been reset

Which **ONE** of the following statements correctly describes:

(1) Containment Spray (CS) pump configuration at the time of the RAS failure.
(2) Required Operator action, in EOP-5, to respond to the RAS failure.

A. (1) CS pumps are running with suction from the RWT.
(2) No Operator Action required.

B. (1) CS pumps are running with suction from the RWT.
(2) Align CS pump suctions to the Containment Sump.

C. (1) CS pumps are stopped.
(2) Align CS pump suctions to the Containment Sump.

D. (1) CS pumps are stopped.
(2) No Operator Action required.

Answer: B

Answer Explanation:

A. Incorrect - With RAS failure, Operator action is required to realign CS suction to the Cntmt Sump.

B. Correct - CS pumps should be running with suction aligned to the RWT. With RAS failure, Operator action is required to realign CS pump suction to the Cntmt Sump.

C. Incorrect - CS pumps are not secured on RAS or when containment pressure is less than CSAS.

D. Incorrect - CS pumps are not secured on RAS or when containment pressure is less than CSAS; With RAS failure, Operator action is required to realign CS suction to the Cntmt Sump.

Question 19 Info			
Topic:	RECALL the operation of ESFAS that includes: Failure of RAS		
Tier/Group:	2/1		
K/A Info:	026 - Containment Spray System (CSS) • K4 Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: • K4.07 Adequate level in containment sump for suction (interlock)		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 RO Audit Exam		
Technical references:	EOP-5, Loss of Coolant Accident		
Comments:	None		

20 **ID: Q25464** **Points: 1.00**

In addition to the "CSAS ACTUATED" annunciator alarm, which of the following conditions is verified to ensure Containment Spray Actuation has occurred per EOP-0, Post Trip Immediate Actions?

A. Operable Containment Air Coolers have shifted to "LOW" speed, Containment Spray Valves have opened and required flow is indicated.

B. Containment Spray Valves open with flow indicated and Condensate Booster pumps tripped.

C. Both MSIVs and MSIV Bypasses are shut, S/G Blowdown isolations are shut, and proper flow is indicated in each spray header.

D. SGFPs have tripped, MSIVs and MFW isolations are shut, and Containment Spray Pumps have started.

Answer: B

Answer Explanation:

A. Incorrect - Containment Coolers shift to "Low" on SIAS Actuation (not CSAS).

B. Correct - EOP-0 Basis Doc states: If pressure continues to rise and exceeds 4.25 PSIG, then CSAS is verified to have actuated to control containment pressure. This check should consist of ensuring that the alarm is received, the Containment Spray Valves are open, Spray flow is indicated and the Condensate Booster Pumps have tripped.

C. Incorrect - These actions do occur, however, EOP-0 Basis Doc states: If pressure continues to rise and exceeds 4.25 PSIG, then CSAS is verified to have actuated to control containment pressure. This check should consist of ensuring that the alarm is received, the Containment Spray Valves are open, Spray flow is indicated and the Condensate Booster Pumps have tripped.

D. Incorrect - These actions do occur, however, EOP-0 Basis Doc states: If pressure continues to rise and exceeds 4.25 PSIG, then CSAS is verified to have actuated to control containment pressure. This check should consist of ensuring that the alarm is received, the Containment Spray Valves are open, Spray flow is indicated and the Condensate Booster Pumps have tripped.

Question 20 Info			
Topic:	CSAS Verification		
Tier/Group:	2/1		
K/A Info:	022 Containment Cooling System (CCS) • A3 Ability to monitor automatic operation of the CCS, including: • A3.01 Initiation of safeguards mode of operation		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	SRO-201-0-8		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – 2008 LOR Quiz		
Technical references:	• EOP-O Technical Basis Doc; • NPOSSO 09-05, Standardization of Verifying ESFAS/AFAS Actuations		
Comments:	None		

21 **ID: Q93000** **Points: 1.00**

Unit-1 is at 100% power when 1-PT-1023, #12 S/G Pressure Channel "C" transmitter output fails high.

Which of the following is **TRUE,** regarding Channel "C", under this condition?

A. SGIS Sensor will **NOT** actuate;
SGIS Block Sensor will **NOT** actuate.

B. ASGT Trip Unit will **NOT** actuate;
SGIS Block Sensor will actuate.

C. AFAS Block Sensor will actuate;
ASGT Trip Unit will **NOT** actuate.

D. AFAS Block Sensor will actuate;
SGIS Sensor will actuate.

Answer: A

Answer Explanation:

A. Correct - S/G Pressure is an input to both SGIS and SGIS Block which actuate on lowering S/G pressure.

B. Incorrect - S/G Pressure is an input to SGIS Block and ASGT. ASGT will actuate, SGIS Block will not actuate.

C. Incorrect - S/G Pressure is an input to AFAS Block and ASGT. AFAS Block will actuate, ASGT will actuate.

D. Incorrect - S/G Pressure is an input to AFAS Block and SGIS. AFAS Block will actuate, SGIS will not actuate.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 21 Info			
Topic:	S/G pressure transmitter impact on ESFAS		
Tier/Group:	2/1		
K/A Info:	013 - Engineered Safety Features Actuation System (ESFAS) • K6 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: • K6.01 Sensors and detectors		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-63-1-3-03		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI-2008 Panel Comp Remediation Exam (October, 2009)		
Technical references:	LD-58; Engineered Safety Features System Description (No. 48)		
Comments:	Modified version of Q20772		

22 **ID: Q23850** **Points: 1.00**

Unit-1 is escalating in power, recovering from a mid-cycle forced outage. The reactor is at approximately 50% power with 11 SGFP in operation. 12 SGFP is out of service for maintenance.

Under these conditions, which **ONE** of the following sets of parameter values on 11 SGFP would support a decision to raise reactor power to 60% in accordance with OP-3, "Normal Power Operations".

A. Suspension flow: 16,200 GPM;
Turbine speed: 5160 RPM;
Suction pressure: 272 PSIG

B. Suction flow: 17,200 GPM;
Turbine speed: 5360 RPM;
Suction pressure: 262 PSIG

C. Suction flow: 15,200 GPM;
Turbine speed: 5060 RPM;
Suction pressure: 242 PSIG

D. Suction flow: 18,200 GPM;
Turbine speed: 5260 RPM;
Suction pressure: 252 PSIG

Answer: A

Answer Explanation:

A. Correct - From AOP-3G: If **ALL** the following conditions are maintained, then one SGFP operation above 440 MWE is permitted:
   - SGFP suction flow rate is below 18,000 GPM
   - SGFP suction pressure is above 250 PSIG
   - SGFP speed is below 5350 RPM

B. Incorrect – SGFPT speed is above the limit.

C. Incorrect - Suction pressure is below minimum limit.

D. Incorrect – SGFP suction flow is out of spec hi.

Question 22 Info			
Topic:	SGFP operating limitations		
Tier/Group:	2/1		
K/A Info:	059 - Main Feedwater System (MFW) • A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: • A1.03 - Power level restrictions for operation of MFW pumps and valves.		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	OI-12A, Feedwater System		
Comments:	None		

23 **ID: Q25461** **Points: 1.00**

When implementing EOP-0 Alternate Actions for an ATWS, which of the following parameters/indications are used to check the reactor has tripped?

A. Delta-T power, Startup Rate, RCS Boron Concentration.

B. NI power, CEA lower electrical limit lights, turbine load.

C. TCB position, Delta-T power, CEAPDS.

D. NI power, Startup Rate.

Answer: D

Answer Explanation:

A. Incorrect - Per EOP-0, a prompt drop in NI Power and a negative startup rate are used to verify the reactor is tripped for a normal trip and for an ATWS.

B. Incorrect - Per EOP-0, a prompt drop in NI Power and a negative startup rate are used to verify the reactor is tripped for a normal trip and for an ATWS.

C. Incorrect - Per EOP-0, a prompt drop in NI Power and a negative startup rate are used to verify the reactor is tripped for a normal trip and for an ATWS.

D. Correct - Per EOP-0, a prompt drop in NI Power and a negative startup rate are used to verify the reactor is tripped for a normal trip and for an ATWS.

Question 23 Info			
Topic:	Indications used to verify a reactor trip has occurred		
Tier/Group:	1/1		
K/A Info:	029 Anticipated Transient Without Scram (ATWS) • EK3 Knowledge of the reasons for the following responses as the apply to the ATWS: • EK3.01 Verifying a reactor trip; methods		
RO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	SRO-201-0-8		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - 2003		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	None		

24 **ID: Q93070** **Points: 1.00**

Unit-1 is operating at 100% power.

- The "CC HEAD TANK LEVEL" annunciator alarmed 5 minutes ago
- CC Head Tank level is 35 inches and lowering slowly

Which ONE of the following would **NOT** be a possible location of Component Cooling system inventory loss?

A. Reactor Vessel Support Cooler.

B. Reactor Coolant Pump Seal Cooler.

C. Component Cooling Heat Exchanger.

D. Reactor Coolant Drain Tank Heat Exchanger.

Answer: B

Answer Explanation:

A. Incorrect – Reactor Vessel support cooler would be a possible source of the leakage from the CCW system.

B. Correct - RCP seal cooler is at higher pressure than CC Head Tank and would cause CC Head Tank level to rise.

C. Incorrect – The Component Cooling Heat Exchanger would be a possible source of the leakage from the CCW system. Salt Water system pressure on the tube side of the heat exchanger is considerably lower than shell side CCW pressure.

D. Incorrect – Reactor Coolant Drain Tank Heat Exchanger would be a possible source of the leakage from the CCW system. Even if the RCDT Pump were running its discharge pressure of 50 PSI max is lower than the normal operating pressure of the CCW system.

Question 24 Info			
Topic:	Given any alarm, associated with the CCW system, identify the most likely cause of the alarm		
Tier/Group:	1/1		
K/A Info:	026 Loss of Component Cooling Water (CCW) • AA1. Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: • AA1.05 The CCWS surge tank, including level control and level alarms, and radiation alarm		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Audit Exam		
Technical references:	1C13-ALM; AOP-7C, Loss of Component Cooling Water		
Comments:	Modified version of question #Q74575		

25 **ID: Q92150** **Points: 1.00**

A loss of P-13000-1 has occurred. DG's do **NOT** repower their respective 4 KV buses and respective 4 KV Bus Alternate Feeder Breakers cannot be closed.

Which answer correctly identifies **ALL** HPSI pumps having the capability of being started from the Control Room?

A. **ONLY** 12 and 22 HPSI's

B. **ONLY** 13 and 23 HPSI's

C. 12, 13, 22, and 23 HPSI's

D. 11, 13, 21, and 23 HPSI's

Answer: C

Answer Explanation:

A. Incorrect - 11 & 21 HPSIs are powered from the Black Bus. Students may pick this answer if they don't recognize the disconnect/power alignment capability of 13 and 23 HPSI's.

B. Incorrect - While 13 and 23 HPSI **can** be electrically aligned to the ZB train power supply via disconnect alignment performed remotely in the Control Room, 12 and 22 HPSI's are powered from the Red Bus via P13000-2 and still have power available as well. Students may select this answer if they don't understand the normal power supply alignment.

C. Correct - 12 and 22 HPSI's are powered from the Red Bus via P13000-2 and still have power available. 13 and 23 HPSI can be electrically aligned to the ZB train power supply via disconnect alignment performed remotely in the Control Room.

D. Incorrect - While 13 and 23 HPSI can be electrically aligned to the ZB train power supply via disconnect alignment performed remotely in the Control Room, 11 & 21 HPSIs are powered from 11 & 21 4KV Busses which are powered from the Black Bus. Students may select this answer if they don't understand the normal power supply alignment.

Question 25 Info			
Topic:	Loss of 500KV Black Bus effects on HPSI Pumps		
Tier/Group:	2/1		
K/A Info:	062 A.C. Electrical Distribution • K2 Knowledge of bus power supplies to the following: • K2.01 Major system loads		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	OI-27C, 4.16 KV System		
Comments:	Modified version of Q75490		

26 **ID: Q93010** **Points: 1.00**

Using provided references:

After a control room evacuation due to a severe fire, the CRS directs you to commence boration on Unit-1.

- Initial RCS boron concentration is 350 ppm
- BAST concentration is 6.75%
- 11 BAST level is 129 inches
- 12 BAST level is 132.5 inches

What is the **MINIMUM** boration time to reach the required RCS boron concentration in accordance with the appropriate AOP?

A. 147-157 minutes

B. 168-178 minutes

C. 303-313 minutes

D. 330-340 minutes

Answer: D

Answer Explanation:

A. Incorrect - This value would be obtained if the student used the curve for two Charging Pumps borating at the stated BAST concentration. Only one Charging Pump would be in operation.

B. Incorrect - This value would be obtained if the student used the curve for two Charging Pumps borating with a BAST concentration of 6.25%. Only one Charging Pump would be in operation.

C. Incorrect - This value is obtained using AOP-9, Attachment 2, for one Charging Pump borating with a BAST concentration of 6.75%.

D. Correct - This value is obtained using AOP-9, Attachment 2, for one Charging Pump borating with a BAST concentration of 6.25%.

Question 26 Info			
Topic:	AOP-9A boration time **CALCULATION**		
Tier/Group:	Generic K & A		
K/A Info:	2.1.25 - Ability to interpret reference materials, such as graphs, curves, tables, etc.		
RO Importance:	3.9		
Proposed references to be provided to applicant:	Unit-1 AOP-9 Attachments, ATTACHMENT 2		
Learning Objective:	LOR-020060320-001		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Panel Exam		
Technical references:	AOP-9A, Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire		
Comments:	Modified version of Q19202		

27 **ID: Q25102** **Points: 1.00**

A reactor trip with a SIAS occurred on Unit-2. While implementing EOP-0 the following indications are noted:

- 2 stuck CEAs.
- One charging pump is operating
- 2-CVC-508-MOV, 22 BAST Gravity Feed is open
- 2-CVC-501-MOV, VCT Outlet is closed
- Pressurizer level is 120 inches and stable
- Pressurizer pressure is 1925 PSIA and lowering
- 21 and 22 S/G levels are -120 inches and lowering
- 21 and 22 S/G pressures are 800 PSIA and lowering
- Containment pressure is 1.0 PSIG and rising
- Containment temperature is 165 ˚F and rising
- P-13000-2 is de-energized

No additional actions have been taken.

Which **ONE** of the following groups of safety functions must be reported as "cannot be met"?

A. Reactivity Control and RCS Pressure/Inventory Control.

B. Reactivity Control and Core/RCS Heat Removal.

C. Vital Auxiliaries and Containment Environment.

D. Core/RCS Heat Removal and Containment Environment.

Answer: D

Answer Explanation:

A. Incorrect - Boration is in progress. Reactivity Control is complete.

B. Incorrect - Boration is in progress. Reactivity Control is complete.

C. Incorrect – Vital Auxiliaries is complete.

D. Correct - S/G pressure and level are not trending in a positive manner, containment pressure and temperature are also trending in the wrong direction.

Question 27 Info			
Topic:	A reactor trip and safe injection has occurred on Unit-2. While implementing EOP-0 the following(2)		
Tier/Group:	2/1		
K/A Info:	103 Containment System • A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: • A1.01 Containment pressure, temperature, and humidity		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	201-0-8-S-02		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	• EOP-0, Post Trip Immediate Actions • NO-1-201, Calvert Cliffs Operating Manual		
Comments:	None		

28 **ID: Q20320** **Points: 1.00**

The Instrument Air compressors do not receive a permissive start signal from the LOCI sequencer.

Which of the following is the reason for this?

A. Service Water cooling to the IA Compressors is isolated by SIAS.

B. To prevent overloading the safety related DGs.

C. The Instrument Air system is not required during a LOCA.

D. During a LOCA, power is unavailable to the air compressors.

Answer: A

Answer Explanation:

A. Correct - SRW is isolated, to the Turbine Building, by SIAS resulting in a loss of cooling to all Turbine Building loads including the I/A and P/A Compressors. From System Description 19, Section 4.4, Compressed Air Operation during SIAS/UV, on page 48: "For a loss of coolant casualty, the instrument and plant air compressors will trip on high temperature because the SIAS signal isolates SRW water from the Turbine Building". "If there was SIAS concurrent with UV, the compressors will load shed but not restart (no cooling water available)"

B. Incorrect - SRW is isolated, to the Turbine Building, by SIAS resulting in a loss of cooling to all Turbine Building loads including the I/A and P/A Compressors. The EDGs are capable of carrying the additional load imposed by the compressors.

C. Incorrect - SRW is isolated, to the Turbine Building, by SIAS resulting in a loss of cooling to all Turbine Building loads including the I/A and P/A Compressors. Key Instrument Air loads are supplied by the SWACs which receive a start signal as a result of a SIAS

D. Incorrect - SRW is isolated, to the Turbine Building, by SIAS resulting in a loss of cooling to all Turbine Building loads including the I/A and P/A Compressors. Power remains available to the compressors as long busses are powered.

Question 28 Info			
Topic:	Why do the instrument air compressors receive a permissive start signal from the Shutdown sequencer		
Tier/Group:	2/1		
K/A Info:	078 Instrument Air System • K1 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: • K1.04 Cooling water to compressor		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-63-1-3-42		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Panel Exam		
Technical references:	SD-019, Compressed Air Systems		
Comments:	None		

29 **ID: Q92751** **Points: 1.00**

Which of the following sets:

(1) Represent the **MINIMUM** conditions requiring "Double Protection" when tagging a mechanical system and;
(2) State the requirements, in accordance with CNG-OP-1.01-1007 Clearance and Safety Tagging, if "Double Protection" is not possible?

A. 1) A piping system that contains fluids greater than 500 PSIG or 200 °F;
2) Shift Manager approval of single boundary isolation is required.

B. 1) A piping system that contains fluids greater than 500 PSIG or 200 °F;
2) GS-Ops Support approval of single boundary isolation is required.

C. 1) A piping system that contains fluids greater than 350 PSIG or 200 °F;
2) Shift Manager approval of single boundary isolation is required.

D. 1) A piping system that contains fluids greater than 350 PSIG or 200 °F;
2) GS-Ops Support approval of single boundary isolation is required.

Answer: A

Answer Explanation:

A. Correct - Per CNG-OP-1.01-1007; the use of two isolation points in series to provide an added measure of protection when the energy source exceeds or could exceed 200 °F or 500 PSIG pressure or contains an explosive, oxidizing gas, or hazardous material for mechanical systems. Authorizing the isolation of equipment per this procedure. The Work Center Senior Reactor Operator (SRO), Fix It Now (FIN) SRO, or Control Room Supervisor (CRS) may perform the functions for the SM described in this procedure as his designee, when the individual is knowledgeable of current plant conditions and designated by the SM. The SM or designee shall: ... Approve the use of single boundary valve use when double valve isolation is required

B. Incorrect - GS-Ops Support is incorrect.

C. Incorrect – 350 PSIG is incorrect

D. Incorrect - 350 PSIG is incorrect GS-Ops Support is incorrect.

Question 29 Info			
Topic:	Apply the Requirements of NO-1-112, Safety Tagging		
Tier/Group:	Generic K & A		
K/A Info:	2.2.13 Knowledge of tagging and clearance procedures		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	CNG-OP-1.01-1007 Clearance and Safety Tagging		
Comments:	Modified version of Q51180		

30 **ID: Q92632** **Points: 1.00**

Unit-1 is has just tripped. The following conditions exist:

- AFAS "A" has **NOT** actuated
- AFAS "B" has actuated
- 11 S/G pressure is 790 PSIA
- 12 S/G pressure is 895 PSIA

Which of the following statements describes the expected plant response?

A. AFW Flow of 300 GPM is initiated to each S/G.

B. AFW Flow of 300 GPM is initiated to 12 S/G only.

C. AFW Flow of 150 GPM is initiated to 12 S/G only.

D. AFW Flow of 150 GPM is initiated to each S/G.

Answer: D

Answer Explanation:

A. Incorrect - AFAS "B" starts only the Steam Driven AFW Pump aligned for auto initiation. Flow will be regulated at 150 GPM each to 11 and 12 S/Gs. Candidate may choose this answer if he assumes the Motor Driven AFW Pump starts as well. This answer would be correct for Unit-2. This answer would be correct if AFAS "A" initiated. This would provide an additional 150 GPM for a total AFW flow of 300 GPM.

B. Incorrect - AFAS "B" starts only the Steam Driven AFW Pump aligned for auto initiation. Flow will be regulated at 150 GPM each to 11 and 12 S/Gs. Candidate may choose this answer if he mistakenly associates AFAS "B" with 12 S/G.

C. Incorrect - AFAS "B" starts only the Steam Driven AFW Pump aligned for auto initiation. Flow will be regulated at 150 GPM each to 11 and 12 S/Gs. Candidate may choose this answer if he mistakenly associates AFAS "B" with 12 S/G.

D. Correct - AFAS "B" starts only the Steam Driven AFW Pump aligned for auto initiation. Flow will be regulated at 150 GPM each to 11 and 12 S/Gs.

Question 30 Info			
Topic:	Assuming Middle of Cycle MTC, and the unit at 100% power, how does an inadvertent AFAS affect react		
Tier/Group:	2/1		
K/A Info:	061 Auxiliary / Emergency Feedwater (AFW) System • K3 Knowledge of the effect that a loss or malfunction of the AFW will have on the following: • K3.02 S/G		
RO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-34-2-3-21		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	SD-036, AFW System Description		
Comments:	None		

31 **ID: Q38849** **Points: 1.00**

With the ADVs in AUTO, how do they function following a Reactor Trip from 50% Reactor power?

A. ADVs will Quick OPEN until T_{AVE} is less than 535 °F then they will modulate to maintain temperature between 535 °F and 540 °F.

B. ADVs will modulate to maintain T_{AVE} between 535 °F and 540 °F.

C. ADVs will modulate to maintain Main Steam Pressure less than 900 PSIG.

D. ADVs will Quick OPEN until T_{AVE} is less than 535 °F then they will modulate to maintain Main Steam pressure less than 900 PSIG.

Answer: B

Answer Explanation:

A. Incorrect - The ADVs will not quick open as the quick open override is not enabled until RRS T_{AVE} exceeds 557 °F which equates to a reactor power of approximately 62%.

B. Correct - The ADVs are controlled by RRS T_{AVE} with the valves beginning to open at a T_{AVE} of approximately 540 °F to lower T_{AVE} to a value of approximately 535 °F.

C. Incorrect - This would be a correct statement for the TBVs. The ADVs are controlled by RRS T_{AVE} with the valves beginning to open at a T_{AVE} of approximately 540 °F to lower T_{AVE} to a value of approximately 535 °F.

D. Incorrect - The ADVs will not quick open as the quick open override is not enabled until T_{AVE} exceeds 557 °F which equates to a reactor power of approximately 62%.

Question 31 Info			
Topic:	ADV control, on a Reactor trip, with an initial Reactor power of 50%		
Tier/Group:	2/1		
K/A Info:	039 Main and Reheat Steam System (MRSS) • K4 Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: • K4.02 Utilization of T-ave. program control when steam dumping through atmospheric relief/dump valves, including T-ave. limits		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	SD-056, Reactor Regulating System		
Comments:	None		

32 **ID: Q24945** **Points: 1.00**

During a reactor startup, as the RO commences withdrawing Regulating Group 2, he notices the reactor is critical.

Which **ONE** of the following describes the actions necessary, per OP-2, Plant Startup from Hot Standby to Minimum Load, to restore shutdown margin?

A. Insert all Shutdown CEAs.

B. Trip the reactor.

C. Insert all Regulating CEAs.

D. Initiate fast boration.

Answer: D

Answer Explanation:

A. Incorrect – This action does not restore shutdown margin.

B. Incorrect – This action does not restore shutdown margin.

C. Incorrect – This action does not restore shutdown margin.

D. Correct – Initiation of boration is the sole method available to restore SDM to within limit. Fast boration is the appropriate method for quickly reestablishing required Shutdown Margin.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 32 Info			
Topic:	During a reactor startup all shutdown groups are fully withdrawn.		
Tier/Group:	1/2		
K/A Info:	024 Emergency Boration • AK3. Knowledge of the reasons for the following responses as they apply to Emergency Boration: • AK3.01 When emergency boration is required		
RO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	203-1-S-06		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - LOR Quiz (February, 2010)		
Technical references:	OP-2, Plant Startup from Hot Standby to Minimum Load		
Comments:	None		

33 **ID: Q92170** **Points: 1.00**

With Unit-1 at 100% power, the Control Room receives various panel alarms. A loss of 1Y09 is diagnosed.

Which **ONE** of the following responses will result from this loss of power?

A. The process indicator on 1-HIC-100, Pressurizer Spray Valve Controller fails downscale.

B. All three Charging pumps will start and will **NOT** cycle automatically on PZR level signals.

C. 1-CC-3832-CV, Component Cooling Containment Supply, fails shut.

D. 1-CVC-501-MOV, VCT Outlet, shuts and 1-CVC-504-MOV, Charging Pump suction from the RWT, opens.

Answer: A

Answer Explanation:

A. Correct - Power is lost to HIC-100 resulting in its indication failing downscale. Its output also fails to zero, resulting in no signal to open the Pressurizer Spray Valves.

B. Incorrect - These actions result from a loss of 1Y10.

C. Incorrect - Component Cooling Containment Supply, 1-CC-3832-CV fails shut on a loss of 11 125V DC Bus.

D. Incorrect - These actions result from a loss of 1Y10.

Question 33 Info			
Topic:	Loss of 1Y09 effects on Pressurizer Pressure control		
Tier/Group:	2/1		
K/A Info:	010 Pressurizer Pressure Control System (PZR PCS) • K2 Knowledge of bus power supplies to the following: • K2.02 Controller for PZR spray valve		
RO Importance:	2.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• AOP-7I, Loss of 4KV, 480 V or 208/120 V Inst Bus Power • Unit-1 Stabilizing Actions Plaque		
Comments:	None		

34 **ID: Q92171** **Points: 1.00**

Unit-2 has been operating at 100% power when a small RCS break occurs. The crew has just transitioned from EOP-0, Post Trip Immediate Actions, to the appropriate Optimal Recovery Procedure.

- RCS pressure is stable at 1200 PSIG.
- Containment Pressure peaked and stabilized at 4.0 PSIG.

Which of the following component indications will be found illuminated, on the Control Room panels?

A. 21 Condensate Booster Pump green lamp;
21 Heater Drain Pump green lamp.

B. 21 Charging Pump red lamp;
21 Boric Acid Pump red lamp.

C. 2-SI-4150-CV, 21 CS HDR VLV, red lamp;
2-SI-4151-CV, 22 CS HDR VLV, red lamp.

D. 21 MSIV green lamp;
21 S/G Feedwater Isolation Valve green lamp.

Answer: B

Answer Explanation:

A. Incorrect - Condensate Booster Pumps and Heater Drain Pumps are verified on Attachment 3, CSAS Verification Checklist, and Attachment 7, SGIS Verification Checklist.

B. Correct - ATTACHMENT (2), Page 3 of 5, SIAS VERIFICATION CHECKLIST, verifies 11, 12 and 13 CHG PPs running and 11 and 12 BA PPs running

C. Incorrect - CS HDR Vlvs are verified on Attachment 3, CSAS Checklist

D. Incorrect - FW ISOL valves are verified on Attachment 3, CSAS Verification Checklist, and Attachment 7, SGIS Verification Checklist.

Question 34 Info			
Topic:	SIAS Verification Checklist		
Tier/Group:	1/1		
K/A Info:	009 Small Break LOCA • EA2 Ability to determine or interpret the following as they apply to a small break LOCA: • EA2.29 CVCS pump indicating lights for determining pump status		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP Attachments, Attachment (2)		
Comments:	None		

35 **ID: Q92850** **Points: 1.00**

Following a Design Basis Large Break LOCA on Unit-2, RAS has actuated, and HPSI has been throttled to 250 GPM per header.

One hour later, running HPSI pump amperage and flow indications are observed to be oscillating.

Which **ONE** of the following actions is preferred to mitigate the HPSI pump amp and flow oscillations per EOP-5, Loss of Coolant Accident?

A. Shut Mini Flow Return to the RWT Isolation MOVs, 2-SI-659 and 2-SI-660.

B. Throttle HPSI flow to minimum per EOP Attachment 10, HPSI Flow.

C. Secure one of the operating HPSI Pumps.

D. Stop both Containment Spray Pumps.

Answer: B

Answer Explanation:

A. Incorrect – Shutting these valves will reduce flow thru the HPSI Pumps, however, the Mini Flow Returns to the RWT, MOV's 2-SI-659 and 2-SI-660, are shut by the RAS signal assuming the lockouts are positioned as directed by the procedure in anticipation of RAS. No information is given to indicate the valves did not perform as designed.

B. Correct - EOP-5, Step IV.S.1.j.(1) specifies: Throttle HPSI flow equally among the four headers to the minimum allowed **PER** ATTACHMENT(10), HIGH PRESSURE SAFETY FLOW.

C. Incorrect - This action would be taken if throttling **PER** ATTACHMENT(10), HIGH PRESSURE SAFETY FLOW, and securing the Containment Spray Pumps were unsuccessful in eliminating indication of cavitation.

D. Incorrect - This action would be taken if throttling **PER** ATTACHMENT(10), HIGH PRESSURE SAFETY FLOW, was unsuccessful in eliminating indication of cavitation.

Question 35 Info			
Topic:	Response to HPSI Cavitation		
Tier/Group:	1/1		
K/A Info:	011 - Large Break LOCA • EK2- Knowledge of the interrelations between the and the following Large Break LOCA: • EK2.02 - Pumps (2.6, 2.7)		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-5, Loss of Coolant Accident		
Comments:	None		

36 **ID: Q92772** **Points: 1.00**

Why is Quench Tank pressure maintained less than 1.5 PSIG while drawing a Pressurizer bubble, per OP-7, Shutdown Operations?

A. Prevents Pressurizer Vent SVs from leaking by.

B. Prevents Pressurizer Safety Valves from unseating.

C. Prevents Reactor Vessel Vent SVs from leaking by.

D. Prevents Power Operated Relief Valves from unseating.

Answer: D

Answer Explanation:

A. Incorrect - Per OI-1B (Quench Tank Operations), Section 6.11 (Quench Tank Lineup for Plant Startup at Low RCS Pressure), Quench Tank pressure is maintained less than 1.5 PSIG to help prevent the PORVs from leaking.

B. Incorrect - Per OI-1B (Quench Tank Operations), Section 6.11 (Quench Tank Lineup for Plant Startup at Low RCS Pressure), Quench Tank pressure is maintained less than 1.5 PSIG to help prevent the PORVs from leaking.

C. Incorrect - Per OI-1B (Quench Tank Operations), Section 6.11 (Quench Tank Lineup for Plant Startup at Low RCS Pressure), Quench Tank pressure is maintained less than 1.5 PSIG to help prevent the PORVs from leaking.

D. Correct – Per OP-7, Sect 6.1.2 Prepare RCS for Drawing Pressurizer Bubble contains a note that states maintaining Quench Tank pressure less than 1.5 PSIG helps prevent PORVs from leaking. Per OI-1B (Quench Tank Operations), Section 6.11 (Quench Tank Lineup for Plant Startup at Low RCS Pressure), Quench Tank pressure is maintained less than 1.5 PSIG to help prevent the PORVs from leaking.

Question 36 Info			
Topic:	Quench Tank parameters for drawing a bubble.		
Tier/Group:	2/1		
K/A Info:	007 Pressurizer Relief Tank/Quench Tank System (PRTS) • K5 Knowledge of the operational implications of the following concepts as the apply to PRTS: • K5.02 Method of forming a steam bubble in the PZR		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• OP-7, Shutdown Operations • OI-1B, Quench Tank operations		
Comments:	None		

37 **ID: Q92190** **Points: 1.00**

With the Unit-1 in Mode 3, maintaining NOP/NOT conditions, an Instrument Air header rupture occurs in the Unit-1 27' East Piping Penetration Room. The leak has been isolated resulting in a complete loss of Instrument Air to all loads **IN AND DOWNSTREAM** of the Unit-1 27' East Piping Penetration Room.

Which **ONE** of the following actions is required, in accordance with the Loss of Instrument Air Abnormal Operating Procedure?

A. Have the ABO manually override ADVs shut.

B. Operate Auxiliary Spray as needed to control RCS pressure.

C. Stop all RCPs then verify Natural Circulation in at least one loop.

D. Take actions for the 1B DG being out of service, due to loss of cooling.

Answer: C

Answer Explanation:

A. Incorrect - ADVs fail shut and the manual override is only to open them (cannot be overridden shut). ADVs are in an adjacent room whose air supply would not be impacted by isolation of the leak.

B. Incorrect – The Auxiliary Spray CV would fail closed on the Loss of Instrument Air to the Unit-1 27' East Piping Penetration Room if that portion of the header is isolated. If not, the normal spray CV's would be available until the RCPs are secured. The stem clarifies that Instrument Air is isolated to the containment, and therefore to the Auxiliary Spray valve, by making reference to Instrument Air loads downstream of the 27' East Piping Penetration Room being isolated as well.

C. Correct - AOP-7D, Loss of Instrument Air specifies: **IF EITHER** of the CC CNTMT SUPPLY and RETURN valves begin to shut **AND** the "CCW FLOW LO" alarms are received on the RCPs, **THEN** Stop **ALL** RCPs **THEN** verify Natural Circulation in at least one loop.

D. Incorrect - 1B DG SRW CV fails open. Cooling is not lost to the EDG.

Question 37 Info			
Topic:	Mode 3 I/A Header Rupture		
Tier/Group:	2/1		
K/A Info:	008 Component Cooling Water System (CCWS) • Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.05 - Effect of loss of instrument and control air on the position of the CCW valves that are air operated		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-020400303-002		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• AOP-7D, Loss of Instrument Air • AOP-3E, Loss of All RCP Flow, Modes 3, 4, or 5		
Comments:	None		

38 **ID: Q19364** **Points: 1.00**

Unit-2 is at 100% power with all 10 trip units bypassed on Channel D RPS for IM Shop wiring modifications, IAW an approved maintenance order. IM determines that the RPS channel must be de-energized to complete the modifications.

What statement best describes the RPS trip logic before and after Channel D RPS is de-energized?

A. 2 of 3 when energized;
1 of 3 when de-energized.

B. 2 of 4 when energized;
2 of 3 when de-energized.

C. 2 of 3 when energized;
2 of 3 when de-energized.

D. 2 of 4 when energized;
1 of 3 when de-energized.

Correct answer changed from 'A' to 'C' in post-exam comment resolution.

Answer: ~~A~~ C

Answer Explanation:

A. Incorrect. ~~Correct~~ Trip logic is 2 of 3 with the Trip Units bypassed while the channel is still energized. De-energizing a channel does not remove~~s~~ the bypass function~~, resulting in that channel being tripped~~. As a result ~~1~~ 2 of 3 remaining Trip Units tripping will cause a reactor trip.

B. Incorrect - Trip logic is 2 of 3 with the Trip Units bypassed while the channel is still energized.

C. Correct. ~~Incorrect~~ - Trip logic is 2 of 3 with the Trip Units bypassed while the channel is still energized. De-energizing a channel does not remove~~s~~ the bypass function~~, resulting in that channel being tripped~~. As a result ~~1~~ 2 of 3 remaining Trip Units tripping will cause a reactor trip.

D. Incorrect – Trip logic is 2 of 3 with the Trip Units bypassed while the channel is still energized.

Question 38 Info			
Topic:	RPS Trip Logic		
Tier/Group:	2/1		
K/A Info:	012 Reactor Protection System (RPS) • Ability to monitor automatic operation of the RPS, including: • A3.01- Individual channel		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-058-1-01		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	System Description 058, Reactor Protective System		
Comments:	None		

39 **ID: Q92210** **Points: 1.00**

Which radiation monitor **MUST** be used to verify the Containment Environment safety function during EOP-0 under **ALL** plant conditions (LOCA, Loss of offsite power, etc.) and what is the basis for use of this instrument?

A. Containment Atmosphere Particulate Monitor (RI-5280);
With 1% failed fuel, will detect a 1 GPM RCS leak within 1 hour.

B. Containment High Range Monitors RI-5317A & B;
Availability during any combination of events.

C. Containment Area Monitors RE-5316A - D;
Powered from vital AC and will be available in all circumstances.

D. Containment Atmosphere Gaseous Monitor (RI-5281);
Provides ability to promptly assess RCS leakage.

Answer: B

Answer Explanation:

A. Incorrect - Containment Atmosphere Particulate Monitor (RI-5280) is isolated on a SIAS.

B. Correct - Any containment radiation monitor can be used to indicate the off normal event. However, as a minimum the Containment High Range Monitors should be checked, based on their availability during any combination of events, including SIAS actuations and LOOP events.

C. Incorrect - RE-5316 A-D are deenergized during power operation.

D. Incorrect - Containment Atmosphere Gaseous Monitor (RI-5281) is powered from MCC-103 which is not backed up by emergency DG power.

Question 39 Info			
Topic:	Which rad monitor should be used to verify Containment Environment safety function during EOP 0?		
Tier/Group:	1/2		
K/A Info:	061 Area Radiation Monitoring (ARM) System Alarms • Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring (ARM) System Alarms: • AK1.01- Detector limitations		
RO Importance:	2.5		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-122-1-3-37		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-0 Technical Basis Document		
Comments:	None		

40 **ID: Q19477** **Points: 1.00**

The 0C DG was slow started from the control room. What action is required to obtain speed control for synchronizing?

A. Depress the Emergency Start Pushbutton.

B. Insert the sync stick in Bkr 152-0701 (07 4KV Bus Tie) and momentarily go to raise or lower on the speed control handswitch.

C. Place the Unit Parallel switch to Parallel.

D. Insert the sync stick in Bkr 152-0703 (0C DG Output Bkr) and momentarily go to raise or lower on the speed control handswitch.

Answer: D

Answer Explanation:

A. Incorrect - Pushing Emergency Start PB will put OC DG in Reset Mode.

B. Incorrect - Bkr 152-0701 is not used in the control scheme for the OC DG.

C. Incorrect - There is no Unit Parallel Switch on the OC DG. Plausible because these controls do exist for the Fairbanks Morse D/Gs and this would be the correct answer for the Fairbanks Morse D/Gs.

D. Correct - IF 0C DG will be paralleled with 07 4KV BUS FDR, 152-0704, from the Control Room, THEN INSERT the Sync Stick for 0C DG OUT BKR, 0-CS-152-0703, to put the governor in the parallel mode. MOMENTARILY PLACE 0C DG SPEED CONTR, 0-CS-0705, to RAISE OR LOWER. Momentary operation of the speed control handswitch is required, per the procedure, to obtain speed control of the engine.

Question 40 Info			
Topic:	OC DG speed control		
Tier/Group:	2/1		
K/A Info:	064 Emergency Diesel Generators (ED/G) • Ability to manually operate and/or monitor in the control room: • A4.06 - Manual start, loading, and stopping of the ED/G		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	DIESELS-22		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	OI-21C 0C Diesel Generator		
Comments:	None		

41 **ID: Q93020** **Points: 1.00**

Unit-1 is at 100% power when an instrument air leak on 1-CVC-515-CV, L/D CNTMT ISOL, results in letdown being isolated.

Which **ONE** of the following describes:
(1) The **immediate** concern with continued operation of the plant in this condition and;
(2) The **preferred** method to mitigate the consequences of this condition per the controlling procedure?

A. (1) Thermal transients on the Chemical and Volume Control system;
(2) Place the backup charging pumps in pull to lock. Manually operate the selected Charging pump, as needed.

B. (1) Continued operation of the charging system will result in exceeding the Pzr level operating band;
(2) Place the selected Charging pump in pull to lock and allow the Backup Charging pump(s) to automatically operate as needed.

C. (1) Thermal transients on the Chemical and Volume Control system;
(2) Operate the Charging pumps, as necessary, and reduce power per OP-3, Normal Power Operation, as needed.

D. (1) Continued operation of the charging system will result in exceeding the Pzr level operating band;
(2) Place the backup charging pumps in pull to lock. Manually operate the selected Charging pump, as needed.

Answer: B

Answer Explanation:

A. Incorrect – OI-2A, Chemical & Volume Control System, specifies use of the Backup Charging Pumps to control Pressurizer Level.

B. Correct – Exceeding the T.S. limit for Pressurizer Level is a concern with L/D secured with Charging remaining in operation. OI-2A directs placing selected Charging Pump in PTL with Backup Charging pump(s) in auto to control Pressurizer level.

C. Incorrect - OI-2A, Chemical & Volume Control System, specifies use of the Backup Charging Pumps to control Pressurizer Level.

D. Incorrect - OI-2A, Chemical & Volume Control System, specifies use of the Backup Charging Pumps to control Pressurizer Level.

Question 41 Info			
Topic:	Loss of Letdown Flow		
Tier/Group:	2/2		
K/A Info:	011 Pressurizer Level Control System (PZR LCS) • A2 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.07 Isolation of letdown		
RO Importance:	3.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	N/A		
Technical references:	1C07-ALM; OI-2A, Chemical and Volume Control System		
Comments:	Modified version of Q14333		

42 **ID: Q92230** **Points: 1.00**

Given the following:

- Both Units are operating at 100% power when a Station Blackout occurs.
- 125 VDC Bus voltages are approaching 105 VDC

Which, if any, DG combinations, when restored, will ultimately restore a Battery Charger to 11, 12, 21 and 22 125 VDC Busses?

A. 1A; 2A

B. 1B; 2B

C. 2A; 2B

D. None of the listed combinations will restore a Battery Charger to each 125 VDC Bus.

Answer: C

Answer Explanation:

A. Incorrect - The 1A & 2A DGs power only the Battery Chargers associated with 11 and 22 125 VDC Busses

B. Incorrect - The 1B & 2B DGs power only the Battery Chargers associated with 12 and 21 125 VDC Busses

C. Correct - The 2A & 2B DGs power one Battery Charger associated with each of the four 125 VDC Busses

D. Incorrect - The 2A & 2B DGs power one Battery Charger associated with each of the four 125 VDC Busses

Question 42 Info			
Topic:	Relationship of DGs and 125 VDC		
Tier/Group:	1/1		
K/A Info:	058 Loss of DC Power • AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: • AK1.01 Battery charger equipment and instrumentation		
RO Importance:	2.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(8)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• AOP-7J, Loss of 120 Volt Vital AC or 125 Volt Vital DC Power • EOP-7, Station Blackout		
Comments:	None		

43 **ID: Q92250** **Points: 1.00**

Given the following conditions:

- Unit-2 was at 100% power with 21 & 22 SRW pumps running
- 23 SRW pump was aligned per normal operation (electrical & mechanical)
- 22 SRW pump tripped on overcurrent
- A LOOP occurred and 2A & 2B EDG started and energized 21 & 24 4 KV buses

One minute later, which SRW pumps, if any, would be operating? (Assume no operator action)

A. None

B. 21 and 23 SRW Pumps

C. 21 SRW pump **ONLY**

D. 23 SRW pump **ONLY**

Answer: B

Answer Explanation:

A. Incorrect - SRW Pumps are started by the Shutdown Sequencer (SDS), on a LOOP.

B. Correct - 23 SRW Pump is normally aligned to 24 4KV Bus and that, with a LOOP, it will start 1 second after sensing the failure of 22 SRW Pump to start.

C. Incorrect - 23 SRW Pump is normally aligned to 24 4KV Bus and that, with a LOOP, it will start 1 second after sensing the failure of 22 SRW Pump to start.

D. Incorrect - 21 SRW will start on the SDS, after 21A DG closes in on 21 4KV bus

Question 43 Info			
Topic:	SRW Pp response to a LOOP		
Tier/Group:	2/1		
K/A Info:	076 Service Water System (SWS) • Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: • K4.02 Automatic start features associated with SWS pump controls		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• EOP-2, Loss of Offsite Power/Loss of Forced Circulation • SD-011 SRW System Description		
Comments:	Modified version of Q20568		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

44 **ID: Q92273** **Points: 1.00**

Given the following:

- The 1A2-11 Starting Air Receiver is tagged out to repair an air leak
- The 1A1 Starting Air Compressor fails
- 1A1-11 and 1A1-12 Starting Air Receiver pressures are 500 PSIG and slowly lowering

Which **ONE** of the following actions, if any, can be taken to maintain 1A DG operability?

A. Emergency start the 1A DG prior to the 1A1-11 and 1A1-12 Starting Air Receiver pressures falling below 290 PSIG.

B. Cross-connect the 1A and OC DG Starting Air Systems prior to the 1A1-11 and 1A1-12 Starting Air Receiver pressures falling below 290 PSIG.

C. No actions can be taken; declare the 1A DG inoperable when the 1A1-11 and 1A1-12 Starting Air Receiver pressures fall below 290 PSIG.

D. Crosstie the 1A1 and 1A2 Starting Air Systems prior to the 1A1-11 and 1A1-12 Starting Air Receiver pressures falling below 290 PSIG.

Answer: D

Answer Explanation:

A. Incorrect – Placing the system in its fail-safe condition (e.g., running) does not, in and of itself, maintain operability.

B. Incorrect - There is no physical means to cross-connect the 1A and OC DG Starting Air Systems. Plausible because, although there is no physical means to cross-connect the starting air systems, the candidate may believe the design is similar to that of the Fairbanks Morse engines where the Starting Air Headers for all three DGs are cross-tied.

C. Incorrect - While this action could be considered at some point, the immediate response would be to crosstie the starting air systems.

D. Correct – The 1A1 and 1A2 Starting Air Systems can be cross tied with either compressor supplying all receivers.

Question 44 Info			
Topic:	1A DG Starting Air Receiver Pressure impact on DG operability		
Tier/Group:	2/1		
K/A Info:	064 Emergency Diesel Generators (ED/G) • Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: • K6.07 - Air receivers		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	1C188-ALM, 1A DG Local Control Panel Alarm Manual		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

45 **ID: Q92270** **Points: 1.00**

Given the following:

- Unit-1 was operating at 100% power, steady state
- A reactor trip occurred and EOP-1, Reactor Trip, has been implemented
- Pressurizer Level Control Channel 1-LIC-110X is selected
- PZR Heater Low Level Cutout Switch is in the X + Y position
- A leak occurs on the variable leg for 1-LT-110X, causing an 80 inch indicated level error
- "PZR CH X LVL" Annunciator is in alarm

Which **ONE** of the following describes:
(1) The effect on the plant this condition would cause and;
(2) What is the preferred method to mitigate this event?

A. (1) All Back-up Charging Pumps stop, Pressurizer Heaters energize;
(2) Shift to 1-LIC-110Y in service.

B. (1) All Back-up Charging Pumps start, Pressurizer Heaters deenergize;
(2) Shift to 1-LIC-110Y in service.

C. (1) All Back-up Charging Pumps start, Pressurizer Heaters energize;
(2) Shift 1-LIC-110X to manual and establish level control.

D. (1) All Back-up Charging Pumps stop, Pressurizer Heaters deenergize;
(2) Shift 1-LIC-110X to manual and establish level control.

Answer: B

Answer Explanation:

A. Incorrect - A leak on the variable leg of a transmitter would cause the indicated level to be lower than the actual level. A lower Pzr level would cause Chg Pumps to start and L/D flow to go to minimum. If indicated level were below 101 inches then the Low Level cutout would deenergize all Pzr heaters.

B. Correct - A leak on the variable leg of a transmitter would cause the indicated level to be lower than the actual level. A lower Pzr level would cause Chg Pumps to start and L/D flow to go to minimum. If indicated level were below 101 inches then the Low Level cutout would deenergize all Pzr heaters. Reference EOP-1, Sect IV.D.1.1. Although answer D may provide a technically viable option for addressing this situation, answer B is "preferred" based on Alarm Manual guidance for this condition, associated procedure use standards (i.e., use procedure guidance if available), and based on reinforcement in the LOI training program as the preferred method of recovery.

C. Incorrect - Pzr heaters would deenergize.

D. Incorrect - A leak on the variable leg of a transmitter would cause the indicated level to be lower than the actual level. A lower Pzr level would cause Chg Pumps to start.

Question 45 Info			
Topic:	Predict the response to a Pzr Lvl Control channel failure.		
Tier/Group:	1/2		
K/A Info:	028 Pressurizer (PZR) Level Control Malfunction • 2.4.6 Knowledge of EOP mitigation strategies.		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-064A2-1		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	1C06-ALM, RCS Control Alarm Manual		
Comments:	None		

46 **ID: Q18935** **Points: 1.00**

During an excess steam demand event, the unaffected SG is maintained within 25 ˚F of CET temperature using its ADV. What is the primary operational implication of this limit during the uncontrolled RCS cooldown?

A. This minimizes the potential for pressurized thermal shock if a heatup of the RCS occurs following an excessive cooldown of the RCS.

B. This minimizes the formation of tube voids, in the affected S/G, after blowdown is complete.

C. This minimizes the RCS cooldown that takes place during blowdown of the affected S/G.

D. This minimizes differential pressure between the S/Gs, thereby allowing reset of the AFAS Block signal.

Answer: A

Answer Explanation:

A. Correct - See EOP-4 Technical Basis Document, Step IV.H.2 (page 27). The 25 ˚F limit is an operational limit associated with PTS mitigation during a cooldown event. Its basis supports the same basis as the broader concept of cooldown limits as referenced in the KA, under which this limit lies. This action sets up the operational controls to support PTS prevention when the uncontrolled cooldown has been completed.

B. Incorrect - S/G tube voiding is determined by RCS pressure being less than saturation pressure for that S/G, and once B/D is complete S/G pressure will be zero.

C. Incorrect - RCS cooldown during the blowdown phase is determined by the size of the leak.

D. Incorrect - AFAS Block will occur, isolating Auxiliary Feedwater flow to the S/G with the lower pressure which is the **affected** S/G.

Question 46 Info			
Topic:	ESDE unaffected S/G temperature limits		
Tier/Group:	2/1		
K/A Info:	039 Main and Reheat Steam System (MRSS) • K5 Knowledge of the operational implications of the following concepts as the apply to the MRSS: • K5.05 Bases for RCS cooldown limits		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-020170410-002		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	EOP-4, Excess Steam Demand Event		
Comments:	None		

47 **ID: Q92290** **Points: 1.00**

Given the following:

- It is 0230 on a Saturday morning
- 21 125V DC Bus has an existing positive ground.

Which **ONE** of the following statements best describes:

(1) What could occur if a negative ground develops on 21 125V DC Bus and;
(2) What actions, if any, are required?

A. (1) Low voltage on system causing an undervoltage trip of 125V DC Bus feeder breakers;
(2) Initiate maintenance to troubleshoot and correct issue.

B. (1) Nothing will be detected, ungrounded systems can withstand multiple grounds with no adverse effects;
(2) No actions are required.

C. (1) Loss of a 125V DC Battery Charger;
(2) Place the Reserve Battery Charger in service.

D. (1) High current flow, caused by the second ground, can cause fuses to blow or protective devices to actuate;
(2) Initiate maintenance to troubleshoot and correct issue.

Answer: D

Answer Explanation:

A. Incorrect - DC loads are protected by fused disconnects with fuse ratings that would protect against a battery drain of sufficient magnitude to lower DC Bus voltage.

B. Incorrect - Ungrounded systems can withstand multiple grounds on the same phase with no affect, the question stem gives a positive and a negative ground. Maintenance is required to eliminate the grounds.

C. Incorrect - Depending on the location of the grounds it would be possible to cause a battery charger to trip off, however, the reserve battery charger would not/could not be placed in-service to replace the tripped one.

D. Correct - A second ground on the opposite polarity creates a current-to-ground flowpath. Uncertainty of operation if second ground occurs (component actuation)

Question 47 Info			
Topic:	Effect of a ground on an ungrounded system		
Tier/Group:	2/1		
K/A Info:	063 DC Electrical Distribution System • A2 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.01 Grounds		
RO Importance:	2.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• System Description 002, 125V DC Distribution System • Ground Training PPt		
Comments:	None		

48 **ID: Q93080** **Points: 1.00**

Unit-1 is operating at 100% power with a normal electrical alignment. In response to a plant transient, the Reactor Trip pushbuttons at 1C05 have been depressed, per EOP-0, Post Trip Immediate Actions. The RO reports the reactor remains at 100% power and all trip breakers remain closed.

Which **ONE** of the following sets of actions will mitigate this condition?

A. Open 11A 480V BUS FDR;
Open 12A 480V BUS FDR.

B. Open 11A 480V BUS FDR;
Open 14A 480V BUS FDR.

C. Open 12A 480V BUS FDR;
Open 13A 480V BUS FDR.

D. Open 13A 480V BUS FDR;
Open 14A 480V BUS FDR.

Answer: C

Answer Explanation:

A. Incorrect –11 and 12 CEDM MG Sets are powered from 480V Load Centers 12A and 13A. EOP-0, Post Trip Immediate Actions direct deenergizing these busses to trip the reactor if the trip pushbuttons do not work.

B. Incorrect - 11 and 12 CEDM MG Sets are powered from 480V Load Centers 12A and 13A. EOP-0, Post Trip Immediate Actions direct deenergizing these busses to trip the reactor if the trip pushbuttons do not work.

C. Correct - 11 and 12 CEDM MG Sets are powered from 480V Load Centers 12A and 13A. EOP-0, Post Trip Immediate Actions direct deenergizing these busses to trip the reactor if the trip pushbuttons do not work.

D. Incorrect - 11 and 12 CEDM MG Sets are powered from 480V Load Centers 12A and 13A. EOP-0, Post Trip Immediate Actions direct deenergizing these busses to trip the reactor if the trip pushbuttons do not work.

Question 48 Info			
Topic:	480V Breaker operation in the Control Room		
Tier/Group:	2/1		
K/A Info:	62 - AC Electrical Distribution System • A4 - Ability to manually operate and/or monitor in the control room: • A4.01 - All breakers (including available switchyard)		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	None		

49 **ID: Q26551** **Points: 1.00**

Given the following:

- RCS Tcold is 530 °F and constant
- RCS Pressure is 1550 PSIA and lowering slowly
- Pressurizer Level is 75 inches and lowering slowly
- Containment Rad Monitors are Clear
- Condenser Off-Gas Alarm has actuated

Which **ONE** of these indications can differentiate the event in progress as a S/G tube leak?

A. Containment Rad Monitors alarms being clear.

B. RCS T_{COLD} is normal and not lowering.

C. RCS subcooling is slowly lowering.

D. Receipt of the Condenser Off-gas alarm.

Answer: D

Answer Explanation:

A. Incorrect - absence of Containment Radiation monitor alarms does nothing to confirm a S/G tube leak.

B. Incorrect - normal Tcold does not confirm a S/G tube leak.

C. Incorrect - loss of subcooling can be indicative of a LOCA and is not unique to S/G tube leaks.

D. Correct - EOP-6 Technical Basis document step IV.J

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 49 Info			
Topic:	Indication of a SGTR vice LOCA		
Tier/Group:	2/1		
K/A Info:	073 Process Radiation Monitoring (PRM) System • 2.4.18 Knowledge of the specific bases for EOPs.		
RO Importance:	3.3		
Proposed references to be provided to applicant:	Steam Tables		
Learning Objective:	SRO-201-6-1-01		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last used – LOI 2008 AOP / EOP Exam (April, 2010)		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	None		

50 **ID: Q92631** **Points: 1.00**

Unit-2 is operating at 100% power. ESFAS Logic cabinet "BL" has been deenergized to support emergent maintenance.

What effect, if any, will this condition have on the Reactor Protective System and/or Main Turbine Trips?

A. A Reactor trip will **NOT** cause a Turbine trip.

B. The RPS / Turbine trip interface will function normally.

C. 2 out of 4 RPS trips on Loss of Load will cause a Turbine trip.

D. Turbine trip logic is reduced to 2 out of 2 Reactor Trip Bus U/V relay actuations.

Answer: A

Answer Explanation:

A. Correct - Per OI-34, Engineered Safety Features Actuation System, Appendix "A": De-energization of the BL Actuation Logic Cabinet renders the Reactor Trip Bus UV turbine trips inoperable.

B. Incorrect - Per OI-34, Engineered Safety Features Actuation System, Appendix "A": De-energization of the BL Actuation Logic Cabinet renders the Reactor Trip Bus UV turbine trips inoperable.

C. Incorrect - The reactor will trip on a 2/4 logic caused by a turbine trip.

D. Incorrect - Per OI-34, Engineered Safety Features Actuation System, Appendix "A": De-energization of the BL Actuation Logic Cabinet renders the Reactor Trip Bus UV turbine trips inoperable.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 50 Info			
Topic:	RPS / TG		
Tier/Group:	2/1		
K/A Info:	012 - System 012 Reactor Protection System • K3 - Knowledge of the effect that a loss or malfunction of the RPS will have on the following: • K3.02 - T/G		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	OI-34, Engineered Safety Features Actuation System		
Comments:	None		

51 **ID: Q93040** **Points: 1.00**

The RCS is in a solid water condition, in preparation for drawing a Pressurizer bubble per OP-7, Shutdown Operations, with the following conditions:

- An RCS overpressure condition occurred
- Power Operated Relief Valves have lifted
- A "SDC PRESS HI" alarm has been received

The cause of the high pressure condition has been corrected and the overpressure condition no longer exists.

Which **ONE** of the following actions is required per the controlling procedure?

A. Manually close the Power Operated Relief Valves.

B. Manually close the Power Operated Relief Valve Block Valves.

C. Check the Power Operated Relief Valves automatically closed.

D. Check the SDC Suction Isolation Valves automatically closed.

Answer: A

Answer Explanation:

A. Correct – When in LTOP conditions, PORVs must be manually closed using the OVERRIDE TO CLOSE handswitch, once opened due to an over-pressure condition, per OP-7.

B. Incorrect – Although plausible as an action for terminating discharge via the PORVs, closing the block valves is not a prescribed action for recovering from an over-pressure condition while in the LTOP mode.

C. Incorrect – PORVs do not automatically close when in LTOP conditions; PORVs must be manually closed using the OVERRIDE TO CLOSE handswitch, once opened due to an over-pressure condition, per OP-7.

D. Incorrect – SDC Suction Isolation Valves are manually closed in response to an over-pressure condition, per Alarm Manual 1C06.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 51 Info			
Topic:	Actions for an overpressure condition when drawing a bubble in the Pressurizer		
Tier/Group:	2/1		
K/A Info:	010 Pressurizer Pressure Control System • 2.2.2 - Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.		
RO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	OP-7-1		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	OP-7, Shutdown Operations		
Comments:	None		

52 **ID: Q92312** **Points: 1.00**

Which of the following sets represents **BOTH** an available indication **AND** control capability during a SBO on Unit-1?

A. RVLMS;
ADVs cannot be operated from 1C43.

B. CEA Mimic;
ADVs cannot be operated from 1C03.

C. RVLMS;
ADVs cannot be operated from 1C03.

D. CEA Mimic;
ADVs cannot be operated from 1C43.

Answer: C

Answer Explanation:

A. Incorrect – RVLMS (PAMS) Channels "A" & "B" are powered from 120V Vital Instrument Busses 11 (1Y01) and 12 (1Y02). They will remain energized. 1Y01 and 1Y02 are powered, via inverters, from 125V DC Busses 11 and 21. 1Y09 is deenergized during an SBO resulting in a loss of control to the Atmospheric Dump Valves. Local control, at 1C43, <u>is</u> established to operate the ADVs to control RCS temperature.

B. Incorrect - The loss of 1Y09, during an SBO, results in a loss of the CEA Mimic. The loss of 1Y09 also results in a loss of control to the Atmospheric Dump Valves. The valves will operate on a quick open signal only. Local control at 1C43 is established to operate the ADVs.

C. Correct - RVLMS (PAMS) Channels "A" & "B" are powered from 120V Vital Instrument Busses 11 (1Y01) and 12 (1Y02) and will remain energized. 1Y01 and 1Y02 are powered, via inverters, from 125V DC Busses 11 and 21. 1Y09 is deenergized during an SBO resulting in a loss of control to the Atmospheric Dump Valves. Local control, at 1C43, is established to operate the ADVs to control RCS temperature.

D. Incorrect - The loss of 1Y09, during an SBO, results in a loss of the CEA Mimic. The loss of 1Y09 also results in a loss of control to the Atmospheric Dump Valves. The valves will operate on a quick open signal only. Local control at 1C43 <u>is</u> established to operate the ADVs.

Question 52 Info			
Topic:	DC powered loads available during a SBO		
Tier/Group:	1/1		
K/A Info:	055 Loss of Offsite and Onsite Power (Station Blackout) • EA2 Ability to determine or interpret the following as they apply to a Station Blackout: • EA2.04 Instruments and controls operable with only dc battery power available		
RO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-002-1-2		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-7, Station Blackout		
Comments:	None		

53 **ID: Q20053** **Points: 1.00**

Wide Range Nuclear Instrumentation (WRNI) Channel "A" experiences a loss of power.

Which **ONE** of the following describes the impact to RPS Channel "A"?

A. SUR trip is enabled.

B. Zero Power Mode Bypass is enabled.

C. CEAPDS PDIL is inhibited.

D. TM/LP signal to CWP is inhibited.

Answer: A

Answer Explanation:

A. Correct – The Flux Trip 2 relay fails to >E-4% on a **loss of power**, enabling SUR trip.

B. Incorrect – The Flux Trip 1 relay fails to >E-4% on a **loss of power**, removing the Zero Power Mode Bypass.

C. Incorrect - The Flux Trip 1 relay fails to >E-4% on a **loss of power**, enabling the CEAPDS PDIL.

D. Incorrect – The Flux Trip 1 relay fails to >E-4% on a **loss of power**, enabling the TM/LP signal to CWP.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 53 Info			
Topic:	Effects on loss of power to the W.R. flux trip relays 1 & 2		
Tier/Group:	1/2		
K/A Info:	032 Loss of Source Range Nuclear Instrumentation • AK2. Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: • AK2.01 Power supplies, including proper switch positions		
RO Importance:	2.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-57-1-5-09		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 Nuclear Instrumentation Exam (May, 2009)		
Technical references:	SD-078A, Nuclear Instrumentation		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

54 **ID: Q19395** **Points: 1.00**

During a feed system auto transfer from low to high power, reactor power reaches 19% before the FRV Bypass valve signal reaches 40%.

Which of the following will occur?

A. FRV position freezes with FBV controlling.

B. FBV position freezes, FRV controls, and the FBV must be manually driven shut.

C. The transfer is completed with feed system in High power mode.

D. The transfer is completed with feed system in Low power mode.

Answer: C

Answer Explanation:

A. Incorrect - FRV would only "freeze" if there were a Transfer Inhibit Signal present. Also, in the High Power Mode the FRV is controlling S/G Level.

B. Incorrect - FRV Bypass would only "freeze" if there were a Transfer Inhibit Signal present. Also, in the High Power Mode the FRV is controlling S/G Level. FRV Bypass is only manually driven shut when performing a Manual Transfer.

C. Correct - System shifts to High Power Mode between 17 & 19%

D. Incorrect - System will be in High Power Mode at 19%, shifts to Low Power between 15 & 13% (Transfer is forced to completion at 13%)

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 54 Info			
Topic:	DFWCS transfer from low to high power		
Tier/Group:	2/2		
K/A Info:	035 Steam Generator System (SGS) • A3 Ability to monitor automatic operation of the S/G including: • A3.01 S/G water level control		
RO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:	LO-045E-1-1		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – 2004 LOR Quiz		
Technical references:	• OI-12A, Feedwater System • SD-045A, Main Feedwater System Description		
Comments:	None		

55 **ID: Q92830** **Points: 1.00**

Given the following conditions:

- Unit-2 is operating at 100% power
- A Loss of Offsite Power occurs coincident with a LOCA
- The 2B DG fails to start

Which **ONE** of the following groups of components will operate in response to the stated conditions?

A. 21 Containment Air Cooler;
22 Charging Pump;
21 Component Cooling Water Pump

B. 23 Charging Pump;
21 Containment Filter;
22 Containment Air Cooler

C. 23 Containment Filter;
23 Containment Air Cooler;
21 Component Cooling Water Pump

D. 21 Charging Pump;
22 Containment Filter;
23 Component Cooling Water Pump

Answer: B

Answer Explanation:

A. Incorrect - 22 Charging Pump is aligned to 480V Bus 24

B. Correct - These loads are normally aligned to 21 480V bus and would receive start signals as the LOCI Sequencer went through its progression

C. Incorrect - 23 CAC is aligned to 480V Bus 24

D. Incorrect - 22 IRU is aligned to 480V Bus 24

Question 55 Info			
Topic:	U-2 IRU Power Supplies		
Tier/Group:	2/2		
K/A Info:	027 - Containment Iodine Removal System (CIRS) • K2 Knowledge of bus power supplies to the following: • K2.01 Fans		
RO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-7-1-5-85		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	OI-27D-2 Station Power 480 Volt System Breaker Lineup		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

56 **ID: Q14531** **Points: 1.00**

Unit-2 is operating normally at 80% reactor power when a Letdown line leak occurs immediately upstream of the Containment penetration.

Which set of the following automatic features could actuate to promptly terminate this event?

1. High Regenerative HX outlet temperature
2. Chemical Volume Control Isolation Signal (CVCIS)
3. Containment Isolation Signal (CIS)
4. Excess Flow Check Valve shuts

A. 1, 2

B. 3, 4

C. 1, 4

D. 2, 3

Answer: C

Answer Explanation:

A. Incorrect - CVCIS would not actuate on a Letdown line break in the Containment.

B. Incorrect - CIS does not promptly provide a shut signal to the Letdown Stops to terminate the event.

C. Correct - Per 2C07-ALM, 2-CVC-515-CV will automatically close on a High Regenerative Heat Exchanger outlet temperature of 470 °F and the Excess Flow Check valve will shut at ~ 200 GPM to isolate a break..

D. Incorrect - CIS does not promptly provide a shut signal to the Letdown Stops to terminate the event.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 56 Info			
Topic:	Plant response to a L/D line break in the West Pen Rm		
Tier/Group:	1/2		
K/A Info:	Combustion Engineering A16 Excess RCS Leakage • AK2. Knowledge of the interrelations between the (Excess RCS Leakage) and the following: •AK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Panel Exam		
Technical references:	• AOP-2A, Excessive Reactor Coolant Leakage • 1C08-ALM, ESFAS 11 Alarm Manual (Windows G-17 & G-18) • 1C07-ALM, Chemical and Volume Control Alarm Manual (Window F-01)		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

57 **ID: Q92350** **Points: 1.00**

Unit-2 is at 30% power MOC when a load rejection occurs. RCS pressure rises to 2420 PSIA, resulting in a reactor trip. The following conditions exist:

- Acoustic Monitor indicates flow through a PORV
- RCS pressure is 2185 PSIA with a lowering trend
- Pressurizer level is 180 inches with a rising trend

Which **ONE** of the following lists actions directed by EOP-0 for regaining control of Pressurizer level and pressure?

A. Place PORV Override handswitches in the "Override To Close" position; Start all available Charging Pumps.

B. Close PORV block MOVs; Lower RCS Pressure to less than 1800 PSIA.

C. Lower RCS Pressure to less than 1800 PSIA; Start all available Charging pumps.

D. Shut PORV Block valves; Place PORV Override handswitches in the "Override To Close" position.

Answer: D

Answer Explanation:

A. Incorrect – Starting all available Charging Pumps with Pressurizer level at 180 inches would help offset the inventory loss due to the leaking PORV but would be a deviation to EOP-0, Post Trip Immediate Actions.

B. Incorrect – Lowering RCS pressure to 1800 PSIA would be a deviation to EOP-0, Post Trip Immediate Actions. Plausible because this is an action, directed by EOP-5 Loss of Coolant Accident, designed to reseat a leaking Pressurizer Safety valve. OP-1, Plant Startup from Cold Shutdown, also contains a step to soak the RCS at ~ 1900 PSIA to ensure proper operation of the Pressurizer Safety valves.

C. Incorrect - Starting all available Charging Pumps with Pressurizer level at 180 inches would help offset the inventory loss due to the leaking PORV but would be a deviation to EOP-0, Post Trip Immediate Actions.

D. Correct - With pressure lowering due to PORV leakage, the PORV block MOV must be verified closed, and the PORV Override HS must be placed in “Override to Close”.

Question 57 Info			
Topic:	Controlling PORV Leakage		
Tier/Group:	1/1		
K/A Info:	008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) • AA1. Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: •AA1.06 Control of PZR level		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-058-1-01		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	Modified version of Q19362		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

58 **ID: Q92351** **Points: 1.00**

With the Unit operating at 100%, which of the conditions would **ONLY** trip the RPS Channel "A" trip units listed below:

- Thermal Margin / Low Pressure
- Axial Power Distribution

A. Loss of a Channel "A" Linear Range Nuclear Instrumentation subchannel (fails to zero).

B. Loss of the Channel "A" Wide Range Nuclear Instrumentation channel HV power supply.

C. Loss of a single Channel "A" T_{COLD} input (fails low).

D. Loss of a single Channel "A" T_{HOT} input (fails high).

Answer: A

Answer Explanation:

A. Correct - Loss of a Channel "A" LRNI sub channel (upper or lower) would cause indicated NI power to go to approximately 50% resulting in trips on Trip Units 7 (TM/LP) and 10 (APD) (because the calculated ASI is extremely high).

B. Incorrect - Loss of the WRNI HV power supply, while causing alarms and abnormal indications, would not cause actuation of any trip units

C. Incorrect - The T_{COLD} inputs to the RPS channel are auctioneered high. Loss of a single Tcold measurement channel would not cause actuation of any trip units

D. Incorrect - The T_{HOT} inputs to the RPS channel are averaged. A single Thot measurement channel, failing high, will result in an indicated DeltaT power of approximately 180%. This will cause Trip Units 1 (Hi Pwr), 7 (TM/LP), & 10 (APD) to trip in addition to causing multiple alarms.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 58 Info			
Topic:	LRNI Subchannel failure		
Tier/Group:	2/2		
K/A Info:	015 - Nuclear Instrumentation System • K6 - Knowledge of the effect of a loss or malfunction on the following will have on the NIS: • K6.01 - Sensors, detectors, and indicators		
RO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-78A-1-2		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• 1C05-ALM, Reactivity Control Alarm Manual • 1C06-ALM, RCS Control Alarm Manual • SD-058, Reactor Protective System		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

59 **ID: Q93090** **Points: 1.00**

Unit-2 is in Mode 2 with a reactor startup in progress. Chemistry reports an unexpected decrease of 30 ppm RCS boron concentration. AOP-1A, Inadvertent Boron Dilution, has been implemented.

Which **ONE** of the following system alignments will result in sustained boration of the RCS at greater than or equal to 40 GPM?

A. Open SI TO CHG HDR, 2-CVC-269-MOV;
Shut REGENERATIVE HX CHARGING INLET, 2-CVC-183;
Open AUX HPSI HDR, 2-SI-617-MOV;
Start 21 Charging Pump.

B. Open BA DIRECT M/U, 2-CVC-514-MOV;
Open 22B LOOP CHARGING, 2-CVC-518-CV;
Start 21 Charging Pump;
Start 21 BA Pump.

C. Open 21 BAST GRAVITY FD, 2-CVC-508-CV;
Shut RWT CHG PP SUCT, 2-CVC-504-MOV,
Open 22B LOOP CHARGING, 2-CVC-518-CV;
Start 21 Charging Pump.

D. Open 21 RWT OUT, 2-SI-4143-MOV;
Open HPSI HDR XCONN, 2-SI-653-MOV;
Open MAIN HPSI HDR, 2-SI-616-MOV;
Start 23 HPSI Pump.

Answer: B

Answer Explanation:

A. Incorrect – The line-up described is a flowpath from the VCT to the RCS. Additional component manipulation would be required to establish boration.

B. Correct - The line-up described establishes a boration flowpath from the BAST to the RCS.

C. Incorrect - The line-up described would not establish a boration flowpath. The VCT outlet MOV must be closed to borate of the RCS.

D. Incorrect - The line-up described would not establish a boration flowpath. RCS normal operating pressure is well above the discharge head of the HPSI Pump.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 59 Info			
Topic:	AOP-1A, preferred boration methodology		
Tier/Group:	2/1		
K/A Info:	004 Chemical and Volume Control System (CVCS) • A4 Ability to manually operate and/or monitor in the control room: • A4.10 Boric acid pumps		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR 202-1A1B-S-07		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	AOP-1A, Inadvertent Boron Dilution (Att 1)		
Comments:	None		

60 **ID: Q92372** **Points: 1.00**

Unit-1 was operating at 100% power when a LOCA occurred. The following conditions exist:

- 12 HPSI Pump was OOS prior to the event
- RCS pressure is 1400 PSIA and slowly lowering
- Containment Pressure is 1.8 PSIG and slowly rising
- EAST ECCS PP RM LVL HI alarm has annunciated on 1C10.
- The ABO reports water level in the East ECCS Pp Room is approximately 10 inches and rising and the source appears to be in the area of the LPSI pump.
- 11 RWT LVL / TEMP Alarm has annunciated on 1C09

(1) What actions must be taken to address these conditions and;
(2) What impact will these actions have on the performance of the Emergency Core Cooling System?

A. (1) Place both LPSI pump, both HPSI pump and both Containment Spray Pump handswitches in PTL and shut both RWT Outlets;
(2) ECCS flow is lost, S/G heat removal remains sufficient.

B. (1) Place 11 LPSI pump, 11 HPSI pump, and 11 Containment Spray Pump handswitches in PTL and shut the associated RWT Outlet;
(2) ECCS flow is reduced to approximately one-half, heat removal capability is inadequate.

C. ((1) Place both LPSI pump, both HPSI pump and both Containment Spray Pump handswitches in PTL and shut both RWT Outlets;
(2) ECCS flow is lost, S/G heat removal capability is inadequate.

D. (1) Place 11 LPSI pump, 11 HPSI pump, 11 Containment Spray Pump handswitches in PTL and shut the associated RWT Outlet;
(2) ECCS flow is reduced to approximately one-half, heat removal capability remains sufficient.

Answer: D

Answer Explanation:

A. Incorrect – Level provided in stem for ECCS Pump room indicates RWT still has sufficient level to support unaffected train. Securing ALL ECCS Pumps would be a wrong choice.

B. Incorrect - Heat removal capability of one SI train meets design criteria.

C. Incorrect – See “A” justification. Based on given conditions, S/G heat removal is adequate.

D. Correct - With given indications the leak is from the RWT (low level alarm with RCS pressure still above pump shutoff head and Containment pressure below CSAS actuation). Pumps taking suction from the affected RWT suct hdr need to be secured to prevent damage. RWT outlet needs to be shut, to isolate the leak.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 60 Info			
Topic:	Loss of ECCS flowpath		
Tier/Group:	2/1		
K/A Info:	006 Emergency Core Cooling System (ECCS) • A2 Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.02 Loss of flow path		
RO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	1C10-ALM, ESFAS 13 Alarm Manual		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

61 **ID: Q14512** **Points: 1.00**

You are performing the duties of the Refueling Control Room Operator (RCRO).

- A core shuffle is in progress
- A series of core-to-core fuel moves is being performed
- The Spent Fuel Handling Machine operator is performing a series of steps, moving **ONLY** new fuel, to set up for later portions of the core load sequence

Which **ONE** of the listed conditions would require core alterations be suspended?

A. Containment Purge is placed in service.

B. Audible count rate is lost in the Control Room.

C. Spent Fuel Pool Ventilation Charcoal filter is bypassed.

D. One of 3 available WRNI channels is declared out of service.

Answer: B

Answer Explanation:

A. Incorrect – Placing Containment Purge does not require suspension of Core Alterations. Plausible because there are conditions (operation of purge with Containment Radiation Monitors inoperable) under which purge operation could cause suspension of Core Alts.

B. Correct – The RCRO verifies audible count rate in the Containment and the Control Room as part of the RCRO turnover process in accordance with NO-1-200, Control of Shift Activities. His responsibilities include monitoring for reactivity changes in the Control Room which is accomplished via monitoring of audible count rate and observation of the two required WRNI channels .

C. Incorrect – The Spent Fuel Pool Ventilation Charcoal filter is only required in service to support movement of recently irradiated fuel assemblies in the Auxiliary Building. The stem of the question clearly states "only" new fuel is being moved in the Spent Fuel Pool Area.

D. Incorrect - The RCRO verifies at least two WRNI channels operable as part of the RCRO turnover process in accordance with NO-1-200, Control of Shift Activities. Tech Specs require 2 source range (WRNI) channels operable during Core Alts..

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 61 Info			
Topic:	Neutron monitoring during refueling evolutions		
Tier/Group:	2/2		
K/A Info:	034 Fuel Handling Equipment System (FHES) • A4 Ability to manually operate and/or monitor in the control room: • A4.02 Neutron levels		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	NO-1-200, Control of Shift Activities, Attachment 26		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

62 **ID: Q28783** **Points: 1.00**

Which of the following is directed by EOP-0, Post Trip Immediate Actions, to prevent an uncontrolled cooldown, in the event of an uncomplicated reactor and turbine trip on Unit-2?

A. Depress the "Reset" button on the MSR control panel.

B. Ensure MSR 2nd Stage Steam Source MOVs shut.

C. Shut Upstream Drain MOVs.

D. Trip the S/G Feed Pumps.

Answer: A

Answer Explanation:

A. Correct - Per U-2 EOP-0 step D.3 basis

B. Incorrect - This verification is directed when performing EOP-0, Post Trip Immediate Actions, on Unit-1.

C. Incorrect - There is no direction to shut Upstream drain valves in EOP-0 and leaking drain valves will have a small effect on RCS temperature immediately after a trip. This is a mitigating strategy in EOP-1 for controlling cooldown.

D. Incorrect – This is an EOP-0 mitigating action for excessive feeding of the Steam Generators, not for controlling cooldown.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 62 Info			
Topic:	Preventing an uncontrolled cooldown on a U-2 Rx trip		
Tier/Group:	2/2		
K/A Info:	045 Main Turbine Generator (MT/G) System • A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: • A1.06 Expected response of secondary plant parameters following T/G trip		
RO Importance:	3.3		
Proposed references to be provided to applicant:	None		
Learning Objective:	SRO-201-0/10.0		
10 CFR Part 55 Content:	55.41(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 OP, AOP-3B, EOP-0 & EOP-1 Exam (Nov, 2009)		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	None		

63 **ID: Q92410** **Points: 1.00**

With both Units operating at 100% power a sustained loss of Spent Fuel Pool Cooling occurs. Which **ONE** of the following actions is taken per the appropriate Abnormal Operating Procedure?

A. Place a second Control Room H & V fan in operation.

B. Place a second Spent Fuel Pool Exhaust fan in operation.

C. Place Unit-1 Shutdown Cooling in service on the SFP.

D. Place the Spent Fuel Pool Charcoal Filters in service.

Answer: D

Answer Explanation:

A. Incorrect – Per OI-22F, the system is **NOT** to be operated with two supply fans running simultaneously.

B. Incorrect - Per AOP-6F, Section VIII, Step A.7.d, Maintain a negative pressure in the Fuel Handling Area by checking that **ONE** of the SFP EXH FANs is running.

C. Incorrect – While OI-3B does have a procedure section to align SDC to the SFP, the prerequisite for doing so is Unit-1 is defueled.

D. Correct - Per AOP-6F, Section VIII, Step A.7.e, Place SFP Charcoal Filters in service. Applicants are expected to recognize need for ventilation filtration in the event of a sustained loss of SFP Cooling.

Question 63 Info			
Topic:	Sustained loss of SFP Clg impact on ventilation systems		
Tier/Group:	2/2		
K/A Info:	033 Spent Fuel Pool Cooling System (SFPCS) • K3 Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: • K3.01 Area ventilation systems		
RO Importance:	2.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	AOP-6F, Spent Fuel Pool Cooling System Malfunctions		
Comments:	None		

64 **ID: Q14479** **Points: 1.00**

At what RCS Cold Leg temperature **must** MPT protection be enabled per Technical Specifications?

	Unit-1	Unit-2
A.	Less than or equal to 369 °F	Less than or equal to 306 °F
B.	Less than or equal to 365 °F	Less than or equal to 301 °F
C.	Less than or equal to 306 °F	Less than or equal to 369 °F
D.	Less than or equal to 301 °F	Less than or equal to 365 °F

Answer: B

Answer Explanation:

A. Incorrect – These values represent the alarm setpoints for enabling (at setpoint and lowering) or disabling (at setpoint and rising) MPT Relief Protection. They are correct for their respective Units.

B. Correct - These values represent the correct values, per T.S. 3.4.12, Low Temperature Overpressure Protection (LTOP) System, at which MPT Relief Protection must be enabled. They are correct for their respective Units. This question matches the K/A as follows: the design feature for LTOP MPT Enable is for the operator to recognize that the required plant conditions are met (operator equivalent of a sensing switch), and then manually enable the protection circuit through the use of keyswitches.

C. Incorrect - These values represent the alarm setpoints for enabling (at setpoint and lowering) or disabling (at setpoint and rising) MPT Relief Protection. Unit-1 and Unit-2 values are reversed.

D. Incorrect - These values represent the correct values, per T.S. 3.4.12, Low Temperature Overpressure Protection (LTOP) System, at which MPT Relief Protection must be enabled. Unit-1 and Unit-2 values are reversed.

Question 64 Info			
Topic:	RCS Overpressure Protection		
Tier/Group:	2/2		
K/A Info:	002 Reactor Coolant System (RCS) • K4 Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: • K4.10 Overpressure protection.		
RO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	LOI Panel Comp (April, 2009)		
Technical references:	OP-5, Plant Shutdown From Hot Standby To Cold Shutdown		
Comments:	None		

65 **ID: Q93100** **Points: 1.00**

Unit-1 is operating in Mode 1. Per the associated Alarm Response Procedures, which **ONE** of the following conditions **REQUIRES** implementation of an Abnormal Operating Procedure?

A. "QUENCH TK •TEMP •LVL •PRESS" annunciates; Safety Injection System Recirc line relief lifts.

B. "12B RCP SEAL •TEMP HI •PRESS" annunciates; 12B Reactor Coolant Pump Upper Seal indicates failed.

C. "LIQUID WASTE DISCH" annunciates; A Liquid Waste Discharge terminates due to high activity.

D. "NON-ESSENTIAL •4KV •13KV MOTOR OVERLOAD" annunciates; One of the running Condensate Booster Pumps trips.

Answer: D

Answer Explanation:

A. Incorrect – RCS Control Alarm Manual, ALM-1C06 contains guidance for evaluating and responding to off-normal Quench Tank parameters, none of which include implementation of an AOP.

B. Incorrect – RCS Control Alarm Manual, ALM-1C06 contains criteria for diagnosing seal failure(s) and corresponding actions, none of which include implementation of an AOP.

C. Incorrect - RMS Alarm Manual, ALM-1C22 specifies verification that discharge valves automatically terminate the release. Implementation of an AOP is not required unless the discharge valves fail to terminate the release.

D. Correct - Condensate & Feedwater Control Alarm Manual, 1(2) C03-ALM, refers the user to AOP-3G, Malfunction of Main Feedwater System, which has actions to be performed. As a minimum, the operator would ensure the standby Condensate Booster Pump automatically started and was not affected by a common mode failure.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 65 Info			
Topic:	General AOP entry criteria		
Tier/Group:	Generic K & A		
K/A Info:	2.4 Emergency Procedures / Plan • 2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.		
RO Importance:	4.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Panel Exam		
Technical references:	2C03-ALM; AOP-3G, Malfunction of Main Feedwater System		
Comments:	None		

66 **ID: 93050** **Points: 1.00**

Which **ONE** of the following is **NOT** an expectation, during EOP-0 implementation, per NO-1-201, Calvert Cliffs Operating Manual?

A. Promptly tying underlying instrument busses or MCCs per appropriate controlling procedures as a parallel action.

B. Valves or pumps not operating upon receipt of an automatic signal may be locally operated to properly position the valve or operate the pump.

C. Draining of the containment sump should be coordinated with the STA to ensure appropriate leakrate data is obtained.

D. If responding to an ATWS, the RO opens the four blue-handled breakers on 2C17 without reference to the procedure.

Answer: C

Answer Explanation:

A. Incorrect – Per NO-1-201, Calvert Cliffs Operating Manual, Attachment 9, EOP/ERPIP Implementation Expectations, it is permissible to promptly tie underlying instrument busses or MCCs as a parallel action.

B. Incorrect – Per NO-1-201, Calvert Cliffs Operating Manual, Attachment 9, EOP/ERPIP Implementation Expectations, valves or pumps not operating upon receipt of an automatic signal may be locally operated to properly position the valve or operate the pump.

C. Correct - Per NO-1-201, Calvert Cliffs Operating Manual, Attachment 9, EOP/ERPIP Implementation Expectations, Do not drain the containment sump during EOP-0, this should be coordinated after the post-EOP-0 procedure is implemented.

D. Incorrect - Per NO-1-201, Calvert Cliffs Operating Manual, Attachment 9, EOP/ERPIP Implementation Expectations, If responding to an ATWS in EOP-0, the RO is expected to immediately open the four blue-handled breakers on 1C17(2C17). This should be done without reference to the EOP-0 plaque.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 66 Info			
Topic:	NO-1-201 Guidelines for implementation of EOPs		
Tier/Group:	Generic K & A		
K/A Info:	2.4 Emergency Procedures / Plan • 2.4.14 Knowledge of general guidelines for EOP usage.		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOI-201-8-8		
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use - LOI 2006 Audit Exam		
Technical references:	NO-1-201, Calvert Cliffs Operating Manual		
Comments:	None		

67 **ID: Q51252** **Points: 1.00**

Unit-1 just completed a Refueling Outage:

- Reactor power is 30% and holding for required testing
- No CEA motion or boration/dilution operations are in progress
- TBV Controller, 1-PIC-4056, is in auto and the setpoint is set at 900 PSIA
- Turbine Bypass Valve, 1-MS-3944-CV, has failed open

What actions are taken to stabilize the plant and Reactor power per AOP-7K, Overcooling Event?

A. Maintain turbine load constant and isolate the TBV to restore T_{COLD} to program; Withdraw CEAs, as necessary, to maintain Reactor power.

B. Lower turbine load to restore T_{COLD} to program; Withdraw CEAs, as necessary, to maintain Reactor power.

C. Maintain turbine load constant and isolate the TBV to restore T_{COLD} to program; Insert CEAs, as necessary, to return Reactor power to the required value.

D. Lower turbine load to restore T_{COLD} to program; Insert CEAs, as necessary, to return Reactor power to the required value.

Answer: B

Answer Explanation:

A. Incorrect – Turbine load would be adjusted to bring T_{COLD} on program.

B. Correct - Per AOP-7K, Overcooling Event in Mode 1 or Two, CEAs should be withdrawn, as necessary, to maintain reactor power if early in core cycle and the overcooling event has been compensated for by adjusting turbine load.

C. Incorrect - Turbine load would be adjusted to bring T_{COLD} on program and CEAs should be withdrawn, as necessary, to maintain reactor power if early in core cycle and the overcooling event has been compensated for by adjusting turbine load.

D. Incorrect - Unit-1 would have a positive MTC given the conditions stated in the stem. CEAs should be withdrawn, as necessary, to maintain reactor power if early in core cycle and the overcooling event has been compensated for by adjusting turbine load.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 67 Info			
Topic:	Initial response to an overcooling event		
Tier/Group:	Generic K & A		
K/A Info:	2.4 - Emergency procedures / Plan • 2.4.11 - Knowledge of abnormal condition procedures.		
RO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Comprehensive Exam (May, 2008)		
Technical references:	AOP-7K, Overcooling Event in Mode 1 or Two		
Comments:	None		

68 **ID: Q19306** **Points: 1.00**

Unit-2 is in Mode 1 and the latest leakage reports are:

- 7.6 GPM - Pressurizer safety valve leakage
- 1.8 GPM - leakage past check valves from the RCS to the SI system
- 0.1 GPM - 21 Steam Generator primary-to-secondary leakage
- 10.6 GPM - total leakage

Which of the following Technical Specification leakage limits are exceeded?

A. Pressure Boundary leakage and Identified leakage

B. Primary to Secondary leakage and Pressure Boundary leakage

C. Primary to Secondary leakage and Unidentified leakage

D. Identified leakage and Unidentified leakage

Answer: C

Answer Explanation:

A. Incorrect - Tech Specs define Pressure Boundary leakage as "LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall". No Pressure Boundary leakage exists. Identified leakage of 9.5 GPM is within the T.S. limit of 10 GPM.

B. Incorrect - 21 S/G Primary to secondary leakage (0.1 GPM x 60 x 24 = 144 GPD) exceeds the T.S. limit of 100 GPD; however no pressure boundary leakage exists. Tech Specs define Pressure Boundary leakage as "LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall".

C. Correct - 21 S/G Primary to secondary leakage (0.1 GPM x 60 x 24 = 144 GPD) exceeds the T.S. limit of 100 GPD. Total leakage of 10.6 GPM minus Identified leakage of 9.5 GPM = 1.1 GPM which exceeds the T.S. limit of 1 GPM unidentified leakage.

D. Incorrect - Total leakage of 10.6 GPM minus Identified leakage of 9.5 GPM = 1.1 GPM which exceeds the T.S. limit of 1 GPM unidentified leakage, however, identified leakage of 9.5 GPM is within the T.S. limit of 10 GPM.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 68 Info			
Topic:	T.S. RCS Leakage - S/G Tube leak & unidentified		
Tier/Group:	1/2		
K/A Info:	037 Steam Generator (S/G) Tube Leak • 2.2.40 Ability to apply Technical Specifications for a system.		
RO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No history of previous use		
Technical references:	T.S. 3.4.13, RCS Operational Leakage; T.S. 1.1, Definitions		
Comments:	None		

69 **ID: Q50731** **Points: 1.00**

A Fire Protection System actuation occurs as evidenced by Control Room annunciation and reports from the field. The appropriate response procedure is implemented and the response team is fully manned.

Which **ONE** of the following is an Operations Technical Advisor (OTA) responsibility at the scene of the fire in accordance with SA-1-101, Fire Fighting?

A. Determine the appropriate fire fighting strategy for plant conditions.

B. Report status of conditions in the area to the Control Room.

C. Make potential EAL declaration recommendations to Shift Manager.

D. Advise the Fire Brigade Leader on use of fire fighting agents.

Answer: B

Answer Explanation:

A. Incorrect - This is a responsibility of the Fire Brigade Leader as defined in SA-1-101, FIRE FIGHTING

B. Correct - This is a responsibility of the Operations Technical Advisor as defined in SA-1-101, FIRE FIGHTING

C. Incorrect - This is a not a specific responsibility of the OTA. The OTA may or may not be an SRO. This is generally the responsibility of the Control Room Supervisor and /or Shift Technical Advisor (STA) with the STA providing a peer-check for any EAL declarations the SM might make.

D. Incorrect - This is a responsibility of the Fire Marshal, if present, as defined in SA-1-101, FIRE FIGHTING

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 69 Info			
Topic:	Operations Technical Advisor responsibilities		
Tier/Group:	2/2		
K/A Info:	086 Fire Protection System (FPS) • 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.		
RO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41 (b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 ESFAS Exam (August, 2009)		
Technical references:	SA-1-101, Fire Fighting		
Comments:	None		

70 **ID: Q54957** **Points: 1.00**

Unit-2 tripped 3 minutes ago. The following conditions exist:

- A loss of Offsite Power has occurred.
- 21 AFW Pump is Tagged out
- 22 AFW Pump tripped on AFAS Actuation
- S/G levels are at (-) 220 inches and lowering
- Pressurizer Level is 95 inches and lowering
- T_{COLD} is 515 ˚F and lowering
- RCS subcooling is 40 ˚F and slowly lowering
- **ONLY** the 1B and 2A EDGs started and loaded
- Containment pressure is 2.2 PSIG and rising

Which of the following is the appropriate Emergency Operating Procedure to mitigate this event upon completion of EOP-0, Post Trip Immediate Actions?

A. EOP-5, Loss Of Coolant Accident.

B. EOP-3, Loss of All Feedwater.

C. EOP-8, Functional Recovery Procedure.

D. EOP-4, Excess Steam Demand Event.

Answer: C

Answer Explanation:

A. Incorrect – A LOCA is occurring making selection of EOP-5 plausible. In addition to the LOCA a Loss of All Feedwater is occurring. A single event diagnosis is not possible requiring implementation of EOP-8.

B. Incorrect – A Loss of All Feedwater is occurring making selection of EOP-3 plausible. In addition to the Loss of All Feedwater a LOCA is occurring. A single event diagnosis is not possible requiring implementation of EOP-8.

C. Correct - Loss of Feed and a LOCA are occurring. No Main or Aux Feed available due to LOOP and loss of AFW flow. A single event diagnosis is not possible requiring implementation of EOP-8.

D. Incorrect – Listed indications could represent an Excess Steam Demand. However, a LOCA and a LOAF are also occurring necessitating implementation of EOP-8 because a single event diagnosis is not possible.

Question 70 Info			
Topic:	Use plant conditions to select the appropriate procedure		
Tier/Group:	1/2		
K/A Info:	CE/E09 – Functional Recovery • EA2 - Ability to determine and interpret the following as they apply to the (Functional Recovery): • EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations.		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2006 Audit Exam		
Technical references:	EOP-0, Post Trip Immediate Actions EOP-8, Functional Recovery Procedure		
Comments:	None		

71 **ID: Q15947** **Points: 1.00**

Which **ONE** of the following choices contains conditions, **ALL** of which require declaring a Fairbanks-Morse diesel generator inoperable? All conditions need not occur simultaneously.

A. Starting air pressure 220 PSIG;
SRW CV manual hand wheel engaged;
Voltage regulator in MANUAL.

B. 120VAC Vital Bus 11 inverter in INV 2;
Voltage regulator in MANUAL;
LOCAL-REMOTE keyswitch in LOCAL.

C. ESFAS test handswitch in NORMAL;
Starting Air pressure 215 PSIG;
Diesel Room Ventilation Fan handswitch in AUTO.

D. Fuel Oil Transfer pump in STOP;
SRW PDIC in MANUAL;
Jacket Cooling Water Temperature 80 °F.

Answer: D

Answer Explanation:

A. Incorrect - Starting Air Receiver pressure is in the normal range and well above the alarm setpoint of 125 PSIG

B. Incorrect - Having the Inverter selector switch in INV 2 does not inop the DG

C. Incorrect - None of the conditions presented will inop the DG

D. Correct - Per OI-21A, Fairbanks Morse DG shall be considered inoperable for any of the following:
- 1B DG Voltage Regulator is selected to MANUAL.
- The 1B DG Room Ventilation Fan is inoperable.
- 1-SRW-1588-PDIC is **NOT** in AUTOMATIC or 1-SRW-1588-CV Manual Hand wheel is engaged.
- 1B DG Fuel Oil Transfer Pump is inoperable.
- 1B DG Jacket Water System temperature is less than 90°F.

Question 71 Info			
Topic:	Conditions that result in the DG being declared OOS		
Tier/Group:	Generic K & A		
K/A Info:	2.2 - Equipment Control • 2.2.37 Ability to determine operability and/or availability of safety related equipment.		
RO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-48-1-2-12		
10 CFR Part 55 Content:	55.41(b)(7)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	Last use – LOI 2008 Diesel Generators Exam (May, 2009)		
Technical references:	OI-21B, 1B Diesel Generator		
Comments:	Improved version of Bank question Q24997 (not modified)		

72 ID: **Q92451** **Points: 1.00**

Unit-1 was operating at 100% power when a Loss of Offsite Power (LOOP) and Steam Generator Tube Rupture (SGTR) occurred. Given the following events and conditions:

- The operators implemented the appropriate Optimal Recovery procedure
- The affected S/G has been identified
- T_{HOT} is 516 °F (slowly lowering)

Why does the optimal recovery procedure direct cooldown to T_{HOT} less than 515°F?

A. Minimizes the differential pressure across the break thereby reducing the leakrate.

B. Establishes natural circulation cooling as soon as possible during the event.

C. Minimizes radiation release to the environment via the affected S/G Main Steam Safety valves.

D. Prevents dilution of the RCS by maintaining S/G pressure lower than RCS pressure.

Answer: C

Answer Explanation:

A. Incorrect - DP across the break would increase as a result of the cooldown unless RCS pressure was lowered simultaneously.

B. Incorrect - A cooldown to 515 °F is not necessary to establish natural circulation conditions

C. Correct - Per the EOP-6 Technical Basis document: The initial cooldown is done prior to isolating the affected S/G. This action reduces the risk of challenging the steam generator safety valves of the affected S/G after it is isolated.

D. Incorrect - Flow from the S/G to the RCS is not a concern. In fact, backflow from the S/G to the RCS is an available method for controlling affected S/G level.

Question 72 Info			
Topic:	Basis for cooldown to < 515 °F prior to isolating affected S/G		
Tier/Group:	Generic K & A		
K/A Info:	2.3 Radiation Control • 2.3.11 Ability to control radiation releases.		
RO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(11)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-6, Steam Generator Tube Rupture Technical Basis Document		
Comments:	None		

73 ID: **Q17948** **Points: 1.00**

The following emergency situations may warrant an individual dose in excess of the established regulatory limit of 5 REM per year:

- Life Saving (voluntary)
- Facility Protection

Which **ONE** of the following represents the limits for dose accumulated by an individual during these emergency situations?

A. Greater than 25 REM for Lifesaving, no upper limit (voluntary);
10 REM for Facility Protection.

B. Greater than 10 REM, not to exceed 25 REM, for Lifesaving (voluntary);
10 REM for Facility Protection.

C. Greater than 25 REM, not to exceed 75 REM, for Lifesaving (voluntary);
25 REM for Facility Protection.

D. Greater than 25 REM for Lifesaving, no upper limit (voluntary);
25 REM for Facility Protection.

Answer: A

Answer Explanation:

A. Correct – Dose limits specified are those outlined in ERPIP 831, Emergency Radiation Exposure Guidance.

B. Incorrect - Dose limits specified in ERPIP 831are: Greater than 25 REM for Lifesaving (voluntary), 10 REM for Facility Protection, and 25 REM for Lifesaving (assigned).

A. Incorrect - Dose limits specified in ERPIP 831are: Greater than 25 REM for Lifesaving (voluntary), 10 REM for Facility Protection, and 25 REM for Lifesaving (assigned).

B. Incorrect - Dose limits specified in ERPIP 831are: Greater than 25 REM for Lifesaving (voluntary), 10 REM for Facility Protection, and 25 REM for Lifesaving (assigned).

Question 73 Info			
Topic:	Emergency dose limits		
Tier/Group:	Generic K & A		
K/A Info:	2.3 Radiation Control • 2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions.		
RO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(12)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No record of previous use		
Technical references:	ERPIP 831, Emergency Radiation Exposure Guidance.		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

74 **ID: Q92570** **Points: 1.00**

What limitations, if any, does NO-1-201, Calvert Cliffs Operating Manual, place on the use of "Working Copies" of technical procedures?

Working copies: .

A. Must be verified current prior to use on subsequent shifts.

B. Must be verified current at least once every twenty four hours.

C. Must **NOT** be used for evolutions lasting longer than one shift.

D. Must complete Procedure Working Copy Coversheet prior to use.

Answer: A

Answer Explanation:

A. Correct - Per No-1-201 Section 5.1.D.2.E.2 (Working Copies) specifies; Evolutions lasting greater than one shift do not require an Attach 7 as long as the procedure user verifies the current Working Copy is still the current approved revision prior to using the procedure at the beginning of the next shift. Otherwise, the procedure user shall complete an Attach 7 for the Working Copy generated. Matches K/A because the ability to verify use of controlled procedures includes knowledge of when a procedure must be verified as a controlled version.

B. Incorrect - Per No-1-201 Section 5.1.D.2.E.2 (Working Copies), Procedure users are responsible for verifying current revision of procedures, if an Attach 7 is not used. Plausible because Attachment 7, Procedure Working Copy Coversheet, if used, must be placed in the PDU basket in the Control Room

C. Incorrect - Per No-1-201 Section 5.1.D.2.E.2 (Working Copies), procedure users are responsible for verifying current revision of procedures, if an Attach 7 is not used.

D. Incorrect - Per No-1-201 Section 5.1.D.2.E.2 (Working Copies), procedure users are responsible for verifying current revision of procedures, if an Attach 7 is <u>**not**</u> used.

Deleted

Q#74 deleted as resolution for post-exam comment.

EXAMINATION ANSWER KEY

LOI 2010 NRC RO Exam

Question 74 Info			
Topic:	Use of NO-1-201 Working Copy Attachment		
Tier/Group:	Generic K & A		
K/A Info:	2.1.21 Ability to verify the controlled procedure copy.		
RO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	NO-1-201, Calvert Cliffs Operating Manual		
Comments:	None		

75 **ID: Q26059** **Points: 1.00**

For each of the post-trip plant conditions listed in Column "A", match the actions required, associated with RCP operation, from Column "B" in accordance with the applicable controlling procedure. Assume ALL RCPs are initially operating and RCS T_{COLD} is 530 °F for each condition listed. (Choices in Column B may be used once, more than once, or not all)

Column A (Plant conditions)	Column B (Actions Required for the RCPs)
1. A. LOCA with RCS pressure at 1700 PSIA	1. No action required
	2. Trip Two RCPs (one in each loop)
2. B. CNTMT pressure is 5.0 PSIG	3. Trip Three RCPs
3. C. SGTR with RCS pressure at 1475 PSIA	4. Trip All Four RCPs
	5. Trip Two RCPs (in the same loop)
4. D. No source of Feed Flow is available	

A. 2, 2, 5, 4

B. 1, 2, 2, 1

C. 2, 4, 2, 4

D. 1, 4, 5, 1

Answer: C

Answer Explanation:

A. Incorrect – CCW flow would be automatically isolated to the Containment with the stated conditions requiring all four RCPs be secured. SGTR RCP operation strategy is same as EOP-O and stated RCS pressure is well above the minimum pump operating limits.

B. Incorrect – RCS pressure dropping to < 1725 PSIA requires implementation of the Trip Two/Leave Two strategy ending up with One RCP in each loop.

C. Correct - RCS pressure dropping to < 1725 PSIA requires implementation of the Trip Two/Leave Two strategy ending up with One RCP in each loop. CCW flow would be automatically isolated to the Containment with the stated conditions requiring all four RCPs be secured. SGTR RCP operation strategy is same as the LOCA strategy and stated RCS pressure is well above the minimum pump operating limits. Loss of all feed flow requires tripping all RCPs to eliminate their heat input to the RCS.

D. Incorrect - RCS pressure dropping to < 1725 PSIA requires implementation of the Trip Two/Leave Two strategy ending up with One RCP in each loop..

Question 75 Info			
Topic:	Interpret procedure guidance for RCP operation & take appropriate action		
Tier/Group:	Generic K & A		
K/A Info:	2.1.20 Ability to interpret and execute procedure steps.		
RO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.41(b)(10)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No record of use on an NRC exam		
Exam Bank History:	No record of previous use		
Technical references:	• EOP-0, Post Trip Immediate Actions • EOP-3, Loss of All Feedwater • EOP-5, Loss of Coolant Accident		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

76 **ID: Q92771** **Points: 1.00**

Resin transfer from 21 CVCS IX to the SRMT is in progress. U-2 Waste Processing Ventilation RMS (2-RI-5410) begins to rise. The RMS is in alarm at 700 CPM and steady.

Similar trends are noted on U-2 WRNGM (2-RIC-5415), now reading 4700 µci/sec and U-2 Main Vent Gaseous (2-RI-5415), reading 20,000 CPM. Neither 2-RIC-5415 nor 2-RI-5415 has reached its alarm setpoint.

Based on these conditions, which of the following describes the required actions in accordance with the appropriate controlling procedure?

A. Verify HP coverage per RWP.

B. Secure Air Sparging of the SRMT.

C. Declare a Radiological Event.

D. Initiate a Reportability Notification.

Answer: C

Answer Explanation:

A. Incorrect - AOP-6C, Accidental Gaseous Waste Release, would be implemented for the elevated RMS readings. Included in the AOP is the direction to involve Radiation Safety but not to verify compliance with requirements of the RWP

B. Incorrect – Action is directed by OI-17A, Solid Waste, but Air Sparging would not be in progress at this point.

C. Correct – AOP-6C, Accidental Gaseous Waste Release, would be implemented for the elevated RMS readings. Included in the AOP is the direction to declare a Radiological Event, as a minimum. Additionally, the criteria for declaring a Radiological Event in ERPIP 3.0 Attachment (19) would be met for an unplanned RMS in alarm indicating significantly different conditions from normal resin transfers

D. Incorrect- A radioactive release is not reportable based on CNG-NL-101-1004. Only releases that exceed Part 20, Table 2, Column 1 limits would need to be submitted as a 60-day LER. Both the WRNGM and the Main Vent RMS are not in alarm, indicating a regulatory limit has not yet been exceeded.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 76 Info			
Topic:	Determine the appropriate actions for a Waste Gas leak		
Tier/Group:	Generic K & A		
K/A Info:	2.3 – Radiation Control • 2.3.14 - Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.		
SRO Importance:	3.8		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(4)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	AOP-6C, Accidental Gaseous Waste Release		
Comments:	Modified version of Q74607		

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

77 **ID: Q50857** **Points: 1.00**

Unit-1 is operating at 100% power with P-13000-2 feeding 13KV Service Bus 11.

A Unit-1 Reactor trip occurs due to 11A RCP experiencing a locked rotor. Immediately thereafter, P-13000-2 deenergizes due to a fault and a steam leak occurs in the turbine building. The crew has implemented EOP-0. The following conditions exist:

- 11 SG level is -80 inches and slowly rising
- 12 SG level is -120 inches and slowly rising
- RCS pressure is 1875 PSIA and rising
- PZR level is 95 inches and rising
- MSIVs are closed per EOP-0 Alternate Action steps
- 1B DG did not start

Based on existing plant conditions, which **ONE** of the following is the correct procedure to implement?

A. EOP-1, Reactor Trip

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

C. EOP-6, Steam Generator Tube Rupture

D. EOP-8, Functional Recovery Procedure

Answer: A

Answer Explanation:

A. Correct - Based on the information given, a reactor trip has occurred due to low RCS flow. Per the EOP-0 Technical Basis Document the HR safety function is met when "at least one RCP is checked to be operating in a loop with an S/G available for heat removal". EOP-1 would be implemented since all safety functions are met.

B. Incorrect - EOP-2 is implemented during a loss of all forced circulation. Since both Loop 12 RCPs are still operating, natural circulation does not exist and EOP-2 is not desired. Plausible due to loss of P-13000-2.

C. Incorrect – Plausible because the candidate may associate the S/G level mismatch with a S/G Tube Leak when in fact the S/G level mismatch is due to the pump configuration of one RCP in Loop 11 and two RCPs in Loop 12. EOP-6 identifies S/G level mismatch as one of the ways to identify the ruptured generator.

D. Incorrect – EOP-8, Functional Recovery Procedure would be implemented if single event diagnosis were not possible. Information given supports diagnosis of an uncomplicated Reactor Trip making selection of EOP-1, Reactor Trip, appropriate.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 77 Info			
Topic:	EOP Transition with 11A secured		
Tier/Group:	1/1		
K/A Info:	CE/E02 - Reactor Trip Recovery • EA2 - Ability to determine and interpret the following as they apply to the (Reactor Trip Recovery) • EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations.		
SRO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	LESSON PLAN 202-2AS-08		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:	Used by LOR during 2008, Session III (average score – 97% for 36 student encounters)		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	None		

78 **ID: Q26669** **Points: 1.00**

Given the following plant conditions:

- A reactor trip has occurred
- All CEAs are inserted with reactor power lowering
- RCS pressure is 1900 PSIA and lowering
- Pzr Level is 140 inches and lowering
- RCS T_{COLD} is 512°F and lowering
- RCS Subcooling is 118 °F and rising slowly
- 11 S/G Pressure is 700 PSIA and lowering
- 12 S/G Pressure is 830 PSIA and steady
- 11 S/G Level is -180 inches and lowering
- 12 S/G Level is -70 inches and rising with AFW feeding 12 S/G
- 11 4KV bus is energized
- 14 4KV bus is deenergized

Based on the information provided, which **ONE** of the following is the correct Optimal Recovery Procedure for this event?

A. EOP-1, Reactor Trip

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

C. EOP-4, Excess Steam Demand Event

D. EOP-5, Loss of Coolant Accident

Answer: C

Answer Explanation:

A. Incorrect - Information provided (Core and RCS Heat Removal Safety Function not met) makes it clear something more than an uncomplicated trip has occurred.

B. Incorrect - Information provided does not support a LOOP or Natural Circulation condition.

C. Correct - An Excess Steam Demand Event is indicated by the S/G differential pressure and the high subcooled margin value.

D. Incorrect - Subcooled Margin is well in excess of the values expected for a LOCA condition.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 78 Info			
Topic:	Given conditions determine the optimal recovery procedure		
Tier/Group:	1/1		
K/A Info:	CE/E05 - Excess Steam Demand • EA2 - Ability to determine and interpret the following as they apply to the (Excess Steam Demand) • EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations.		
SRO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:	No history of previous use		
Technical references:	EOP-0, Post Trip Immediate Actions		
Comments:	None		

79 **ID: Q92473** **Points: 1.00**

Both units are operating at 100% power. Due to voltage regulator concerns on U-1, the generator is operating with a 1.0 Power Factor. Additionally, STP O-8A-1 is in progress with the 1A DG paralleled to its respective 4KV bus and has been at full load for 30 minutes.

A system event occurs resulting in a "11 SRW HDR PRESS LO" and "U-1 4KV ESF MOTOR OVERLOAD" alarms. 11 SRW header pressure indicates 30 PSIG and steady. The appropriate procedure has been implemented.

The following conditions exist:

- Main Turbine Thrust Bearing Metal temperature is 193 °F and slowly rising
- Main Turbine Journal Bearing Metal temperature is 225 °F and slowly rising
- Generator Hydrogen temperature is 50 °C and slowly rising

What action(s) should you, as the CRS, direct be taken for the event?

A. Shutdown the 1A DG.

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

C. Reduce MVAR load to "0" to reduce Main Transformer heat loads.

D. Reduce MVAR load, as necessary, to maintain generator temperature.

Answer: B

Answer Explanation:

A. Incorrect - The 1A DG is cooled by a self-contained cooling system, so is unaffected.

B. Correct - Per AOP-7B Section V.A.1 exceeding the Main Turbine Thrust Bearing metal temperature limit of 190 °F is criteria for tripping the reactor and implementing EOP-0. AOP-7B specifies "with the approval of the SM/CRS" for tripping the reactor and implementation of EOP-0.

C. Incorrect - MVARS are required to be reduced zero to "reduce Main Generator Heating". With the generator operating with a 1.0 Power Factor, there is no reactive load being carried by the machine. There is no need to lower MVARs since they are already zero.

D. Incorrect - MVARS are required to be reduced zero to "reduce Main Generator Heating" with power reduced as required to maintain Main Generator temperatures. With the generator operating with a 1.0 Power Factor, there is no reactive load being carried by the machine. There is no need to lower MVARs since they are already zero.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 79 Info			
Topic:	Actions necessary on a loss of 11 SRW header		
Tier/Group:	1/1		
K/A Info:	062 - Loss of Nuclear Service Water • AA2 - Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: • AA2.04 - The normal values and upper limits for the temperatures of the components cooled by SWS		
SRO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	202-7-S-05		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☒ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	AOP-7B, LOSS OF SERVICE WATER		
Comments:	Modified version of Q39867		

80 **ID: Q92790** **Points: 1.00**

Unit-1 has just been shutdown to Mode 3 at NOP/NOT. Unit- 2 is operating at 100% power.

A fault occurs, isolating the Red Bus.

Which **ONE** of the following describes a correct procedure selection and strategy?

A. On Unit-2, complete EOP-0, Post Trip Immediate Actions then implement EOP-2, Loss of Offsite Power/Loss of Forced Circulation;
Manually control ADVs, from 2C03, to establish an RCS heat sink.

B. On Unit-1, implement AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power;
Tie 1Y09 to 1Y10.

C. On Unit-1, implement AOP-3E, Loss of All RCP Flow, Modes 3, 4, or 5;
Use TBVs to maintain T_{COLD} between 525 °F and 535 °F.

D. On Unit-2, complete EOP-0, Post Trip Immediate Actions, then implement EOP-1, Reactor Trip;
Use TBVs or ADVs to maintain T_{COLD} between 525 °F and 535 °F.

Answer: A

Answer Explanation:

A. Correct – EOP-2 is implemented due to the loss of forced circulation. TBVs will not be available, only ADVs will be available for heat removal.

B. Incorrect – Tying 1Y09 to 1Y10 is an action for loss of 11 4KV Bus. 11 4KV Bus remains energized.

C. Incorrect – RCPs will remain running and AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power will be implemented for the loss of 14 4KV Bus.

D. Incorrect - EOP-2 is implemented due to the loss of forced circulation. TBVs will not be available, only ADVs will be available for heat removal. EOP-1 Safety Functions were not all met.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 80 Info			
Topic:	RCS Heat Removal status		
Tier/Group:	1/2		
K/A Info:	CE/A13 - Natural Circulation Operations • AA2 - Ability to determine and interpret the following as they apply to the (Natural Circulation Operations) • AA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.		
SRO Importance:	3.7		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-2, LOSS OF OFFSITE POWER / LOSS OF FORCED CIRCULATION		
Comments:	None		

81 **ID: Q92474** **Points: 1.00**

Given both units operating at full power, which **ONE** of the following conditions results in the shortest duration Technical Specification Limiting Condition for Operation Completion Time and is the required action?

A. Unit-1 CNTMT PAL inner door seal leakage exceeds the T.S. limit; Verify the outer door is closed.

B. Unit-1 RCS leak rate unidentified leakage is above the T.S. limit: Reduce LEAKAGE to within limits.

C. Unit-2 CNTMT avg temperature is steady at 121°F; Reduce CNTMT avg temperature to less than or equal to 120°F.

D. Unit-2 BL ESFAS Logic Cabinet is removed from service; Restore affected Logic channel to OPERABLE status.

Answer: A

Answer Explanation:

A. Correct - The Unit-1 CNTMT PAL inner door seal leakage is covered by T.S. 3.6.2. Action "A.1" requires "Verify the OPERABLE door is closed in the affected air lock" with a completion time of 1 hour.

B. Incorrect - RCS Operational Leakage is governed by T.S. 3.4.13. Action "A" requires "Reduce RCS Leakage to within limits" with a completion time of 4 hours.

C. Incorrect - Containment Air Temperature is governed by T.S. 3.6.5.Action "A" requires "Restore containment average air temperature to within limit" with a completion time of 8 hours.

D. Incorrect - The ESFAS System Logic Cabinet is governed by T.S. 3.3.5. Action "C" requires "Restore affected Manual Actuation channel and Actuation Logic channel to OPERABLE status" with a completion time of 48 hours.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 81 Info			
Topic:	Loss of Cntmt Integrity / 1 Hour Tech Specs		
Tier/Group:	1/2		
K/A Info:	069 - Loss of Containment Integrity • 2.2.39 - Knowledge of less than or equal to one hour Technical Specification action statements for systems.		
SRO Importance:	4.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	Tech Spec Sections: 3.4, RCS; 3.5, ECCS & 3.6, Containment Systems		
Comments:	None		

82 **ID: Q51174** **Points: 1.00**

Given the following:

- A major transient occurred, resulting in an automatic reactor trip and SIAS
- EOP-5, Loss of Coolant Accident, has been entered
- RCS pressure is 1550 PSIA and lowering slowly
- RCS temperature is 515 °F and stable

Five minutes later, the following conditions are observed:

- SG 11 pressure is 450 PSIA and lowering
- RCS temperature is 440 °F and lowering
- RCS pressure is 1350 PSIA and lowering

Which **ONE** of the following describes the correct strategy for the current plant conditions?

A. Remain in EOP-5, Loss of Coolant Accident. Refer to EOP-4, Excess Steam Demand Event, for actions required to isolate 11 S/G and terminate the RCS cooldown.

B. Transition to EOP-4, Excess Steam Demand Event, to isolate the SG 11 and stabilize RCS temperature.

C. Implement EOP-8, Functional Recovery Procedure, and isolate 11 S/G by use of the appropriate Core and RCS Heat Removal Success Path.

D. Implement EOP-8, Functional Recovery Procedure, and isolate 11 S/G by use of the appropriate RCS Pressure and Inventory Control Success Path.

Answer: C

Answer Explanation:

A. Incorrect - Conditions stated (multiple events in progress) are entry criteria for the Functional Recovery Procedure which will correctly assess and prioritize actions to address jeopardized safety functions. EOP-8 will provide the actions required to address both the LOCA and the ESDE.

B. Incorrect - Conditions stated (multiple events in progress) are entry criteria for the Functional Recovery Procedure which will correctly assess and prioritize actions to address jeopardized safety functions. Transitioning to EOP-4 will not address the in-progress LOCA.

C. Correct - Conditions stated (multiple events in progress) are entry criteria for the Functional Recovery Procedure which will correctly assess and prioritize actions to address jeopardized safety functions. The appropriate Core & RCS Heat Removal success path will provide direction for this event.

D. Incorrect - Conditions stated (multiple events in progress) are entry criteria for the Functional Recovery Procedure which will correctly assess and prioritize actions to address jeopardized safety functions. The appropriate <u>Core & RCS Heat Removal</u> success path will provide direction for this event.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 82 Info			
Topic:	Given plant conditions recognize the success paths and order of their priority.		
Tier/Group:	1/2		
K/A Info:	CE/E09 - Functional Recovery • EA2 - Ability to determine and interpret the following as they apply to the (Functional Recovery) • EA2.2 - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.		
SRO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:	No history of previous use		
Technical references:	NO-1-201, CALVERT CLIFFS OPERATING MANUAL; EOP-8, Functional Recovery Procedure		
Comments:	None		

83 **ID: Q92490** **Points: 1.00**

Unit-2 was operating at 100% power when an event occurred. The following conditions exist 10 minutes into the event:

- RCS pressure is 37 PSIA
- Pressurizer level is 0 inches
- CETs indicate 265 °F
- S/G levels are -40 inches and rising slowly
- S/G pressures are 900 PSIA and steady
- Containment pressure is 12 PSIG and slowly rising
- RWT level is 28 feet and lowering

45 minutes into the event, you are giving another Transient Brief for the EOP in use. Which **ONE** of the following is the primary heat removal strategy to brief with the crew?

A. Steam Generators with AFW and ADVs

B. LPSI flow, from the RWT

C. Containment Spray flow, through the Shutdown Cooling Heat Exchanger

D. HPSI flow, from the Containment Sump

Answer: D

Answer Explanation:

A. Incorrect – Given plant conditions, a LOCA is in progress. EOP, Loss of Coolant Accident; directs that the SGs be cooled to below RCS pressure, but this is not the primary heat removal method.

B. Incorrect - Given plant conditions, a LOCA is in progress. Based on RWT trend, the RWT is lowering at ~1'/min (Initial level of 38' and level at 28' in 10 mins). With low RCS pressure, SI flow will not significantly vary as time continues. At 45 mins, the RWT should be empty and RAS actuated. This will trip the LPSI pumps and they will not be available for heat removal.

C. Incorrect - Given plant conditions, a LOCA is in progress. EOP, Loss of Coolant Accident; does not direct the alignment of CS pumps through the SDC HX. CS pumps are verified in operation, but their function is not to provide the primary heat removal method, but rather to minimize containment pressure.

D. Correct - Given plant conditions, a LOCA is in progress. EOP, Loss of Coolant Accident; directs that the HPSI pumps be aligned to the containment sump once RAS has actuated. Based on RWT trend, the RWT is lowering at ~1'/min (Initial level of 38' and level at 28' in 10 mins). With low RCS pressure, SI flow will not significantly vary as time continues. At 45 mins, the RWT should be empty and RAS actuated.

EXAMINATION ANSWER KEY

Question 83 Info			
Topic:	HPSI Pump cavitation question for SRO		
Tier/Group:	1/1		
K/A Info:	011 - Large Break LOCA ▪ 2.2.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.		
SRO Importance:	4.4		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-033480602-002		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-5, Loss of Coolant Accident		
Comments:	None		

84 **ID: Q92510** **Points: 1.00**

Unit-1 was operating at 100% power when Instrument Air (IA) header pressure began lowering due to a rupture of the IA header in the turbine building. IA header pressure continued to lower and has stabilized at 35 PSIG as read on 1C13. All systems operated as designed and Operator actions, if needed, were taken.

Which **ONE** of the following describes the appropriate controlling procedure and necessary actions to mitigate the event?

A. OP-3, Normal Power Operation, and isolate the Turbine Bypass Valves to prevent an excessive cooldown.

B. AOP-3G, Malfunction of Main Feedwater System, and pin the Feedwater Regulating Valve to maintain SG levels.

C. AOP-7D, Loss of Instrument Air, and close both Steam Generator Feed Pump Miniflow manual isolation valves.

D. EOP-0, Post Trip Immediate Actions, initiate Auxiliary Feedwater Water, and operate ADVs.

Answer: D

Answer Explanation:

A. Incorrect - The TBVs are not isolated during a loss of IA as the valves due to fail open.

B. Incorrect – The FRVs are not pinned when IA pressure lowers to 35 PSIG as the unit is tripped. SG levels are maintained by taking EOP-0 actions to isolate MFW and initiate AFW.

C. Incorrect – AOP-7D is the correct procedure that is implemented immediately as IA pressure is lowering. However, once IA pressure reaches 50 PSIG, AOP-7D directs that the unit be tripped and EOP-0 be implemented.

D. Correct - AOP-7D is the correct procedure that is implemented immediately as IA pressure is lowering. However, once IA pressure reaches 50 PSIG, AOP-7D directs that the unit be tripped and EOP-0 be implemented. In EOP-0, alternate actions are required for Core and RCS Heat Removal since MFW is excessive due to FRV valves failing as is or lost as IA impacts various high level dumps, requiring initiation of AFW. ADVs are available since the stem indicates that actions were taken as IA pressure lowered, which includes starting the SWACs. This would provide IA supply to the ADVs.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 84 Info			
Topic:	ADVs supplied by SWACs		
Tier/Group:	2/2		
K/A Info:	041 - Steam Dump System (SDS)/Turbine Bypass Control • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: • A2.03 - Loss of IAS		
SRO Importance:	3.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• AOP-7D, Loss of Instrument Air • EOP-0, Post Trip Immediate Actions		
Comments:	None		

85 **ID: 40688** **Points: 1.00**

Unit-1 is at 100% power, EOC, when the following occur:

- Reactor power promptly lowers to 92% and continues to slowly lower
- PZR Pressure simultaneously lowered to 2200 PSIA
- RCS T_{COLD} has dropped to 541 ˚F
- No CVCS operations are in progress

Of the provided options:

1. Which of the following procedures would address this set of plant conditions, and;
2. Which of the actions is required, by the selected procedure?

A. (1) AOP-7K, Overcooling Event in Mode One or Two
(2) Adjust Turbine to restore T_{COLD} to program

B. (1) AOP-1B, CEA Malfunction
(2) Adjust Turbine to restore T_{COLD} to program

C. (1) AOP-7K, Overcooling Event in Mode One or Two
(2) Withdraw CEAs, as necessary, to restore T_{COLD} to program

D. (1) AOP-1B, CEA Malfunction
(2) Withdraw CEAs, as necessary, to restore T_{COLD} to program

Answer: B

Answer Explanation:

A. Incorrect – For the given plant conditions, boration, as allowed by the AOP, would be ineffective in restoring T_{COLD} to program given the initial conditions. Dilution operations are not directed by the procedure as a method of restoring T_{COLD} to program.

B. Correct –This action is directed by the AOP and would be effective in restoring T_{COLD} to program.

C. Incorrect – AOP-1B cautions "Do NOT use CEAs to control RCS temperature". Plausible because AOP-1B allows use of CEAs, to adjust power, during realignment of the dropped CEA.

D. Correct - For the given plant conditions, TBV operation, as allowed by the AOP, would be ineffective in restoring T_{COLD} to program given the initial conditions.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 85 Info			
Topic:	S/D CEA Alignment		
Tier/Group:	1/2		
K/A Info:	003 - Dropped Control Rod • 2.4.11 - Knowledge of abnormal condition procedures.		
SRO Importance:	4.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:	Last used in May, 2009 LOR quiz		
Technical references:	AOP-1B, CEA Malfunction		
Comments:	None		

86 **ID: Q38119** **Points: 1.00**

Using provided references

Given the following plant conditions:

- A lightning strike in the switchyard results in loss of all three high lines and a dual unit trip @ 1035.
- 1B DG failed to start
- The 0C DG was started @ 1039.
- SMECO is in a normal line-up.
- At 1051 the PPO reports they are ready to close the 0C disconnects to 14 4KV Bus.

What, if any, EAL classification is warranted for Unit-1?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. No EAL classification is warranted

Answer: B

Answer Explanation:

A. Incorrect - Per ERPIP 3.0, Attachment (1), Emergency Action Level Criteria, an Alert per H.A.2.1.2 is the appropriate call for AC power capability to 4KV Vital busses on either Unit being reduced to ONLY one DG for >15 minutes. An Unusual Event would be appropriate if both Vital 4 KV Busses were powered by their respective DGs. In this case only 11 4KV Bus was powered by its respective DG for a period of at least 16 minutes.

B. Correct – Per ERPIP 3.0, Attachment (1), Emergency Action Level Criteria, an Alert per H.A.2.1.2 is the appropriate call for AC power capability to 4KV Vital busses on either Unit being reduced to ONLY one DG for >15 minutes.

C. Incorrect - Per ERPIP 3.0, Attachment (1), Emergency Action Level Criteria, an Alert per H.A.2.1.2 is the appropriate call for AC power capability to 4KV Vital busses on either Unit being reduced to ONLY one DG for >15 minutes. To reach Site Area Emergency criteria, both 4KV Vital Busses would have to be deenergized for >15 minutes.

D. Incorrect - Per ERPIP 3.0, Attachment (1), Emergency Action Level Criteria, an Alert per H.A.2.1.2 is the appropriate call for AC power capability to 4KV Vital busses on either Unit being reduced to ONLY one DG for >15 minutes.

EXAMINATION ANSWER KEY

Question 86 Info			
Topic:	LOOP EAL Declaration		
Tier/Group:	1/1		
K/A Info:	055 - Loss of Offsite and Onsite Power (Station Blackout) • 2.4.40 - Knowledge of SRO responsibilities in emergency plan implementation.		
SRO Importance:	4.5		
Proposed references to be provided to applicant:	ERPIP 3.0, Attachment (1)		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	ERPIP 3.0		
Comments:	None		

87 **ID: Q92730** **Points: 1.00**

SFP Charcoal Filters have been declared inoperable. Fuel movement within the SFP is desired.

What is the **MINIMUM** time that the fuel to be moved must have been out of a critical reactor before fuel movement per OI-25A, Spent Fuel Handling Machine, may commence?

A. Greater than 100 hours.

B. Greater than 32 days.

C. Greater than 92 days.

D. Greater than 184 days.

Answer: B

Answer Explanation:

A. Incorrect - A candidate, unsure of the correct duration, may be familiar with 100 hours (minimum time shutdown before fuel movement) and consider this a reasonable choice as an answer.

B. Correct - As defined in the **Tech Spec Bases** for T.S. 3.7.11, Spent Fuel Pool Exhaust Ventilation System (SFPEVS). *"The SFPEVS is designed to mitigate the consequences of a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 32 days)".*

C. Incorrect - A candidate, unsure of the correct duration, may be familiar with 92 days (quarterly surveillance interval from the Tech Specs) and consider this a reasonable choice as an answer.

D. Incorrect - A candidate, unsure of the correct duration, may be familiar with 184 days (semi-annual surveillance interval from the Tech Specs) and consider this a reasonable choice as an answer.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 87 Info			
Topic:	Definition of Recently Irradiated Fuel		
Tier/Group:	Generic K & A		
K/A Info:	2.1 - Conduct of Operations • 2.1.42 - Knowledge of new and spent fuel movement procedures.		
SRO Importance:	3.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(7)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	Tech Spec 3.7.11		
Comments:	None		

88 **ID: Q44148** **Points: 1.00**

Why must Linear Heat Rate be maintained less than 22 KW/FT, as described in the **BASIS** for T.S. Safety Limit 2.1.1.2, Linear Heat Rate?

A. Limits fuel clad temperature to 2200 °F.

B. Prevents exceeding fuel centerline temperature limits.

C. Prevents exceeding DNBR limits.

D. Maintains Site Boundary dose within limits.

Answer: B

Answer Explanation:

A. Incorrect - Per the Technical Specification Bases for T.S. 3.2.1, Liner Heat Rate (LHR), "The limitation on the LHR ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F".

B. Correct - Per the Technical Specification Bases for T.S. 2.1.1, Reactor Core SLs, "the restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate below the level at which fuel centerline melting occurs". The Safety limit of 22 KW/FT is significantly less conservative than the COLR limit of 14.3 KW/FT.

C. Incorrect - Per the Technical Specification Bases for T.S. 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, "the limits placed on departure from nucleate boiling (DNB) related parameters ensure that these parameters will not be less conservative than were assumed in the analyses, and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed".

D. Incorrect - Per the Technical Specification Bases for T.S. 3.4.15, RCS Specific Activity, "the RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a SGTR accident".

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 88 Info			
Topic:	Basis for COLR LHR limit		
Tier/Group:	1/1		
K/A Info:	2.2 – Equipment Control • 2.2.38 - Knowledge of conditions and limitations in the facility license.		
SRO Importance:	4.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(1)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	Tech Spec 2.1.1, Reactor Core Safety Limits		
Comments:	None		

89 **ID: Q92614** **Points: 1.00**

A fire exists in the Unit-2 45' West Electrical Penetration Room. Which of the following lists documents that must be reviewed, per ERPIP 3.0, to assist the Fire Brigade in firefighting efforts?

A. AOP-11 Series;
Fire Strategies Manual;
Interactive Cable Analysis.

B. AOP-11 Series;
Interactive Cable Analysis;
ES-013, Loss of Power Effects /Load List.

C. AOP-9 Series;
Fire Strategies Manual;
Plant Area Fire Strategy Templates (Maps).

D. AOP-9 Series;
Plant Area Fire Strategy Templates (Maps);
ES-013, Loss of Power Effects /Load List.

Answer: C

Answer Explanation:

A. Incorrect - AOP-11 is not a series and is for Control Room Evacuation for non-fires. It is not included in the list of documents to review contained in ERPIP 3.0, Attachment 16.

B. Incorrect - AOP-11 is not a series and is for Control Room Evacuation for non-fires. AOP-11 is not included in the list of documents to review contained in ERPIP 3.0, Attachment 16. ES-013, Loss of Power Effects /Load List, while a good potential reference, is not included in the list of documents to review contained in ERPIP 3.0, Attachment 16.

C. Correct - All listed resources are listed in ERPIP 3.0 Attachment 16 and are available in the Control Room.

D. Incorrect - ES-013, Loss of Power Effects /Load List, while a good potential reference, is not included in the list of documents to review contained in ERPIP 3.0, Attachment 16.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 89 Info			
Topic:	Resources to assist the CR in firefighting efforts		
Tier/Group:	Generic K & A		
K/A Info:	2.4 - Emergency Procedures / Plan • 2.4.26 - Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage.		
SRO Importance:	3.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• SA-1-101, FIRE FIGHTING • ERPIP 3.0, Attachment (16), Fire in the Protected Area, ISFSI, or MPF		
Comments:	None		

90 **ID: Q93060** **Points: 1.00**

Unit-1 has implemented EOP-5, Loss of Coolant Accident, due to a small break LOCA inside the containment concurrent with a loss of offsite power. 14 4KV Bus failed to energize from its respective DG.

Which **ONE** of the following describes:
(1) The impact of these events on the Steam Generators and;
(2) The strategy for managing current plant conditions per EOP-5?

A. (1) The ADVs are **NOT** available remotely to support RCS heat removal;
(2) Operate 13 HPSI on 11 4KV Bus, to establish adequate heat removal.

B. (1) Condensate Booster Pumps are **NOT** available as a feed source;
(2) Operate 13 HPSI on 11 4KV Bus, to establish adequate heat removal.

C. (1) Motor Driven AFW Pump is **NOT** available to support RCS heat removal;
(2) Establish RCS heat removal via natural circulation.

D. (1) Main Feedwater Pumps are **NOT** available as a feed source;
(2) Establish RCS heat removal via natural circulation.

Answer: D

Answer Explanation:

A. Incorrect – The ADVs are available for heat removal. HPSI flow out the break maintains RCS inventory with heat removal via the S/Gs providing the ability to cooldown the RCS to SDC entry conditions.

B. Incorrect - Forced circulation is not available for RCS heat removal, however HPSI flow out the break maintains RCS inventory with heat removal via the S/Gs providing the ability to cooldown the RCS to SDC entry conditions. EOP-5 does not drive starting 13 HPSI pump if 11 HPSI pump starts and functions properly. No information is provided stating 11 HPSI pump is not operating correctly.

C. Incorrect – The motor driven AFW Pump is available. AFW and the ADVs are used to establish heat removal and cooldown the RCS to SDC entry conditions.

D. Correct - Main Feedwater Pumps are not available as a feed source, AFW and the ADVs are used to establish heat removal and cooldown the RCS to SDC entry conditions.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 90 Info			
Topic:	Small Break LOCA heat removal		
Tier/Group:	2/2		
K/A Info:	035 – Steam Generator • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.06 – Small Break LOCA		
SRO Importance:	4.6		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-5, Loss of Coolant Accident		
Comments:			

91 **ID: Q92651** **Points: 1.00**

Unit-1 is operating at 100% power with Group 5 CEAs at 131 inches when the pulse counting position indication system is lost due to a power supply malfunction. It has become apparent the TRM restoration time will not be met.

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

A. Initiate a Condition Report for a Reactivity Management event.

B. Contact Systems Engineering to complete a Functionality Assessment.

C. Initiate the Event Notification Worksheet for a Licensee Event Report.

D. Contact Generation Dispatcher to inform of plant status.

Answer: B

Answer Explanation:

A. Incorrect – Loss of the pulse counting position indication system does not classify as a Reactivity Management event per CNG-OP-3.01-1000, Reactivity Management.

B. Correct - For SSCs that are not expressly subject to Tech Specs and that are determined to be degraded, assess the reasonable expectation of functionality.

C. Incorrect - There are no criteria stated that meet the threshold for notification of a LER

D. Incorrect - There is not enough information given to ascertain whether a power reduction is imminent.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 91 Info			
Topic:	TRM requirements for OOS CEA Position Indication		
Tier/Group:	2/2		
K/A Info:	014 - Rod Position Indication System (RPIS) • 2.4.30 - Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.		
SRO Importance:	4.1		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	Technical Requirements Manual; CNG-OP-1.01-1002, Conduct of Operability Determination/Functionality Assessments		
Comments:	None		

92 **ID: Q92670** **Points: 1.00**

During a Reactor startup, with power at 1 x $10^{-4\%}$, a Turbine Bypass valve fails partially open. T_{COLD} **approaches** the Technical Specification limit.

Which **ONE** of the following:
(1) Is the basis for the T_{COLD} Tech Spec temperature limit and;
(2) Describes the correct procedure to address this event?

A. (1) Ensures operation within the bounds of the existing accident analyses;
(2) Enter the EOP for excess steam demand events.

B. (1) Minimizes the possibility of violating DNB limits;
(2) Enter the AOP for overcooling events.

C. (1) Ensures operation within the existing instrumentation ranges and accuracies;
(2) Enter the EOP for excess steam demand events.

D. (1) Ensures operation within the bounds of the existing accident analyses;
(2) Enter the AOP for overcooling events.

Answer: D

Answer Explanation:

A. Incorrect - The basis for the minimum temperature for criticality is correct T.S. Bases 3.4.2, however, the correct procedure to implement would be AOP-7K which would ultimately direct tripping the reactor and implementing EOP-0. EOP-4 would be the correct choice if the plant was operating in Mode-3

B. Incorrect - The basis for the minimum temperature for criticality is incorrect, however, the correct procedure to implement **would be** AOP-7K which would ultimately direct tripping the reactor and implementing EOP-0

C. Incorrect - The basis for the minimum temperature for criticality is incorrect. Additionally, the correct procedure to implement would be AOP-7K which would ultimately direct tripping the reactor and implementing EOP-0

D. Correct - The basis for the minimum temperature for criticality is correct per T.S. Bases 3.4.2 **and** the correct procedure to implement would be AOP-7K which would ultimately direct tripping the reactor and implementing EOP-0.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 92 Info			
Topic:	Minimum temperature for criticality		
Tier/Group:	Generic K & A		
K/A Info:	2.1 - Conduct of Operations • 2.1.32 - Ability to explain and apply system limits and precautions.		
SRO Importance:	4.0		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	T.S. 3.4.2, RCS Minimum Temperature for Criticality		
Comments:	None		

93 **ID: Q15858** **Points: 1.00**

Using references provided:

Unit-1 is in Mode 1. System Engineering has determined that 4KV Bus 14 Normal and Alternate Feeder breakers may not trip on an undervoltage when required. What action is required?

A. Enter TS 3.8.1, Action B, for 1B DG out of service.

B. Enter TS 3.8.9, Action A, for both breakers out of service.

C. Enter TS 3.8.9, Action B, for both breakers out of service.

D. Enter TS 3.8.1, Action E, for 1B DG being out of service.

Answer: A

Answer Explanation:

A. Correct - The 1B DG would be inoperable to 4KV Bus 14. Normal and Alternate Feeder Breakers being open are part of the logic circuit that must be completed for the 1B DG to close in on and power up 4KV Bus 14.

B. Incorrect - T.S. 3.8.9 requires OPERABLE AC electrical power distribution subsystems. From the Basis doc: "OPERABLE AC electrical power distribution subsystems require the associated buses, load centers, motor control centers, and distribution panels to be energized to their proper voltages". By this definition 4KV Bus 14 is operable.

C. Incorrect - T.S. 3.8.9, Condition B represents one or more 120V AC vital buses without power.

D. Incorrect - LCO 3.8.1.E is not applicable to 4KV Bus 14. CREVS and CRETS components are powered from 4 KV Busses 11 & 24.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 93 Info			
Topic:	Evaluate T.S. for 4kv feeder breaker problem (References required)		
Tier/Group:	Generic K & A		
K/A Info:	2.2 - Equipment Control • 2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.		
SRO Importance:	4.2		
Proposed references to be provided to applicant:	T.S.3.8.1 & 3.8.9 & associated Bases		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☒ Bank	☐ Modified	☐
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:	Last used for May, 2009 panel comp (average score – 71% over 38 student encounters since 2002)		
Technical references:	Tech Specs 3.8.1, AC Sources-Operating & 3.8.9 Distribution Systems-Operating		
Comments:	None		

94 **ID: Q92690** **Points: 1.00**

Unit -1 was operating at 100% power when the following events and conditions occurred:

- 1-RE-1752A/B/C/D (11/12/13/14 CAR Suction RAD MONs) are in alarm and indicating a leakrate of 28 GPD and stable
- 1-RIC-5421A (N16 RAD MONITOR) indicates a leakrate of 31 GPD and stable
- 1-RI-4014 (Unit 1 S/G B/D RMS) is elevated
- 1-RIC-4095 (Unit 1 S/G B/D Recovery RMS) is elevated

As Control Room Supervisor, which **ONE** of the following are you required to direct?

A. Implement AOP-2A, Excessive RCS Leakage

B. Secure S/G Blowdown per OI-8A, Blowdown System

C. Implement AOP-10, Abnormal Secondary Chemistry Conditions

D. Trip the reactor, perform EOP-0, Reactor Trip and implement EOP-6, Steam Generator Tube Rupture

Answer: C

Answer Explanation:

A. Incorrect - An RCS leak of 31 GPD (.02 GPM) is below the threshold for implementation of AOP-2A, Excessive RCS Leakage. AOP-10, Abnormal Secondary Chemistry Conditions, specifies implementation of AOP-2A **IF** the SG tube leakage reaches the operational limit of 50 GPD through any one SG.

B. Incorrect - The S/G Blowdown System RMSs, while elevated, have yet to reach a value where manual or automatic action is required per plant procedure. The decision to secure Blowdown under these circumstances would be based on recommendations from the Chemistry folks.

C. Correct - AOP-10, Abnormal Secondary Chemistry Conditions, Attachment 1 (UNIT 1 ACTIONS FOR SG TUBE LEAKAGE GREATER THAN 5 GPD) is written to address SG tube leakage of between 5 GPD and 50 GPD.

D. Incorrect - An RCS leak of 31 GPD (.02 GPM) is below the threshold for implementation of AOP-2A, Excessive RCS Leakage. AOP-2A, Excessive RCS Leakage is the procedure that would direct shutdown and/or a reactor trip for S/G tube leakage reaching the appropriate threshold.

Question 94 Info			
Topic:	Use RMS indications to evaluate RCS leakage		
Tier/Group:	Generic K & A		
K/A Info:	2.3 - Radiation Control • 2.3.5 - Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.		
SRO Importance:	2.9		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(4)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• 1C22-ALM, RMS Alarm Manual • AOP-10, Abnormal Secondary Chemistry Conditions • AOP-2A, Excessive RCS Leakage		
Comments:	None		

95 **ID: Q92810** **Points: 1.00**

U-1 is operating at 100% power when a plant trip occurs. All safety functions of EOP-0 have been reported. The following conditions exist:

- All CEAs are fully inserted.
- All electrical busses are energized from their normal power supplies.
- Pressurizer level is 88 inches and lowering slowly
- Pressurizer pressure is 1875 PSIA and lowering slowly
- T_{AVG} is 530 °F and lowering slowly
- ADVs and TBVs are shut
- RCS subcooling is 100 °F and rising slowly
- Main feedwater is being supplied to both steam generators
- 11 S/G level is -90 inches and lowering slowly
- 12 S/G level is -50 inches and rising
- Containment pressure is 1.5 PSIG and rising slowly
- 11 Main Steam Line Radiation Monitor reads 4.6 E-6 R/hr
- 12 Main Steam Line Radiation Monitor reads 2.2 E-4 R/hr

Which **ONE** of the following must be implemented based on existing plant conditions?

A. EOP-4, Excess Steam Demand Event

B. EOP-5, Loss of Coolant Accident

C. EOP-6, Steam Generator Tube Rupture

D. EOP-8, Functional Recovery Procedure

Answer D

Answer Explanation:

A. Incorrect - Plant conditions indicate a steam leak or an RCS leak is removing decay heat. Containment parameters indicate the location of this leak. A SG tube leak also exists on 12 SG based on MSLRM indications. Based on a tube leak occurring with either an RCS leak or a steam leak in containment, EOP-8 would be implemented..

B. Incorrect - Plant conditions indicate a steam leak or an RCS leak is removing decay heat. Containment parameters indicate the location of this leak. A SG tube leak also exists on 12 SG based on MSLRM indications. Based on a tube leak occurring with either an RCS leak or a steam leak in containment, EOP-8 would be implemented..

C. Incorrect - Plant conditions indicate a steam leak or an RCS leak is removing decay heat. Containment parameters indicate the location of this leak. A SG tube leak also exists on 12 SG based on MSLRM indications. Based on a tube leak occurring with either an RCS leak or a steam leak in containment, EOP-8 would be implemented..

D. Correct - Plant conditions indicate a steam leak or an RCS leak is removing decay heat. Containment parameters indicate the location of this leak. A SG tube leak also exists on 12 SG based on MSLRM indications. Based on a tube leak occurring with either an RCS leak or a steam leak in containment, EOP-8 would be implemented..

Question 95 Info			
Topic:	SRO responsibilities for AOPs during C/D		
Tier/Group:	2/1		
K/A Info:	2.1 - Conduct of Operations • 2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.		
SRO Importance:	4.4		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	• NO-1-200, Control of Shift Activities; NO-1-201, Calvert Cliffs Operating Manual • EOP-0, Post Trip Immediate Actions		
Comments:	None		

96 **ID: Q20870** **Points: 1.00**

Unit-2 is operating at 100% power. 23 HPSI pump has been declared inoperable due to a ground on phases A & C of the motor.

What T.S. action, if any, is required?

A. Align 22 HPSI pump to the Main HPSI header, within 1 hour, and declare the ECCS subsystem operable.

B. Enter the applicable T.S. LCO and restore 23 HPSI pump to service within the allowed completion time.

C. Align 22 HPSI pump to the Aux HPSI header, within 1 hour, and declare the ECCS subsystem operable.

D. No T.S. LCO action is required as at least 100% of the ECCS flow equivalent to a single operable ECCS train is available.

Answer: B

Answer Explanation:

A. Incorrect - 22 HPSI is not an acceptable substitute for 23 HPSI because it shares a common suction header with 21 HPSI. Redundancy would remain compromised.

B. Correct - Tech Spec 3.5.2, Action A, allows an out of service time of 72 hours assuming 21 HPSI remains operable.

C. Incorrect - 22 HPSI is not an acceptable substitute for 23 HPSI because it shares a common suction header with 21 HPSI. Redundancy would remain compromised.

D. Incorrect - Tech Specs require the redundancy of two 100% capable trains with an allowed out of service time, for one train, of 72 hours providing 100% of the ECCS flow equivalent to a single operable ECCS train is available. Plausible because the applicant may believe having 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available in the form of 21 HPSI Pump satisfies the LCO.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 96 Info			
Topic:	Determine actions for 23 HPSI OOS		
Tier/Group:	2/1		
K/A Info:	006 - Emergency Core Cooling System (ECCS) • 2.2.22 - Knowledge of limiting conditions for operations and safety limits.		
SRO Importance:	4.7		
Proposed references to be provided to applicant:	None		
Learning Objective:	CRO-7-1-5-94		
10 CFR Part 55 Content:	55.43(b)(2)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:	September, 2005 Panel comp (average score – 86% for 7 student encounters)		
Technical references:	Tech Spec 3.5.2, ECCS - Operating		
Comments:	None		

97 **ID: Q92692** **Points: 1.00**

During a Large Break LOCA:

(1) How is the Main Steam system affected and;
(2) What is the EOP strategy for the condition where S/G pressure is greater than RCS pressure?

A. (1) CSAS actuation will cause the MSIVs to shut;
(2) Cool the S/Gs using the ADVs.

B. (1) CSAS actuation will cause the MSIVs to shut;
(2) Bypass the MSIVs and cool the S/Gs using the TBVs.

C. (1) CIS actuation will cause the MSIVs to shut;
(2) Cool the S/Gs using the ADVs.

D. (1) SGIS actuation will cause the MSIVs to shut;
(2) Bypass the MSIVs and cool the S/Gs using the TBVs.

Answer: A

Answer Explanation:

A. Correct - CSAS will actuate on a Large Break LOCA and provides an automatic closure signal to the MSIVs. EOP-5, Loss of Coolant Accident, specifies: IF S/G pressure is greater than RCS pressure, THEN commence S/G cooldown using TURB BYP valves OR ADVs.

B. Incorrect - EOP-5, Loss of Coolant Accident, specifies "IF S/G pressure is greater than RCS pressure, THEN commence S/G cooldown using TURB BYP valves OR ADVs".

Plausibility - EOP-8, Functional Recovery Procedure (HR-2, S/G Heat Sink with SIS Operation), provides direction for bypassing the MSIVs and use of the TBVs in the event the ADVs are not available.

C. Incorrect - CIS does not provide a signal to automatically close the Main Steam Isolation Valves (MSIVs).

D. Incorrect - EOP-5, Loss of Coolant Accident, specifies "IF S/G pressure is greater than RCS pressure, THEN commence S/G cooldown using TURB BYP valves OR ADVs". MSIV Bypasses are not an option provided in EOP-5.

Plausibility - EOP-8, Functional Recovery Procedure (HR-2, S/G Heat Sink with SIS Operation), provides direction for bypassing the MSIVs and use of the TBVs in the event the ADVs are not available.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 97 Info			
Topic:	Affect of LOCA on Main Steam		
Tier/Group:	2/1		
K/A Info:	039 - Main and Reheat Steam System (MRSS) • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations • A2.01 - Flow paths of steam during a LOCA		
SRO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	EOP-5, Loss of Coolant Accident		
Comments:	None		

98 **ID: Q92693** **Points: 1.00**

Unit 1 is in MODE 3. The following conditions exist:

- RCS Pressure is 2250 PSIA
- T_{COLD} is 530 °F
- S/G pressure is 880 PSIG
- 13 AFW Pump out of service
- A loss of Instrument Air is in progress

(1) What effect will there be on the AFW system?
(2) What is the correct action to address this condition?

A. (1) There would be no remote speed control of 11 or 12 AFW Pp;
(2) Station an Operator locally to control the steam driven AFW pump speed, to maintain AFW Pp speed at a constant 4500 rpm.

B. (1) All AFW components are supplied by the Salt Water Air system, thus there is no impact on the AFW system;
(2) Direct an Operator at 1C04 to operate a steam driven AFW pump to maintain S/G level.

C. (1) There would be no remote speed control of 11 or 12 AFW Pp;
(2) Station an Operator locally to control the steam driven AFW pump speed, to maintain AFW discharge pressure at approximately 980 PSIG.

D. (1) There would be no control of the AFW Flow Control Valves from 1C04;
(2) Direct an Operator at 1C43 to operate the AFW Flow Control Valves to maintain S/G level.

Answer: C

Answer Explanation:

A. Incorrect - 11 & 12 AFW Pump speed cannot be controlled from the Control Room, due to the loss of Instrument Air, requiring an operator be stationed to manually control AFW Pump speed to maintain **AFW Pump discharge pressure 100 PSI greater than S/G pressure** per AOP-7D.

B. Incorrect - The Salt Water Air Compressors due not provide a backup supply of air to 11 & 12 AFW Pumps. 11 & 12 AFW Pump speed cannot be controlled from the Control Room, due to the loss of Instrument Air, requiring an operator be stationed to manually control AFW Pump speed to maintain AFW Pump discharge pressure 100 PSI greater than S/G pressure per AOP-7D.

C. Correct - 11 & 12 AFW Pump speed cannot be controlled from the Control Room, due to the loss of Instrument Air, requiring an operator be stationed to manually control AFW Pump speed to maintain AFW Pump discharge pressure 100 PSI greater than S/G pressure per AOP-7D.

D. Incorrect - AOP-7D, Loss of Instrument Air specifies: Control the AFW Pump speed with the Local Speed Adjust Knob, to maintain AFW Pump discharge approximately 100 PSIG greater than SG pressure. Adjust the SG FLOW CONTRs (IA supplied by the Sal Water Air Compressors) to maintain SG level between (-) 24 and (+) 30 inches and trending to zero inches.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 98 Info			
Topic:	AFW Pp speed controlled w/o instrument air		
Tier/Group:	2/1		
K/A Info:	061 - Auxiliary / Emergency Feedwater (AFW) System • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.07 - Air or MOV failure		
SRO Importance:	3.5		
Proposed references to be provided to applicant:	None		
Learning Objective:			
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☐ Bank	☐ Modified	☒ New
Cognitive level:	☐ Memory or Fundamental ☒ Comprehension or Analysis		
Last NRC Exam used on:	N/A		
Exam Bank History:	None		
Technical references:	AOP-7D, Loss of Instrument Air		
Comments:	None		

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

99 **ID: Q92710** **Points: 1.00**

Unit-1 is operating at 60% power when a loss of 4KV Bus 13 occurs.

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

A. (1) Loss of 12 and 13 Condensate Pumps;
(2) Commence a Rapid Power Reduction, to lower Condensate Header flow to less than 8000 GPM.

B. (1) Loss of 13 Condensate Pump and 13 Condensate Booster Pump;
(2) Bypass the Condensate Precoat Filters and Condensate Demineralizers and Verify 11 or 12 Condensate Pump and 11 or 12 Condensate Booster Pumps running.

C. (1) Loss of lube oil to both SGFPs;
(2) Immediately trip the Reactor and implement EOP-0. After completion of the Reactivity Safety Function, trip both SGFPs.

D. (1) Loss of 12 Heater Drain Pump, 13 Condensate Booster Pump and 13 & 14 CAR Pumps;
(2) No Stabilizing actions are necessary

Answer: A

Answer Explanation:

A. Correct - 12 and 13 Condensate Pps are lost necessitating a power reduction to get Condensate Header flow to less than the capacity of a single Condensate Pp.

B. Incorrect - While 13 Condensate Pp and 13 Condensate Booster Pump are lost, 12 Condensate Pp is also lost; necessitating a power reduction to get Condensate Header flow to less than the capacity of a single Condensate Pp.

C. Incorrect - Each SGFP has an Oil Pp powered from MCC-106 and one powered from MCC-116; therefore lube oil will not be lost with a loss of MCC-116 (13 4KV bus).

D. Incorrect - While the listed loads are in fact lost, the loss of two Condensate Pumps necessitates reducing power to get Condensate Header flow to less than the capacity of a single Condensate Pump.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 99 Info			
Topic:	Loss of a non-vital 4KV Bus 13 at 60% power		
Tier/Group:	2/1		
K/A Info:	062 - AC Electrical Distribution System • A2 - Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: • A2.01 - Types of loads that, if de-energized, would degrade or hinder plant operation		
SRO Importance:	3.9		
Proposed references to be provided to applicant:	None		
Learning Objective:	AOP-7I-03		
10 CFR Part 55 Content:	55.43(b)(5)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:			
Technical references:	• AOP-7I, Loss of 4KV, 480 Volt or 208/120 Volt Instrument Bus Power • Unit-1 Immediate Actions From 100% Power (Stabilizing Actions Plaque) Operator Aid		
Comments:	None		

100 **ID: Q37531** **Points: 1.00**

At 0800, EOP-8 was entered and a Site Area Emergency was declared. Because no Optimal Recovery Procedure was appropriate, the Technical Support Center staff was asked to provide a new procedure for this situation. It is now 1452.

When may you:

(1) Exit the current procedure and;
(2) Implement the new procedure developed by the Technical Support Center?

A. (1) When the new procedure's Intermediate Safety Function Acceptance Criteria are met and the new procedure has been approved;
(2) At the Shift Manager's direction.

B. (1) When the EOP-8 Safety Function Acceptance Criteria are met;
(2) Upon direction by the Technical Support Center-Director.

C. (1) The new procedure has been approved;
(2) Upon direction by the Technical Support Center-Director

D. (1) When the EOP-8 Safety Function Acceptance Criteria are met and the new procedure has been approved;
(2) At the Shift Manager's direction.

Answer: D

Answer Explanation:

A. Incorrect - When the EOP-8 Safety Function Acceptance Criteria are met is correct per EOP-8 Rev 29, step V.G.1.

B. Incorrect – Part (1) is partially correct. An approved procedure is required along with direction from the SM or TSC Director to implement it.

C. Incorrect - Part (1) is partially correct EOP-8 Safety Function Acceptance Criteria must be met as well. Part (2) is correct.

D. Correct - Per EOP-8 Rev 29, step V.G.1.

EXAMINATION ANSWER KEY

LOI 2010 NRC SRO Exam

Question 100 Info			
Topic:	Tech Supported generated procedures		
Tier/Group:	Generic K & A		
K/A Info:	2.2 - Equipment Control • 2.2.5 - Knowledge of the process for making design or operating changes to the facility.		
SRO Importance:	3.2		
Proposed references to be provided to applicant:	None		
Learning Objective:	LOR-042040404-001		
10 CFR Part 55 Content:	55.43(b)(3)		
Question source:	☒ Bank	☐ Modified	☐ New
Cognitive level:	☒ Memory or Fundamental ☐ Comprehension or Analysis		
Last NRC Exam used on:	No history of use on previous NRC exams		
Exam Bank History:	No record of previous use		
Technical references:	EOP-8, Functional Recovery Procedure		
Comments:	None		

NRC Written RO Exam Key for Calvert Cliffs Nuclear Power Plant
Exam Administered on August 11, 2010

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

NRC Written SRO Exam Key for Calvert Cliffs Nuclear Power Plant
Exam Administered on August 11, 2010

1.	A	**36.**	**D**	71.	D
2.	B	**37.**	**C**	72.	C
3.	A	**38.**	~~**A**~~ C	73.	A
4.	C	**39.**	**B**	~~74.~~	~~A~~ deleted
5.	B	**40.**	**D**	75.	C
6.	**C**	41.	B	**76.**	**C**
7.	**D**	42.	C	**77.**	**A**
8.	**A**	43.	B	**78.**	**C**
9.	**B**	44.	D	**79.**	**B**
10.	**D**	45.	B	**80.**	**A**
11.	C	**46.**	**A**	81.	A
12.	D	**47.**	**D**	82.	C
13.	A	**48.**	**C**	83.	D
14.	D	**49.**	**D**	84.	D
15.	C	**50.**	**A**	85.	B
16.	**B**	51.	A	**86.**	**B**
17.	**A**	52.	C	**87.**	**B**
18.	**C**	53.	A	**88.**	**B**
19.	**B**	54.	C	**89.**	**C**
20.	**B**	55.	B	**90.**	**D**
21.	A	**56.**	**C**	91.	B
22.	A	**57.**	**D**	92.	D
23.	D	**58.**	**A**	93.	A
24.	B	**59.**	**B**	94.	C
25.	C	**60.**	**D**	95.	D
26.	**D**	61.	B	**96.**	**B**
27.	**D**	62.	A	**97.**	**A**
28.	**A**	63.	D	**98.**	**C**
29.	**A**	64.	B	**99.**	**A**
30.	**D**	65.	D	**100.**	**D**
31.	B	**66.**	**C**		
32.	D	**67.**	**B**		
33.	A	**68.**	**C**		
34.	B	**69.**	**B**		
35.	B	**70.**	**C**		

Note: Questions 76 thru 100 are SRO-only.