Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 K1.05	
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause- effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency generator fuel oil supply system

Question: RO #1

While monitoring the monthly surveillance run of the A EDG on 102', you discover a significant leak on the discharge piping of the fuel oil supply header, spraying onto the engine.

What action must be taken to shut down the EDG AND minimize the fuel oil leak to prevent a possible fire/explosion?

A. Have the Control Room unload the EDG and perform normal 10 minute cooldown and then open power supply breaker to the Standby Fuel Oil (AP402) Pump.

B. Push the STOP pushbutton on the 1AC421 Panel and then open the power supply breaker to the Standby Fuel Oil (AP402) Pump.

C. Push either one of the Emergency STOP pushbuttons on the 1AC421 Panel.

D. Push both of the Emergency STOP pushbuttons on the 1AC421 Panel.

Proposed Answer: D

Explanation (Optional):

A: Have the Control Room unload the EDG and perform normal 10 minute cooldown and then open power supply breaker to the Standby Fuel Oil (AP402) Pump. **INCORRECT:** This will NOT immediately stop the standby fuel oil pump and will allow additional fuel oil to be deposited on the engine/floor. To open the breaker will require exiting the engine room and going up to 130' to MCC 52-411

B: Push the STOP pushbutton on the 1AC421 Panel and then open the power supply breaker to the Standby Fuel Oil (AP402) Pump. **INCORRECT:** This will stop the engine but will NOT immediately stop the standby fuel oil pump and will allow additional fuel oil to be deposited on the engine/floor. To open the breaker will require exiting the engine room and going up to 130' to MCC 52-411

C: Push either one of the Emergency STOP pushbuttons on the 1AC421 Panel. **INCORRECT:** Depressing either Emergency STOP pushbutton will have no effect on the shutdown of the standbyfuel oil pump, BOTH buttons are required to be depressed simultaneously

D: Push both of the Emergency STOP pushbuttons on the 1AC421 Panel **CORRECT:** Depressing both Emergency STOP pushbuttons simultaneously will shutdown (lockout) all of the engine mounted equipment, including the standby fuel oil pump (AP402) which discharges to the fuel oil header, limiting fuel oil discharge from the header

Technical Reference(s): NOH04EDG000C, page 55 rev 02 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EDG000E016, Given plant conditions resulting in a diesel engine shutdown, summarize/identify the automatic actions which occur when diesel speed decreases below 125 rpm.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	211000	K1.01	
	Importance Rating	3.0		

Knowledge of the physical connections and/or cause- effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Core spray line break detection: Plant-Specific

Question: RO #2

An ATWS has been in progress for roughly ten (10) minutes

Then:

OHA B3-B5 CORE SPRAY LINE BREAK is received;

A control panel walk-down reveals:

- Core Spray Loop A flow is 0 gpm
- Core Spray Loop B flow is 0 gpm
- HPCI Injection flow is 0 gpm
- RCIC Injection flow is 600 gpm

What caused the overhead alarm to be received?

A. RCIC Injection

B. SLC Injection

C. Core Spray Loop Inboard Injection valve (F005A) leakage

D. Core Spray Loop Inboard Injection valve (F005B) leakage

Proposed Answer: B

Explanation (Optional):

A: RCIC Injection, **INCORRECT:** RCIC has no direct interface with the Core Spray system

B: SLC Injection, **CORRECT:** The SLC system injects into the "A" Core Spray sparger and due to the arrangement of the Δp cell for the leak detection instrumentation, the OHA B3-B5 will be received during a SLC initiation.

C: Core Spray Loop Inboard Injection valve (F005A) leakage, **INCORRECT:** Due to the arrangement of the Δp cell for the leak detection instrumentation only a leak downstream of the F006A/B check valves would be detected. The F005A is upstream prior to reaching the check valve.

D: Core Spray Loop Inboard Injection valve (F005B) leakage **INCORRECT:** Due to the arrangement of the Δp cell for the leak detection instrumentation only a leak downstream of the F006A/B check valves would be detected. The F005B is upstream prior to reaching the check valve.

Technical Reference(s): NOH01SLCYSC (Attach if not previously provided)
NOH01CSSYS0C page 14/22/23

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)
SLCSYSE004 Given plant conditions, summarize/identify the interrelationship between the following Systems and the Standby Liquid Control System.
b. Core Spray

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	239002	K2.01	
	Importance Rating	2.8		

Knowledge of electrical power supplies to the following: SRV solenoids

Question: RO #3

The plant is at rated power when several overhead alarms are received in the main Control Room

A field report from the Aux Building Equipment Operator is that the 10D440 125 VDC bus has been damaged and de-energized.

How does this affect (if any) the ability to use the ADS SRVs?

A. "A" (B logic) pilot solenoids ONLY will lose actuator power

B. "B" (D logic) pilot solenoids ONLY will lose actuator power

C. Both "A" AND "B" (B AND D logic) pilot solenoids will lose actuator power

D. Neither "A" NOR "B" (B NOR D logic) pilot solenoids will lose actuator power

Proposed Answer: B

Explanation (Optional):

A: "A" pilot solenoids ONLY will lose actuator power, **INCORRECT**; ADS valves "A" only, pilot solenoids are powered from "B" channel 125 VDC 1BD417, which is fed from 10D420 125 VDC Unit Sub-Station.

B: "B" pilot solenoids ONLY will lose actuator power, **CORRECT**; ADS valves "B" only, pilot solenoids are powered from D channel 125 VDC 1DD417, which is fed from 10D440 125 VDC Unit Sub-Station.

C: Both "A" AND "B" pilot solenoids will lose actuator power ,**INCORRECT**; ADS valves "B" only, pilot solenoids are powered from D channel 125 VDC 1DD417, which is fed from 10D440 125 VDC Unit Sub-Station. ADS valves "A" only, pilot solenoids are powered from "B" channel 125 VDC 1BD417, which is fed from 10D420 125 VDC Unit Sub-Station.

D: Neither "A" NOR “B” pilot solenoids will lose actuator power , **INCORRECT**; ADS valves "B" only, pilot solenoids are powered from D channel 125 VDC 1DD417, which is fed from 10D440 125 VDC Unit Sub-Station. ADS valves "A" only, pilot solenoids are powered from "B" channel 125 VDC 1BD417, which is fed from 10D420 125 VDC Unit Sub-Station.

Technical Reference(s): NOH04ADSSYSC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ADSSYSE07, From memory, evaluate the interrelationship between the Automatic Depressurization System and the following, IAW available Control Room references: 125 VDC Class 1E Distribution System (As available)

Question Source: Bank # 56198

Modified Bank # (Note changes or attach parent)

New

Question History: Audit 1999

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	218000	K3.02	
	Importance Rating	4.5		

Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on the following: Ability to rapidly depressurize the reactor

Question: RO #4

Which of the following is the MINIMUM number of Safety Relief Valves (SRV) that must be opened during an Emergency Depressurization and the reason for that minimum number?

A. 4 SRVs will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow.

B. 4 SRVs provide the minimum depressurization rate required to ensure the low pressure ECCS systems inject soon enough to minimize the amount of time water level is below the top of active fuel.

C. 5 SRVs will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow.

D. 5 SRVs provide the minimum depressurization rate required to ensure the low pressure ECCS systems inject soon enough to minimize the amount of time water level is below the top of active fuel.

Proposed Answer: C

Explanation (Optional):

A: 4 SRVs will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow. **INCORRECT,** not sufficient IAW EPG/SAG calculations for Hope Creek Station

B: 4 SRVs provide the minimum depressurization rate required to ensure the low pressure ECCS systems inject soon enough to minimize the amount of time water level is below the top of active fuel. **INCORRECT,** not sufficient IAW EPG/SAG calculations for Hope Creek Station

C: 5 SRVs will remove all decay heat from the core at a pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow. **CORRECT** HC.OP-EO.ZZ-0202, Step ED-9 Bases, Minimum Number of SRVs required for ED - [Reference EPG/SAG

Appendix B section 17.22] - The Minimum Number of SRVs Required for Emergency Depressurization (MNSRED) is the least number of SRVs which corresponds to a Minimum Steam Cooling Pressure (MSCP) sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow at the corresponding MSCP. The MNSRED is utilized to ensure the RPV will depressurize and remain depressurized when emergency depressurization is required.

D: 5 SRVs provide the minimum depressurization rate required to ensure the low pressure ECCS systems inject soon enough to minimize the amount of time water level is below the top of active fuel. **INCORRECT,** not sufficient IAW EPG/SAG calculations for Hope Creek Station

Technical Reference(s): EOP-202, ED-9 Bases, Minimum Number of SRVs required for ED - [Reference EPG/SAG Appendix B section 17.22] (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP202E003, Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank # 53392

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	262001	K3.01	
	Importance Rating	3.5		

Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: Major System Loads

Question: RO #5

Given:

- All Circulating Water pumps are running
- All Circulation Water pump discharge valves (HV-2152A-D) are OPEN FULL

Then:

- Power is lost to Bus 10A502

Assuming NO operator actions are performed on the circulating water system.

Power to 10A502 has been restored for one minute, what is the Circulating Water system configuration?

A. AP501 and CP501 are running, HV-2152A and HV-2152C are in the OPEN FULL position

B. AP501 and CP501 are running, HV-2152A and HV-2152C are in the OPEN/CL MID position

C. BP501 and DP501 are running, HV-2152B and HV-2152D are in the OPEN FULL position

D. BP501 and DP501 are running, HV-2152B and HV-2152D are in the OPEN/CL MID position

Proposed Answer: B

Explanation (Optional):

A: AP501 and CP501 are running, HV-2152A and HV-2152C are in the fully OPEN position, **INCORRECT**, In the event of a loss of a 10A502 bus power failure, power is lost to all of the circulating water pump HPUs and circulating water pumps B & D. B & D circulating water pump

discharge valves fully shut and initially the A & C discharge valves remain as is. When power is subsequently restored A &C discharge valves move to the OPEN MID position.

B: AP501 and CP501 are running, HV-2152A and HV-2152C are in the OPEN MID position, **CORRECT,** In the event of a loss of a 10A502 bus power failure, power is lost to all of the circulating water pump HPUs and circulating water pumps B & D. B & D circulating water pump discharge valves fully shut and initially the A & C discharge valves remain as is. When power is subsequently restored A &C discharge valves move to the OPEN MID position.

C: BP501 and DP501 are running, HV-2152B and HV-2152D are in the fully OPEN position, **INCORRECT,** In the event of a loss of a 10A502 bus power failure, power is lost to all of the circulating water pump HPUs and circulating water pumps B & D. B & D circulating water pump discharge valves fully shut and initially the A & C discharge valves remain as is. When power is subsequently restored A &C discharge valves move to the OPEN MID position.

D: BP501 and DP501 are running, HV-2152B and HV-2152D are in the OPEN MID position, **INCORRECT,** In the event of a loss of a 10A502 bus power failure, power is lost to all of the circulating water pump HPUs and circulating water pumps B & D. B & D circulating water pump discharge valves fully shut and initially the A & C discharge valves remain as is. When power is subsequently restored A &C discharge valves move to the OPEN MID position.

Technical Reference(s): OP-SO.DA-0001 Sect 3.3.9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: CIRCWAE005, Given procedure HC.OP-SO.DA-0001, Circulating Water System Operation, explain the bases for the precautions and limitations IAW HC.OP-SO.DA-0001. (As available)

Question Source: Bank # 56919

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K3.01	
	Importance Rating	4.0	

Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE COOLANT INJECTION SYSTEM will have on following: Reactor water level control: BWR-2,3,4

Question: RO #6

A reactor level transient occurs scramming the plant:

- HPCI starts to allow for injection to the vessel
- RCIC is C/T for maintenance
- RPV level starts to increase
- The HPCI Aux Oil Pump seizes (trips) as the HPCI turbine nears rated speed
- RPV level swells up to 58"

Subsequently, RPV level then drops to -40"

With NO operator action, what is the response of HPCI and reactor vessel level restoration?

A. HPCI will trip when the Aux Oil Pump trips and will NOT restart and vessel level will lower.

B. HPCI will trip on High Level and will restart on low level to restore vessel level.

C. HPCI will trip on High Level and will NOT restart and vessel level will lower.

D. HPCI will remain running throughout these events and vessel level will be restored.

Proposed Answer: C

Explanation (Optional):

A: HPCI will trip when the Aux Oil Pump trips and will not restart and vessel level will lower. **INCORRECT**, when sufficient oil pressure has been developed by the shaft driven oil pump, the Aux oil pump would normally be cycled off, as speed reaches 1450-1650 rpm. At rated speed (≈ 4150 rpm) the Aux oil pump will not be running

B: HPCI will trip on High Level and will restart on low level to restore vessel level. **INCORRECT,** level 8 (+54") will cause HPCI to trip, however the Aux Oil pump is needed to allow pump to initially come up to speed until the shaft driven oil pump takes over (1450-1650 rpm). With a seized Aux Oil pump there is no oil pressure to get the turbine steam supply valves (FV-4879 and FV-4880) opened.

C: HPCI will trip on High Level and will not restart and vessel level will lower. **CORRECT,** level 8 (+54") will cause HPCI to trip and the Aux Oil pump is needed to allow pump to initially come up to speed until the shaft driven oil pump takes over (1450-1650 rpm). With a seized Aux Oil pump there is no oil pressure to get the turbine steam supply valves (FV-4879 and FV-4880) opened.

D: HPCI will remain running throughout these events and vessel level will be restored. **INCORRECT,** level 8 (+54") will cause HPCI to trip and the Aux Oil pump is needed to allow pump to initially come up to speed until the shaft driven oil pump takes over (1450-1650 rpm). With a seized Aux Oil pump there is no oil pressure to get the turbine steam supply valves (FV-4879 and FV-4880) opened.

Technical Reference(s): OP-SO.BJ-0001, pages 11, 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: HPCI00E010, Given plant conditions, determine the sequence of events following receipt of a HPCI turbine trip signal IAW control room references (As available)

Question Source: Bank #

Modified Bank # 55744 (Note changes or attach parent)

New

Question History: NRC 10/99

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	205000	K4.03	
	Importance Rating	3.8		

Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: Low reactor water level: Plant-Specific

Question: RO #7

The plant is shutdown and preparing to place "B" Loop of RHR in Shutdown Cooling.

Which one of the lists below contains ONLY valves that have interlocks to prevent draining the Reactor Vessel while aligning Shutdown Cooling?

A. HV-F004B (RHR pump torus suction - LPCI mode) and HV-F007B (RHR pump Minimum Flow Valve)

B. HV-F008 (RHR Shutdown Cooling Suction Valve) and HV-F017B (RHR LPCI Injection Valve)

C. HV-F016B (RHR Containment Spray Outboard Valve) and HV-F021B (RHR Containment Spray Inboard Valve)

D. HV-F024B (RHR Loop B Test Return Valve) and BC-HV-F027B (Torus Spray Valve)

Proposed Answer: D

Explanation (Optional):

A: BC-HV-F004B (RHR pump torus suction - LPCI mode) BC-HV-F007B (RHR pump Minimum Flow Valve) **INCORRECT**, while the F004 is interlocked to prevent being opened simultaneously with the F006, the F007 is not interlocked to prevent vessel drain down during SDC operations

B: BC-HV-F008B (RHR Shutdown Cooling Suction Valve) BC-HV-F017B (RHR LPCI Injection Valve) **INCORRECT,** while the F008 is NOT interlocked to prevent a drain down, however it will isolate on a low RPV level @ 12.5", the F017 is not interlocked to prevent vessel drain down, but interlocked to prevent opening above RPV pressure of 450 psig and an LPCI initiation signal present

C: BC-HV-F016B (RHR Containment Spray Outboard Valve) BC-HV-F021B (RHR Containment Spray Inboard Valve) INCORRECT, the F016 and F021 are not interlocked to prevent vessel

drain down, but interlocked to allow both valves to be opened only with: a LPCI initiation signal present and High drywell pressure and its respective LPCI injection valve HV-F017A(B) 100% CLOSED.

D: BC-HV-F024B (RHR Loop B Test Return Valve) BC-HV-F027B (Torus Spray Valve) **CORRECT,** both the F024and 27 as well as the F004 all must be closed to allow for the HV-F006 to be opened for Shutdown Cooling, this prevents a vessel drain down.

Technical Reference(s): OP.SO.BC-001/002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RHRSYSE006, Concerning the interlocks associated with the shutdown cooling suction valves (F006A/B) and the Torus suction (F004A/B), pump test return (F024A/B), and Torus spray isolation (F027A/B) valves. a. Summarize the bases for the interlocks, IAW available control room references. (As available)

Question Source: Bank # 53709

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

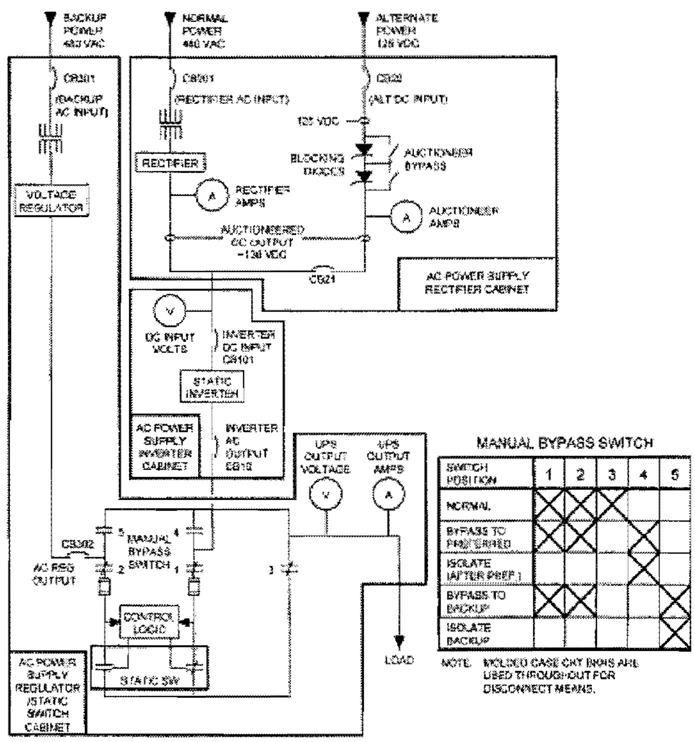
Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	262002	K4.01	
	Importance Rating	3.1		

Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies

Question: RO #8

Given:

- A refueling outage is in progress
- 10A401 4KV 1E bus is de-energized for pre-planned maintenance.
- The operator places the 1AD481 inverter Manual Bypass Control Switch from "NORM" to the "ISOLATE ALTERNATE" position.



SELECT the effect this will have on the power supply to the load (1AJ481).

A. Normal power 480 VAC will supply power to the load.

B. Alternate power 125 VDC will supply power to the load.

C. Backup power 480 VAC will supply power to the load.

D. NO power will be supplied to the load.

Proposed Answer: D

Explanation (Optional):

A: Normal power 480 VAC will supply power to the load. **INCORRECT,** Normal 480 VAC is not available because 10A401 is de-energized and contacts 3 and 4 are open.

B: Alternate power 125 VDC will supply power to the load. **INCORRECT,** Contacts 1, 2, 3, and 4 are open. No possible way to provide output from the inverter.

C: Backup power 480 VAC will supply power to the load. **INCORRECT,** 10A401 bus is de-energized which is the source of Back-up AC.

D: NO power will be supplied to the load. **CORRECT,** Exhibit 2 of SO-PN-0001 shows that in Isolate after Alt, that the static switch output will be disconnected from the load (contact 3 open), and the supply from Backup 480 will be connected to the load (contact 5 closed and 4 is open) However with 10A401>10B411-33, Backup 480 for AJ481 is de-energized.

Technical Reference(s): NOH01EAC00 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: 1EAC00E024 Provided a drawing of the UPS: (Figs. 6,7,8) c. Predict the response caused by operating the manual bypass switch IAW Att. 2 of the Lesson Plan. (As available)

Question Source: Bank # 53942

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	300000	K5.01	
	Importance Rating	2.5		

Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Air compressors

Question: RO #9

Given:

- A loss of coolant accident has occurred.
- The Reactor Auxiliaries Cooling System (RACS) has been restored.

Which of the following describes the availability/response of the Emergency Instrument Air Compressor (EIAC) for these conditions should instrument air header pressure begin lowering?

A. The EIAC will automatically start on instrument air header pressure less than 85 psig if the LOCA signal is cleared.

B. The EIAC will NOT automatically start but can be started locally after relieving intercooler pressure.

C. The EIAC is NOT available until the LOCA signal is cleared, PCIS reset, and the 1E breaker is closed.

D. The EIAC is NOT available until the Non-1E breaker is closed and instrument air pressure is less than 85 psig.

Proposed Answer: C

Explanation (Optional):

A: The EIAC will automatically start on instrument air header pressure less than 85 psig if the LOCA signal is cleared. **INCORRECT**: 1E breaker must be reset and manually closed first

B: The EIAC will NOT automatically start but can be started locally after relieving intercooler pressure. **INCORRECT**: The EIAC is not available until the 1E breaker is reset (LOCA signal) and closed and instrument air pressure is less than 85 psig.

C: The EIAC is NOT available until the LOCA signal is cleared, PCIS reset, and the 1E breaker is closed. **CORRECT**: The EIAC is not available until the LOCA signal is cleared, PCIS reset, and the 1E breaker closed manually.

D: The EIAC is NOT available until the Non-1E breaker is closed and instrument air pressure is less than 85 psig. **INCORRECT**: Not until 1E breaker is reset (LOCA signal) manually

Technical Reference(s): OP-SO.KB-0001, Interlocks 3.3.5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: INSAIRE015, From memory, determine the response of the emergency instrument air compressor to the following conditions. (As available)
a. Loss of Offsite Power (LOP)
b. Loss of Coolant Accident (LOCA)
c. Compressor intercooler pressure > 5 psig and a start signal received

Question Source: Bank # 56532

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2010

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K5.01	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION : Hydrogen generation during battery charging.

Question: RO #10

Which one of the following statements explains why a loss of battery room ventilation can result in a significant safety hazard to plant personnel.

A loss of the battery room ventilation system can cause...

A. the accumulation of H_2 while the battery is being charged could lead to an explosion hazard.

B. the accumulation of excess H_2SO_4 in the battery room(s) could lead to a hazardous atmosphere.

C. the accumulation of CO_2 while the battery is being charged could cause displacement of oxygen which is a suffocation hazard.

D. high humidity in the battery rooms can result in an increased chance of receiving an electrical shock.

Proposed Answer: A

Explanation (Optional):

A: the accumulation of H_2 while the battery is being charged which could lead to an explosion hazard. **CORRECT**, A loss of the battery room ventilation system can cause the accumulation of hydrogen while the battery is being charged which could lead to an explosion hazard.

B: the accumulation of H_2SO_4 in the battery room(s) which could lead to a hazardous atmosphere **INCORRECT**, H_2SO_4 is sulfuric acid, which is contained inside the batteries themselves and pose no safety hazard as long as the acid stays within the confines of the battery

C: the accumulation of CO_2 while the battery is being charged, this could cause displacement of oxygen which is a suffocation hazard. **INCORRECT**, CO2 is not given off from the battery at any

condition of operation.

D: high humidity in the battery rooms can result in an increased chance of receiving an electrical shock. **INCORRECT**, the shock hazard in the room is not a function of room humidity.

Technical Reference(s): NOH01DCELEC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: DCELECE005, Summarize the interrelationship(s) between 125VDC 1E/N1E Power Systems and the following: (As available)
a. 480VDC 1E/N1E Power Supply
b. Auxiliary Building Ventilation System

Question Source: Bank # 68755

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	261000	K6.01	
	Importance Rating	2.9		

Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : A.C. electrical distribution

Question: RO #11

Given:

- The plant is operating at 100% power
- A Loss of Offsite power occurs
- Drywell pressure is 5 psig and rising
- "A" Emergency Diesel Generator fails to start

Which one of the following describes the effect on FRVS after 3 minutes?
(Assume NO operator action)

A. Only 3 Recirc Fans and NO Vent Fan start

B. Only 3 Recirc Fans and one (1) Vent Fan start

C. Only 4 Recirc Fans and NO Vent Fan start

D. Only 4 Recirc Fans and one (1) Vent Fan start

Proposed Answer: D

Explanation (Optional):

A: Only 3 Recirc Fans and NO Vent Fan start, **INCORRECT,** A vital bus (AG400-EDG) powers A/EVH213 FRVS Recirc and AV206 Vent fan. The B/C/DVH213 Recirc Fans will start as well as the AUTO start of FVH213 Recirc fan and BV206 Vent Fan after Low Flow from AVH213/AV206 Fans

B: Only 3 Recirc Fans and one (1) Vent Fan start, **INCORRECT,** A vital bus (AG400-EDG) powers A/EVH213 FRVS Recirc and AV206 Vent fan. The B/C/DVH213 Recirc Fans will start as well as the AUTO start of FVH213 Recirc fan and BV206 Vent Fan after Low Flow from AVH213/AV206

Fans

C: Only 4 Recirc Fans and NO Vent Fan start, **INCORRECT,** A vital bus (AG400-EDG) powers A/EVH213 FRVS Recirc and AV206 Vent fan. The B/C/DVH213 Recirc Fans will start as well as the AUTO start of FVH213 Recirc fan and BV206 Vent Fan after Low Flow from AVH213/AV206 Fans

D: Only 4 Recirc Fans and one (1) Vent Fan start, **CORRECT,** A vital bus (AG400-EDG) powers A/EVH213 FRVS Recirc and AV206 Vent fan. The B/C/DVH213 Recirc Fans will start as well as the AUTO start of FVH213 Recirc fan and BV206 Vent Fan after Low Flow from AVH213/AV206 Fans

Technical Reference(s): OP-SO.GU-001, OP-AB.ZZ-135 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: SECCONE012, Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, identify the status of the Secondary Containment by evaluation of the controls/ instrumentation/alarms. (As available)

Question Source: Bank # 68889

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2002

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 K6.04	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Condensate storage and transfer system

Question: RO #12

Given:

- The Reactor Core Isolation Cooling (RCIC) is operating in Full Flow Recirc
- The RCIC flow controller is in "Automatic"
- RCIC turbine speed is 2450 rpm

Which of the following describes the response of RCIC turbine speed and system flow if the operator throttles the RCIC Test Bypass To CST Isolation Valve (F022) in the "open" direction for the given conditions?

(Compare the conditions after they stabilize to before the valve was throttled.)

A. RCIC turbine speed lowers, System flow remains unchanged

B. RCIC turbine speed lowers, System flow goes down

C. RCIC turbine speed raises, System flow remains unchanged.

D. RCIC turbine speed raises, System flow goes up

Proposed Answer: A

Explanation (Optional):

A: RCIC turbine speed lowers, System flow remains unchanged **CORRECT,** With flow controller in Auto, opening the CST valve reduces the resistance to flow (system flow initially increases for that given turbine speed), flow controller reduces turbine speed to return back to the selected flow setpoint.

B: RCIC turbine speed lowers, System flow goes down **INCORRECT,** With flow controller in Auto, opening the CST valve reduces the resistance to flow (system flow initially increases for that

given turbine speed), flow controller reduces turbine speed to return back to the selected flow setpoint.

C: RCIC turbine speed raises, System flow remains unchanged **INCORRECT,** With flow controller in Auto, opening the CST valve reduces the resistance to flow (system flow initially increases for that given turbine speed), flow controller reduces turbine speed to return back to the selected flow setpoint.

D: RCIC turbine speed raises, System flow goes up **INCORRECT,** With flow controller in Auto, opening the CST valve reduces the resistance to flow (system flow initially increases for that given turbine speed), flow controller reduces turbine speed to return back to the selected flow setpoint.

Technical Reference(s): NOH04RCIC00 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RCIC000E023, Given any of the following and appropriate control room reference material, evaluate and determine the effect on the RCIC system of the following IAW the RCIC System Lesson Plan: (As available)

a. A given valve opening or closure

Question Source: Bank # 55683

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 1999

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 14

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	215005	A1.07	
	Importance Rating	3.0		

Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: APRM (gain adjustment factor)

Question: RO #13

Given:

- The Mode Switch is in the STARTUP/HOT STANDBY position.
- All APRMs are indicating ≈ 1%.
- Reactor power is ≈ mid-scale on Range 7 of the IRMs.
- Reactor Recirc Pump speeds are at minimum.
- The "A" APRM is in bypass for an I & C surveillance

Then:

- During the surveillance the Tech inadvertently fails the APRM "C" upscale.

What is the response from RPS?

A. Half scram when "C" APRM indicates 14%.

B. Half scram when "C" APRM indicates 69%.

C. Full scram when "C" APRM indicates 14%.

D. Full scram when "C" APRM indicates 69%.

Proposed Answer: A

Explanation (Optional):

A: Half scram when "C" APRM indicates 14%. **CORRECT**, Scram setpoint is 14% with the Mode Switch in Startup. APRM Scram setpoints are always active. ½ Scram will occur on RPS A2.

B: Half scram when "C" APRM indicates 69% **INCORRECT,** Scram setpoint is 14% with the Mode Switch in Startup. APRM Scram setpoints are always active. ½ Scram will occur on RPS A2. The Upscale Thermal trip is 0.57(W-Δw) + 58% and would not be reached prior to half scram from the 14% setpoint. Min flow 20% x .57 resulting in 11 + 58% = 69%

C: Full scram when "C" APRM indicates 14% **INCORRECT,** Scram setpoint is 14% with the Mode Switch in Startup. APRM Scram setpoints are always active. ½ Scram will occur on RPS A2 .

D: Full scram when "C" APRM indicates 69% **INCORRECT,** Scram setpoint is 14% with the Mode Switch in Startup. APRM Scram setpoints are always active. ½ Scram will occur on RPS A2. The Upscale Thermal trip is 0.57(W-Δw) + 58% and would not be reached prior to half scram from the 14% setpoint. Min flow 20% x .57 resulting in 11 + 58% = 69%

Technical Reference(s): OP-SO.SE-0001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: APRM00E009, From memory, IAW Technical Specifications, determine the rod blocks and/or scrams initiated by the APRM System, IAW available references. (As available)

Question Source: Bank # 80586

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2, 6

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	209001	A1.03	
	Importance Rating	3.8		

Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Reactor water level

Question: RO #14

Given:

- A loss of coolant accident has occurred
- Reactor water level reached -140" and is currently rising
- Reactor pressure is 350 psig and slowly lowering
- All plant systems responded as designed

Which of the following describes the capabilities for the Core Spray System (CS) Inboard Injection Valves (F005A & B) and the Outboard Injection Valves (F004A & B), and how will vessel level makeup be affected?

A. None of the four F004A/B OR F005A/B valves may be overridden closed and vessel level will continue to rise.

B. ONLY the F004A/B valves may be overridden closed and vessel injection from Core Spray will be terminated.

C. ONLY F005A/B valves may be overridden closed and vessel injection from Core Spray will be terminated.

D. Core Spray is NOT presently injecting because all four F004A/B OR F005A/B valves are closed AND vessel level will continue to rise.

Proposed Answer: C

Explanation (Optional):

A: None of the four F004A/B OR F005A/B valves may be overridden closed and vessel level will continue to rise. **INCORRECT,** The F005 valves may be overridden closed with initiation signal present by the Auto Open Ovrd pushbuttons.

B: ONLY the F004A/B valves may be overridden closed and vessel injection from Core Spray will be terminated. **INCORRECT,** The F005 valves may be overridden closed with initiation signal present by the Auto Open Ovrd pushbuttons.

C: ONLY the F005A/B valves may be overridden closed and vessel injection from Core Spray will be terminated. **CORRECT,** The F005 valves may be overridden closed with initiation signal present by the Auto Open Ovrd pushbuttons.

D: Core Spray is NOT presently injecting because all four F004A/B OR F005A/B valves are closed AND vessel level will continue to rise. **INCORRECT,** With reactor pressure 350 psig and dropping and shutoff head ≈ 385 psig, the injection valves would be open and pumps would be injecting . Normal lineup for the F005A/B is closed and the F004A/B is open

Technical Reference(s): OP-SO.BE-0001, M-52-0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: CSSYS0E013, Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Control Room, assess the status of the Core Spray System or its components by evaluation of the controls/instrumentation/alarms. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	215003 A2.06	
Importance Rating	3.0	

Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty Range Switch

Question: RO #15

A reactor startup is in progress:

IRM channel "C" chart recorder is presently indicating "35" on Range 4 and continues to read "35", even after the range switch is moved to Range 5.

What should the chart recorder show when the Range switch for the IRM channel "C" is taken to Range 5 AND what actions are required for this condition?

A. The indication should have dropped down to "3.5" on the 0-40 range. Terminate control rod withdrawal, bypass the "C" IRM AND verify the rod block has reset.

B. The indication should have jumped up to "350" on the 0-125 range. Terminate control rod withdrawal, bypass the "C" IRM AND reset the half-scram.

C. The indication should have dropped down to "1.12" on the 0-40 range. Terminate control rod withdrawal, bypass the "C" IRM AND verify the rod block has reset.

D. The indication should have jumped up to "112" on the 0-125 range. Terminate control rod withdrawal, bypass the "C" IRM AND reset the half-scram.

Proposed Answer: A

Explanation (Optional):

A: The indication should have dropped down to "3.5" on the 0-40 range, as an additional attenuator {x .1} is placed in service in the circuitry for the new range of indication. **CORRECT,** When ranged up to range 5 the percentage is 35/125 (from range 4) When converting the denominator

to 40ths it becomes 1.12/40 of the scale.

B: The indication should have jumped up to "350" on the 0-125 range, as an additional attenuator {x 10} is placed in service in the circuitry for the new range of indication. **INCORRECT,** When ranged up to range 5 the percentage is 35/125 (from range 4) When converting the denominator to 40ths it becomes 1.12/40 of the scale.

C: The indication should have dropped down to "1.12" on the 0-40 range, as an additional attenuator {x .1} is placed in service in the circuitry for the new range of indication. **INCORRECT,** When ranged up to range 5 the percentage is 35/125 (from range 4) When converting the denominator to 40ths it becomes 1.12/40 of the scale.

D: The indication should have jumped up to "112" on the 0-125 range, as an additional attenuator {x 10} is placed in service in the circuitry for the new range of indication. **INCORRECT,** When ranged up to range 5 the percentage is 35/125 (from range 4) When converting the denominator to 40ths it becomes 1.12/40 of the scale.

Technical Reference(s): NOH01IRMSYS, page 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: IRMSYSE005, Given a scenario of applicable operating conditions, determine the parameter setpoint, and bypass conditions for each IRM signal which will initiate a rod block and/or reactor scram, IAW control room procedures. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2, 6

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 A2.01	
	Importance Rating	2.7	

Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded

Question: RO #16

Given:

- Special low power nuclear testing is in progress
- Shorting Links have been removed
- The 'C' SRM high Voltage power supply fails.
- 'C' SRM counts drop to zero.

What action(s) is/are required IAW plant procedures?

A. Bypass the “C” SRM because a Withdraw rod block (only) exists.

B. Bypass the “C” SRM because an Insert AND Withdraw rod block exist.

C. Bypass the “C” SRM, reset A2 RPS because a Withdraw Rod Block AND a Trip of RPS Channel A2 exist.

D. LOCK the Mode Selector Switch in Shutdown because a Withdraw Rod Block AND a Full Scram exist.

Proposed Answer: A

Explanation (Optional):

A: Bypass the “C” SRM because a Withdraw rod block (only) exists. **CORRECT**, SO.SE-0001 SRM downscale and SRM INOP do not input into RPS, Limitations step 3.2.2 and 3.2.3.

B: Bypass the “C” SRM because an Insert AND Withdraw rod block exist. **INCORRECT,** Only the RWM inputs a insert block, SO.SE-0001 SRM downscale and SRM INOP do not input into RPS, Limitations step 3.2.2 and 3.2.3.

C: Bypass the "C" SRM, reset A2 RPS because a Withdraw Rod Block AND a Trip of RPS Channel A2 exist. **INCORRECT,** No input to RPS from SRM downscale and/or SRM INOP, SO.SE-0001 SRM downscale and SRM INOP do not input into RPS, Limitations step 3.2.2 and 3.2.3.

D: LOCK the Mode Selector Switch in Shutdown because a Withdraw Rod Block AND a Full Scram exist. **INCORRECT,** No input to RPS from SRM downscale and/or SRM INOP, SO.SE-0001 SRM downscale and SRM INOP do not input into RPS, Limitations step 3.2.2 and 3.2.3.

Technical Reference(s): OP-SO.SE-0001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ABIC04E004, Explain the reasons for how plant/system parameters respond when implementing Neutron Monitoring. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2, 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A3.06	
	Importance Rating	3.7	

Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Indicating lights and alarms

Question: RO #17

Given:

- The reactor is shutdown
- All control rods fully inserted.
- Reactor pressure is 385 psig and stable
- RPV level is being held relatively constant
- Preparation for entry into Mode 4 are on-going

Then, 'C' RHR Loop receives a spurious LOCA level 1 signal and the OHA A6-A4 RHR LPCI LOOP C INITIATED is received.

Which of the following describes the status of "C" RHR?

A. BC-HV F017C (LPCI Injection valve) indicates open and the 'C' RHR minimum flow control valve (HV-F007C) will indicate open

B. BC-HV F017C (LPCI Injection valve) indicates open and the 'C' RHR minimum flow control valve (HV-F007C) will indicate closed

C. BC-HV F017C (LPCI Injection valve) indicates closed and the 'C' RHR minimum flow control valve (HV-F007C) will indicate open

D. BC-HV F017C (LPCI Injection valve) indicates closed and the 'C' RHR minimum flow control valve (HV-F007C) will indicate closed.

Proposed Answer: A

Explanation (Optional):

A: BC-HV F017C (LPCI Injection valve) indicates open and the 'C' RHR minimum flow control valve (HV-F007C) will indicate open. **CORRECT,** The F017C will be open as the reactor pressure permissive of < 450 psig is satisfied with an injection signal present. The F007C will still be open as system flow is < 1270 gpm, because reactor pressure is above the shutoff head of the pump (366 psig) and will not be injecting to the vessel.

B: BC-HV F017C (LPCI Injection valve) indicates open and the 'C' RHR minimum flow control valve (HV-F007C) will indicate closed. **INCORRECT,** The F017C will be open as the reactor pressure permissive of < 450 psig is satisfied with an injection signal present. The F007C will still be open as system flow is < 1270 gpm, because reactor pressure is above the shutoff head of the pump (366 psig) and will not be injecting to the vessel.

C: BC-HV F017C (LPCI Injection valve) indicates closed and the 'C' RHR minimum flow control valve (HV-F007C) will indicate open. **INCORRECT,** The F017C will be open as the reactor pressure permissive of < 450 psig is satisfied with an injection signal present. The F007C will still be open as system flow is < 1270 gpm, because reactor pressure is above the shutoff head of the pump (366 psig) and will not be injecting to the vessel.

D: BC-HV F017C (LPCI Injection valve) indicates closed and the 'C' RHR minimum flow control valve (HV-F007C) will indicate closed. **INCORRECT,** The F017C will be open as the reactor pressure permissive of < 450 psig is satisfied with an injection signal present. The F007C will still be open as system flow is < 1270 gpm, because reactor pressure is above the shutoff head of the pump (366 psig) and will not be injecting to the vessel.

Technical Reference(s): NOH01RHRSYSC,pages 14,24, 35 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RHRSYSE011, Given a labeled drawing of, or access to the Residual Heat Removal System controls/indication on 10C650: a. Explain the function of each indicator IAW available control room references. b. Assess plant conditions which will cause the indicators to light or extinguish IAW available control room references. c. Determine the effect of each control on the RHR System IAW available control room references.
d. Assess plant conditions or permissives required for the control switches / pushbuttons to perform their intended functions IAW available control room references. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7, 8
	55.43	

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	400000	A3.01	
	Importance Rating	3.0		

Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

Question: RO #18

Given:

- The plant is operating at rated power.
- TACS return accumulator (AT-412) develops a leak in its instrument tubing
- Sensed pressure in the transmitters drops to 20 psig.

Which one of the following describes the plant response and actions required? (Assume NO operator actions are successful).

A. TACS valves HV-2522E/F CLOSE. TACS supply and return valves (HV-2522/2496) from both SACS loops will be open. If the TACS loss is sustained, reduce Reactor Recirculation pumps to minimum speed and lock the mode switch in shutdown.

B. TACS valves HV-2522E/F CLOSE. TACS supply and return valves (HV-2522/2496) from both SACS loops will be open. If the TACS loss is sustained, lock the mode switch in shutdown and then trip both Reactor Recirculation pumps.

C. TACS valves HV-2522E/F remain OPEN. TACS supply and return valves (HV-2522/2496) from both SACS loops close. If the TACS loss is sustained, lock the mode switch in shutdown and then trip both Reactor Recirculation pumps.

D. TACS valves HV-2522E/F remain OPEN. TACS supply and return valves (HV-2522/2496) from both SACS loops close. If the TACS loss is sustained, reduce Reactor Recirculation pumps to minimum speed and lock the mode switch in shutdown.

Proposed Answer: A

Explanation (Optional):

A: TACS valves HV-2522E/F CLOSE. TACS supply and return valves (HV-2522/2496) from both SACS loops will be open. If the TACS loss is sustained, reduce Reactor Recirculation pumps to minimum speed and lock the mode switch in shutdown. **CORRECT, AB.COOL-0002, Retainment Override** Condition - Complete and Sustained Loss of TACS
Action – Ia. Reduce Reactor Recirculation Pumps to Minimum Speed. Ib. LOCK the Reactor Mode Switch in SHUTDOWN.
Interlocks section: Low TACS pressure at supply OR return accumulators (22 psig) causes TACS HV-2522E/F to close. Low SACS to TACS flow from In-Service loop (9,900 gpm) In-service loop SACS pumps transfer to MAN. Standby Loop SACS Pumps Start IF in AUTO. TACS supply and return valves on standby SACS loop open IF respective pump AND valve control is in AUTO.

B: TACS valves HV-2522E/F CLOSE. TACS supply and return valves (HV-2522/2496) from both SACS loops will be open. If the TACS loss is sustained, lock the mode switch in shutdown and then trip both Reactor Recirculation pumps. **INCORRECT**

C: TACS valves HV-2522E/F remain OPEN. TACS supply and return valves (HV-2522/2496) from both SACS loops close. If the TACS loss is sustained, lock the mode switch in shutdown and then trip both Reactor Recirculation pumps. **INCORRECT**

D: TACS valves HV-2522E/F remain OPEN. TACS supply and return valves (HV-2522/2496) from both SACS loops close. If the TACS loss is sustained, reduce Reactor Recirculation pumps to minimum speed and lock the mode switch in shutdown. **INCORRECT**

Technical Reference(s): OP-AB.COOL-0002, AR.ZZ-0029 page 99, OP-SO.EG-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ABCOL2E004, Explain the reasons for how plant/system parameters respond when implementing Safety Auxiliaries Cooling System. (As available)

Question Source: Bank # 119057

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	4, 7
	55.43	

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	223002	A4.02	
	Importance Rating	3.9		

Ability to manually operate and/or monitor in the control room: Manually initiate the system (PCIS)

Question: RO #19

The plant is in OPCON 4:

- Primary Containment has been de-inerted
- Containment Atmosphere Control System (CACS) is aligned to purge the drywell and suppression chamber
- The 'B' and 'C' Reactor Building Ventilation Supply and Exhaust Fans are running
- The 'A' Reactor Building Ventilation Supply and Exhaust Fans are in AUTO

Then,

- An operator arms and depresses the 'D' Channel PCIS Manual Initiation pushbutton on 10C651C for a surveillance test.

Two minutes later AND after plant condition(s) stabilize, what would be the final status of the containment purge lineup and the Reactor Building Ventilation System (RBVS)?

A. There will be NO effect on the containment purge lineup or Reactor Building Ventilation fans.

B. There will be NO effect on the containment purge lineup. All Reactor Building Ventilation fans will be tripped.

C. Drywell and Suppression Chamber purge supply and exhaust lines will be isolated, all Reactor Building Ventilation fans will be tripped.

D. Drywell and Suppression Chamber purge supply and exhaust lines will be isolated, the running Reactor Building Ventilation fans will remain in service.

Proposed Answer: C

Explanation (Optional):

A: There will be NO effect on the containment purge lineup or Reactor Building Ventilation fans. **INCORRECT**, Manual initiation of the 'D' Channel PCIS closes the GS-HV-4950, 4962, 4979, and 4980. These valves isolate the purge supply and exhaust lines. While the 'D' channel does not directly trip the running RBVS supply and exhaust fans, it will isolate the Reactor Building Ventilation System supply and exhaust lines, which will result in all running fans tripping on low flow after a 90 second time delay.

B: There will be NO effect on the containment purge lineup, all Reactor Building Ventilation fans will be tripped. **INCORRECT**. Manual initiation of the 'D' Channel PCIS closes the GS-HV-4950, 4962, 4979, and 4980. These valves isolate the purge supply and exhaust lines.

C: Drywell and Suppression Chamber purge supply and exhaust lines will be isolated, all Reactor Building Ventilation fans will be tripped. **CORRECT.** Manual initiation of the 'D' Channel PCIS closes the GS-HV-4950, 4962, 4979, and 4980. These valves isolate the purge supply and exhaust lines. While the 'D' channel does not directly trip the running RBVS supply and exhaust fans, it will close the GU-HD-9414B and 9370B. These valves isolate the Reactor Building Ventilation System supply and exhaust lines, which will result in all running fans tripping on low flow after a 90 second time delay. The 'A' RBVS supply and exhaust fans are directly tripped (load shed) by the 'D' Channel PCIS signal.

D: Drywell and Suppression Chamber purge supply and exhaust lines will be isolated, the running Reactor Building Ventilation fans will remain in service. **INCORRECT,** While the 'D' channel does not directly trip the running RBVS supply and exhaust fans, it will isolate the Reactor Building Ventilation System supply and exhaust lines, which will result in all running fans tripping on low flow after a 90 second time delay.

Technical Reference(s): OP-SO.GS-0001,OP-SO.SM-0001 OP-ST.SM-0001, M-57/76 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: M-57-1, M-76-1

Learning Objective: INERT0E012, Given plant parameters, analyze plant parameters for conditions that will automatically isolate the Containment inerting and makeup flowpaths. (As available)

Question Source: Bank # 62574

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2009

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	7, 8
	55.43	

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

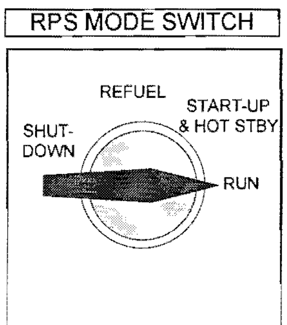
Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	212000	A4.01	
	Importance Rating	4.6		

Ability to manually operate and/or monitor in the control room: Provide manual SCRAM signal(s)

Question: RO #20

The CRS directs a manual scram from 56% power. The RO places the Mode Switch in the SHUTDOWN position.



Of the following, what was the first scram signal seen by the RPS logic?

A. 120/125 on Range 5 of the IRMs.

B. > 14% power as sensed by the APRMs.

C. > 2 x 10^5 CPS as sensed by the SRMs.

D. Rx Mode switch in the shutdown position.

Proposed Answer: B

Explanation (Optional):

A: 120/125 on Range 5 of the IRMs. **INCORRECT,** IRMs are taken to Range 3 after mode switch is taken to run, per procedure IO-003, on range 5, readings would be downscale.

B: > 14% power as sensed by the APRMs. **CORRECT,** > 14% power as sensed by the APRMs- Mode switch placed in STARTUP/HOT STBY with power greater than 14%.

C: > 2 x 10^5 CPS as sensed by the SRMs. **INCORRECT,** SRMs are withdrawn and less than the scram setpoint.

D: Rx Mode switch in the shutdown position. **INCORRECT,** 14% power as sensed by the APRMs- Mode switch placed in STARTUP/HOT STBY with power greater than 14%. Even though a back-up scram signal would be generated in the Shutdown position the Up-scale (>14%) signal would generated first.0

Technical Reference(s): OP-SO.SE-0001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RPS000E004, From memory, identify the parameters which initiate a Reactor Scram, list the scram initiation setpoints for each identified parameter, and determine when the parameter is bypassed. (As available)

Question Source: Bank # 56367

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6, 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	259002	2.1.30	
	Importance Rating	4.4		

Conduct of Operations: Ability to locate and operate components, including local controls. (Reactor Water Level Control)

Question: RO #21

Given:

- Reactor power is 98%
- Plant conditions are normal
- All three feed pumps are in service
- OHA B3-F3 'RFP TURBINE AUTO XFR TO MANUAL' has alarmed
- The overhead alarm (B3-F3) is associated with the 1AP101 'A' RFPT
- The 1AP101 'A' RFPT has had a complete control signal failure

Select how speed for the affected RFP can be controlled.

A. The speed can ONLY be varied locally (1AC-132).

B. Can be immediately controlled in Manual using the PDS.

C. The speed can be varied using the INC SPEED and DEC SPEED buttons.

D. Can be controlled manually using the PDS following a 30 second time delay.

Proposed Answer: C

Explanation (Optional):

A: The speed can ONLY be varied locally. **INCORRECT,** the INC SPEED and DEC SPEED pushbuttons on 10C651B allow speed control from the Main Control Room.

B: Can be immediately controlled in Manual using the PDS. **INCORRECT,** the PDS is not available with a Control Signal Failure condition.

C: The speed can be varied using the INC SPEED and DEC SPEED buttons. **CORRECT**, RFP TURBINE AUTO XFER TO MANUAL occurs if a RFP experiences a Control Signal Failure, or following an RRCS Runback (ATWS). Since plant conditions are normal (given conditions), an ATWS/RRCS Runback can be ruled out. In the case of a Control Signal Failure, the affected RFP can only be controlled using the INC SPEED and DEC SPEED pushbuttons.

D: Can be controlled manually using the PDS following a 30 second time delay. **INCORRECT,** the PDS is not available with a Control Signal Failure condition.

Technical Reference(s): OP-AB.RPV-004; OP-AR.ZZ-0007, Attachment F3. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: FWCONTE011, From memory, describe the response of the respective RFPT if it senses a Control Signal Failure, IAW Available Control Room References. (As available)

Question Source: Bank # 53564

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	211000	2.1.32	
	Importance Rating	3.8		

Ability to explain and apply system limits and precautions. (SLC)

Question: RO #22

IAW HC.OP-IS.BH-0001, SLC Pump AP208 Inservice Test, which of the following is a procedural precaution and why?

The improper sequence of valve operations when flushing the pump can...

A. result in dilution of the boron concentration in the SLC test tank.

B. introduce air into the suction piping and result in pump air binding.

C. result in dilution of the boron concentration in the pump discharge piping.

D. introduce air into the pump discharge piping resulting in possible relief valve lifting

Proposed Answer: B

Explanation (Optional):

A: result in dilution of the boron concentration in the SLC test tank. **INCORRECT**, IAW OP-IS.BH-001, step 3.1.6. "The sequence of valve operations when flushing the SLC Pump suction piping is important. They are in this order to preclude introducing air into the SLC Pump suction." The SLC test tank is normally only filled with demin water and dilution is not a concern.

B: introduce air into the suction piping and result in pump air binding. **CORRECT**, IAW OP-IS.BH-001, step 3.1.6. "The sequence of valve operations when flushing the SLC Pump suction piping is important. They are in this order to preclude introducing air into the SLC Pump suction."

C: result in dilution of the boron concentration in the pump discharge piping. **INCORRECT,** IAW OP-IS.BH-001, step 3.1.6. "The sequence of valve operations when flushing the SLC Pump suction piping is important. They are in this order to preclude introducing air into the SLC Pump suction." The discharge piping is normally filled with demin water and would only have boron solution if the pumps had been in-service drawing suction from the main SLC tank.

D: introduce air into the pump discharge piping resulting in possible relief valve lifting. **INCORRECT**, IAW OP-IS.BH-001, step 3.1.6. "The sequence of valve operations when flushing

the SLC Pump suction piping is important. They are in this order to preclude introducing air into the SLC Pump suction." The SLC pumps are positive displacement pumps and would pump the air out of the discharge piping and into the RPV. The relief valves would not open as the air would be compressed by the pumps and not cause the relief valves to lift.

Technical Reference(s): OP-IS.BH-0001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: SLCSYSE003, Given P&ID's determine the following flowpaths for the Standby Liquid Control System. a. Standby Liquid Control System Injection flowpath, b. Standby lineup c. Standby Liquid Control System testing flowpath (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 1, 5

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	212000	2.4.11	
	Importance Rating	4.0		

Emergency Procedures / Plan: Knowledge of abnormal condition procedures. (RPS)

Question: RO #23

Given:

- The Reactor is operating at 20% power.
- Main Generator output is ≈ 256 MWe.
- Main Turbine 1st Stage Pressure is 75.8 psig.
- RPS Bus 'B' is de-energized due to a ground fault
- Crew has entered OP-AB.IC-003, Reactor Protection System

Then:

- The main turbine trips due to failed reactor level 8 logic inputs.

Which of the following describes the plant's response and the required actions?

A. Both Reactor Recirc Pumps trip, Lock the Mode Switch in shutdown and enter EOP-101, RPV Control.

B. ONLY the 'B' Reactor Recirc Pump trips, enter OP-AB.RPV-003, Recirculation System/Power Oscillations.

C. Main Turbine bypass valves cycle open on the initial pressure transient, enter EOP-101A, ATWS RPV Control on the failure to scram.

D. ONLY the reactor scrams, Lock the Mode Switch in shutdown enter EOP-101 RPV Control

Proposed Answer: A

Explanation (Optional):

A: Both Reactor Recirc Pumps trip, Lock the Mode Switch in shutdown and enter EOP-101,

RPV Control. **CORRECT,** A precaution in HC.OP-SO.SB-0001 warns that transfer of an RPS power supply will result in EOC-RPT actuation and a recirc pump trip if the Turbine Stop valves are closed. This is due to the momentary loss of power to the RPS bus during the transfer. The only way to prevent this is to bypass the EOC-RPT trip with the Recirc Pump Trip System Disable switches. IAW OP-IO.ZZ-0003, the Recirc Pump Trip System Disable switches are placed in NORMAL immediately after synchronizing and loading the Main Turbine, and prior to placing feedwater in Single Element control on the Master level controller. The initial conditions for this question have the plant at a point where the switches would already be in NORMAL. EOP-101 entry is required after LOCKING the mode switch in shutdown due to having NO Rx Recirc pumps running and the reactor is critical, per immediate operator action of OP-AB.RPV-003

B: ONLY the 'B' Reactor Recirc Pump trips, enter OP-AB.RPV-003, Recirculation System/Power Oscillations. **INCORRECT,** A precaution in HC.OP-SO.SB-0001 warns that transfer of an RPS power supply will result in EOC-RPT actuation and a recirc pump trip if the Turbine Stop valves are closed. This is due to the momentary loss of power to the RPS bus during the transfer. The only way to prevent this is to bypass the EOC-RPT trip with the Recirc Pump Trip System Disable switches. IAW HC.OP-IO.ZZ-0003, the Recirc Pump Trip System Disable switches are placed in NORMAL immediately after synchronizing and loading the Main Turbine, and prior to placing feedwater in Single Element control on the Master level controller. The initial conditions for this question have the plant at a point where the switches would already be in NORMAL.

C: Main Turbine bypass valves cycle open on the initial pressure transient, enter EOP-101A, ATWS RPV Control on the failure to scram. **INCORRECT,** While the main turbine bypass valves will initially cycle open on the trip of the turbine to maintain reactor pressure stable, the reactor will not scram, this is because there is NO scram signal present. No need to enter the ATWS EOP for a failure to scram, there was no scram signal to fail.

D: Only the reactor scrams, Lock the Mode Switch in shutdown enter EOP-101 RPV Control, **INCORRECT,** the reactor will not scram, this is because there is NO scram signal present. EOP-101 entry is required after LOCKING the mode switch in shutdown due to having No Recirc pumps running and the reactor is critical, per immediate operator action of OP-AB.RPV-003

Technical Reference(s): HC.OP-SO.SB-001, PN1-C71-1020-006 Shts 9/11/13/15, OP-IO.ZZ-003, OP-AB.RPV-003 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RPS000E014, Given labeled diagrams/drawings of the RPS trip logics, explain the coincidence requirements necessary to generate a reactor scram. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	6, 7
	55.43	

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	264000	A4.04	
	Importance Rating	3.7		

Ability to manually operate and/or monitor in the Control Room: Manual start, loading, and stopping of emergency generator: Plant Specific

Question: RO #24

Given:

- An NCO attempts to manually start the B (1BG400) Emergency Diesel Generator (EDG) from the 10C651E panel.
- Ten (10) seconds after the pushbutton on 10C651E was depressed the EDG speed is only 100 rpm.
- DIESEL ENG PNL A/B/C/D C423 alarm is received.

Which of the following describes the current status of the B EDG?

The B EDG starting air solenoid valves are ____________________.

A. closed because the B EDG has successfully started.

B. closed and the B EDG will continue to roll down and stop.

C. open because the start signal is still present and speed has NOT exceeded 125 rpm.

D. open and the B EDG will continue to roll for another 5 seconds unless its speed reaches 125 rpm.

Proposed Answer: B

Explanation (Optional):

A: closed because the B EDG has successfully started. **INCORRECT,** The Start Failure Relay will have actuated. The EDG will stop.

B: closed and the B EDG will continue to roll down and stop. **CORRECT,** The start failure circuit will close the air start valves 7 seconds after the start signal. With the engine running >40 RPM but less than 125, the Start Failure Crankshaft Rotating alarm will be in. The air start circuits are disabled until the Engine Shutdown relays are reset.

C: open because the start signal is still present and speed has NOT exceeded 125 rpm. **INCORRECT,** The Start Failure Relay will have actuated. The start valves will be closed until the SFR is manually reset.

D: open and the B EDG will continue to roll for another 5 seconds unless its speed reaches 125 rpm. **INCORRECT**, The start valves will be closed until the SFR is manually reset. The Start Failure Relay will have actuated 7 seconds after the start signal if rpm not >125.

Technical Reference(s): NOH04EDG000C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EDG000E007, Given plant conditions, determine if the Diesel Generator will trip under manual and/or automatic start conditions. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7, 8

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	239002	A3.01	
	Importance Rating	3.8		

Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: SRV operation after ADS actuation

Question: RO #25

Given:

- A transient has occurred causing the plant to scram
- An ADS actuation was required IAW EOP-202 Emergency RPV Depressurization
- All five of the ADS SRVs were NOT able to be opened
- CRIDS is unavailable.

During the post-transient panel walkdown, how can the operator validate which of the ADS SRVs did and did NOT open?

A. Its Acoustic Monitor is placed in the "T" mode

B. Its Acoustic Monitor is placed in the "C" mode

C. The Tailpipe Temperature recorder is placed in "DATA" mode

D. The Tailpipe Temperature recorder is placed in "COUNT" mode

Proposed Answer: B

Explanation (Optional):

A: Its Acoustic Monitor is placed in the "T" mode, **INCORRECT,** this switch position will display in the LED window, the current setpoint in decibels (Threshold) at which the Acoustic Monitor would indicate that the chosen SRV is open.

B: Its Acoustic Monitor is placed in the "C" mode, **CORRECT,** this switch position will display in the LED window, the number of times (Counts) the Acoustic Monitor would indicate that the chosen SRV has exceeded the threshold value, since the last time the counter was reset.

C: The Tailpipe Temperature recorder is placed in "DATA" mode, **INCORRECT,** the temperature recorder will display/record the temperature of each SRV tailpipe. Because it takes time for the

tailpipe temperature to lower once an SRV has closed, the recorder is not suitable to display opening/closing cycles

D: The Tailpipe Temperature recorder is placed in "COUNT" mode, **INCORRECT,** the temperature recorder will display/record the temperature of each SRV tailpipe. Because it takes time for the tailpipe temperature to lower once an SRV has closed, the recorder is not suitable to display opening/closing cycles

Technical Reference(s): NOH01ADSSYSC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ADSSYSE004, Given a labeled diagram/drawing of, or access to, the Automatic Depressurization System controls/indication bezel, IAW available Control Room references: a. Explain the function of each indicator. b. Assess plant conditions which will cause the indicator to light or extinguish. (As available)

Question Source: Bank #

Modified Bank # 55821 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7, 8

55.43

Comments:

Following an MSIV isolation/scram from 100% power, the Lo-Lo Set SRV's cycled several times. CRIDS is unavailable. How does the operator know how many times each SRV opened?

A. SRV Accoustic Monitor is placed in the "C" mode

B. SRV Accoustic Monitor is placed in the "T" mode

C. SRV Tailpipe Temp recorder is placed in "DATA" mode

D. SRV Tailpipe Temp recorder is placed in "RECORD" mode

Answer: A

Associated objective(s):

Audit9_99 Audit Exam 9/99 Raw

Question 1 Details

Question Type:	Multiple Choice
Topic:	AUTOMATIC DEPRESSURIZATION SYSTEM - Plant-Specific
System ID:	30621
User ID:	Q55821
Status:	Active
Must Appear:	No
Difficulty:	0.00
Time to Complete:	3
Point Value:	1.00
Cross Reference:	Audit Exam 9/99
User Text:	218000A3.03
User Number 1:	3.70
User Number 2:	3.80
Comment:	Audit Exam 9/99 Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: ADS valve acoustical monitor noise: Plant-Specific

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 K3.04	
	Importance Rating	3.6	

Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Adequate core cooling

Question: RO #26

The Unit has experienced a trip of all feed pumps from 100% power.

- HPCI is inoperable.
- RCIC and CRD are in operation.
- Reactor water level is -150 inches and slowly lowering.

RCIC is operating in AUTOMATIC with the following control board indications:

- Pump Suction Pressure 15 psig
- Pump Discharge Pressure 225 psig
- Turbine Inlet Pressure 910 psig
- Turbine Exhaust Pressure 10 psig
- Turbine Speed 2200 rpm
- Flow controller setting 600 gpm

Given these plant conditions, which ONE of these actions is required for RCIC, including the reason?

A. IMMEDIATELY secure RCIC. RCIC is not injecting into the Rx Vessel.

B. Continue to run RCIC. Raise discharge pressure by throttling the discharge valve to restore RPV level.

C. Continue to run RCIC. Raise turbine speed by placing the flow controller in MANUAL and adjusting the flow controller to restore RPV level.

D. IMMEDIATELY secure RCIC. The low suction pressure trip failed.

Proposed Answer: C

Explanation (Optional):

A: IMMEDIATELY secure RCIC. RCIC is not injecting into the Rx Vessel. **INCORRECT**, Discharge pressure is not sufficient to overcome reactor pressure, however the ability to take manual control of the system to raise discharge pressure still exists. No immediate need to secure RCIC as pump speed is above the minimum for oil pressure and exhaust line check valve concerns.

B: Continue to run RCIC. Raise discharge pressure by throttling the discharge valve to restore RPV level. **INCORRECT,** Discharge pressure is not sufficient to overcome reactor pressure, however the ability to take manual control of the system to raise discharge pressure still exists. If the discharge valve is throttled, normally the system in auto will raise speed to compensate for the reduced flow rate, however, flow setting is set at 600, so the flow controller in automatic is not functioning correctly and manual control is required to raise discharge flow.

C: Continue to run RCIC. Raise turbine speed by placing the flow controller in MANUAL and adjusting the flow controller to restore RPV level. **CORRECT,** Discharge pressure is not sufficient to overcome reactor pressure, however the ability to take manual control of the system to raise discharge pressure still exists. If the discharge valve is throttled, normally the system in auto will raise speed to compensate for the reduced flow rate, however, flow setting is set at 600, so the flow controller in automatic is not functioning correctly and manual control is required to raise discharge flow. With vessel level lowering and HPCI and feed water not available, every reasonable attempt to inject with RCIC should be taken.

D: IMMEDIATELY secure RCIC. The low suction pressure trip failed. **INCORRECT,** no indication of a system trip exists, also suction pressure is 15 psig, well above the trip setpoint of < 20" Hg VAC 2 second td

Technical Reference(s): OP-SO.BD-001, Section 3.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RCIC00E022, Given RCIC turbine control system failures, evaluate and determine the effect on the RCIC system. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	215001	K1.08	
	Importance Rating	2.5		

Knowledge of the physical connections and/or cause- effect relationships between TRAVERSING IN-CORE PROBE and the following: Reactor pressure vessel: (Not-BWR1)

Question: RO #27

- A TIP System trace is being performed by the On-duty Reactor Engineer
- An 'A' Channel NSSSS manual isolation signal is received

In an attempt to isolate the related RPV penetrations, what is the TIP system response (if any)?

A. No automatic actions occur when only an 'A' Channel NSSSS isolation signal is received.

B. The TIP Shear Valve automatically fires to cut the detector cable and seal the guide tube.

C. The TIP Guide Tube Ball Valve automatically closes, cutting the detector cable and sealing the guide tube.

D. The TIP detectors not in the "in-shield" position will automatically withdraw to their "in-shield" position and the TIP Guide Tube Ball Valves automatically close.

Proposed Answer: D

Explanation (Optional):

A: No automatic actions occur when only an 'A' Channel NSSSS isolation signal is received. **INCORRECT,** manual initiation of NSSSS Channel "A" will cause isolation of affected systems, including TIP. Per OP-SO.SM-001

B: The TIP Shear Valve automatically fires to cut the detector cable and seal the guide tube. **INCORRECT,** the Shear Valves must be manually initiated.

C: The TIP Guide Tube Ball Valve automatically closes, cutting the detector cable and sealing the guide tube. **INCORRECT -** the Ball Valve will not close with the cable inside the valve.

D: The TIP detectors not in the "in-shield" position will automatically withdraw to their "in-shield" position and the TIP Guide Tube Ball Valves automatically close. **CORRECT,** manual initiation of NSSSS Channel "A" will cause isolation of affected systems, including TIP.

Technical Reference(s): NOH01TIPS00. OP-SO.SM-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: TIPS00E006, From memory explain the response of the TIP System following the receipt of an isolation signal from the Nuclear Steam Supply Shutoff System. (As available)

Question Source: Bank # 53710

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 2, 6

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	286000	K2.02	
	Importance Rating	2.9		

Knowledge of electrical power supplies to the following: Pumps (Fire Protection)

Question: RO #28

Given:

- The plant is operating at 100% power
- A loss of MCC 00B590 occurs.
- There is no fire.

Based on this, it will be necessary to:

A. secure the motor driven fire pump to prevent pumping down the fire water storage tank.

B. secure the diesel driven fire pump to prevent pumping down the fire water storage tank.

C. manually start the motor driven fire pump due to a loss of the diesel driven fire pump battery chargers.

D. manually start the diesel fire pump due to a loss of the motor driven fire pump power supply.

Proposed Answer: B

Explanation (Optional):

A: secure the motor driven fire pump to prevent pumping down the fire water storage tank, **INCORRECT,** with the loss of the 00B590 MCC the motor driven fire pump has no power supply

B: secure the diesel driven fire pump to prevent pumping down the fire water storage tank, **CORRECT,** -The diesel driven fire pump auto starts on a loss of power to the motor driven fire pump. Since there is no fire, the fire water storage tank is being pumped down due to the

discharge relief valve lifting with no system flow demand.

C: manually start the electric fire pump due to a loss of the diesel driven fire pump battery chargers, **INCORRECT,** The battery chargers for the diesel driven fire pump are powered from 1AJ483

D: manually start the diesel fire pump due to a loss of the motor driven fire pump power supply, **INCORRECT,** - The diesel driven fire pump auto starts on a loss of power to the motor driven fire pump.

Technical Reference(s): E-0013 sht 2, OP-AR.QK-0002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: FIRPROE008, Summarize the effect that a loss of power would have on the following: Electric Motor Driven Fire Pump. Diesel Driven Fire Pump. (As available)

Question Source: Bank # 111405

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 K3.02	
	Importance Rating	3.1	

Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on following: Reactor water level

Question: RO #29

Given:

- Core Alterations are in progress
- RHR Loop "A" is in Shutdown Cooling
- RPV Coolant Temperature is 100° F
- RHR Loop "B" is in Fuel Pool Cooling Assist
- Both Fuel Pool Cooling Pumps are C/T for repairs
- RWCU is in service with blowdown flow at 25 gpm for level control

Then:

- A controller malfunction causes blowdown valve (BG-HV-F033) to stroke full open.

With NO Operation action taken, what will be the affect from the conditions above?

A. Cavity level will continue to lower until isolated when level reaches -38".

B. Cavity level will continue to lower until isolated when level reaches -129".

C. Cavity level will lower until the weirs are uncovered, then will remain at the level of the weirs.

D. Skimmer Surge Tank level will lower until the weirs are uncovered then will remain at the level of the weirs.

Proposed Answer: A

Explanation (Optional):

A: Cavity level will continue to lower until isolated when level reaches -38". **CORRECT,** RWCU takes a suction from the bottom head and receives an isolation signal at -38". This isolation is not part of the SDC NSSSS isolations that are defeated per IO-5.

B: Cavity level will continue to lower until isolated when level reaches -129". **INCORRECT,** RWCU takes a suction from the bottom head and receives an isolation signal at -38". This isolation is not part of the SDC NSSSS isolations that are defeated per IO-5.

C: Cavity level will lower until the weirs are uncovered, then will remain at the level of the weirs. **INCORRECT,** The mass loss is from the RPV. Cavity weirs only direct water to the Skimmer Surge Tank.

D: Skimmer Surge Tank level will lower until the weirs are uncovered then will remain at the level of the weirs. **INCORRECT,** The Skimmer Surge Tank receives water from the weirs. When level drops below the weirs, tank level will continue to lower and be maintained by makeup from CST.

Technical Reference(s): OP-SO.BG-001, OP-IO.ZZ-005, OP-GP.SM-001, M-44-1 Sht 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RWCU00E018, Given a set of plant conditions evaluate the effects of RWCU System blowdown operation on the RHX and NRHX's IAW the RWCU System Lesson Plan. (As available)

Question Source: Bank # 56398

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 3, 7

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	202002	K4.06	
	Importance Rating	3.1		

Knowledge of RECIRCULATION FLOW CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Recirculation pump adequate NPSH: Plant-Specific

Question: RO #30

Given :

- The plant was operating at 98% power when a pressure transient occurred, causing RPV pressure to peak at 1050 psig.
- Reactor pressure is now under control with H SRV cycling.
- RPV level reached -20 inches and is recovering to normal.
- NO operator actions have been taken.

Which of the following correctly describes the status of the Reactor Recirculation Pumps for the current conditions?

Both Recirculation Pumps:

A. have tripped off.

B. are running at 30 % speed.

C. are running at 45 % speed.

D. are running at the electrical low speed stops.

Proposed Answer: B

Explanation (Optional):

A: have tripped off. **INCORRECT,** RPV pressure reached 1050 psig NOT 1071 psig. RPV level reached -20 inches NOT -38". Recirc pump RPT Breakers trip at 1071 psig/ -38".

B: are running at 30 % speed. **CORRECT,** 1050 psig will cause reactor scram but not ATWS-RPT actuation at 1071. 30 % speed limiter (full runback) is sealed-in.

C: are running at 45 % speed. **INCORRECT,** 1050 psig will cause reactor scram but not ATWS-RPT actuation at 1071. 30 % speed limiter (full runback) is sealed-in.

D: are running at the electrical low speed stops. **INCORRECT,** 1050 psig will cause reactor scram but not ATWS-RPT actuation at 1071. 30 % speed limiter (full runback) is sealed-in. Manual operator action needed to reduce to minimum speed.

Technical Reference(s): NOH01RECCON, OP-SO.BB-002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RECCONE015 , Explain the purpose of each recirc pump runback and list the signals that will generate each runback IAW available control room references (As available)

Question Source: Bank # 55914

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6, 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	241000 K5.05	
	Importance Rating	2.8	

Knowledge of the operational Implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM : Turbine inlet pressure vs. reactor pressure

Question: RO #31

Which one of the following describes the pressure rise at the Main Turbine inlet and reactor steam dome as power is increased from main generator synchronization to full rated thermal power?

Assume turbine loading is at a constant ramp rate.

Main Turbine inlet pressure rise is ____________ and reactor steam dome pressure rise is ____________.

A. Linear; Linear.

B. Linear; Non-Linear.

C. Non-Linear; Linear.

D. Non-Linear; Non-Linear.

Proposed Answer: B

Explanation (Optional):

A: Linear; Linear. **INCORRECT,** Main turbine inlet pressure rise is linear. Reactor pressure rise is not a direct linear correlation, it is non-linear.

B: Linear; Non-Linear. **CORRECT,** Linear; Non-Linear. Correct. Main turbine inlet pressure rises from 905 to 935 psig at 3.33% steam flow to 1psig rise. Rx pressure rises from 905 to 1005 psig. Reactor pressure rises higher due to the increased differential pressure caused by steam line flow increases.

C: Non-Linear; Linear. **INCORRECT,** Main turbine inlet pressure rise is linear. Reactor pressure rise is not a direct linear correlation, it is non-linear.

D: Non-Linear; Non-Linear. **INCORRECT,** Main turbine inlet pressure rise is linear. Reactor pressure rise is not a direct linear correlation, it is non-linear.

Technical Reference(s): NOH01EHCLOG (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EHCLOGE002, Given plant conditions evaluate the cause-effect relationship between the pressure regulating system and the following: Reactor Power, Reactor Pressure, Steam Flow, Reactor Water Level (As available)

Question Source: Bank # 80590

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2003

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 7,14

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	201002	K6.01	
	Importance Rating	2.5		

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR MANUAL CONTROL SYSTEM : Select matrix power

Question: RO #32

Given:

- The plant is performing the control rod exercise surveillance
- The Nuclear Control Operator (RO) selects control rod 34-19 on the rod select matrix
- Only one half of the selected rod push button illuminates when depressed
- The 'ACTIVITY CONTROLS DISAGREE" light is illuminated

Which of the following describes what has failed and how that affects the ability to move control rods?

A. The selected control rod activity control card is in the scan mode and rod motion is allowed.

B. The selected control rod activity control card is in the scan mode and rod motion is NOT allowed.

C. Only one of the two RMCS transmitter cards has successfully selected the control rod and rod motion is allowed.

D. Only one of the two RMCS transmitter cards has successfully selected the control rod and rod motion is NOT allowed.

Proposed Answer: D

Explanation (Optional):

A: The selected control rod activity control card is in the scan mode and rod motion is allowed. **INCORRECT**, this is a transmitter card failure

B: The selected control rod activity control card is in the scan mode and rod motion is NOT allowed. **INCORRECT**, this is a transmitter card failure

C: Only one of the two RMCS transmitter cards has successfully selected the control rod and rod motion is allowed. **INCORRECT**, this is a transmitter card failure

D: Only one of the two RMCS transmitter cards has successfully selected the control rod and rod motion is NOT allowed. **CORRECT**, this is a transmitter card failure, the ACTIVITY CONTROLS DISAGREE will prevent rod motion. Only one of the two RMCS transmitter cards has successfully selected the control rod and rod motion is not allowed.

Technical Reference(s): OP-SO.SF-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: MANCONE003, Given a labeled diagram/drawing of, or access to, the Reactor Manual Control System controls/indication bezel, summarize the following: The function of each indicator. The condition which will cause the indicator to light or extinguish. The effect of each control on the Reactor Manual Control System. The conditions or permissives required for the control switches to perform their intended function. (As available)

Question Source: Bank # 53362

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2, 6

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	239001	A1.08	
	Importance Rating	3.8		

Ability to predict and/or monitor changes in parameters associated with operating the MAIN AND REHEAT STEAM SYSTEM controls including: Reactor pressure

Question: RO #33

During a reactor plant startup Main Turbine Sealing Steam will automatically transfer from Auxiliary Steam at approximately 60 psig reactor pressure.

This is accomplished by:

A. 4th stage extraction becoming available to the Steam Seal Evaporator.

B. The Auxiliary Steam Header PCV (PV-2038) closing as the Steam Seal Evaporator feeds the sealing header.

C. Flow from the turbine valve stem leak-off (MSV, TCV, CIV, and Bypass Valves) becoming sufficient to supply the seal header.

D. Turbine seals being supplied by Main Steam via H. P. turbine seal leak-off as the HP Turbine Shell pressurizes during shell warming (60-100 psig)

Proposed Answer: B

Explanation (Optional):

A: 4th stage extraction becoming available to the Steam Seal Evaporator. **INCORRECT,** extraction steam comes from the #4 feed water heater which gets supplied from the 8th stage low pressure steam as shown on P&IDS M-01/M-02/M-29

B: The Auxiliary Steam Header PCV (PV-2038) closing as the Steam Seal Evaporator feeds the sealing header. **CORRECT**, Per IOP-003 step 5.3.22, "Main Turbine Sealing Steam will automatically transfer from Auxiliary Steam to Main Steam at approximately 60 psig."

C: Flow from the turbine valve stem leak-off (MSV, TCV, CIV, and Bypass Valves) becoming sufficient to supply the seal header. **INCORRECT,** steam seal leak-off is directed back to the main condenser as shown on P&IDS M-01/M-29

D: Turbine seals being supplied by Main Steam via H. P. turbine seal leak-off as the HP Turbine Shell pressurizes during shell warming (60-100 psig) **INCORRECT,** seal leak-off is directed either back to the main condenser, to the #3 feed water heaters, or the steam packing exhauster NOT to the sealing steam header as shown on P&ID M-29

Technical Reference(s): OP-IO.ZZ-003, NOH01SEALSTC, P&IDS M-01/M-02/M-29 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: SEALSTE006, Given plant conditions, explain the sources of evaporator heating steam. (As available)

Question Source: Bank # 55017

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	288000 A2.05	
	Importance Rating	2.6	

Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Extreme outside weather conditions: Plant-Specific

Question: RO #34

Given:

- The plant is operating at 100% power.
- RCIC has just been placed in service for a quarterly surveillance.
- Outside air temperature is 5°F.

Then:

- Boiler operator reports a trip of the Auxiliary Boiler.
- Heating steam header pressure is dropping rapidly.

Roughly 10 minutes later, the Plant operator reports:

- OHA B1A1 RCIC Turbine Trip in alarm
- FC-HV-F007 Inboard Steam Supply Isolation valve going closed.
- FC-HV-F008 Outboard Steam Supply Isolation valve going closed.

(1) What is the cause of the present RCIC condition AND (2) what actions are required ?

A. (1) RCIC Pump Room High Temperature. (2) Declare RCIC INOP AND be in at least HOT SHUTDOWN within the next 12 hours ONLY.

B. (1) RCIC Pump Room High Temperature. (2) Declare RCIC INOP AND restore RCIC to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours ONLY.

C. (1) RCIC Room Ventilation Duct High Differential Temperature. (2) Enter AB.MISC-003, Loss of Auxiliary Steam and consider securing reactor building ventilation and place the RCIC and HPCI Room Coolers in service AND monitor outside air temperatures to ensure adequate margin to prevent freezing heating/cooling coils

D. (1) RCIC Room Ventilation Duct High Differential Temperature. (2) Enter AB.MISC-003, Loss of Auxiliary Steam and secure reactor building ventilation and place FRVS in service and place the RCIC and HPCI Room Coolers in service.

Proposed Answer: D

Explanation (Optional):

A: (1) RCIC Pump Room High Temperature. (2) Declare RCIC INOP AND be in at least HOT SHUTDOWN within the next 12 hours ONLY. **INCORRECT**, No other OHAs are in that support this condition. RCIC Tech Spec is incomplete. Also requires entry into AB.MISC-003 Loss of Auxiliary Steam

B: (1) RCIC Pump Room High Temperature. (2) Declare RCIC INOP AND restore RCIC to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours ONLY. **INCORRECT,** No other OHAs are in that support this condition. RCIC Tech Spec is incomplete. Also requires entry into AB.MISC-003 Loss of Auxiliary Steam

C: (1) RCIC Room Ventilation Duct High Differential Temperature. (2) Enter AB.MISC-0003, Loss of Auxiliary Steam and consider securing reactor building ventilation and place the RCIC and HPCI Room Coolers in service AND monitor outside air temperatures to ensure adequate margin to prevent freezing heating/cooling coils. **INCORRECT,** During boiler trips and subsequent loss of heating steam to RBVS the reactor building supply temperatures will lower. The reduction in supply temperature may cause the differential temperature isolations for HPCI and RCIC to be challenged. RCIC system trips on any isolation signal and high differential temperature for RCIC

room ventilation can/does occur with low outside air temperature and elevated room temperatures (RCIC in service for surveillance), Also requires entry into AB.MISC-003 Loss of Auxiliary Steam, Condition A for temperature < 40°F. The actions listed are for Condition B for temperatures above 40°F.

D: RCIC Room Ventilation Duct High Differential temperature. **CORRECT,** During boiler trips and subsequent loss of heating steam to RBVS the reactor building supply temperatures will lower. The reduction in supply temperature may cause the differential temperature isolations for HPCI and RCIC to be challenged. RCIC system trips on any isolation signal and high differential temperature for RCIC room ventilation can/does occur with low outside air temperature and elevated room temperatures (RCIC in service for surveillance), requires entry into AB.MISC-003 Loss of Auxiliary Steam, Condition A, for temperature < 40°F.

Technical Reference(s): HC.OP-AB.MISC-003, Note 3. NOH04RCIC00, (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ABMSC3E004, Explain the reasons for how plant/system parameters respond when implementing Loss of Auxiliary Steam. (As available)

Question Source: Bank #

Modified Bank # 118792 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	201001	A3.03	
	Importance Rating	2.7		

Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: System pressure

Question: RO #35

The CRD system is in operation with the reactor operating at 43% power. The CRD flow control valve is in automatic. The RO takes the Drive Water Pressure Control Throttle Valve (HV-F003) to open for two seconds.

Which one of the following describes how parameters will stabilize when the transient is over?

A. Drive water Δp will rise; Cooling water flow will lower.

B. Drive water Δp will rise; Cooling water flow will remain the same.

C. Drive water Δp will lower; Cooling water flow will lower.

D. Drive water Δp will lower; Cooling water flow will remain the same.

Proposed Answer: D

Explanation (Optional):

A: Drive water Δp will rise; Cooling water flow will lower. **INCORRECT**, Δp will lower as the F003 (PCV) is opened and the FCV will restore flow to same value system flow should remain constant

B: Drive water Δp will rise; Cooling water flow will remain the same. **INCORRECT,** Δp will lower as the F003 (PCV) is opened

C: Drive water Δp will lower; Cooling water flow will lower. **INCORRECT,** the FCV will restore flow to same value system flow should remain constant

D: Drive water Δp will lower; Cooling water flow will remain the same. **CORRECT,** as the F003 (PCV) will lower Δp, and initially raise flow. The FCV will respond and restore normal flow to the system.

Technical Reference(s):	NOH04CRDYD	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective:	CRDHYDE007, Given a drawing, explain how drive/cooling water differential pressure is controlled.	(As available)

Question Source:	Bank #	53508	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:

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

10 CFR Part 55 Content:	55.41	2, 6
	55.43	

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	226001	A4.15	
	Importance Rating	3.6		

Ability to manually operate and/or monitor in the control room: Suppression chamber pressure: Mark-I-II (RHR/LPCI: Containment Spray Mode)

Question: RO #36

During a transient, Suppression Chamber Spray was initiated due to:

- Suppression Chamber Pressure 9.0 psig
- Suppression Pool Level 102"
- Drywell Pressure 7 psig
- Drywell Temperature 228°F

Conditions degrade and the order to place Drywell Spray in service has been given to the RO.

(1) What valves must be realigned to get into Drywell Spray AND (2) when is Drywell Spray required to be secured?

A. (1) Close the HV-F024 Loop Test Return valve AND HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve AND HV-F021 Inboard Containment Spray valve. (2) When drywell pressure approaches 0 psig.

B. (1) Close the HV-F024 Loop Test Return valve OR HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve OR HV-F021 Inboard Containment Spray valve. (2) When drywell pressure approaches 0 psig.

C. (1) Close the HV-F024 Loop Test Return valve AND HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve AND HV-F021 Inboard Containment Spray valve. (2) When suppression chamber pressure approaches 0 psig.

D. (1) Close the HV-F024 Loop Test Return valve OR HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve OR HV-F021 Inboard Containment Spray valve. (2) When suppression chamber pressure approaches 0 psig.

Proposed Answer: A

Explanation (Optional):

A: Close the HV-F024 Loop Test Return valve AND HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve AND HV-F021 Outboard Containment Spray valve. When drywell pressure approaches 0 psig. **CORRECT,** per OP-AB.ZZ-0001 Att 2. to get into drywell spray would require ensuring F024 and F027 are closed and then open both the F016 and F021, per EOP-102 retainment override PCC-1, IF: Drywell sprays have been initiated, THEN BEFORE drywell pressure reaches 0 psig, TERMINATE drywell sprays.

B: Close the HV-F024 Loop Test Return valve OR HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve OR HV-F021 Outboard Containment Spray valve. When drywell pressure approaches 0 psig. **INCORRECT,** per OP-AB.ZZ-0001 Att 2. to get into drywell spray would require ensuring F024 and F027 are closed and then open BOTH the F016 and F021, per EOP-102 retainment override PCC-1, IF: Drywell sprays have been initiated, THEN BEFORE drywell pressure reaches 0 psig, TERMINATE drywell sprays.

C: Close the HV-F024 Loop Test Return valve AND HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve AND HV-F021 Outboard Containment Spray valve. When suppression chamber pressure approaches 0 psig. **INCORRECT,** per OP-AB.ZZ-0001 Att 2. to get into drywell spray would require ensuring F024 and F027 are closed and then open both the F016 and F021, per EOP-102 retainment override PCC-1, IF: Drywell sprays have been initiated, THEN BEFORE drywell pressure reaches 0 psig, TERMINATE drywell sprays. If suppression chamber sprays were still in service, then would terminate on suppression chamber pressure before 0 psig.

D: Close the HV-F024 Loop Test Return valve OR HV-F027 Suppression Chamber Spray valve, then open HV-F016 Outboard Containment Spray valve OR HV-F021 Outboard Containment Spray valve. When suppression chamber pressure approaches 0 psig. **INCORRECT,** per OP-AB.ZZ-0001 Att 2. to get into drywell spray would require ensuring F024 and F027 are closed and then open BOTH the F016 and F021, per EOP-102 retainment override PCC-1, IF: Drywell sprays have been initiated, THEN BEFORE drywell pressure reaches 0 psig, TERMINATE drywell sprays. If suppression chamber sprays were still in service, then would terminate on suppression chamber pressure before 0 psig.

Technical Reference(s): EOP-102, OP-AB.ZZ-001 Att. 2/3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RHRSYSE014, Given a copy/mimic of the RHR System controls on 10C650A, predict proper RHR System response during the LPCI mode of operation to include the following, IAW available control room references:
f. Determine the operator actions required to initiate Torus/containment spray during LPCI mode of operation, IAW available control room references. (As available)

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

Question History:

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

10 CFR Part 55 Content:	55.41	7, 8
	55.43	

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	259001	2.4.2	
	Importance Rating	4.5		

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Question: RO #37

Given:

- The plant is starting up following a refueling outage IAW HC.OP-IO.ZZ-003
- Reactor Power is 13%
- The Mode Switch is in RUN
- Preparations are being made to synchronize the Main Generator
- A RFPT is in service feeding the vessel and B RFPT is on min flow, per OP-SO.AE-001
- A & B PCPs are in service
- A & B SCPs are in service
- The Main Turbine is @ 1800 rpm

A Bailey card malfunction causes AB-HV-1006 RFPT HP STM SPLY ISLN MOV to fail shut.

Since HV-1006 can NOT be reopened what actions (if any) are required?

A. NO additional actions are required because reactor vessel level lowers and returns to normal as feedwater flow is restored.

B. Reactor Operator LOCKS the mode switch in Shutdown because an entry condition exists for EOP-101 ONLY.

C. Reactor Operator LOCKS the mode switch in Shutdown because an entry condition exists for AB.ZZ-000 ONLY.

D. Reactor Operator LOCKS the mode switch in Shutdown because an entry condition exists for EOP-101 and AB.ZZ-000, with EOP-101 taking precedent.

Proposed Answer: D

Explanation (Optional):

A: (1) The 'A' RFPT governor valve ramps open in an attempt to maintain RFPT speed. Reactor vessel level lowers and returns to normal as feedwater flow is restored. (2) NO additional actions are required. **INCORRECT,** The 'A' RFPT governor valve ramps open in an attempt to maintain RFPT speed. Since no steam is available to the RFPT, feedwater flow is lost, and the Reactor scrams on low level.

B: (1) The 'A' RFPT governor valve ramps open in an attempt to maintain RFPT speed. Since no steam is available to the RFPT, feedwater flow is lost, and the Reactor Operator LOCKS the mode switch in Shutdown. (2) An entry condition exists for EOP-101 ONLY. **INCORRECT**, will also require entry into AB.ZZ-000 for the reactor scram, however EOP-101 takes precedent/priority

C: (1) The 'A' RFPT governor valve ramps open in an attempt to maintain RFPT speed. Since no steam is available to the RFPT, feedwater flow is lost, and the Reactor Operator LOCKS the mode switch in Shutdown. (2) An entry condition exists for AB.ZZ-000 ONLY. **INCORRECT,** The 'A' RFPT governor valve ramps open in an attempt to maintain RFPT speed. Since no steam is available to the RFPT, feedwater flow is lost, and the Reactor scrams on low level. Requires entry into AB.ZZ-000 for the reactor scram, and EOP-101, with EOP-101 taking priority.

D: (1) The 'A' RFPT governor valve ramps open in an attempt to maintain RFPT speed. Since no steam is available to the RFPT, feedwater flow is lost, and the Reactor Operator LOCKS the mode switch in Shutdown. (2) An entry condition exists for EOP-101 and AB.ZZ-000, with EOP-101 taking precedent. **CORRECT,** The 'A' RFPT governor valve ramps open in an attempt to maintain RFPT speed. Since no steam is available to the RFPT, feedwater flow is lost, and the reactor is scrammed on low level. Requires entry into AB.ZZ-000 for the reactor scram, and EOP-101, with EOP-101 taking priority.

Technical Reference(s): EOP-101, AB.ZZ-000 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: FEED00E018, From memory, summarize/identify the sources of steam to the RFPT's and when each source is used, IAW available Control Room References. EO101PE002, State the three entry conditions for the Reactor/Pressure Vessel (RPV) Control Emergency Operating Procedure. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7, 10
	55.43	

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000 A1.04	
	Importance Rating	2.6	

Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including: System venting

Question: RO #38

A Reactor cooldown/depressurization from normal operations is in progress.

Which RPV level indication below will have the LEAST accurate direct reading when the reactor depressurization is complete and why? (actual level –vs- indicated level)

A. Narrow Range, because it is calibrated hot and as the reactor depressurizes the disparity to actual level is greater when compared to the disparity of wide range.

B. Wide Range, because it is calibrated hot and as the reactor depressurizes the disparity to actual level is greater when compared to the disparity of narrow range.

C. Upset Range, because it is calibrated hot and as the reactor depressurizes the disparity to actual level is greater when compared to the disparity of narrow AND wide range.

D. Shutdown Range, because it is calibrated cold and as the reactor depressurizes the disparity to actual level is greater when compared to hot conditions.

Proposed Answer: B

Explanation (Optional):

A: Narrow Range, because it is calibrated hot and as the reactor depressurizes the disparity to actual level is greater when compared to the disparity of wide range. **INCORRECT,** Narrow Range is calibrated for saturated conditions at 1000 psig and as reactor pressure/temperature are reduced the disparity of actual level to indicated level widens, but not as much as Wide Range. See temperature compensation curves from IOP-0004, Att 7.

B: Wide Range, because it is calibrated hot and as the reactor depressurizes the disparity to actual level is greater when compared to the disparity of narrow range. **CORRECT**, Wide Range is calibrated for saturated conditions at 1000 psig and as reactor pressure/temperature are reduced

the disparity of actual level to indicated level widens much further than Narrow Range. See temperature compensation curves from IOP-0004, Att 7.

C: Upset Range, because it is calibrated hot and as the reactor depressurizes the disparity to actual level is greater when compared to the disparity of narrow AND wide range. **INCORRECT,** Upset Range is calibrated for saturated conditions at 1000 psig, however, due to it's scale 0-180", the disparity of actual level to indicated level during cooldown/depressurization is negligible. See temperature compensation curves from IOP-0004, Att 7.

D: Shutdown Range, because it is calibrated cold and as the reactor depressurizes the disparity to actual level is greater when compared to hot conditions. **INCORRECT,** Shutdown Range is calibrated for 120°F, 0 psig, whereas Wide Range is calibrated for saturated conditions at 1000 psig. See temperature compensation curves from IOP-0004, Att 7.

Technical Reference(s): NOH04RXINSTC, IOP-0004 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RXINSTE018, Given changes in the following parameters, evaluate the affect on each RPV level indication:
a. Reactor Pressure, b. Drywell Temperature, c. Steam Flow (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 14

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AK1.01	
	Importance Rating	2.5	

Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Classifications by type

Question: RO #39

An electrical fire has been reported in the 'A' Emergency Diesel Generator (EDG) room. The automatic fire suppression system actuated as designed. Fire response personnel are planning to enter the room to confirm that the fire is extinguished.

Which one of the following describes the fire class and the potential hazards associated with entering the room?

A. Class A; suffocation hazard due to CO_2 discharge.

B. Class B; electrocution hazard due to water deluge.

C. Class C; suffocation hazard due to CO_2 discharge.

D. Class C; electrocution hazard due to water deluge.

Proposed Answer: C

Explanation (Optional):

A: Class A; suffocation hazard due to CO_2 discharge. **INCORRECT**, Class A fire pertains to paper/ wood products and this is an electrical fire, which is a Class C fire.

B: Class B; electrocution hazard due to water deluge. **INCORRECT,** Because the actual generator and its control panels as well as fuel and lubricating oil are housed in the room, the room is NOT protected by water deluge. Class B fire pertains to flammable/combustible liquids. This is an electrical fire, which is a Class C fire.

C: Class C; suffocation hazard due to CO_2 discharge. **CORRECT,** Because the actual generator and its control panels as well as fuel and lubricating oil are housed in the room, the room is NOT protected by water deluge and is protected by CO_2. Suffocation is a major concern due to the ability of Cardox to displace oxygen in an enclosed space. This is an electrical fire, which is a

Class C fire.

D: Class C; electrocution hazard due to water deluge. **INCORRECT**, It is a Class C fire however, because the actual generator and its control panels as well as fuel and lubricating oil are housed in the room, the room is NOT protected by water deluge.

Technical Reference(s): NOH01FIRPRO (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: FIRPROE016, Describe fire detection and suppression provided for the Emergency Diesel Generator Rooms. (As available)

Question Source: Bank # 84220

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295006	AK1.02	
	Importance Rating	3.4		

Knowledge of the operational implications of the following concepts as they apply to SCRAM : Shutdown margin

Question: RO #40

Following a reactor scram all rods are at position "00" except one that is at position "24."

Which of the following describes the capability of the reactor to remain shutdown?

A. Design basis shutdown margin is met, therefore the reactor will remain shutdown under all conditions.

B. Design basis shutdown margin is NOT met, therefore it CANNOT be assured that the reactor will remain shutdown under all conditions.

C. Control rods are inserted to or beyond the Maximum Sub-critical Banked Withdrawal limit, therefore the reactor will remain shutdown under all conditions.

D. Control rods are NOT inserted to or beyond the Maximum Sub-critical Banked Withdrawal limit, therefore it CANNOT be assured the reactor will remain shutdown under all conditions.

Proposed Answer: A

Explanation (Optional):

A: Design basis shutdown margin is met, therefore the reactor will remain shutdown under all conditions. **CORRECT**, Design basis shutdown margin is met, therefore the reactor will remain shutdown under all conditions. IAW Tech Specs definition, “SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F and xenon free” and HC.OP-EO.ZZ-0101 Bases step RC-1 retention Override page 3.

B: Design basis shutdown margin is NOT met, therefore it CANNOT be assured that the reactor will remain shutdown under all conditions. **INCORRECT**, Design basis shutdown margin is met, therefore the reactor will remain shutdown under all conditions. IAW Tech Specs definition and

HC.OP-EO.ZZ-0101 Bases step RC-1 retention Override page 3.

C: Control rods are inserted to or beyond the Maximum Sub-critical Banked Withdrawal limit, therefore the reactor will remain shutdown under all conditions. **INCORRECT,** Maximum Sub-critical Banked Withdrawal Position is defined as: "All control rods are inserted to or beyond position 02." One rod is withdrawn past position 24. The reactor will remain shutdown under all conditions as Shutdown Margin is met.

D: Control rods are NOT inserted to or beyond the Maximum Sub-critical Banked Withdrawal limit, therefore it CANNOT be assured the reactor will remain shutdown under all conditions. **INCORRECT**, Maximum Sub-critical Banked Withdrawal Position is defined as: "All control rods are inserted to or beyond position 02." One rod is withdrawn past position 24. The reactor will remain shutdown under all conditions as Shutdown Margin is met.

Technical Reference(s): EOP-101 Bases, Tech Specs definitions 1.40 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: AB0000E003, State five (5) methods by which the operator can verify a successful scram action. (As available)

Question Source: Bank # 61270

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 1998

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 1, 2

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295023	AK1.03	
	Importance Rating	3.7		

Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS : Inadvertent criticality

Question: RO #41

What are the operational implications (if any) of inadvertent criticality during refueling?

A. NONE, because it is physically impossible to reach criticality during refueling.

B. Exceeding 10CFR20 dose rates in the reactor building because secondary containment is no longer intact.

C. Exceeding 10CFR100 dose rates on the refuel floor because primary containment is no longer intact.

D. More difficult to control and monitor reactor physics due to the configuration of the reactor core and instrumentation during refueling.

Proposed Answer: D

Explanation (Optional):

A: NONE, because it is physically impossible to reach criticality during refueling. **INCORRECT**, By ignoring procedural guidance and overriding various rod blocks it is physically possible to achieve an inadvertent criticality or withdraw sufficient rods to achieve criticality

B: Exceeding 10CFR20 dose rates in the reactor building because secondary containment is no longer intact. **INCORRECT,** 10CFR20 dose rates are considered for normal power plant operations and not for an inadvertent criticality during a refueling. Dose rates would only rise if criticality were achieved during the refueling-assuming no refueling accident resulting in damaged fuel. With the various rod withdraw blocks, refuel bridge interlocks, and procedural restrictions in place it is NOT possible to achieve criticality

C: Exceeding 10CFR100 dose rates on the refuel floor because primary containment is no longer intact. **INCORRECT,** 10CFR100 dose rates on the refuel floor already take into account refueling activities with primary containment not intact. With the various rod withdraw blocks, refuel bridge

interlocks, and procedural restrictions in place it is NOT possible to achieve criticality

D: More difficult to control and monitor reactor physics due to the configuration of the reactor core and instrumentation during refueling. **CORRECT,** due to removal of detectors or disconnecting cables/amplifiers for the nuclear instrumentation, the RO is challenged to keep track of the operable instrumentation and control reactivity with various fuel bundles/control rods that can/are removed from the core.

Technical Reference(s): 10CFR20, 10CFR100, OP-SO.KE-001 section 3.3.1 Interlocks (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ACCANLE007, Given access to a copy of 10 CFR 100, summarize the maximum allowable dose and time limitations at the Site Boundary and for the Low Population Zone IAW 10 CFR 100. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3, 6

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295019	AK2.14	
	Importance Rating	3.2		

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Plant air systems

Question: RO #42

Given:

- A complete loss of service and instrument air has occurred.
- Service and Instrument Air receivers are depressurized.
- Reactor level and pressure are being maintained by HPCI and RCIC.
- The temporary diesel driven air compressor is NOT available.

Then:

- Maintenance has reported that the 10K107 Service Air Compressor is now ready to be placed back in service.
- The Coast-Down Timer has been reset.
- The Post-Lube Timer has been reset.

Will the Reactor Operator in the Control Room be able to place the 10K107 back in service with the present plant conditions and why?

A. NO. There is NO instrument air available to run the Auxiliary Oil pump to start the Service Air Compressor and the start-up oil pressure interlock can NOT be satisfied.

B. NO. There is NO service air available to run the Auxiliary Oil pump to start the Service Air Compressor and the start-up oil pressure interlock can NOT be satisfied.

C. YES. The Auxiliary Oil pump is motor driven and the start-up oil pressure interlock will be satisfied.

D. YES. The Auxiliary Oil pump is NOT needed to start the Service Air Compressor.

Proposed Answer: D

Explanation (Optional):

A: NO. There is NO instrument air available to run the Auxiliary Oil pump to start the Service Air Compressor and the start-up oil pressure interlock can NOT be satisfied. **INCORRECT**, the Auxiliary Oil pump is driven by instrument air, there is no air to drive the pump and the main oil pump will start without the aux oil pump being in service prior to the start.

B: NO. There is NO service air available to run the Auxiliary Oil pump to start the Service Air Compressor and the start-up oil pressure interlock can NOT be satisfied. **INCORRECT,** the Auxiliary Oil pump is driven by instrument air, there is no air to drive the pump and the main oil pump will start without the aux oil pump being in service prior to the start.

C: YES. The Auxiliary Oil pump is motor driven and the start-up oil pressure interlock will be satisfied. **INCORRECT,** the Auxiliary Oil pump is driven by instrument air, there is no air to drive the pump and the main oil pump will start without the aux oil pump being in service prior to the start.

D: YES. The Auxiliary Oil pump is NOT needed to start the Service Air Compressor. **CORRECT,** the Auxiliary Oil pump is driven by instrument air, there is no air to drive the pump and the main oil pump will start without the aux oil pump being in service prior to the start.

Technical Reference(s): OP-SO.KA-001, NOH01SERAIR (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: SERAIRE009, Concerning the Main and Auxiliary Oil Pumps: Discuss their type, capacity, and purpose IAW available references. Given a system diagram, determine the lube oil flowpath IAW available references. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295004	AK2.02	
	Importance Rating	3.0		

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: Batteries

Question: RO #43

Regarding the battery chargers for the following DC Busses:

- RCIC 250 VDC bus 10D460
- 125 VDC bus 10D420

Which one of the following describes how their respective bus power is affected following the loss of their respective charger(s)?

The batteries for the ___(1)___ are designed to supply their loads for ___(2)___ .

	(1)	(2)
A.	250 VDC 125 VDC	Four (4) hours Four (4) hours
B.	250 VDC 125 VDC	Two (2) hours Two (2) hours
C.	250 VDC 125 VDC	Two (2) hours Four (4) hours
D.	250 VDC 125 VDC	Four (4) hours Two (2) hours

Proposed Answer: A

Explanation (Optional):

A: 250 VDC - 4 hours, 125 VDC - 4 hours, **CORRECT**, Per Table 5.3 of AB.ZZ-0135, each battery bus is designed for a 4 hour capacity (1885 AH @ 8 hours 10D420) (330 AH @ 8 hours 10D460)
B: 250 VDC - 2 hours, 125 VDC - 2 hours, **INCORRECT**, Per Table 5.3 of AB.ZZ-0135, each battery bus is designed for a 4 hour capacity (1885 AH @ 8 hours 10D420) (330 AH @ 8 hours 10D460)
C: 250 VDC - 2 hours, 125 VDC - 4 hours, **INCORRECT,** Per Table 5.3 of AB.ZZ-0135, each battery bus is designed for a 4 hour capacity (1885 AH @ 8 hours 10D420) (330 AH @ 8 hours 10D460)
D: 250 VDC - 4 hours, 125 VDC - 2 hours, **INCORRECT,** Per Table 5.3 of AB.ZZ-0135, each battery bus is designed for a 4 hour capacity (1885 AH @ 8 hours 10D420) (330 AH @ 8 hours 10D460)

Technical Reference(s): OP-AB.ZZ-0135 Table 5.3 Battery Bank Capacities (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: 0AB135E004, Explain the reasons for how plant/system parameters respond when implementing, Station Blackout/Loss Of Offsite Power Diesel Generator Malfunction, Abnormal Operating Procedure. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295031	EK2.14	
	Importance Rating	3.9		

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Emergency generators

Question: RO #44

A plant transient has occurred:

- HPCI and RCIC started automatically to restore level
- Reactor level is presently being held @ 35" by RCIC
- HPCI was secured by the operator
- Reactor pressure is being controlled by SRVs
- LOCA load shed non-1E breakers are open (10C650E)
- LOCA load shed 1E breakers are closed (10C650E)

(Assume NO additional operator actions)

With the above conditions, what is the status of the emergency diesel generators?

A. Due to the non-1E breaker status, the diesel is running unloaded

B. Due to the 1E breaker status the diesel is running unloaded

C. Due to the non-1E breaker status, the diesel is running loaded.

D. Due to the 1E breaker status, the diesel is NOT running.

Proposed Answer: D

Explanation (Optional):

A: Due to the non-1E breaker status, the diesel is running unloaded. **INCORRECT**, The emergency diesel generators will NOT be running, either loaded/unloaded. Because reactor vessel level only dropped to ≤ -38", the non-1E breakers have LOCA load shed and the 1E breakers remain closed.

B: Due to the 1E breaker status the diesel is running unloaded. **INCORRECT**, The emergency diesel generators will NOT be running, either loaded/unloaded. Because reactor vessel level only dropped to ≤ -38", the non-1E breakers have LOCA load shed and the 1E breakers remain closed.

C: Due to the non-1E breaker status, the diesel is running loaded. **INCORRECT**, The emergency diesel generators will NOT be running, either loaded/unloaded. Because reactor vessel level only dropped to ≤ -38", the non-1E breakers have LOCA load shed and the 1E breakers remain closed.

D: Due to the 1E breaker status, the diesel is NOT running. **CORRECT,** The emergency diesel generators will NOT be running, either loaded/unloaded. Because reactor vessel level only dropped to ≤ -38", the non-1E breakers have LOCA load shed and the 1E breakers remain closed.

Technical Reference(s): OP-SO.SM-0001 tables 19/20, OP-SO.KJ-0001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EDG000E006, From memory, recall the auto start signals associated with the Diesel Generators, including: a. Setpoints for LOP and Degraded Voltage signals. b. Setpoints for LOCA signals. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295003	AK3.01	
	Importance Rating	3.3		

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Manual and auto bus transfer

Question: RO #45

The 10A401 bus is being supplied by its normal infeed breaker (52-40108) when an undervoltage condition occurs on that infeed.

An AUTOMATIC TRANSFER can still occur to the alternate power supply with which of the following conditions?

A. Alternate infeed (52-40101) voltage <92% of nominal.

B. Alternate infeed (52-40101) "Auto Close Block" control illuminated.

C. Associated DG test lockout exists due to reverse power.

D. Associated bus lockout (overcurrent or differential overcurrent).

Proposed Answer: C

Explanation (Optional):

A: Alternate infeed voltage <92% of nominal. **INCORRECT**, This condition would signify an undervoltage condition on the alternate infeed breaker and a bus transfer would not occur. The infeed to be transferred to must be above the 94% nominal voltage.

B: Alternate infeed "Auto Close Block" control illuminated. **INCORRECT**, Once the bus voltage is < 70% undervoltage, undervoltage relays trip. IF bus voltage decreases to < 70%, regardless of the feeder voltage conditions, the alternate supply breaker will close following a 0.7 second time delay, provided: Auto Close Block is NOT selected. And Associated feeder voltage is ≥ 94%.

C: Associated DG test lockout exists due to reverse power. **CORRECT**, While in TEST, an Emergency Diesel Generator trips upon receipt of any of the following signals: Bus Overcurrent EDG Reverse Power, EDG Low Field Current, EDG Over Excitation. Only the EDG Output breaker would be tripped and a transfer to the alternate infeed breaker could occur.

D: Associated bus lockout (overcurrent or differential overcurrent). **INCORRECT**, Switchgear Bus Overcurrent condition: Trips the supplying breaker AND blocks closing of the alternate breaker. Trips AND locks out load/motor feeder breakers. Initiates breaker failure protection.

Technical Reference(s): OP-SO.PB-0001, OP-SO.KJ-001 Interlocks 3.3.4. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: 1EAC00E025, Given a labeled diagram/drawing (Figs. 2,3,4) of, or access to, the 1E AC Power System controls/indication bezels, a. Predict when each indicator will light. b. Assess plant conditions that will cause the indicators to light or extinguish. c. Predict the effect of each control switch on the 1E AC Power System. d. Assess the plant conditions or permissives required for the control switches to perform their intended function IAW the Lesson Plan. (As available)

Question Source: Bank # 54043

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7, 10

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295026	EK3.05	
	Importance Rating	3.9		

Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor SCRAM

Question: RO #46

Step SP/T-5 and SP/T-6 of the Suppression Pool Temperature Control section of HC.OP-EO.ZZ-102, Primary Containment Control, directs the initiation of a Reactor scram prior to reaching 110°F in the Suppression Pool.

Which of the following describes the reason for initiating a Reactor scram at this suppression pool temperature?

A Reactor scram is required prior to reaching 110°F:

A. To prevent exceeding the Suppression Pool design temperature.

B. To ensure enough time is available to inject the Hot Shutdown Boron Weight of boron into the RPV before Suppression Pool temperature exceeds the Heat Capacity Temperature Limit.

C. To limit the heat input into the Primary Containment as much as possible, since the heat capacity of the Suppression Pool is lost with Suppression Pool temperature above 110°F.

D. To ensure enough time is available to inject the Hot Shutdown Boron Weight of boron into the RPV before Suppression Pool temperature exceeds the Pressure Suppression Pressure Limit.

Proposed Answer: B

Explanation (Optional):

A: To prevent exceeding the Suppression Pool design temperature. **INCORRECT**, The design temperature of the Torus is 310°F which is not in jeopardy at 110°F.

B: This is the Suppression Pool temperature at which the initiation of a Reactor scram is required by Technical Specifications. CORRECT, Tech Specs 3.6.2.1 action b.2. states "With Suppression Pool average temperature greater than 110°F, place the Reactor Mode switch in

the shutdown position...". Per EOP Limits Conv. The Boron Injection Initiation Temperature (BIIT) is the greater of: The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit. The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

C: To limit the heat input into the Primary Containment as much as possible, since the heat capacity of the Suppression Pool is lost with Suppression Pool temperature above 110°F. **INCORRECT,** The heat capacity of the SP is reduced at 110°F but not lost until 212°F.

D: To ensure enough time is available to inject the Hot Shutdown Boron Weight of boron into the RPV before Suppression Pool temperature exceeds the Pressure Suppression Pressure Limit. **INCORRECT,** The requirement is to inject the HSBW before SP temperature reaches the HCTL.

Technical Reference(s): HC.OP-EO.ZZ-0102, Steps SP/T-5 & SP/T-6, Tech Spec 3.6.2.1.a.2)b) OP-EO.ZZ-Limits-Conv 5/56 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP102E009, Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW available control room references. (As available)

Question Source: Bank # 55982

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EK3.03	
	Importance Rating	4.1	

Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Lowering reactor water level

Question: RO #47

What describes why and to what RPV level is lowered to during performance of OP-EO.ZZ-0101A, ATWS - RPV Control?

A. The level is lowered to below the level of the feedwater spargers to maximize Core Inlet Sub-cooling.

B. The level is lowered to a few inches above the level of the feedwater spargers to maximize Core Inlet Sub-cooling.

C. The level is lowered to below the level of the feedwater spargers to minimize Core Inlet Sub-cooling

D. The level is lowered to a few inches above the level of the feedwater spargers to minimize Core Inlet Sub-cooling

Proposed Answer: C

Explanation (Optional):

A: The level is lowered to below the level of the feedwater spargers to maximize Core Inlet Sub-cooling. **INCORRECT**, the reason is to reduce core inlet sub-cooling

B: The level is lowered to a few inches above the level of the feedwater spargers to maximize Core Inlet Sub-cooling. **INCORRECT**, the reason is to reduce core inlet sub-cooling and it is lowered below the spargers

C: The level is lowered to below the level of the feedwater spargers to minimize Core Inlet Sub-cooling, **CORRECT**, IAW EOP-101A bases, If the reactor is not shutdown, RPV water level must be controlled to minimize core inlet subcooling, thereby preventing or mitigating the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

D: The level is lowered to a few inches above the level of the feedwater spargers to minimize Core Inlet Sub-cooling, **INCORRECT**, level is lowered to below the spargers, not above them, If the reactor is not shutdown, RPV water level must be controlled to minimize core inlet subcooling,

thereby preventing or mitigating the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

Technical Reference(s): EOP 101A Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EO101AE006, Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank # 120300

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295021	AA1.04	
	Importance Rating	3.7		

Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING : Alternate heat removal methods

Question: RO #48

The plant was operating at 100% power when a Loss of Offsite Power occurred. Shortly thereafter, Control Room abandonment was required and control was transferred to the Remote Shutdown Panel.

Which of the following describes the alignment for Alternate Shutdown Cooling from the Remote Shutdown Panel (10C399) under these conditions?

A. Injection from 'B' RHR through the LPCI line, maintaining at least two SRVs open with reactor pressure at least 50 psig above suppression chamber pressure.

B. Injection from 'A' RHR through the LPCI line, maintaining at least two SRVs open with reactor pressure at least 50 psig above suppression chamber pressure.

C. Injection from 'B' RHR through the LPCI line, maintaining at least one SRV open with reactor pressure at least 50 psig above suppression chamber pressure.

D. Injection from 'A' RHR through the LPCI line, maintaining at least one SRV open with reactor pressure at least 50 psig above suppression chamber pressure.

Proposed Answer: A

Explanation (Optional):

A: Injection from 'B' RHR through the LPCI line, maintaining at least two SRVs open with reactor pressure at least 50 psig above suppression chamber pressure. **CORRECT,** OP-IO.ZZ-008 Attachment 10 directs opening two SRVs and injecting with 'B' RHR through the F017 as necessary to maintain reactor pressure at least 50 psig above suppression pool pressure

B: Injection from 'A' RHR through the LPCI line, maintaining at least two SRVs open with reactor pressure at least 50 psig above suppression chamber pressure. **INCORRECT,** OP-IO.ZZ-008 Attachment 10, 'A' RHR is placed in suppression pool cooling and injection is with 'B' RHR. ('A' RHR has no controls for the HV-F003 or F048 at the RSP, so controlling injection and cooldown rate would be difficult).

C: Injection from 'B' RHR through the LPCI line, maintaining at least one SRV open with reactor pressure at least 50 psig above suppression chamber pressure. **INCORRECT,** OP-IO.ZZ-008 Attachment 10 directs opening two SRVs and injecting with 'B' RHR through the F017 as necessary to maintain reactor pressure at least 50 psig above suppression pool pressure. OP-IO.ZZ-008 Attachment 10, additionally a third SRV would be opened if reactor pressure stabilized more than 160 psig above suppression pool pressure.

D: Injection from 'A' RHR through the LPCI line, maintaining at least one SRV open with reactor pressure at least 50 psig above suppression chamber pressure. **INCORRECT,** OP-IO.ZZ-008 Attachment 10 directs opening two SRVs and injecting with 'B' RHR through the F017 as necessary to maintain reactor pressure at least 50 psig above suppression pool pressure. OP-IO.ZZ-008 Attachment 10, 'A' RHR is placed in suppression pool cooling and injection is with 'B' RHR. ('A' RHR has no controls for the HV-F003 or F048 at the RSP, so controlling injection and cooldown rate would be difficult). Additionally, a third SRV would be opened if reactor pressure stabilized more than 160 psig above suppression pool pressure.

Technical Reference(s): OP-IO.ZZ-008 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: IOP008E006, Analyze plant conditions and parameters to determine if plant operation is in accordance with the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications. (As available)

Question Source: Bank #

Modified Bank # 62226 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AA1.07	
	Importance Rating	3.1	

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Nuclear boiler instrumentation system

Question: RO #49

Given:

- The reactor is operating on the 100% rod line
- "B" Reactor Recirc pump just tripped
- "A" Reactor Recirc receives a FULL runback

Conditions now are:

- Reactor power is 57%
- "B" Recirc Loop flow is 1.1 Kgpm/Hr (R614-B31)
- "A" Recirc Loop flow is 15.4 Kgpm/Hr (R614-B31)
- Core Flow is 24.6 x 10^6 lbsm/Hr (FR-R613-B21)
- Jet Pump Loop B flow 9.8 x 10^6 lbsm/Hr (FL-R611B-B21)
- Jet Pump Loop A flow 32.3 x 10^6 lbsm/Hr (FL-R611A-B21)
- Observed APRM noise is stable at 3% peak to peak
- OPRMs are operable

What operator action is required?

A. Lock the Mode switch in SHUTDOWN

B. Enter IO-006 Power Changes During Operation ONLY for Single Loop Operations

C. Perform actions IAW Enhanced Stability Guidelines (ESG)

D. Perform actions IAW Standard Power Reduction Instructions (SPRI)

Proposed Answer: C

Explanation (Optional):

A: Lock the Mode switch in SHUTDOWN, **INCORRECT**, From AB.RPV-003 Immediate Operator actions: OPRM's INOPERABLE AND in REGION 1 of Figure 1, Plant is in Region 1 of the power to flow map, however OPRMs are operable and exiting the region is permissible/required IAW the Enhanced Stability Guidance (ESG) per RE-AB.ZZ-001

B: Enter IO-006 ONLY for Single Loop Operations, **INCORRECT,** this procedure will be entered, however other actions are required to exit Region 1 of the power to flow map.

C: Perform actions IAW Enhanced Stability Guidelines (ESG), **CORRECT,** Per AB.RPV-003, Condition B3. IF OPRM's are OPERABLE, THEN **PERFORM** the following: A. IF in REGION 1 of Figure 2 THEN **EXIT** REGION 1 IAW Enhanced Stability Guidance.(ESG) (RE.ZZ-0001)

D: Perform actions IAW Standard Power Reduction Instructions (SPRI), **INCORRECT,** these actions would be taken if both reactor recirc pumps were in service. Per AB.RPV-003, Condition B3. IF OPRM's are OPERABLE, THEN **PERFORM** the following: A. IF in REGION 1 of Figure 2 THEN **EXIT** REGION 1 IAW Enhanced Stability Guidance.(ESG) (RE.ZZ-0001)

Technical Reference(s): AB.RPV-003, RE-AB.ZZ-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: AB.RPV-003, power to flow maps

Learning Objective: RECIRCE011, Given a set of conditions and a drawing of, or access to, the controls, instrumentation and/or alarms located in the Main Control Room, assess the status of the Recirc System or its components by evaluation of the controls/instrumentation/alarms (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AA1.02	
	Importance Rating	2.9	

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : Reactor/turbine pressure regulating system

Question: RO #50

Given:

- The plant was operating at 90% power
- Toxic gas concerns have required the Main Control Room to be evacuated
- HC.OP-AB.HVAC-002 Condition C has been completed for all control room actions
- The transfer of controls to the Remote Shutdown Panel (10C399) have been completed

Which of the following systems are available for reactor vessel pressure control from the Remote Shutdown Panel (10C399)?

A. SRV's F, H & M AND RHR Shutdown Cooling

B. Reactor Feed Pumps AND Reactor Recirculation

C. High Pressure Coolant Injection AND LO-LO SET SRVs

D. SRV's A and E and Reactor Core Isolation Cooling

Proposed Answer: A

Explanation (Optional):

A: SRV's F, H & M and RHR Shutdown Cooling, **CORRECT**, IO-8 initiates RPV cooldown with SRV's F,H, & M until Shutdown Cooling can be established.

B: Reactor Feed Pumps and Reactor Recirculation, **INCORRECT**, Subsequent action C of HVAC-002 Trips the main turbine and closes the MSIVs. Recirc pumps are manually tripped and discharge valves closed.

C: High Pressure Coolant Injection and LO-LO SET SRVs, **INCORRECT,** HPCI cannot be controlled from the Remote Shutdown Panel (10C399). LO LO SET SRVs are only controlled

from the Control Room.

D: SRV's A and E and Reactor Core Isolation Cooling, **INCORRECT,** SRV's A and E are only available in the lower relay room on 102' and are not available from the RSP (10C399)

Technical Reference(s): IO.ZZ-008 AB.HVAC-002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: IOP008E006, Analyze plant conditions and parameters to determine if plant operation is in accordance with the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications. (As available)

Question Source: Bank # 68860

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2002

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295024	EA2.01	
	Importance Rating	4.2		

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell pressure

Question: RO #51

Given:

- The reactor has scrammed (all control rods are at position 00) on high drywell pressure.
- Reactor pressure is 35 psig.
- Reactor level is -90 inches rising.
- Suppression pool level is 75 inches.
- Suppression pool temp is 120°F.
- Suppression chamber temp is 100°F.
- Suppression chamber press is 15 psig.
- Drywell temp is 280°F.
- Drywell pressure is 17 psig.

To control the primary containment under these conditions the operator should monitor and control hydrogen concentration in the Supp Chamber and the Drywell and:

A. place one loop of RHR in drywell spray and the other loop of RHR in drywell and suppression chamber spray.

B. place one loop of RHR in suppression pool cooling and the other loop of RHR in drywell and suppression chamber spray.

C. place one loop of RHR in suppression pool cooling and suppression chamber spray and the other loop of RHR in drywell spray.

D. place one loop of RHR in suppression pool cooling and the other loop of RHR in drywell spray, and vent the suppression chamber.

Proposed Answer: C

Explanation (Optional):

A: place one loop of RHR in drywell spray and the other loop of RHR in drywell and suppression chamber spray. **INCORRECT**, Drywell Spray is always on a loop by itself, if available the second loop is placed in Suppression Pool Cooling/spray to prevent inadvertent bypass of the containment on the operating pump trip. Never place both loops in Drywell Spray this may exceed the makeup capacity of the vacuum breakers and draw the containment negative.

B: place one loop of RHR in suppression pool cooling and the other loop of RHR in drywell and suppression chamber spray. **INCORRECT,** Drywell Spray is always on a loop by itself, if available the second loop is placed in Suppression Pool Cooling/spray to prevent inadvertent bypass of the containment on the operating pump trip.

C: place one loop of RHR in suppression pool cooling and suppression chamber spray and the other loop of RHR in Drywell Spray. **CORRECT,** adequate core cooling is assured, Suppression Pool temperature requires Suppression Pool Cooling/spray, DWSIL curve is satisfied Drywell Spray is appropriate.

D: place one loop of RHR in suppression pool cooling and the other loop of RHR in drywell spray, and vent the suppression chamber. **INCORRECT,** venting the containment is only requires if pressure cannot be maintained below the design limit of 65 psig, and only after attempts to lower pressure with Drywell Spray.

Technical Reference(s): EOP-102 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP-102 DW/P leg and DW/T leg

Learning Objective: EO102PE007, Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW available control room references. (As available)

Question Source: Bank # 80609

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2003

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 9, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295038	EA2.04	
	Importance Rating	4.1		

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : Source of off-site release

Question: RO #52

Given:

- A plant shutdown is in progress
- North Plant Vent RMS is in HIGH alarm
- South Plant Vent RMS is reading 4.5 e^{+2} µCi/sec
- FRVS Vent RMS is reading 6.5e^{-2} µCi/sec
- FRVS is NOT in service

Which of the following is the source of the high alarm?

A. Service Area Exhaust System

B. Solid Radwaste Exhaust System

C. Radwaste Area Exhaust System

D. Turbine Building Exhaust System

Proposed Answer: B

Explanation (Optional):

A: Service Area Exhaust System, **INCORRECT**, Discharges to South Plant Vent Stack, See AB.CONT-004, Condition C.

B: Solid Radwaste Exhaust System, **CORRECT,** Discharges to North Plant Vent Stack, See AB.CONT-004, Condition D.

C: Radwaste Area Exhaust System, **INCORRECT,** Discharges to South Plant Vent Stack, See AB.CONT-004, Condition C.

D: Turbine Building Exhaust System, **INCORRECT,** Discharges to South Plant Vent Stack, See AB.CONT-004, Condition C.

Technical Reference(s): AB.CONT-004 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ABCNT4E001, Recognize abnormal indications/alarms and/or procedural requirements for implementing Radioactive Gaseous Release. (As available)

Question Source: Bank # 55868

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10, 11

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AA2.02	
	Importance Rating	3.5	

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Voltage outside the generator capability curve.

Question: RO #53

Given:

- The plant is operating at 100% power
- Generator load is 1290 MWe
- Generator H_2 pressure is 73 psi

A transient on the grid results in:

- Generator MVARs are 40
- Generator load remains 1290 MWe

Based on this:

A. Raise generator MWe

B. No actions are required

C. Raise generator H_2 pressure

D. MVARs must be raised above the current value

Proposed Answer: D

Explanation (Optional):

A: Raise generator MWe **INCORRECT**, IAW Att 1 of OP-SO.MA-001, presently below the minimum excitation line of the generator capability curve, if megawatts electric were raised would put the generator further away from the minimum curve. Requires that megavar load be raised to get within the curve, at least to 80 megavars

B: No actions are required, **INCORRECT**, IAW Att 1 of OP-SO.MA-001, presently below the minimum excitation line of the generator capability curve, if megawatts electric were raised would put the generator further away from the minimum curve. Requires that megavar load be raised to get within the curve, at least to 80 megavars

C: Raise generator H_2 pressure, **INCORRECT**, IAW Att 1 of OP-SO.MA-001, presently below the minimum excitation line of the generator capability curve, if hydrogen pressure was raised would more affect the generators megawatt maximum versus megavar minimum. Also present generator hydrogen pressure is 73 psig which is normal at this power level. Requires that megavar load be raised to get within the curve, at least to 80 megavars

D: MVARs must be raised above the current value, **CORRECT**, IAW Att 1 of OP-SO.MA-001, presently below the minimum excitation line of the generator capability curve. Requires that megavar load be raised to get within the curve, at least to 80 megavars

Technical Reference(s): OP-AR.ZZ-015 , D2038 - Generator electrical malfunction
OP-SO.MA-001 Main Generator & Exciter Operation & Switching (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Generator capability curve

Learning Objective: ABBOP4E005, Interpret and apply charts, graphs and tables contained within Grid Disturbances. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 10

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295005	2.1.31	
	Importance Rating	4.6		

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (Main Turbine Generator Trip)

Question: RO #54

Due to a main turbine vibration problem with a generator load of 110 MWe, a manual turbine trip is performed.

(1) Which of the following describes when the operator is REQUIRED to open the generator output breakers IAW OP-SO.AC-001 for the given conditions AND (2) what will be the 500KV switchyard alignment immediately after the main generator has been taken off-line? (Assume the generator has not already tripped on reverse power.)

A. (1) Within 15 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are open.

B. (1) Within 45 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are open.

C. (1) Within 15 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are closed.

D. (1) Within 45 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are closed.

Proposed Answer: A

Explanation (Optional):

A: 1) Within 15 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are open. **CORRECT -** IAW OP-SO.AC-001, step 3.1.15, procedure caution calls for operator actions within 15 seconds of the turbine trip at low power. Low power is considered below 150 Mwe. Step 5.4.12 has the operator trip open the 500KV 2-6 and 5-6 breakers.

B: 1) Within 45 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are open. **INCORRECT -** IAW OP-SO.AC-001, step 3.1.15, procedure caution calls for operator actions within 15 seconds of the turbine trip at low power. Low power is considered below 150 Mwe. Step 5.4.12 has the operator trip open the 500KV 2-6 and 5-6 breakers.

C: 1) Within 15 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are closed. **INCORRECT -** IAW OP-SO.AC-001, step 3.1.15, procedure caution calls for operator actions

within 15 seconds of the turbine trip at low power. Low power is considered below 150 Mwe. Step 5.4.12 has the operator trip open the 500KV 2-6 and 5-6 breakers. The generator output breakers are NOT reclosed until the main generator disconnect is opened which requires manual switching in the 500 KV switchyard.

D: (1) Within 45 seconds of the turbine trip. (2) 500 KV breakers Sect. 2-6 and 5-6 are closed. **INCORRECT -** IAW OP-SO.AC-001, step 3.1.15, procedure caution calls for operator actions within 15 seconds of the turbine trip at low power. Low power is considered below 150 Mwe. Step 5.4.12 has the operator trip open the 500KV 2-6 and 5-6 breakers. The generator output breakers are NOT reclosed until the main generator disconnect is opened which requires manual switching in the 500 KV switchyard.

Technical Reference(s): OP-SO.AC-001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: MNTURBE017, Explain how the Main Turbine interrelates with each of the following: a. Nuclear Boiler, b. Main Steam System, c. Steam Sealing System, d. Extraction Steam System, e. Main Turbine Lube Oil System, f. Reactor Feedwater System, g. Condensate System, h. EHC Control Logic System, i. EHC Control Oil System, j. Instrument Air System, k. Nuclear Boiler Instrumentation, l. Non-1E AC Power System, m. Bently TSI Vibration Monitoring System, n. Main Generator, o. Reactor Protection System, p. Reactor Recirculation System, q. Stator Water Cooling System (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295030	2.2.12	
	Importance Rating	3.7		

Equipment Control: Knowledge of surveillance procedures.

Question: RO #55

Which one of the following is the basis for the Technical Specifications, Minimum Suppression Chamber Water Volume in Operation Conditions 1, 2, and 3 and how is it documented?

A. Ensures that a sufficient supply of water is available in the event of a LOCA to permit recirculation cooling flow to the core AND is documented via OP-DL.ZZ-026, Surveillance Log

B. Ensures adequate heat capacity that, with the Minimum CST Volume, Long Term Cooling is available for the design basis accident AND is documented via OP-DL.ZZ-026, Surveillance Log.

C. Ensures that a sufficient supply of water is available in the event of a LOCA to permit recirculation cooling flow to the core AND is documented via OP-DL.ZZ-003, Control Room Console Log.

D. Ensures adequate heat capacity that, with the Minimum CST Volume, Long Term Cooling is available for the design basis accident AND is documented via OP-DL.ZZ-003, Control Room Console Log.

Proposed Answer: A

Explanation (Optional):

A: Ensures that a sufficient supply of water is available in the event of a LOCA to permit recirculation cooling flow to the core AND is documented daily via OP-DL.ZZ-0026, Surveillance Log. **CORRECT,** See Tech Spec Bases 4.6.2, OP-DL.ZZ-0026 Attach 1a, torus level is taken on the midnight shift once a day in Modes 1, 2 and 3. Allows for a full power blowdown of the reactor contents and then reflood capability with LPCI/Core Spray

B: Ensures adequate heat capacity that, with the Minimum CST Volume, Long Term Cooling is available for the design basis accident AND is documented via OP-DL.ZZ-0026, Surveillance Log. **INCORRECT,** See Tech Spec Bases 4.6.2, OP-DL.ZZ-0026 Attach 1a, torus level is taken on the midnight shift once a day in Modes 1, 2 and 3. While maintaining level will keep the HCTL intact, the CST is NOT mentioned in the bases as a concern for torus water level.

C: Ensures that a sufficient supply of water is available in the event of a LOCA to permit recirculation cooling flow to the core AND is documented via OP-DL.ZZ-0003, Control Room

Console Log. **INCORRECT,** See Tech Spec Bases 4.6.2, OP-DL.ZZ-0026 Attach 1a, torus level is taken on the midnight shift once a day in Modes 1, 2 and 3. Allows for a full power blowdown of the reactor contents and then reflood capability with LPCI/Core Spray

D: Ensures adequate heat capacity that, with the Minimum CST Volume, Long Term Cooling is available for the design basis accident AND is documented via OP-DL.ZZ-0003, Control Room Console Log. **INCORRECT,** See Tech Spec Bases 4.6.2, OP-DL.ZZ-0026 Attach 1a, torus level is taken on the midnight shift once a day in Modes 1, 2 and 3. While maintaining level will keep the HCTL intact, the CST is NOT mentioned in the bases as a concern for torus water level.

Technical Reference(s): Tech Spec Bases 4.6.2, OP-DL.ZZ-0026 Att 1a. (Attach if not previously provided)
LS-HC-1000-1001 SFCP 3/4 6-1

Proposed References to be provided to applicants during examination: none

Learning Objective: TECSPCE008, Given specific plant operating conditions, and a copy of the Hope Creek Generating Station Technical Specifications, determine the following: a. If a Limiting Condition for Operation has been exceeded. b. If a Limiting Safety System Setting has been reached and/or exceeded. c. If a Safety Limit has been violated. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3, 8, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295028	2.2.42	
	Importance Rating	3.9		

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question: RO #56

Which one of the following identifies the bases for the Drywell Average Air Temperature Limiting Condition for Operation (LCO)?

In the event of a DBA, initial drywell average air temperature is assumed to be less than or equal to:

A. 150° F so that the resultant peak accident temperature is maintained below 310° F during main steam line break conditions and is consistent with the safety analysis.

B. 150° F so that the containment peak air temperature does NOT exceed the design temperature of 340° F during LOCA conditions and is consistent with the safety analysis.

C. 135° F so that the resultant peak accident temperature is maintained below 310° F during main steam line break conditions and is consistent with the safety analysis.

D. 135°F, so that the containment peak air temperature does NOT exceed the design temperature of 340° F during LOCA conditions and is consistent with the safety analysis.

Proposed Answer: D

Explanation (Optional):

A: 150° F so that the resultant peak accident temperature is maintained below 310° F during main steam line break conditions and is consistent with the safety analysis. **INCORRECT**, See answer "D"

B: 150° degrees F so that the containment peak air temperature does NOT exceed the design temperature of 340° F during LOCA conditions and is consistent with the safety analysis. **INCORRECT**, See answer "D"

C: 135° F so that the resultant peak accident temperature is maintained below 310° F during main steam line break conditions and is consistent with the safety analysis. **INCORRECT**, See answer "D"

D: 135° F, so that the containment peak air temperature does NOT exceed the design temperature of 340° F during LOCA conditions and is consistent with the safety analysis. **CORRECT**, Per T/S 3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE, The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis. The 135°F average temperature is conducive to normal and long term operation.

Technical Reference(s): Tech Spec Bases 3/4.6.1.7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: PRICONE009, Given a Scenario of applicable operating conditions and access to Technical Specifications:
a. Select those sections that are applicable to the Primary Containment Structure IAW HCGS Technical Specifications. b. Evaluate Primary Containment Structure operability and determine required actions based upon system operability IAW HCGS Technical Specifications. (SRO/STA ONLY)
c. Explain the bases for those technical specification items associated with the Primary Containment Structure IAW HCGS technical specifications. (As available)

Question Source: Bank # 115528

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2005

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 9, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295018 AA2.03	
Importance Rating	3.2	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Cause for partial or complete loss

Question: RO #57

Given:

The Unit is in OPCON 1 at 100% power.
Multiple CRIDS high temperature alarms are in for both 'A' and 'B' Reactor Recirculation Pumps

These computer points are trending up on both 'A' and 'B' Recirc pumps:

- Pump Motor oil temperature
- Upper Motor bearing temperature
- Lower Motor bearing temperature
- #1 Seal Cavity temperature
- #2 Seal Cavity temperature

The plant is currently stable

Which one of the following malfunctions could explain this combination of indications, and under these circumstances, what is the MAXIMUM amount of time that plant procedures permit continued Recirculation Pump operations?

A. Inadvertent isolation of RACS cooling water flow to the Recirculation Pumps. Restore cooling within 10 minutes, or trip both Recirculation Pumps.

B. Inadvertent isolation of Chilled Water flow to the Recirculation Pumps. Restore cooling within 10 minutes, or trip both Recirculation Pumps.

C. Inadvertent isolation of RACS cooling water flow to the Recirculation Pumps. Restore cooling within 20 minutes, or trip both Recirculation Pumps.

D. Inadvertent isolation of Chilled Water flow to the Recirculation Pumps. Restore cooling within 20 minutes, or trip both Recirculation Pumps.

Proposed Answer: A

Explanation (Optional):

A: Inadvertent isolation of RACS cooling water flow to the Recirculation Pumps. Restore cooling within 10 minutes, or trip both Recirculation Pumps. **CORRECT**, RACS cooling supplies cooling water to the specified loads. HC.OP-AB.COOL-003 "RACS" requires that the recirc pumps be tripped if RACS cooling Cannot be restored within 10 minutes.

B: Inadvertent isolation of Chilled Water flow to the Recirculation Pumps. Restore cooling within 10 minutes, or trip both Recirculation Pumps. **INCORRECT** - Chilled Water normally supplies cooling to the Recirc Pump Motor AIR coolers. A loss of chilled water would result in Motor WINDING high temperature indications and alarms, NOT the indicated alarms.

C: Inadvertent isolation of RACS cooling water flow to the Recirculation Pumps. Restore cooling within 20 minutes, or trip both Recirculation Pumps. **INCORRECT**, RACS cooling supplies cooling water to the specified loads. HC.OP-AB.COOL-003 "RACS" requires that the recirc pumps be tripped if RACS cooling Cannot be restored within 10 minutes.

D: Inadvertent isolation of Chilled Water flow to the Recirculation Pumps. Restore cooling within 20 minutes, or trip both Recirculation Pumps. **INCORRECT** - Chilled Water normally supplies cooling to the Recirc Pump Motor AIR coolers. A loss of chilled water would result in Motor WINDING high temperature indications and alarms, NOT the indicated alarms.

Technical Reference(s): M-13-0, AB.COOL-003 (Attach if not previously provided)

NOH01RACS00C

Proposed References to be provided to applicants during examination: none

Learning Objective: RECIRCE011, Given a set of conditions and a drawing of, or access to, the controls, instrumentation and/or alarms located in the Main Control Room, assess the status of the Recirc System or its components by evaluation of the controls/instrumentation/alarms IAW available control room references (As available)

Question Source: Bank # 84247

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2007

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 14

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295025	EK3.02	
	Importance Rating	3.9		

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : Recirculation pump trip: Plant-Specific

Question: RO #58

Given:

- It is near the end of a fuel cycle.
- Main Turbine Stop Valves (TSVs) are being tested to validate the EOC-RPT setpoints.
- Two TSVs initiate an EOC-RPT signal at 10% closed.
- Two TSVs initiate an EOC-RPT signal at 5% closed.

Which of the following is a safety implication (if any) of this condition if reactor pressure were to exceed 1100 psig?

A. There are no safety implications because the TSV EOC-RPT trip is a 1-out-of-2 logic.

B. Reactor safety has been enhanced by the overly conservative trip value for TSV closure.

C. Void reactivity feedback may exceed control rod reactivity if these TSVs close at power.

D. There will be an excessive thermal margin upon EOC-RPT actuation if these TSVs close at power.

Proposed Answer: C

Explanation (Optional):

A: There are no safety implications because the TSV EOC-RPT trip is a 1-out-of-2 logic. **INCORRECT,** The TSV closure uses various combinations, not 1 of 2 twice.

B: Reactor safety has been enhanced by the overly conservative trip value for TSV closure. **INCORRECT,** setpoint is 5% +2% not 10%

C: Void reactivity feedback may exceed control rod reactivity if these TSVs close at power. **CORRECT,** If the TSVs generate an EOC-RPT signal at 10% closed vice the nominal 5% closed (7% allowable) then an excess positive reactivity will be added due to delayed recirc pump trip

D: There will be an excessive thermal margin upon EOC-RPT actuation if these TSVs close at power. **INCORRECT,** If the TSVs generate an EOC-RPT signal at 10% closed vice the nominal _ 5% closed (7% allowable) then an excess positive reactivity will be added upon TSV closure.

Technical Reference(s): Technical Specification 3.3.4.2 and Table 3.3.4.2-2 T/S Bases 3 / 4.3.4, TS 3.3.4.2 & Table 3.3.4.2-2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: MNTURBE018, Given a scenario of applicable operating conditions and access to Technical Specifications:
a. Select those sections applicable to the Main Turbine. b. Evaluate Main Turbine operability and determine required action(s) based upon inoperability.
c. Explain the bases for those technical specification items associated with the Main Turbine (SRO ONLY). (As available)

Question Source: Bank # 80657

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2003

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295010 AK1.03	
	Importance Rating	3.2	

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : Temperature increases

Question: RO #59

Given:

- The plant was operating at 100%
- An electrical malfunction caused a loss of ALL the #2 Drywell Cooling fans (A2V212-H2V212).
- Power is still available to the #1 Drywell Cooling fans (A1V212-H1V212).
- Drywell pressure is 1.0 psig and rising 0.1 psig every five minutes
- There is NO evidence of elevated coolant system leakage in the drywell
- Turbine Building Chilled Water is in service with a supply temperature of 44°F and steady.

Which of these actions is NOT appropriate for the listed conditions?

A. VENT the Drywell.

B. ALIGN RACS to the Chilled Water System for Drywell Cooling.

C. ENSURE all of the 'A' AND 'B' Drywell Fan Cooling Coils are Open.

D. ENSURE all of the #1 Drywell Cooling Fans (A1V212-H1V212) are running in Fast Speed.

Proposed Answer: B

Explanation (Optional):

A: VENT the Drywell. **INCORRECT**, HC.OP-AB.CONT-0001 directs venting the drywell if drywell pressure is >0.75 psig. This would be an appropriate action.

B: ALIGN RACS to the Chilled Water System for Drywell Cooling. **CORRECT,** Turbine Building chilled water has NOT been lost to the drywell. RACS supply temperature is typically 70°F and procedurally limited to a low of 45°F. Aligning RACS to cool the drywell would cause drywell pressure to rise more and thus is NOT an appropriate action.

C: ENSURE all of the 'A' AND 'B' Drywell Fan Cooling Coils are Open. **INCORRECT,** With limited air flow due to the loss of ½ of the drywell cooling fans, it is critical to ensure all drywell cooling coils are open. The 'A' and 'B' coils are in series with air flow and are not associated with a particular fan, so both sets of cooling coils are still effectively capable of cooling the drywell. This would be an appropriate action.

D: ENSURE all of the #1 Drywell Cooling Fans (A1V212-H1V212) are running in Fast Speed. **INCORRECT,** With the loss of all the #2 Drywell Cooling Fans, it is critical to ensure all remaining fans are in service, this would be an appropriate action.

Technical Reference(s): OP-AB.CONT-0001 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: DWVENTE009, Regarding HC.OP-AB.CONT-0001, Drywell Pressure, be able to discuss the following items: a. Given the procedure, state the subsequent operator actions and Retainment Override including the reason for each action. b. Given a set of conditions determine if a high drywell pressure conditions exists. (As available)

Question Source: Bank # 113596

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 9,10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295009	AK2.02	
	Importance Rating	3.9		

Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: Reactor water level control

Question: RO #60

Following a reactor scram, setpoint setdown logic is designed to ____________.

A. automatically lower the DFCS Master Level Controller setpoint to prevent a vessel overfeed when vessel level is ≤ 12.5" for at least one second

B. remove the level signal from DFCS so feed flow will match steam flow to prevent vessel overfeed when vessel level is ≤ 12.5" for at least one second

C. automatically lower the DFCS Startup Level Controller setpoint to prevent vessel overfeed when vessel level is ≤ 15" for at least one second

D. remove the total steam flow signal so feed flow will vary due to any deviation between actual and desired level only, to quickly restore level to normal when vessel level is ≤ 15" for at least one second.

Proposed Answer: A

Explanation (Optional):

A: Automatically lower the DFCS Master Level Controller setpoint to prevent a vessel overfeed when vessel level is ≤ 12.5" for at least one second. **CORRECT** - The Master Controller Setpoint is electrically lowered to +14" when vessel level is below 12.5" for at least one second.

B: Remove the level signal from DFCS so feed flow will match steam flow to prevent vessel overfeed when vessel level is ≤ 12.5" for at least one second. **INCORRECT** - The level input remains so water level will be controlled automatically by DFCS. The Master Controller Setpoint is electrically lowered to +14" when vessel level is below 12.5" for at least one second.

C: Automatically lower the DFCS Startup Level Controller setpoint to prevent vessel overfeed when vessel level is ≤ 15" for at least one second. **INCORRECT** - The Master Controller Setpoint is electrically lowered to +14" when vessel level is below 12.5" for at least one second.

D: Remove the total steam flow signal so feed flow will vary due to any deviation between actual and desired level only, to quickly restore level to normal when vessel level is ≤ 15" for at least one second. **INCORRECT -** The Master Controller Setpoint is electrically lowered to +14" when vessel level is below 12.5" for at least one second.

Technical Reference(s): NOH04FWCONTC, H-1-AE-ECS-0128-03C/X (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: FWCONTE004, From memory, state the purpose of the setpoint setdown unit and describe how it accomplishes its purpose, IAW Available Control Room References (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295034	EK3.03	
	Importance Rating	4.0		

Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Personnel evacuation

Question: RO #61

The Unit is in OPCON 1 with irradiated fuel moves in progress on the refueling floor. Ventilation systems are lined up in the normal lineup to support plant conditions.

Then:

- An irradiated fuel bundle is dropped from the full up position.
- The bundle impacts other irradiated bundles in the fuel pool.
- A large amount of bubbles are observed rising from the dropped and impacted bundles.
- The NEW FUEL CRITICALITY RAD HI OHA E6-A4 is in alarm.
- The Filtration and Recirculation Ventilation System (FRVS) has initiated.
- The Reactor Building Ventilation System (RBVS) has isolated.
- The Refuel Floor evacuation alarm is sounding.

Why is the Refuel Floor required to be evacuated for the above conditions?

A. The release from the damaged fuel contains O^{19} (Oxygen) AND H^{3} (Hydrogen-Tritium) and eventually enough of it will saturate the Refuel Floor and create an explosive atmosphere.

B. The release from the damaged fuel contains H^{3} (Hydrogen-Tritium) AND Kr^{90} (Krypton) and eventually enough of it will saturate the Refuel Floor and create an explosive and highly radioactive airborne atmosphere.

C. The release from the damaged fuel contains O^{19} (Oxygen) AND Xe^{135} (Xenon) and eventually enough of it will saturate the Refuel Floor and create a highly radioactive airborne and explosive atmosphere.

D. The release from the damaged fuel contains Kr^{90} (Krypton) AND Xe^{135} (Xenon) and eventually enough of it will saturate the Refuel Floor and create a highly radioactive airborne atmosphere.

Proposed Answer: D

Explanation (Optional):

A: The release from the damaged fuel contains O^{19} (Oxygen) AND H^{3} (Hydrogen-Tritium) and eventually enough of it will saturate the Refuel Floor and create an explosive atmosphere. **INCORRECT**, Oxygen and Hydrogen are by-products of the radiolytic decomposition of water and would be expected in the steam carryover during power operation in the reactor. They would not be expected from damaged fuel bundles, which would carry fission product gases. With FRVS in service enough air mixing and exhaust would prohibit sufficient Oxygen to accumulate to present an explosion hazard.

B: The release from the damaged fuel contains H^{3} (Hydrogen-Tritium) AND Kr^{90} (Krypton) and eventually enough of it will saturate the Refuel Floor and create an explosive and highly radioactive airborne atmosphere. **INCORRECT,** Hydrogen is a by-product of the radiolytic decomposition of water and would be expected in the steam carryover during power operation in the reactor. It would not be expected from damaged fuel bundles, which would carry fission product gases, which would be both Xenon and Krypton. With FRVS in service enough air mixing and exhaust would prohibit sufficient Hydrogen to accumulate to present an explosion hazard.

C: The release from the damaged fuel contains O^{19} (Oxygen) AND Xe^{135} (Xenon) and eventually enough of it will saturate the Refuel Floor and create a highly radioactive airborne and explosive atmosphere. **INCORRECT,** Oxygen is a by-product of radiolytic decomposition of water and would be expected in the steam carryover during power operation in the reactor. It would not be expected from damaged fuel bundles, which would carry fission product gases, which would be both Krypton and Xenon. With FRVS in service enough air mixing and exhaust would prohibit sufficient Oxygen to accumulate to present an explosion hazard.

D: The release from the damaged fuel contains Kr^{90} (Krypton) AND Xe^{135} (Xenon) and eventually enough of it will saturate the Refuel Floor and create a highly radioactive airborne atmosphere. **CORRECT,**

Technical Reference(s): NOH01CHEMREC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: CHEMREE005, Given plant conditions involving fuel damage, appraise the radiological concerns associated with the following, IAW the Student Handout: a. Post accident sampling, b. Access to plant buildings, c. Transport of radioactive fluids SRO ONLY (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	11, 12
	55.43	

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295035	EA1.02	
	Importance Rating	3.8		

Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: SBGT/FRVS

Question: RO #62

Given:

The plant is operating at 100%.

OHA E6-A3 "REFUEL FLR EXH RAD ALARM/TRBL" alarms
OHA E6-A5 "RB EXH RADIATION ALARM/TRBL" alarms
OHA E6-C5 "RBVS & WING AREA HVAC PNL 10C382" alarms.

The Reactor Operator reports these readings from the RM11:

1SP-RE-4856A	1.9E-3 μCi/cc	Refuel Flr Exh Channel "A"
1SP-RE-4856B	1.0E-3 μCi/cc	Refuel Flr Exh Channel "B"
1SP-RE-4856C	1.1E-3 μCi/cc	Refuel Flr Exh Channel "C"

1SP-RE-4857A	9.90E-4 μCi/cc	Rx Bldg Exh Channel "A"
1SP-RE-4857B	1.05E-3 μCi/cc	Rx Bldg Exh Channel "B"
1SP-RE-4857C	1.11E-3 μCi/cc	Rx Bldg Exh Channel "C"

What is Reactor Building Δp and what is controlling it?

A. -.25" WG, Maintained by RBVS (Reactor Building Ventilation System)

B. -.25" WG, Maintained by FRVS (Filtration, Recirculation Ventilation System)

C. -.55" WG, Maintained by RBVS (Reactor Building Ventilation System)

D. -.55" WG, Maintained by FRVS (Filtration, Recirculation Ventilation System)

Proposed Answer: D

Explanation (Optional):

A: -.25” WG, Maintained by RBVS (Reactor Building Ventilation System), **INCORRECT**, RBVS would be isolated due to the hi radiation signal from RB Exh rad monitors B/C channels and FRVS would be maintaining the building Δp @-.55 WG. The -.25 WG is the Tech Spec limit for secondary containment operability.

B: -.25” WG, Maintained by FRVS (Filtration, Recirculation Ventilation System), **INCORRECT**, RBVS would be isolated due to the hi radiation signal from RB Exh rad monitors B/C channels and FRVS would be maintaining the building Δp @-.55 WG. The -.25 WG is the Tech Spec limit for secondary containment operability.

C: -.55” WG, Maintained by RBVS (Reactor Building Ventilation System), **INCORRECT,** RBVS would be isolated due to the hi radiation signal from RB Exh rad monitors B/C channels and FRVS would be maintaining the building Δp @-.55 WG. The -.25 WG is the Tech Spec limit for secondary containment operability.

D: -.55” WG, Maintained by FRVS (Filtration, Recirculation Ventilation System), **CORRECT**, RBVS would be isolated due to the hi radiation signal from RB Exh rad monitors B/C channels and FRVS would be maintaining the building Δp @-.55 WG. The -.25 WG is the Tech Spec limit for secondary containment operability.

Technical Reference(s): OP-SO.GR-001, OP-AR.ZZ-019 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: SECCONE007, From memory, briefly describe how a slight negative pressure is maintained in the reactor building during abnormal (accident) conditions and state the value at which it is maintained. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7, 9

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295036	EA2.02	
	Importance Rating	3.1		

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : Water level in the affected area

Question: RO #63

Given:

- The plant has reduced power to 95% due to condenser high vacuum.
- The Reactor Building Operator (RBEO) reports that when they open the water-tight door into the RACS room, water spills over the lip of the door.
- OHA A2-D2 RACS PUMP ROOM FLOODED has alarmed.

Investigation reveals that ALL systems remain normally aligned.

Which of the following describes action(s) to be taken for the above conditions?

A. Enter EO.ZZ-103/4, Reactor Building & Rad Release Control AND Runback Recirc to minimum AND LOCK the mode switch in SHUTDOWN.

B. Enter EO.ZZ-103/4, Reactor Building & Rad Release Control AND commence a normal plant shutdown AND enter IO.ZZ-004, Shutdown From Rated Power To Cold Shutdown.

C. Ensure isolation of EA-HV-2203 LOOP A RACS HX HDR SUP AND EA-HV-2204 LOOP B RACS HX HDR AND enter AB.COOL-003 Reactor Auxiliary Cooling.

D. Ensure isolation of EA-HV-2207 RACS HX HDR INLET VLV AND EA-HV-2346 RACS HX HDR OUTLET VLV AND enter AB.COOL-005 Total Loss of Station Service Water.

Proposed Answer: C

Explanation (Optional):

A: Enter EO.ZZ-0103/4, Reactor Building & Rad Release Control AND Runback Recirc to minimum AND LOCK the mode switch in SHUTDOWN. **INCORRECT**, the RACS room is not one the

areas listed on Table 2 of EOP-103/4 and no actions from EOP-103/4 should be taken.

B: Enter EO.ZZ-0103/4, Reactor Building & Rad Release Control AND commence a normal plant shutdown AND enter IO.ZZ-0004, Shutdown From Rated Power To Cold Shutdown. **INCORRECT,** the RACS room is not one the areas listed on Table 2 of EOP-103/4 and no actions from EOP-103/4 should be taken.

C: Ensure EA-HV-2203 LOOP A RACS HX HDR SUP AND EA-HV-2204 LOOP B RACS HX HDR AND enter AB.COOL-003 Reactor Auxiliary Cooling. **CORRECT,** IAW OP-SO.ED-001, Service Water valves EA-HV-2203/4 isolate on RACS room flooded, as activated from LSH-2365A/B. Either level switch will cause RACS Pump Room flooded OHA. Room flooded is an entry condition for AB.COOL-003

D: Ensure EA-HV-2207 RACS HX HDR INLET VLV AND EA-HV-2346 RACS HX HDR OUTLET VLV AND enter AB.COOL-0001 Station Service Water. **INCORRECT,** IAW OP-SO.ED-001, Service Water valves EA-HV-2203/4 isolate on RACS room flooded, as activated from LSH-2365A/B. Either level switch will cause RACS Pump Room flooded OHA. The EA-HV-2207/2346 are isolated from LSH-2365C which does NOT cause the OHA. Room flooded is an entry condition for AB.COOL-003 not AB.COOL-005.

Technical Reference(s): AB.COOL-001 AB.COOL-003, OP-AR.ZZ-002, EOP-103/4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: SERWATE006, Give a brief description of what happens to the Station Service Water System on the following plant conditions: a. LOCA, b. LOP, c. Pipe Break - In RACS HX Room - Safety-related piping. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295008	2.1.27	
	Importance Rating	3.9		

Conduct of Operations: Knowledge of system purpose and / or function. (High Reactor Water Level)

Question: RO #64

A malfunction of the Digital Feedwater Level Control System has resulted in a rising reactor water level. The Reactor Feedwater Pumps are automatically tripped on a high reactor water level signal to prevent:

A. main steam line piping and hanger damage due to filling the main steam lines.

B. feedwater line damage due to increasing pump discharge flow rate and head.

C. feed pump turbine damage due to water erosion of feed pump turbine nozzles.

D. damage to main steam drain lines from potential two phase flow.

Proposed Answer: A

Explanation (Optional):

A: main steam line piping and hanger damage due to filling the main steam lines. **CORRECT**, Feed pumps are tripped on level 8 (+54") to prevent overfilling the reactor and flooding the main steam lines which could cause damage to the main turbine.

B: feedwater line damage due to increasing pump discharge flow rate and head. **INCORRECT,** discharge pressure exceeding design limits is prohibited by a high discharge pressure (1700 psi) trip of the RFPT, reactor vessel level has no effect on discharge pressure of the feed pump

C: feed pump turbine damage due to water erosion of turbine nozzles. **INCORRECT,** the RFPT are not a susceptible to water erosion as the main turbine due to path LP and HP steam must take to get to the feed pump turbine. Turbine nozzle erosion is a concern for the main turbine.

D: damage to main steam drain lines from potential two phase flow. **INCORRECT**, due to recent events, two-phase flow is a concern on main steam line drains, however this is not dependent on reactor vessel level and can occur at any vessel level.

Technical Reference(s): NOH04RXINSTC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RXINSTE010, Given a list of reactor vessel pressure and/or level setpoints, determine the automatic action that occurs (As available)

Question Source: Bank # 68855

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2002

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295020 AK3.06	
	Importance Rating	3.3	

Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Suppression pool water level response.

Question: RO #65

Given:

HPCI is presently running on min flow with its injection valves closed.

Then:

A false isolation signal is received and the BJ-HV-F042 Torus Suction, strokes full closed after the BJ-HV-F004 CST Suction strokes full open. All other containment parameters are normal.

What are the consequences (if any) of the BJ-HV-F042 stroking closed with the above conditions?

A. NO consequences, system will continue to run on min flow until secured.

B. Torus level will start to rise because CST water is now being diverted to the torus.

C. Torus level will start to lower because torus water is now being diverted to the CST.

D. System will trip on low suction pressure.

Proposed Answer: B

Explanation (Optional):

A: NO consequences, system will continue to run on min flow until secured. **INCORRECT**, With suction on the CST and min flow discharging to the Torus, Torus level will begin to rise and actions must be taken to maintain level within the Tech Spec limits 74.5" – 78.5".

B: Torus level will start to rise because CST water is now being diverted to the torus. **CORRECT**, With suction on the CST and min flow discharging to the Torus, Torus level will begin to rise and

actions must be taken to maintain level within the Tech Spec limits 74.5" – 78.5".

C: Torus level will start to lower because torus water is now being diverted to the CST. **INCORRECT**, With suction on the CST and min flow discharging to the Torus, Torus level will begin to rise and actions must be taken to maintain level within the Tech Spec limits 74.5" – 78.5".

D: System will trip on low suction pressure. **INCORRECT**, With suction on the CST and min flow discharging to the Torus, Torus level will begin to rise and actions must be taken to maintain level within the Tech Spec limits 74.5" – 78.5". With CST suction valve open there will be adequate suction pressure and system will not trip.

Technical Reference(s): OP-SO.BJ-001, NOH01HPCI00 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: HPCI00E012, Given plant conditions, determine the HPCI System response to any of the following IAW control room references: a. Low CST level (HPCI in operation) b. High Suppression Pool level (HPCI in operation) c. Loss of 250 VDC d. Loss of 480 VAC e. Loss of 125 VDC (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.26	
	Importance Rating	3.4	

Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

Question: RO #66

Given:

The plant is operating at 100% power.
An electrical transient results in a loss of Bus 10A401.

IAW SA-AA-129 Electrical Safety, which of the following is required to perform a Low Voltage test prior to work on the 10A401 bus?

Test an approved voltage-meter on:

A. 10A402 before verifying 10A401 de-energized.

B. 10A120 before and after verifying 10A401 de-energized.

C. 10B410 before and after verifying 10A401 de-energized.

D. 10B420 before and after verifying 10A401 de-energized.

Proposed Answer: D

Explanation (Optional):

A: 10A402 before verifying 10A401 de-energized. **INCORRECT**, INSPECT voltage-testing devices for proper operation. PROVE the testing device before and after use by using an approved tester or on a conductor energized at the same or lower voltage than the conductor being tested for no-voltage. 10A402 is a 4160 volt bus and should not be used to perform a low voltage check as it is higher than 600 volts and is not low voltage.

B: 10A120 before and after verifying 10A401 de-energized. **INCORRECT** INSPECT voltage-testing devices for proper operation. PROVE the testing device before and after use by using an

approved tester or on a conductor energized at the same or lower voltage than the conductor being tested for no-voltage. 10A402 is a 4160 volt bus and should not be used to perform a low voltage check as it is higher than 600 volts and is not low voltage.

C: 10B410 before and after verifying 10A401 de-energized. **INCORRECT** INSPECT voltage-testing devices for proper operation. PROVE the testing device before and after use by using an approved tester or on a conductor energized at the same or lower voltage than the conductor being tested for no-voltage. 10B410 is a Unit Sub-Station powered from the 10A401 bus and as such will not be energized to perform the first part of the live-dead-live-check of the voltmeter.

D: 10B420 before and after verifying 10A401 de-energized. **CORRECT** INSPECT voltage-testing devices for proper operation. PROVE the testing device before and after use by using an approved tester or on a conductor energized at the same or lower voltage than the conductor being tested for no-voltage. 10B420 is a Unit Sub-Station powered from the 10A402 bus and is 480 volt which is low voltage (<600 volt)

Technical Reference(s): SA-AA-129, ELECTRICAL SAFETY, (Attach if not previously provided)

OP-AA-109-115

Proposed References to be provided to applicants during examination: none

Learning Objective: NOC01BKRDLDCE5, Perform an electrical Zero Voltage (LIVE-DEAD-LIVE) test using Digital Multimeter Electrical Test Equipment to verify the absence or presence of electrical potential on non racking breakers >=230v to <600 volt (As available)

Question Source: Bank # 110088

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.29	
	Importance Rating	4.1	

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Question: RO #67

The Unit is at 100% power. After a system outage, untagging of the HPCI system is being performed to restore HPCI operability.

The breaker for the Lube Oil Cooling Wtr Valve, BJ-HV-F059, must be closed during performance of the untagging.

Which ONE of the following describes the required verification activities which must be completed before the system can be considered operable?

A. Independent verification is required and must be performed by a licensed operator and be documented.

B. Independent verification is required and can be performed by any qualified operator and must be documented.

C. Only concurrent verification is required and must be performed by a licensed operator and be documented.

D. Only concurrent verification is required and can be performed by any qualified operator and must be documented.

Proposed Answer: B

Explanation (Optional):

A: Independent verification is required and must be performed by a licensed operator and be documented. **INCORRECT,** ANY qualified operator can perform the independent verification, per OP-AA-109-115, step 4.7.5.

B: Independent verification is required and can be performed by any qualified operator and must be documented. **CORRECT,** per OP-AA-109-115, step 4.7.5., perform Independent Verification,

Qualified Operator, WCCS • When releasing Safety Tags, perform independent verifications on systems listed in OP-AA-108-101-1002 Att 12, HPCI is one of the systems listed on Att 12

C: Only concurrent verification is required and must be performed by a licensed operator and be documented. **INCORRECT,** OP-AA-109-115, step 4.7.5., perform Independent Verification, Qualified Operator, WCCS • When releasing Safety Tags, perform independent verifications on systems listed in OP-AA-108-101-1002 Att 12, HPCI is one of the systems listed on Att 12

D: Only concurrent verification is required and can be performed by any qualified operator and must be documented. **INCORRECT,** OP-AA-109-115, step 4.7.5., perform Independent Verification, Qualified Operator, WCCS • When releasing Safety Tags, perform independent verifications on systems listed in OP-AA-108-101-1002 Att 12, HPCI is one of the systems listed on Att 12

Technical Reference(s): OP-AA-109-115, OP-AA-108-101-1002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NA0015E006, Identify tagging rules and conditions IAW the Safety Tagging Procedure, OP-AA-109 and the SAP/WCM Tagging Operations Procedure, OP-AA-109-115. (As available)

Question Source: Bank # 116310

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.6	
	Importance Rating	3.0	

Knowledge of the process for making changes to procedures.

Question: RO #68

Given:

- It is August and the plant has been de-rated (95%) due to environmental conditions.
- The A SJAE is stalling and must be swapped.
- Condenser vacuum is degrading very slowly.

While briefing the SJAE swap, a critical step in the procedure was found to be missing.

Which of the following describes the requirement, if any, to continue the evolution?

A. A procedure change request is required and an on-the-spot-change can be made.

B. Obtain verbal concurrence from the CRS to change the sequence of steps and continue.

C. Complete the evolution as written then perform a permanent revision change after the evolution is complete.

D. A full procedure revision to the SJAE operating procedure is required. An on-the-spot change CAN NOT be performed.

Proposed Answer: A

Explanation (Optional):

A: A procedure change request is required and an on-the-spot-change can be made. **CORRECT,** An on-the-spot-change (OTSC) may used. See step 4.1.5 of OP-AA-101-111-1003.

B: Obtain verbal concurrence from the CRS to change the sequence of steps and continue. **INCORRECT,** Written documentation is required (On-the-spot change) See step 4.1.5 of OP-AA-101-111-1003.

C: Complete the evolution as written then perform a permanent revision change after the evolution is complete. **INCORRECT,** If an error is found in the procedure actions must be taken to correct the issue before proceeding. See step 4.1.5 of OP-AA-101-111-1003.

D: A full procedure revision to the SJAE operating procedure is required. An on-the-spot change CAN NOT be performed. **INCORRECT,** An on-the-spot change may be used. See step 4.1.5 of OP-AA-101-111-1003.

Technical Reference(s): OP-AA-101-111-1003, HU-AA-104-101, AD-AA-101-101 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ADMPROE003, From Memory Describe the conditions under which performance of a procedure should be stopped, and the necessary subsequent actions IAW OP-AA-101-111-1003, and HU-AA-104-101 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.39	
	Importance Rating	3.9	

Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Question: RO #69

The plant is operating at 75% power
It is 06:17 and you note while taking logs:

- Drywell average temperature is 137° F.
- Drywell Pressure is 1.2 psig.
- Suppression Pool water level is 74".

From the readings above what actions are required to satisfy Technical Specifications?

A. Restore suppression pool level within normal band by 07:17

B. Restore drywell pressure within normal band by 07:17

C. Restore both drywell average temperature AND suppression pool level by 18:17

D. Restore both drywell average temperature AND drywell pressure by 18:17

Proposed Answer: A

Explanation (Optional):

A: Suppression Pool Level ONLY. (2) 07:17, **CORRECT,** per T/S 3.6.2.1, "a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."

B: Drywell Pressure ONLY. (2) 07:17, **INCORRECT,** per T/S 3.6.1.6, "With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours." Action is required within 1 hour

if DW pressure exceeds 1.5 psig

C: (1) Drywell Average Temperature AND Suppression Pool Level. (2) 18:17, **INCORRECT,** per T/S 3.6.1.7, "With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours." Per T/S 3.6.2.1, "a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours."

D: (1) Drywell Average Temperature AND Drywell Pressure. (2) 18:17, **INCORRECT**, per T/S 3.6.1.7, "With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Per T/S 3.6.1.6, "With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours." Action is required within 1 hour if DW pressure exceeds 1.5 psig.

Technical Reference(s): Tech Specs, 3.6.1.6, 3.6.1.7, and 3.6.2.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: TECSPCE013, Given specific plant operating conditions which require operator actions within 1 hour From Memory select the correct Technical Specification action(s). (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.11	
	Importance Rating	3.8	

Ability to control radiation releases.

Question: RO #70

During a declared ALERT due to high offsite release rate following a plant transient, the Turbine Building Equipment Operator reports that he has discovered that the Turbine Building Ventilation System is shutdown.

Under these conditions, Turbine Building Ventilation will...

A. remain shutdown to minimize the potential for release from the plant.

B. be restarted to provide an elevated, monitored release point for radioactive material.

C. be restarted, but only if it will result in no additional release of radioactive material.

D. be restarted or left shutdown based on area temperatures AND the potential for additional release.

Proposed Answer: B

Explanation (Optional):

A: remain shutdown to minimize the potential for release from the plant. **INCORRECT**, this may cause an unmonitored ground level release

B: be restarted to provide an elevated, monitored release point for radioactive material. **CORRECT,** per EOP-104 bases, RR-4, "Operating HVAC preserves building accessibility and discharges radioactivity through an elevated monitored release point.

C: be restarted, but only if it will result in no additional release of radioactive material. **INCORRECT,** not restarting could cause an unmonitored release

D: may be restarted or left shutdown based on area temperatures and the potential for additional release. **INCORRECT,** not restarting could cause an unmonitored release

Technical Reference(s): EOP-103/4 Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP103E006, Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step. (As available)

Question Source: Bank # 56131

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10, 11, 12

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.12	
	Importance Rating	3.2	

Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question: RO #71

Radiation Protection technicians have surveyed the Refuel Floor Reactor Head Laydown Area during an outage and obtained the following results:

Area Dose Rates one foot from the source: 72 mR/Hr
Airborne Concentration: 0.15 DAC
Smear Results: 750 Dpm/100 Cm^2 gamma

Based on these results the area will be posted as a:

I. Radiation Area
II. High Radiation Area
III. Very High Radiation Area
IV. Contaminated Area
V. Airborne Radioactivity Area

A. I and V

B. II and IV

C. III and IV

D. I, IV, and V

Proposed Answer: A

Explanation (Optional):

A: I, and V, **CORRECT**, Radiation area = >5 mRem/hr to 80 mRem/hr. Airborne Radioactivity area = >10% or .10 DAC.
B: II, and IV, **INCORRECT**, Not a Contamination Area; Contamination Area= >1000 dpm/100cm2. Not a Hi Rad area <80mr/hr
C: III, and IV, **INCORRECT**, Very High Radiation Area = >400 Rads/Hr. Not a Contamination Area; Contamination Area= >1000 dpm/100cm2
D: I, IV, and V, **INCORRECT**, < 80 mRem/hr.

Technical Reference(s): RP-AA-460 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NOH04ADM024E002, From Memory State the definition of the following terms: Contaminated Area, High Contamination Area, High Radiation Area, Locked High Radiation Area, Radiation Area Restricted Area, Very High Radiation Area, Airborne Radioactivity Area, Declared Pregnant Woman (DPW), Total Effective Dose Equivalent (TEDE) IAW RP-AA-203, Exposure Control and Authorization, RP-AA-270, Prenatal Radiation Exposure, RP-AA-376, Radiological Postings, Labeling and Marking (As available)

Question Source: Bank # 76884

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10, 11, 12

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.22	
	Importance Rating	3.6	

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Question: RO #72

Hope Creek is experiencing an ATWS.

Why are you required to inhibit the automatic initiation of ADS?

This prevents ________________

A. large irregular neutron flux oscillations.

B. causing brittle fracture of the reactor vessel.

C. a power excursion due to low pressure ECCS injection

D. exceeding 110° F Suppression Pool Temperature before boron injection

Proposed Answer: C

Explanation (Optional):

A: large irregular neutron flux oscillations, **INCORRECT**, during an ATWS oscillations of neutron flux would be expected, however this is not the concern with uncontrolled injection from RHR and Core Spray.

B: causing brittle fracture of the reactor vessel, **INCORRECT,** with the reactor not shutdown, the overriding concern for uncontrolled injection from RHR and Core Spray is the power excursion caused by the relatively cool water being injected onto a critical core. Brittle fracture of the fuel cladding is more of a concern for this condition.

C: a power excursion due to low pressure ECCS injection, **CORRECT,** ADS is required to be inhibited when either reactor vessel level drops below -129" or during an ATWS. This action will prohibit the uncontrolled injection of large amounts of water from RHR and Core Spray post-blowdown, which during an ATWS would cause a power excursion fro the injection of relatively cool water on a critical reactor core.

D: exceeding 110° F Suppression Pool Temperature before boron injection, **INCORRECT,** this concern is addressed at step LP-11, for when to lower reactor vessel level in an attempt to lower reactor power. Inhibiting ADS has no initial effect on torus temperature.

Technical Reference(s): EOP-101A Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EO101AE006, Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank #

Modified Bank # 30560 (Note changes or attach parent)

New

Question History: NRC 2005

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 1, 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.28	
	Importance Rating	3.2	

Knowledge of procedures relating to a security event.

Question: RO #73

IAW OP-HC-108-101-1002, Key Control, security keys can only be used with the permission of the ____(1)______ AND they are located _____(2)_____ ?

A. (1) Security Supervisor; (2) in the Shift Manager's Office

B. (1) Security Supervisor; (2) in the Remote Shutdown Panel

C. (1) SM/CRS; (2) on the Equipment Operator Building Watch key rings

D. (1) SM/CRS; (2) in the Shift Manager's Office AND Remote Shutdown Panel

Proposed Answer: D

Explanation (Optional):

A: (1) Security Supervisor; (2) in the Shift Manager's Office, **INCORRECT**, IAW OP-HC-108-101-1002, 4.3.2. Some key rings are designated to have sensitive keys such as a security key to open card reader doors. These keys are to be used for job related activities only. 1. Security keys should be used only with the permission of the SM/CRS. 2. Per attachment 2, the Shift Manager has six security keys in the SM key locker.

B: (1) Security Supervisor; (2) in the Remote Shutdown Panel, **INCORRECT**, IAW OP-HC-108-101-1002, 4.3.2. Some key rings are designated to have sensitive keys such as a security key to open card reader doors. These keys are to be used for job related activities only. 1. Security keys should be used only with the permission of the SM/CRS. 2. 2. Per attachment 4, there are five security keys in the Remote Shutdown Panel key locker

C: (1) SM/CRS; (2) on the Equipment Operator Building Watch key rings, **INCORRECT**, IAW OP-HC-108-101-1002, 4.3.2. Some key rings are designated to have sensitive keys such as a security key to open card reader doors. These keys are to be used for job related activities only. 1. Security keys should be used only with the permission of the SM/CRS. 2. Security keys are no longer contained on the building watch key rings, however they can be assigned to the building watches as needed for specific job-related activities per step 4.3.2.

D: (1) SM/CRS; (2) in the Shift Manager's Office AND Remote Shutdown Panel, **INCORRECT**, IAW OP-HC-108-101-1002, 4.3.2. Some key rings are designated to have sensitive keys such as a security key to open card reader doors. These keys are to be used for job related activities only. 1. Security keys should be used only with the permission of the SM/CRS. 2. Per attachment 2, the Shift Manager has six security keys in the SM key locker and attachment 4, there are five security keys in the Remote Shutdown Panel key locker.

Technical Reference(s): OP-HC-108-101-1002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NOH04ADM065E002, From memory State when a security key can be used and what needs to be done if a security key is lost. OP-HC-108-101-1002. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.7	
	Importance Rating	3.5	

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Question: RO #74

When signing onto a High Radiation RWP (Radiation Work Permit), you are acknowledging:

A. you know the contamination levels throughout the plant.

B. you have permission to remove items from a Contaminated Area.

C. that a RP (Radiation Protection) brief is required prior to entering the High Rad Area.

D. all barriers, barricades, doors and/or locks used to secure High Rad Areas are secure after entering these areas.

Proposed Answer: C

Explanation (Optional):

A: you know the contamination levels throughout the plant. **INCORRECT**, All workers are responsible for controlling the spread of contamination by: Knowing the contamination levels in and around your work area.

B: you have permission to remove items from a Contaminated Area. **INCORRECT**, All workers are responsible for controlling the spread of contamination by: Knowing the contamination levels in and around your work area. Contacting Radiation Protection before removing any items from a contaminated area.

C: That a RP (Radiation Protection) brief is required prior to entering the High Rad Area. **CORRECT**, By electronically signing the RWP you verify you have read and understand the RWP and will comply with all of these provisions. Ensuring prior to each entry into these areas, that a detailed High Radiation Area Pre-Job Brief, including low dose waiting areas, tasks allowed, alarm set points and proper control of barricades and postings is provided by Radiation Protection personnel.

D: all barriers, barricades, doors and/or locks used to secure High Rad Areas are secure after entering these areas. **INCORRECT**, All workers are responsible for high radiation areas controls by: Ensure all barriers, barricades, doors and/or locks used to secure these areas are secure

prior to exiting these areas.

Technical Reference(s): RP-AA-403, (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NOH04ADM024E001, From Memory Describe what the worker is acknowledging when signing a RWP prior to use. IAW RP-AA-403, Administration of the Radiation Work Permit Program (As available)

Question Source: Bank # 113168

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10, 12

55.43

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.18	
	Importance Rating	3.3	

Knowledge of the specific bases for EOPs.

Question: RO #75

Following a reactor scram with a Station Blackout, the plant is being depressurized using the Safety Relief Valves (SRV).

Which of the following is the reason why the depressurization should be accomplished with "sustained" SRV openings?

A. Due to the loss of 1E bus power, this reduces probability of a RCIC isolation on high exhaust pressure.

B. Due to the loss of 1E bus power, this avoids high local temperatures which may result from insufficient suppression pool cooling.

C. Due to pneumatic supply (PCIG and instrument air) being lost to the SRVs, sustained SRV opening should be utilized to maximize cooldown rate, including exceeding 100°F/hr before valve operation is lost.

D. Due to pneumatic supply (PCIG and instrument air) being lost to the SRVs, this ensures the SRV accumulator pneumatic supply is available and adequate for later use if the Emergency Operating Procedures require SRVs be opened for rapid depressurization of the RPV (e.g. Emergency Depressurization).

Proposed Answer: D

Explanation (Optional):

A: Due to the loss of 1E bus power, this reduces probability of a RCIC isolation on high exhaust pressure. **INCORRECT**, While torus pressure and temperature is expected to rise, it should not be high enough to exceed the high exhaust pressure trip of RCIC

B: Due to the loss of 1E bus power, this avoids high local temperatures which may result from insufficient suppression pool cooling. **INCORRECT**, This is the reason why a opening sequence is followed and a label of the location of the SRVs is installed just above the controls for the SRVs.

C: Due to pneumatic supply (PCIG and instrument air) being lost to the SRVs, sustained SRV opening should be utilized to maximize cooldown rate, including exceeding 100°F/hr before valve operation is lost. **INCORRECT** , OP-EO.ZZ-0101, StepRC/P-6 Bases "Sustained SRV opening, instead of permitting the valves to cycle, conserves accumulator pressure when the source of pressure to the SRV pneumatic supply is isolated or otherwise out of service. Such action to reduce the number of cycles on the SRVs prolongs SRV availability should more degraded conditions later required SRVs be opened for rapid depressurization of the RPV. However, the SRVs are operated so that the cooldown rate LCO (typically 100°F/hr) is not exceeded."

D: Due to pneumatic supply (PCIG and instrument air) being lost to the SRVs, this ensures the SRV accumulator pneumatic supply is available and adequate for later use if the Emergency Operating Procedures require SRVs be opened for rapid depressurization of the RPV (e.g. Emergency Depressurization). **CORRECT**, OP-EO.ZZ-0101, StepRC/P-6 Bases
Sustained SRV opening, instead of permitting the valves to cycle, conserves accumulator pressure when the source of pressure to the SRV pneumatic supply is isolated or otherwise out of service. Such action to reduce the number of cycles on the SRVs prolongs SRV availability should more degraded conditions later required SRVs be opened for rapid depressurization of the RPV.

Technical Reference(s): EOP-101 Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EO101PE005, Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step. (As available)

Question Source: Bank # 53437

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 10

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	K/A #	295005	AA2.07	
	Importance Rating			3.6

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Reactor water level

Question: SRO #76

Given:

The plant is operating at 100% reactor power.

Then:

- OHA A7-B5, RPV LEVEL 7 and A7-A5, RPV LEVEL 8 are received.
- Reactor Narrow Range Level indicators are:
  R606A-C32 @ 58"
  R606B-C32 @ 57.5"
  R606C-C32 @ 58.5"
- Main Generator output remains ≈ 1275 MWe

As the CRS, what actions are you directing?

A. Lock the Mode Switch in Shutdown and enter AB.ZZ-0000, Reactor Scram

B. Manually trip the Main Turbine and enter EOP-101A, ATWS RPV Control.

C. Lock the Mode Switch in Shutdown and enter EOP-101A, ATWS RPV Control.

D. Take manual control of feedwater flow and lower reactor water level to its normal band using AB-RPV-0004 RPV Level Control.

Proposed Answer: A

Explanation (Optional):

A: Lock the Mode Switch in Shutdown and enter AB-0000 Reactor Scram, (because the turbine remains on line above its trip setpoint.) **CORRECT** Retainment override of RPV -0004 requires locking the Mode Switch in Shutdown >50 inches RPV Level. Since level is above this value, the retainment action applies. After the scram EOP-101 entry not expected on +12.5 entry and AB-0000 Scram actions after tripping the turbine. Automatic Main turbine trip did not occur. Per OP-AA-101-111-1003 section 4.5.5, trip the main turbine upon failure of the automatic action. However the reactor must be scrammed first as directed by AB-RPV-0004 at >+50 inches.

B: Manually trip the Main Turbine and enter EOP-101A ATWS, (because the turbine remains on line above its trip setpoint and a scram should have occurred). **INCORRECT**, Not a failure to scram condition. The ECG tech bases 5.1.2.a / 5.1.2.b describes a failure for RPS to actuate or Failure of CRD to insert rods. RPS is not challenged if the turbine has not tripped.

C: Lock the Mode Switch in Shutdown and enter EOP-101A, ATWS,(because the turbine remains on line above its trip setpoint and a scram should have occurred.) **INCORRECT**, Not a failure to scram condition. The ECG tech bases 5.1.2.a / 5.1.2.b describes a failure for RPS to actuate or Failure of CRD to insert rods. RPS is not challenged if the turbine has not tripped.

D: Take manual control of feedwater flow and slowly lower reactor water level to its normal band using AB-RPV-0004 RPV Level Control, (because digital feedwater level control is malfunctioning and not controlling reactor level at the 35" setpoint.) **INCORRECT**, Retainment override of RPV -0004 requires locking the Mode Switch in Shutdown >50 inches RPV Level. Since level is above this value, the retainment action applies.

Technical Reference(s): AR.ZZ-0005 Att, A5 and B5, AB.RPV-0004, OP-AA-101-111-1003 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ABBOP2E001, Recognize abnormal indications/alarms and/or procedural requirements for implementing Main Turbine. ABRPV4E001, Recognize abnormal indications/alarms and/or procedural requirements for implementing Reactor Level Control. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	295030	EA2.01
Importance Rating		4.2

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Suppression pool level

Question: SRO #77

The plant operating at 100% power.

Then:

Suppression Pool Level is 75" and dropping rapidly ≈1"/minute.

What are your actions and by what torus level are you required to order these actions?

A. Scram the reactor then emergency depressurize the RPV IAW EOP-202, Emergency Depressurization when torus level can NOT be maintained above 55"

B. Runback Recirc to minimum then scram the reactor IAW EOP-102, Primary Containment Control. when torus level can NOT be maintained above 55"

C. Runback Recirc to intermediate AND scram the reactor IAW EOP-102, Primary Containment Control. when torus level can NOT be maintained above 38.5"

D. Secure RCIC regardless of adequate core coverage IAW OP-SO.BD-0001, RCIC System Operating procedure when torus level can NOT be maintained above 30"

Proposed Answer: B

Explanation (Optional):

A: Scram the reactor and emergency depressurize the RPV when torus level is $\leq$ 55". **INCORRECT**, the action to emergency depressurize the RPV comes from EOP-102 step SP/L-7 and is required at 38.5" torus level NOT 55". Scramming the reactor is appropriate at 55" per step SP/L-6.

B: Runback Recirc to minimum and scram the reactor when torus level is ≤ 55". **CORRECT**, Running reactor Recirc to minimum and scramming the reactor is appropriate at 55" per step SP/L-6 EOP-102.

C: Runback Recirc to intermediate and scram the reactor when torus level is ≤ 38.5". **INCORRECT**, The action to emergency depressurize the RPV comes from EOP-102 step SP/L-7 and is required at 38.5" torus level NOT running Recirc back to intermediate. Running reactor Recirc to minimum and scramming the reactor is appropriate at 55" per step SP/L-6 EOP-102.

D: Isolate RCIC regardless on adequate core coverage when torus level is ≤ 30". **INCORRECT**, the action taken at torus level of 30" or less is to isolate HPCI regardless of adequate core cooling is assured, step SP/L-11 of EOP-102.

Technical Reference(s): EOP-102 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: PRICONE009, Given a Scenario of applicable operating conditions and access to Technical Specifications: a. Select those sections that are applicable to the Primary Containment Structure IAW HCGS Technical Specifications. b. Evaluate Primary Containment Structure operability and determine required actions based upon system operability IAW HCGS Technical Specifications. (SRO/STA ONLY) c. Explain the bases for those technical specification items associated with the Primary Containment Structure IAW HCGS technical specifications. (As available)

Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295006	AA2.02
	Importance Rating		4.4

Ability to determine and/or interpret the following as they apply to SCRAM : Control rod position

Question: SRO #78

When are you (1) required to exit AB.ZZ-0000(Q), "Reactor Scram," and enter EO.ZZ-0101A, ATWS - RPV Control," and (2) how is this accomplished?

A. (1) When all control rods are found at notch "02" (2) Based on indications from the Four Rod Display.

B. (1) If all control rods are full in, with the exception of one at notch "02" (2) Based on indications using the Full Core Display.

C. (1) If all control rods are full in, with the exception of one at notch "48" (2) Based on indications using Rod Worth Minimizer.

D. (1) When all control rods are inserted to notch "02", with the exception of one at notch "20" (2) Based on indications using CRIDS.

Proposed Answer: D

Explanation (Optional):

A: (1) When all control rods are found at notch "02" (2) Based on indications from the Four Rod Display. **INCORRECT,** With all control rods at position "02" the reactor is considered shutdown under all conditions and there is no reason to exit AB-0000 and enter EOP-101A.

B: (1) If all control rods are full in, with the exception of one at notch "02" (2) Based on indications using the Full Core Display. **INCORRECT,** With all control rods full in and one rod at position "02" the reactor is considered shutdown under all conditions and there is no reason to exit AB-0000 and enter EOP-101A.

C: (1) All control rods full in, with the exception of one at Notch "48" (2) Based on indications using Rod Worth Minimizer. **INCORRECT,** With all control rods full in and one at position "48" the reactor is considered shutdown under all conditions and there is no reason to exit AB-0000 and enter EOP-101A.

D: (1) When all control rods are inserted to notch "02", with the exception of one at notch "20" (2) Based on indications using CRIDS. **CORRECT,** With all control rods at position "02" and one at "20" the reactor is NOT considered shutdown under all conditions and exiting AB-0000 and entering EOP-101A is required as this is a retainment override step in AB-000.

Technical Reference(s): AB.ZZ-0000, EOP-101A Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EO101AE003, Explain the significance of "Maximum Subcritical Banked Withdrawal Position" and state its value. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	K/A #	295021	2.4.34	
	Importance Rating			4.1

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (Loss of Shutdown Cooling)

Question: SRO #79

Given:

- The plant is operating at 100% power.
- All systems are aligned for normal operations.
- A fire started under Control Room console 10C651 causing a reactor scram.
- The Control Room has been evacuated because of extreme smoke conditions.
- The reactor has been depressurized to less than 80 psig with SRV's and RCIC.
- "B" RHR was in Suppression Pool Cooling prior to the pump tripping due to failed control power circuit in the Remote Shutdown Panel (10C399).

What describes the system to be used to achieve Cold Shutdown, and what is the maximum cooldown rate that is permitted?

A. IAW HC.OP-SO.BC-003, RHR Alternate Cooling Modes, place Alternate Decay Heat Removal in service using "C" RHR from the field with a cooldown rate NOT to exceed 100°F/Hour.

B. IAW HC.OP-SO.BC-003, RHR Alternate Cooling Modes, place Alternate Decay Heat Removal in service using "D" RHR from the field with a cooldown rate NOT to exceed 100°F/Hour.

C. IAW HC.OP-IO.ZZ-008, Shutdown from Outside the Control Room, place Shutdown Cooling in-service using "A" RHR from the field with a cooldown rate NOT to exceed 90°F/Hour.

D. IAW HC.OP-IO.ZZ-008, Shutdown from Outside the Control Room, place Shutdown Cooling in-service using "B" RHR from the field with a cooldown rate NOT to exceed 9 0°F/Hour.

Proposed Answer: C

Explanation (Optional):

A: IAW HC.OP-SO.BC-0003, RHR Alternate Cooling Modes, place Alternate Decay Heat Removal in service using "C" RHR from the field with a cooldown rate NOT to exceed 90°F/Hour. **INCORRECT**, Alternate Decay Heat Removal is not available to be placed in service using IO.ZZ-0008, guidance exists in AB.RPV-0009, but not all equipment can be manipulated from the field and require access to the main control room to complete. Cooldown rate is set @ ≤ 90°F/Hour per step 5.10.7 of IO.ZZ-0008. Guidance in IO-008 is following a loss of power, that condition is not mentioned in the stem of the question.

B: IAW HC.OP-SO.BC-0003, RHR Alternate Cooling Modes, place Alternate Decay Heat Removal in service using "D" RHR from the field with a cooldown rate NOT to exceed 90°F/Hour. **INCORRECT,** Alternate Decay Heat Removal is not available to be placed in service using IO.ZZ-0008, guidance exists in AB.RPV-0009, but not all equipment can be manipulated from the field and require access to the main control room to complete. Cooldown rate is set @ ≤ 90°F/Hour per step 5.10.7 of IO.ZZ-0008. Guidance in IO-008 is following a loss of power, that condition is not mentioned in the stem of the question.

C: IAW HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room, place Shutdown Cooling in-service using "A" RHR from the field with a cooldown rate NOT to exceed 100°F/Hour. **CORRECT,** A RHR is available to be placed in service using IO.ZZ-0008, all necessary equipment can be manipulated from the field and requires NO access to the main control room to complete. Cooldown rate is set @ ≤ 90°F/Hour per step 5.10.7 of IO.ZZ-0008

D: IAW HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room, place Shutdown Cooling in-service using "B" RHR from the field with a cooldown rate NOT to exceed 100°F/Hour. **INCORRECT,** Cooldown rate is set @ ≤ 90°F/Hour per step 5.10.7 of IO.ZZ-0008 also not guidance to place in service from the breaker contained in procedure.

Technical Reference(s): IO.ZZ-008 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: IOP008E004, Apply Precautions, Limitations and Notes while executing the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295004	2.1.23
	Importance Rating		4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation (Partial or Complete Loss of DC power)

Question: SRO #80

Given:

- At 07:30 Main generator output reached 1280 MWe.
- At 09:00 a common mode failure has disabled both chargers supplying the 1AD411 battery.
- Applicable Abnormal procedures have been entered.
- At 10:00 12 hr Maintenance has reported the battery chargers cannot be restored for at least 24 hours.

What crew actions are required by plant procedures?

A. Immediately Declare HPCI Inoperable and enter a 72 hour LCO.

B. Immediately Declare HPCI Inoperable and enter a 14 day LCO.

C. Commence a reactor shutdown at 11:00 and contact Station Duty Manager to make applicable notifications IAW Significant Event Reporting, OP-AA-106-101.

D. Commence a reactor shutdown at 23:00 and contact Station Duty Manager to make applicable notifications IAW Significant Event Reporting, OP-AA-106-101.

Proposed Answer: C

Explanation (Optional):

A: Immediately Declare HPCI Inoperable and enter a 72 hour LCO. **INCORRECT,** HPCI is a 250 VDC battery 10D421 and is not required to be declared INOP. This spec would only apply if one LPCI subsystem or Core Spray subsystem was also INOP.

B: Immediately Declare HPCI Inoperable and enter a 14 day LCO. **INCORRECT,** HPCI is a 250 VDC battery 10D421 and is not required to be declared INOP. Battery spec is more limiting and takes into account the other systems that are impacted by its loss.

C: Commence a reactor shutdown at 11:00 and contact Station Duty Manager to make applicable notifications IAW Significant Event Reporting - OP-AA-106-101. **CORRECT,** Tech Spec 3.8.2.1.a "With any 125 v battery and/or al associated chargers of the above required D.C. electrical power sources inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours". The 2 hour spec ends at 11:00. Per OP-AA-106-101, Significant Event Reporting, the Station Duty Manager (SDM) is required make notifications IAW Attachment 1, for a forced entry into a shutdown LCO of 72 hours or less.

D: Commence a reactor shutdown at 23:00 and contact Station Duty Manager to make applicable notifications IAW Significant Event Reporting - OP-AA-106-101. **INCORRECT,** Tech Spec 3.8.2.1.a "With any 125 v battery and/or al associated chargers of the above required D.C. electrical power sources inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours". The 2 hour spec ends at 11:00. Per OP-AA-106-101, Significant Event Reporting, the Station Duty Manager (SDM) is required make notifications IAW Attachment 1, for a forced entry into a shutdown LCO of 72 hours or less.

Technical Reference(s): Tech Specs 3.8.2.1, OP-AA-106-101 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Tech Spec 3.8.2.1

Learning Objective: EO101LE006, Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulation prescribed by that step. (As available)

Question Source: Bank #

Modified Bank # 116202 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2, 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	K/A #	295024	2.4.41	
	Importance Rating			4.6

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications. (High Drywell Pressure)

Question: SRO #81

The plant is operating at 100% power

Then:

- A station blackout occurs and HPCI and RCIC are NOT available

Five (5) minutes post scram conditions are:

• Reactor Scram	Complete
• Reactor Pressure	650 psig and stable
• Reactor level	-165” dropping 2”/Hr.
• Drywell pressure	2.5 psig rising 0.1psi/min
• Drywell temperature	190°F rising 1°/30 min.
• Suppression pool level	80” rising 0.05”/Hr
• Suppression chamber temperature	140°F rising 1°F/Hr

What is the ECG classification at this time?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: C

Explanation (Optional):

A: Unusual Event **INCORRECT**, See SAE explanation

B: Alert, **INCORRECT,** See SAE explanation

C: Site Area Emergency **CORRECT,** With reactor vessel level less than 161" (-165") and NO available feed requires a declaration of a SAE, per Fission Product Barrier table EAL 3.1.1a [3 points], EAL 3.2.1.b or 3.2.2.b [4 points] total of 7 points

D: General Emergency **INCORRECT,** see SAE explanation

Technical Reference(s): ECG Section 3.0 Barrier Table (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Barrier table Sect 3.0 ECG

Learning Objective: SOB200, ECG/E-Plan/Fire & Medical Questions (As available)

Question Source: Bank #

Modified Bank # 56171 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031	EA2.04
	Importance Rating		4.8

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Adequate core cooling

Question: SRO #82

A LOCA has occurred resulting in:

- All control rods are fully inserted.
- Reactor pressure is 50 psig.
- Drywell pressure is 3.5 psig.
- Compensated Reactor Water level is -140" and steady.
- A RHR is injecting into the reactor vessel at 10,000 gpm.
- Remaining Low Pressure ECCS pumps are now available.
- NO other pumps are injecting.

What additional actions (if any) are required?

A. NO other actions are required IAW EOP-101 RPV Control.

B. Inject to the vessel with B RHR at full flow IAW EOP-101 RPV Control.

C. Inject to the vessel with B, C and D RHR pumps at full flow along with one loop of Core Spray IAW EOP-206 RPV Flooding.

D. Inject to the vessel with B, C and D RHR pumps at full flow along with both loops of Core Spray IAW EOP-206 RPV Flooding.

Proposed Answer: B

Explanation (Optional):

A: NO other actions are required IAW EOP-101 RPV Control. **INCORRECT**, Per EOP-101 step RC/L-5, "Restore AND Maintain RPV level above -129 in. RPV level control may be augmented by 1 or more Alternate Injection Systems (Table 3)" With level at -140" an additional injection system is required and B RHR is a Table 2 subsystem and should be used to restore level.

B: Inject to the vessel with B RHR at full flow IAW EOP-101 RPV Control. **CORRECT,** Per EOP-101 step RC/L-5, "Restore AND Maintain RPV level above -129 in. RPV level control may be augmented by 1 or more Alternate Injection Systems (Table 3)" With level at -140" an additional injection system is required and B RHR is a Table 2 subsystem and should be used to restore level.

C: Inject to the vessel with B, C and D RHR pumps at full flow along with one loop of Core Spray IAW EOP-206 RPV Flooding. **INCORRECT,** NO entry exists into EOP-206. Accurate level indication is available as stated in the stem, "compensated level is -140". No actions are to be performed IAW EOP-206.

D: Inject to the vessel with B, C and D RHR pumps at full flow along with both loops of Core Spray IAW EOP-206 RPV Flooding. **INCORRECT,** NO entry exists into EOP-206. Accurate level indication is available as stated in the stem, "compensated level is -140". No actions are to be performed IAW EOP-206.

Technical Reference(s): EOP-101, EOP-206 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: INADCCE006, Given a set of plant conditions, evaluate the capability of existing emergency core cooling systems to maintain adequate core cooling IAW the Student Handout. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295013	AA2.01
	Importance Rating		4.0

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression Pool Temperature.

Question: SRO #83

The plant is operating at 100% power with HPCI pump testing in progress. Which one of the following requires HPCI to be shutdown AND what actions are required for that temperature to satisfy Tech Specs?

A. Average Suppression Pool Temp reaches 95°. Restore average temperature to ≤ 95°F within 24 hours.

B. Average Suppression Pool Temp reaches 105°F. Restore average temperature to ≤ 95°F within 24 hours.

C. Any one Suppression Pool Temp sensor reaches 110°F. LOCK the mode switch in shutdown and place a loop of RHR in torus cooling.

D. Any one Suppression Pool Temp sensor reaches 120°F. Depressurize the RPV to < 200 psig within 12 hours.

Proposed Answer: B

Explanation (Optional):

A: Average Suppression Pool Temp reaches 95°. Restore average temperature to ≤ 95°F within 24 hours. **INCORRECT,** IAW T/S 3.6.2.1.a.2, Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to: a) 105°F during testing which adds heat to the suppression chamber. b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95° within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

B: Average Suppression Pool Temp reaches 105°F. Restore average temperature to ≤ 95°F within 24 hours. **CORRECT**, IAW T/S 3.6.2.1.a.2, Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to: a) 105°F during testing which adds heat to the suppression chamber. b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95° within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

C: Any one Suppression Pool Temp sensor reaches 110°F. LOCK the mode switch in shutdown and place a loop of RHR in torus cooling. **INCORRECT,** IAW T/S 3.6.2.1.a.2, Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to: a) 105°F during testing which adds heat to the suppression chamber. b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER. With the suppression chamber average water temperature greater than 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

D: Any one Suppression Pool Temp sensor reaches 120°F. Depressurize the RPV to < 200 psig within 12 hours. **INCORRECT,** IAW T/S 3.6.2.1.a.2, Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to: a) 105°F during testing which adds heat to the suppression chamber. b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER. 3.Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

Technical Reference(s): Tech Spec 3.6.2.1.a.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP102E004, From memory, recall the reason why average suppression pool temperature is used for determining the entry condition and subsequent actions IAW available control room references. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	295032	2.1.20
Importance Rating		4.6

Conduct of Operations: Ability to interpret and execute procedure steps. (High Secondary Containment Area Temperature)

Question: SRO #84

Given:

- The reactor is operating at rated power
- RCIC is in service for its quarterly surveillance test
- RCIC room temperature rises to 125° F.
- The RCIC room cooler fans have both seized

The operators complete the RCIC surveillance and secure RCIC.
All surveillance data is satisfactory, however, RCIC room temperature is still ≈ 123 °F.

Then:

- Annunciator B1-A3 (HPCI PUMP ROOM FLOODED) alarms
- The Reactor Building Operator reports the leak is coming from the HPCI CST suction line between the isolation valve (BJ-HV-F004) and check valve 1-BJ-V006
- HPCI Room 4111 water level is 1.5 inches
- Operators close the CST suction valve to HPCI, BJ-HV-F004- HPCI CST SUCTION VALVE isolating the leak.
- The Torus suction valve to HPCI, BJ-HV-F042- HPCI TORUS SUCTION VALVE CAN NOT be opened.

What actions are required?

A. Enter TS LCO for HPCI AND plant operation may continue for 14 days.

B. Enter TS LCO for RCIC AND plant operation may continue for 14 days.

C. Enter TS LCO for HPCI and RCIC AND a plant shutdown is required.

D. Enter the TS LCO for RCIC and HPCI being inoperable AND per EOP 103/4-Reactor Building and Rad Release Control, Runback reactor recirculation, manually scram the reactor.

Proposed Answer: C

Explanation (Optional):

A: Enter TS LCO for HPCI AND plant operation may continue for 14 days. **INCORRECT** - RCIC surveillance was satisfactory, however 2 room coolers INOPERABLE and IAW OP-HC-108-115-1001 Exhibit 2, RCIC must be declared INOP. HPCI has NO suction available, therefore a TS LCO is also required.

B: Enter TS LCO for RCIC AND plant operation may continue for 14 days. **INCORRECT** - RCIC surveillance was satisfactory, however 2 room coolers INOPERABLE and IAW OP-HC-108-115-1001 Exhibit 2, RCIC must be declared INOP. HPCI has NO suction available, therefore a TS LCO is also required.

C: Enter TS LCO for HPCI and RCIC AND a plant shutdown is required. **CORRECT** - RCIC surveillance was satisfactory, however 2 room coolers INOPERABLE and IAW OP-HC-108-115-1001 Exhibit 2, RCIC must be declared INOP. HPCI has NO torus suction and must be declared INOP and a normal plant shutdown per Tech Specs (3.0.3) is required.

D: Enter the TS LCO for RCIC and HPCI being inoperable AND per EOP 103/4-Reactor Building and Rad Release Control, Runback reactor recirculation, manually scram the reactor. **INCORRECT -** 2 areas are not above max safe, immediate shutdown NOT required. Primary system not discharging into containment. RCIC surveillance was satisfactory, however 2 room coolers INOPERABLE and IAW OP-HC-108-115-1001 Exhibit 2, RCIC must be declared INOP. HPCI has NO torus suction and must be declared INOP and a normal plant shutdown per Tech

Specs (3.0.3) is required.

Technical Reference(s): EOP-103/4 Tech Spec 3.5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # 119525 (Note changes or attach parent)

New

Question History: NRC 2010

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2, 5

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295017	2.4.30
	Importance Rating		4.1

Emergency Procedures / Plan; 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. (High Off-site Release Rates)

Question: SRO #85

Given:

- The Unit is in OPCON 1 at 100% power.
- A large leak develops in the RWCU system.
- OHA D3-B3 RWCU STM LK ISLN TIMER INITIATED alarms.
- South Plant Vent activity begins to rise.

Then:

- OHA C1-A2 RWCU DIFF FLOW HI alarms.
- The RBEO reports steam in the RWCU Heat Exchanger room
- Reactor Operator's attempts to secure the RWCU are unsuccessful.
- Total Plant Vent release rate is now 4.5 E+3μCi/cc and rising

20 Minutes later:

- Both RWCU pumps have just been tripped at their respective 480 breakers.
- BG-HV-F001, BG-HV-F004 and AE-HV-F039 are now all closed. (breaker override)
- South Plant Vent Noble Gas is in High alarm.
- Total Plant Vent release rate is now 5.0 E+3μCi/cc and slowly stabilizing
- Radiation Protection is unable to perform a dose assessment

IAW the ECG, (1) which one of the following is the HIGHEST level event classification that applies to the event described above and (2) who is required to be notified within 15 minutes?

	(1)	(2)
A.	Alert	NJ and Delaware State Police ONLY
B.	Alert	NJ and Delaware State Police AND LAC Township
C.	Site Area Emergency	NJ and Delaware State Police ONLY
D.	Site Area Emergency	NJ and Delaware State Police AND LAC Township

Proposed Answer: C

Explanation (Optional):

A: Alert, NJ and Delaware State Police ONLY, **INCORRECT**, Section 3 EALs. With the failure of RWCU to isolate and attempts to close the valves failed. EAL 3.2.3.a [3 points] and 3.3.4.a [2 points] add up to an SAE of 5 points, NJ and Delaware State Police are required to be notified within 15 minutes per ECG Att. 6 Primary Communicator

B: Alert, NJ and Delaware State Police AND LAC Township **INCORRECT**, Section 3 EALs. With the failure of RWCU to isolate and attempts to close the valves failed. EAL 3.2.3.a [3 points] and 3.3.4.a [2 points] add up to an SAE of 5 points, NJ and Delaware State Police are required to be notified within 15 minutes and LAC Township within 30 minutes per ECG Att. 6 Primary Communicator

C: Site Area Emergency, NJ and Delaware State Police ONLY, **CORRECT**, Section 3 EALs. With the failure of RWCU to isolate and attempts to close the valves failed. EAL 3.2.3.a [3 points] and 3.3.4.a [2 points] add up to an SAE of 5 points, NJ and Delaware State Police are required to be

notified within 15 minutes per ECG Att. 6 Primary Communicator

D: Site Area Emergency, NJ and Delaware State Police AND LAC Township **INCORRECT** Section 3 EALs. With the failure of RWCU to isolate and attempts to close the valves failed. EAL 3.2.3.a [3 points] and 3.3.4.a [2 points] add up to an SAE of 5 points, NJ and Delaware State Police are required to be notified within 15 minutes and LAC Township within 30 minutes per ECG Att. 6 Primary Communicator

Technical Reference(s): ECG EALs and RALs, Technical Basis 6.0, 11.3, Sect 3.0 Barrier Table (Attach if not previously provided)

Proposed References to be provided to applicants during examination: ECG, Sect 11.3, Tech Bases pages 1-5 only (RALs) and Barrier Table Sect 3.0

Learning Objective: SOB200, ECG/E-Plan/Fire & Medical Questions (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2, 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	206000	A2.04
	Importance Rating		3.0

Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. failures: BWR-2,3,4

Question: SRO #86

The plant is at 100% power:

The Reactor Building Equipment Operator investigates an inadvertent trip of the 52-212053 for the FD-HV-F003 Steam Supply Outboard Isolation valve.

30 minutes later,

The on-shift electrician reports that the motor overloads are damaged and must be replaced.

Replacement and retest of the motor overloads will take two (2) hours.

What affect will this have on the HPCI system AND what Tech Spec action (if any) is required?

A. NO affect on the system, HPCI is operable AND NO Tech Spec action is required.

B. HPCI will NOT function to inject to the vessel AND must be declared inoperable per T/S 3.5.1 ECCS AND the FD-HV-F003 can be manually operated.

C. HPCI will function to inject to the vessel, T/S 3.6.3 Primary Containment must be entered and the FD-HV-F003 can be manually operated.

D. HPCI will NOT function to inject to the vessel, AND T/S 3.5.1 ECCS AND 3.6.3 Primary Containment must be entered because the FD-HV-F003 can NOT function automatically.

Proposed Answer: C

Explanation (Optional):

A: NO affect on the system, HPCI is operable AND NO Tech Spec action is required. **INCORRECT**, the system is affected in as much as the steam isolation valve FD-HV-F003 is powered from 10B212 and will not isolate on any isolation signal. T/S for containment isolation 3.6.3, must be entered as the penetration must be isolated within 4 hours or restore power back to the F003 valve.

B: HPCI will NOT function to inject to the vessel AND must be declared inoperable per T/S 3.5.1 ECCS AND the FD-HV-F003 can be manually operated. **INCORRECT,** the system will still initiate and inject to the RPV. Steam isolation valve FD-HV-F003 is powered from 10B212 and will not isolate on any isolation signal. T/S for containment isolation 3.6.3, must be entered as the penetration must be isolated within 4 hours or restore power back to the F003 valve.

C: HPCI will function to inject to the vessel, T/S 3.6.3 Primary Containment must be entered and the FD-HV-F003 can be manually operated. **CORRECT,** the system will still initiate and inject to the RPV. Steam isolation valve FD-HV-F003 is powered from 10B212 and will not isolate on any isolation signal. T/S for containment isolation 3.6.3, must be entered as the penetration must be isolated within 4 hours or restore power back to the F003 valve.

D: HPCI will NOT function to inject to the vessel, AND T/S 3.5.1 AND 3.6.3 must be entered because the equipment powered from 10B212 can NOT function automatically. **INCORRECT,** the system will still initiate and inject to the RPV. Steam isolation valve FD-HV-F003 is powered from 10B212 and will not isolate on any isolation signal. T/S for containment isolation 3.6.3, must be entered as the penetration must be isolated within 4 hours or restore power back to the F003 valve.

Technical Reference(s): Tech Spec 3.5.1, 3.6.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: HPCI00 E018, Given plant conditions and access to Technical Specifications: a. Select those sections, which are applicable to the HPCI System IAW HCGS technical specifications. b. Evaluate HPCI System operability and required actions based upon system operability IAW HCGS technical specifications. **(SRO Only).** c. Explain the bases for those technical specification items associated with the HPCI System IAW HCGS technical specifications. **(SRO Only)**>> (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2, 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	262001	A2.04
	Importance Rating		4.2

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Types of loads that, if de-energized, would hinder plant operation.

Question: SRO #87

Given:

The plant has sustained a station blackout.
A Reactor Coolant System leak inside the drywell has reduced reactor pressure to 150 psig.
Reactor water level has dropped to -120 inches and continues to lower at ~ 2"/minute

Maintenance reports that they can restore the "A" EDG in ~ 20 minutes and the "C" EDG in ~ 10 minutes, but they only have enough people to work on one EDG at a time.

For the stated systems, which EDG do you direct them to restore and why?

A. Restore "A" EDG first because in 5 minutes Core Spray injection is required per EOP-101.

B. Restore "C" EDG first because in 5 minutes Core Spray injection is required per EOP-101.

C. Restore "A" EDG first to make the SRVs available, because in 35 minutes an emergency depressurization is required per EOP-202.

D. Restore "C" EDG first to make the SRVs available, because in 35 minutes an emergency depressurization is required per EOP-202.

Proposed Answer: A

Explanation (Optional):

A: Restore "A" EDG first because in 5 minutes Core Spray injection is required per EOP-101. **CORRECT**, In 5 minutes RPV level will be -130" and per EOP-101 step ALC-2 "Restore and maintain RPV level above -129" with 1 or more preferred injection systems (Table 1)". For injection from A Core Spray pump to be possible it is necessary to have power available to the

HV-F005A Injection valve, which is powered from the 10A401 bus, which is restored via the A EDG

B: Restore "C" EDG first because in 5 minutes Core Spray injection is required per EOP-101. **INCORRECT –** In 5 minutes RPV level will be -130" and per EOP-101 step ALC-2 "Restore and maintain RPV level above -129" with 1 or more preferred injection systems (Table 1)". For injection from C Core Spray pump to be possible it is necessary to have power available to the HV-F005A Injection valve, which is powered from the 10A401 bus, which is restored via the A EDG, not the C EDG.

C: Restore "A" EDG first to make the SRVs available, because in 35 minutes an emergency depressurization is required per EOP-202. **INCORRECT-** In 35 minutes RPV level will be -190" and per EOP-101 step ALC-10 "Before RPV level reaches -185" and ALC-11 Emergency RPV Depressurization is required" The ADS SRVs are powered from B and D channels and by not having either A or C EDG to power their respective busses, has no effect on the operation of the SRVs ability to depressurize the RPV.

D: Restore "C" EDG first to make the SRVs available, because in 35 minutes an emergency depressurization is required per EOP-202. **INCORRECT-** In 35 minutes RPV level will be -190" and per EOP-101 step ALC-10 "Before RPV level reaches -185" and ALC-11 Emergency RPV Depressurization is required" The ADS SRVs are powered from B and D channels and by not having either A or C EDG to power their respective busses, has no effect on the operation of the SRVs ability to depressurize the RPV.

Technical Reference(s): EOP-101, NOH01CSSYS0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP-101, step ALC-2 thru ALC-18 only

Learning Objective: 1EAC00E003, Given a list of electrical loads (motor/unit substations), choose which are powered from the 1E 4.16KV switchgear. (As available)

Question Source: Bank #

Modified Bank # 110606 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	261000	2.2.44
	Importance Rating		4.4

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. (SGTS)

Question: SRO #88

Given:

A loss of both EHC pumps requires a manual scram
HPCI and RCIC receive an auto start signal during the transient
HC.OP-EO.ZZ-0101 is entered
The reactor is confirmed shutdown
All RPV parameters are now stable

Due to the response of the ventilation systems in the reactor building, as the CRS, what actions (if any) are you directing?

A. There is no affect on the RBVS due to this transient, continue monitoring reactor building Δp IAW OP-SO.GR-001, Reactor Building Ventilation System.

B. Verify FRVS Auto Initiation IAW OP-AB.ZZ-001, Transient Plant Conditions, and secure both A and B FRVS Vent fans per OP-SO.GR-001, Reactor Building Ventilation Operation

C. Verify RBVS Isolation IAW the OP-SO.GR-001, Reactor Building Ventilation System and place FRVS in-service IAW OP-SO.GU-001, Filtration, Recirculation and Ventilation Operation.

D. Verify RBVS Isolation and FRVS Auto Initiation IAW OP-SO.SM-001, Isolation System Operation and secure E/F FRVS Recirc fans per OP-SO.GU-001, Filtration, Recirculation and Ventilation Operation

Proposed Answer: D

Explanation (Optional):

A: There is no affect on the RBVS due to this transient, continue monitoring reactor building Δp IAW OP-SO.GR-001, Reactor Building Ventilation System. **INCORRECT,** All RBVS supply and exhaust fans trip and inlet and outlet dampers close due to LOCA level 2 (-38") OP-SO.SM-001 shows the isolation/start signals for both RBVS and FRVS. The OP-SO.GU-001 has guidance to secure recirc fans E/F post isolation/initiation.
Verify FRVS Auto Initiation IAW OP-AB.ZZ-001, Transient Plant Conditions, and secure both A and B FRVS Vent fans per OP-SO.GR-001, Reactor Building Ventilation Operation.

B: **INCORRECT,** All RBVS supply and exhaust fans trip and inlet and outlet dampers close due to LOCA level 2 (-38") OP-SO.SM-001 shows the isolation/start signals for both RBVS and FRVS. The OP-SO.GU-001 has guidance to secure recirc fans post isolation/initiation not the vent fans.

C: Verify RBVS Isolation IAW the OP-SO.GR-001, Reactor Building Ventilation System and place FRVS in-service IAW OP-SO.GU-001, Filtration, Recirculation and Ventilation Operation. **INCORRECT,** All RBVS supply and exhaust fans trip and inlet and outlet dampers close due to LOCA level 2 (-38") OP-SO.SM-001 shows the isolation/start signals for both RBVS and FRVS. The OP-SO.GU-001 has guidance to secure recirc fans E/F post isolation/initiation. Placing FRVS in-service would be redundant as the system has already initiated on the level 2.

D: Verify RBVS Isolation and FRVS Auto Initiation IAW OP-SO.SM-001, Isolation System Operation and secure E/F FRVS Recirc fans per OP-SO.GU-001, Filtration, Recirculation and Ventilation Operation **CORRECT,** All RBVS supply and exhaust fans trip and inlet and outlet dampers close due to LOCA level 2 (-38") OP-SO.SM-001 shows the isolation/start signals for both RBVS and FRVS. The OP-SO.GU-001 has guidance to secure recirc fans E/F post isolation/initiation.

Technical Reference(s): OP-SO.GR-001, OP-SO.GU-001 (Attach if not previously provided)
OP-AB.ZZ-001, OP-SO.SM-001

Proposed References to be provided to applicants during examination: none

Learning Objective: RBVENTE011, Given plant conditions, summarize/identify how the Reactor Building Ventilation System responds to a LOP and/or LOCA. (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	K/A #	215004	2.2.38	
	Importance Rating			4.5

Knowledge of the conditions and limitations in the facility license (SRM)

Question: SRO #89

Given:

- Refueling is in progress with 550 fuel bundles remaining in the vessel
- The reactor mode switch is locked in REFUEL
- Source Range Monitors (SRMs) A, B, and C are operable
- SRM D is inoperable and bypassed
- Shutdown margin has been verified
- All control rods are at position 00

With the refueling platform unloaded over the vessel, the count rate on SRM B drops to 1.5 cps

Core Alterations…

A. are permitted in quadrants A and C ONLY.

B. can continue if NO fuel movement occurs.

C. must be formally suspended.

D. can continue with NO restrictions.

Proposed Answer: C

Explanation (Optional):

A: Are permitted in quadrants A and C ONLY, **INCORRECT**, The T/S limits (3.9.2.b) for minimum operable SRMs is NOT met. Requires an SRM in the quadrant being loaded and the adjacent quadrant. With both B and D SRM inoperable, NO operable adjacent SRM exists and core

alterations must be formally suspended,

B: can continue if NO fuel movement occurs, **INCORRECT**, The T/S limits (3.9.2.b) for minimum operable SRMs is NOT met. Requires an SRM in the quadrant being loaded and the adjacent quadrant. With both B and D SRM inoperable, NO operable adjacent SRM exists and core alterations must be formally suspended,

C: must be formally suspended, **CORRECT**, The T/S limits (3.9.2.b) for minimum operable SRMs is NOT met. Requires an SRM in the quadrant being loaded and the adjacent quadrant. With both B and D SRM inoperable, NO operable adjacent SRM exists and core alterations must be formally suspended,

D: can continue with NO restrictions, **INCORRECT**, The T/S limits (3.9.2.b) for minimum operable SRMs is NOT met. Requires an SRM in the quadrant being loaded and the adjacent quadrant. With both B and D SRM inoperable, NO operable adjacent SRM exists and core alterations must be formally suspended,

Technical Reference(s): IO-009, Tech Spec 3.9.2.b (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: TECSPCE010, Given specific plant operating conditions and a copy of the Hope Creek Generating Station Technical Specifications, evaluate plant/system operability and determine required actions (if any) to be taken. (As available)

Question Source: Bank #

Modified Bank # 54838 (Note changes or attach parent)

New

Question History: NRC 2009

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2, 7

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	223002	A2.09
	Importance Rating		3.7

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System initiation.

Question: SRO #90

Given:

- The plant is in Refuel and preparing to transition to Cold Shutdown
- OP-IO.ZZ-001 Refueling to Cold Shutdown has just been entered.
- NO actions from OP-IO.ZZ-001 have been performed yet.
- B RHR is in Shutdown Cooling.

Then:

- The BB-PT-N078A Rx Pressure transmitter fails upscale.
- OHA C8-A5 NSSSS INBD ISLN SYS OUT OF SVCE alarms
- OHA C8-C5 NSSSS TRIP UNIT TROUBLE alarms

What are the required actions for the failed transmitter and why?

A. Enter AB.RPV-009, for the loss of Shutdown Cooling and place A RHR in Shutdown Cooling.

B. Enter AB.RPV-009, for the loss of Shutdown Cooling and restore B RHR in Shutdown Cooling.

C. Enter AB.CONT-002, due to the containment isolation and restore/reset isolated equipment.

D. Enter AR.ZZ-012, OHA Window Box C8 due to the transmitter failure and enter Tech Spec 3.3.2., Isolation Actuation Instrumentation

Proposed Answer: D

Explanation (Optional):

A: Enter AB.RPV-009 Shutdown Cooling, for the loss of Shutdown Cooling and place A RHR in Shutdown Cooling. **INCORRECT,** OP-GP.SM-001 is still active when exiting IOP-009, the procedure was performed per IOP-005 step 5.4.44 which is the prerequisite procedure for IOP-009 and RHR valves F008/9/15A-B will NOT isolate.

B: Enter AB.RPV-009 Shutdown Cooling, for the loss of Shutdown Cooling and restore B RHR in Shutdown Cooling. **INCORRECT,** OP-GP.SM-001 is still active when exiting IOP-009, the procedure was performed per IOP-005 step 5.4.44 which is the prerequisite procedure for IOP-009 and RHR valves F008/9/15A-B will NOT isolate.

C: Enter AB.CONT-002 Primary Containment, due to the containment isolation and restore/reset isolated equipment. **INCORRECT,** OP-GP.SM-001 is still active when exiting IOP-009, the procedure was performed per IOP-005 step 5.4.44 which is the prerequisite procedure for IOP-009 and RHR valves F008/9/15A-B will NOT isolate.

D: Enter AR.ZZ-0012 Overhead Alarm Response, due to the transmitter failure and enter Tech Spec 3.3.2., **CORRECT,** OP-GP.SM-001 is still active when exiting IOP-009, the procedure was performed per IOP-005 step 5.4.44 which is the prerequisite procedure for IOP-009 and RHR valves F008/9/15A-B will NOT isolate. Alarm response has direction to enter the applicable Tech Spec for the failed transmitter.

Technical Reference(s): OP-IO.ZZ-001, OP-IO.ZZ-005, OP-IO.ZZ-009 and OP.GP-SM-001, OP.AB-RPV-009 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OP-IO.ZZ-001 sect 5.2 ONLY

Learning Objective: NSSSS0E004, Given a list of NSSSS isolation signals, explain the plant conditions and/or operator actions necessary for automatic and/or manual bypass of the isolation signal. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2,5

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2012
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	259001	A2.03
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of condensate pump(s)

Question: SRO #91

Given:

- The reactor is operating at 3400 MWt
- A RFPT is C/T due to pump coupling replacement
- B and C RFPT are feeding the RPV in 3 element control
- All primary condensate pumps are running
- All secondary condensate pumps are running
- OPRMs are INOP
- Feedwater flow ≈ 89%
- Core Flow is ≈ 90%

Then

- The C Secondary Condensate pump trips
- B and C RFPT remain in AUTO
- Intermittent APRM upscale alarms are received
- Core flow is now 48%

What procedure entry and action is required for the present plant conditions?

A. IAW AB.RPV-004 Reactor Level Control, order the Reactor Operator to take manual control of feedwater and control reactor water level between level 4 and level 7.

B. IAW EOP-101 RPV Control, order the Reactor Operator to LOCK the mode in SHUTDOWN when reactor water level drops ≤ 12.5".

C. IAW RPV-001, Reactor Power, order the Reactor Operator to insert control rods to clear the upscale alarms.

D. IAW RPV-003, Recirculation System/Power Oscillations, order the Reactor Operator to LOCK the mode switch in SHUTDOWN, for entering region 1 of the power to flow map.

Proposed Answer: C

Explanation (Optional):

A: IAW AB.RPV-004 Reactor Level Control, order the Reactor Operator to take manual control of feedwater and control reactor water level between level 4 and level 7. **INCORRECT**, with feedwater flow > 75% a trip of a Secondary Condensate pump will cause an intermediate runback (45%) of the reactor recirculation pumps. While the operators may take manual control of feedwater, because the failure listed in the stem is not a failure of automatic feed water control, there is no requirement to take manual control.

B: IAW EOP-101 RPV Control, order the Reactor Operator to LOCK the mode in SHUTDOWN when reactor water level drops ≤ 12.5". **INCORRECT,** reactor water level should not reach 12.5" as an intermediate runback on reactor recirculation is initiated by the C Secondary Condensate pump tripping with feed water flow >75%. NO requirement to scram the reactor on low level.

C: IAW RPV-001, Reactor Power, when APRM Upscale alarms are received, order the Reactor Operator to insert control rods to clear the upscale alarms. **CORRECT,** with feedwater flow >75% a trip of a Secondary Condensate pump will cause an intermediate runback (45%) of the reactor recirculation pumps. If APRM upscale alarms come in, per RPV-001, the requirement is to insert control rods to clear the upscale alarms.

D: IAW RPV-003, Recirculation System/Power Oscillations, order the Reactor Operator to LOCK the mode switch in SHUTDOWN, for entering region 1 of the power to flow map. **INCORRECT,**

with feedwater flow > 75% a trip of a Secondary Condensate pump will cause an intermediate runback (45%) of the reactor recirculation pumps. Power is expected to drop to ≈60-65% and would not enter into region 1 of the power to flow map. No requirement to lock the mode switch in shutdown.

Technical Reference(s): AB.RPV-001, AB.RPV-003, AB.RPV-004 and EOP-101 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: FEED00E026, From memory, given plant conditions, summarize/identify the three (3) possible RFPT runback signals. IAW available Control Room References. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		2
K/A #	226001	2.2.42
Importance Rating		4.6

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (RHR/LPCI: CTMT Spray Mode)

Question: SRO #92

Given:

- The plant is operating at 100% power.
- OP-IS.BC-0101 RESIDUAL HEAT REMOVAL SUBSYSTEM A VALVES - INSERVICE TEST surveillance is being performed
- While stroke timing the BC-HV-F048A RHR Heat Exchanger Bypass, the valve is stroked full open and CAN NOT be reclosed.
- A local investigation and attempt to stroke the valve reveals the valve is mechanically bound in place and will NOT move.

Which one of the following Technical Specification actions is applicable as a result of this condition?

A. Declare the LPCI mode of A RHR INOP AND provided that at least one CORE SPRAY subsystem is OPERABLE, restore the inoperable LPCI subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

B. Declare ONLY the Suppression Pool Cooling mode of A RHR INOP AND restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

C. Declare ONLY the Suppression Pool Spray mode of A RHR INOP AND restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

D. Declare BOTH the Suppression Pool Cooling and Spray modes of A RHR INOP AND restore the inoperable loop of Suppression Pool Cooling to OPERABLE status within 72 hours AND Suppression Pool Spray with 7 days OR be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Proposed Answer: D

Explanation (Optional):

A: Declare the LPCI mode of A RHR INOP AND provided that at least one CORE SPRAY subsystem is OPERABLE, restore the inoperable LPCI subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. **INCORRECT**, With the heat exchanger valve failed open LPCI is not INOP. Valve gets an open signal on an initiation signal for LPCI.

B: Declare ONLY the Suppression Pool Cooling mode of A RHR INOP AND restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. **INCORRECT**, With the heat exchanger valve failed open Suppression Pool Cooling AND Spray are both INOP. See T/S 3.6.2.2 and 3.6.2.3. For operability, flow through the heat exchanger (bypass closed) is required.

C: Declare ONLY the Suppression Pool Spray mode of A RHR INOP AND restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. **INCORRECT**, With the heat exchanger valve failed open Suppression Pool Cooling AND Spray are both INOP. See T/S 3.6.2.2 and 3.6.2.3. For operability, flow through the heat exchanger (bypass closed) is required.

D: Declare BOTH the Suppression Pool Cooling and Spray modes of A RHR INOP AND restore the inoperable loop of Suppression Pool Cooling to OPERABLE status within 72 hours AND Suppression Pool Spray with 7 days OR be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. **CORRECT**, With the heat exchanger valve failed open Suppression Pool Cooling AND Spray are both INOP. See T/S 3.6.2.2 and 3.6.2.3. For operability, flow through the heat exchanger (bypass closed) is required.

Technical Reference(s): Tech Specs 3.5.1, 3.6.2.2 and 3.6.2.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Tech Spec sections 3.5.1-3.5.2 only, 3.6.2.2 – 3.6.4.2 only, see Q#69 & #93

Learning Objective: RHRSYSE013, Given Plant Conditions and access to Technical Specifications: Select those sections which are applicable to the Residual Heat Removal System IAW HCGS Technical Specifications. Evaluate Residual Heat Removal System operability and determine required actions based upon system operability IAW HCGS Technical Specifications. (SRO Only) Explain the bases for those Technical Specifications associated with the Residual Heat Removal System IAW HCGS Technical Specifications. (SRO Only) (As available)

Question Source: Bank #

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

Question History:

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	223001	2.2.37
	Importance Rating		4.6

Equipment Control: Ability to determine operability and / or availability of safety related equipment. (Primary Containment System and Auxiliaries)

Question: SRO #93

While investigating the source of excessive N^2 makeup to the Drywell,

- Operators discover that The Drywell Purge Isolation Valve GS-HV-4952 was wired wrong during the last refuel outage.
- The GS-HV-4952 has been open since startup 35 days ago.

In addition,

- Maintenance reports that the Drywell Purge Exhaust valve GS-HV-4950 limit switches were not set correctly and the valve is not full closed.

Maintenance has closed both valves and restored them to operable status in 3 hours.

What impact will this have on plant operation?

A. The Drywell can NOT be de-inerted until Op Cond 4.

B. The Drywell can ONLY be de-inerted in Op Cond 2 or 3.

C. The Isolation Valve GS-HV-4952 can ONLY be used for purging.

D. The Exhaust Valve GS-HV-4950 can now be used for pressure control.

Proposed Answer: A

Explanation (Optional):

A: The Drywell cannot be de-inerted until Op Cond 4, **CORRECT**, T/S 3.6.1.8 limit open time on the DW purge valves to 500 hours per 365 days. The time allowed has expired 35 days x 24

hours/day = 840 hours. Therefore, valve opening is not allowed in Op Cond 1, 2 or 3 unless for the purpose of pressure control thru the 2 inch bypass valve.

B: The Drywell can only be de-inerted in Op Cond 2 or 3, **INCORRECT,** T/S 3.6.1.8 limit open time on the DW purge valves to 500 hours per 365 days. The time allowed has expired 35 days x 24 hours/day = 840 hours. Therefore, valve opening is not allowed in Op Cond 1, 2,or 3 unless for the purpose of pressure control thru the 2 inch bypass valve.

C: The Isolation Valve GS-HV-4952 can only be used for purging, **INCORRECT,** T/S 3.6.1.8 limit open time on the DW purge valves to 500 hours per 365 days. The time allowed has expired 35 days x 24 hours/day = 840 hours. Therefore, valve opening is not allowed in Op Cond 1, 2,or 3 unless for the purpose of pressure control thru the 2 inch bypass valve, purging in NOT allowed in the present condition.

D: The Bypass Valve GS-HV-4950 can now be used for pressure control, **INCORRECT,** T/S 3.6.1.8 limit open time on the DW purge valves to 500 hours per 365 days. The time allowed has expired 35 days x 24 hours/day = 840 hours. Therefore, valve opening is not allowed in Op Cond 1, 2,or 3 unless for the purpose of pressure control thru the 2 inch bypass valve, purging in NOT allowed in the present condition. Pressure control is permitted " Valves open for pressure control are not subject to the 500 hours per 365 day limit, provided the 2 inch bypass lines are being utilized" the GS-HV-4950 is NOT the 2 inch bypass line.

Technical Reference(s): M-57-1 sheet 1, Tech Spec 3.6.1.8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: INERT0E015, Given a scenario of applicable operating conditions and access to Technical Specifications:
a. Select those sections which are applicable to the Containment Inerting and Purge System. b. Evaluate Containment Inerting and Purge System operability and determine required actions based upon system inoperability. (SRO Only)
c. Explain the bases for those Technical Specification sections associated with the Containment Inerting a Purge System. (As available)

Question Source: Bank # 56386

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.43	
	Importance Rating		4.3

Ability to use procedures to determine the effects on reactivity of plant changes, such as RCS temperature, secondary plant, fuel depletion, etc.

Question: SRO #94

Given:

- The turbine had been operating for 183 days when it was inadvertently tripped by an I/C surveillance.
- The reactor has been scrammed for 5 hours.
- All other plant equipment responded as required.
- No start-up restraints exist and a quick turn around has been approved IAW OP-AA-108-114 Post Transient Review.
- You are the Reactivity Management SRO (RMSRO) for the restart.
- The Reactor Operator, who performed the previous cold startup, reports to you that control rods 26-27, 26-35, 34-27 and 34-35 are less reactive than expected and has stopped the startup awaiting your decision.

As the Reactivity Management SRO (RMSRO) you should direct the startup be...

A. continued. The center of the core is less reactive due to the buildup of xenon.

B. suspended. The center of the core is more reactive due to the depletion of xenon.

C. continued. The center of the core is less reactive due to the depletion of xenon.

D. suspended. The center of the core is more reactive due to starting up at a higher temperature.

Proposed Answer: A

Explanation (Optional):

A: continued. The center of the core is less reactive due to the buildup of xenon. **CORRECT**, Since the production of xenon-135 and its precursor, iodine-135, is dependent on the thermal neutron flux, the production at any local point in the reactor core depends on the thermal neutron flux at

that core location. Since the neutron flux is not uniform over the core volume, the production and removal rates, and the isotopic concentrations, xenon-135 and iodine-135 are also not uniform over the core volume. Thus, xenon-135 has a local reactivity effect, which tends to change the thermal neutron flux shape. Where local xenon concentration is relatively high, the flux is depressed, and where local xenon concentration is relatively low, flux tends to be greater. These effects are due to the poisoning effect of Xe-135 on the neutron life cycle. As normal flux distribution tends to be highest at the center and bottom or the core, Xe concentration will also be highest there following a shutdown.

B: suspended. The center of the core is more reactive due to the depletion of xenon. **INCORRECT**, Since the production of xenon-135 and its precursor, iodine-135, is dependent on the thermal neutron flux, the production at any local point in the reactor core depends on the thermal neutron flux at that core location. Since the neutron flux is not uniform over the core volume, the production and removal rates, and the isotopic concentrations, xenon-135 and iodine-135 are also not uniform over the core volume. Thus, xenon-135 has a local reactivity effect, which tends to change the thermal neutron flux shape. Where local xenon concentration is relatively high, the flux is depressed, and where local xenon concentration is relatively low, flux tends to be greater. These effects are due to the poisoning effect of Xe-135 on the neutron life cycle. As normal flux distribution tends to be highest at the center and bottom or the core, Xe concentration will also be highest there following a shutdown.

C: continued. The center of the core is less reactive due to depletion of xenon. **INCORRECT,** Since the production of xenon-135 and its precursor, iodine-135, is dependent on the thermal neutron flux, the production at any local point in the reactor core depends on the thermal neutron flux at that core location. Since the neutron flux is not uniform over the core volume, the production and removal rates, and the isotopic concentrations, xenon-135 and iodine-135 are also not uniform over the core volume. Thus, xenon-135 has a local reactivity effect, which tends to change the thermal neutron flux shape. Where local xenon concentration is relatively high, the flux is depressed, and where local xenon concentration is relatively low, flux tends to be greater. These effects are due to the poisoning effect of Xe-135 on the neutron life cycle. As normal flux distribution tends to be highest at the center and bottom or the core, Xe concentration will also be highest there following a shutdown. Moderator temperature s an overall effect on the core not localized such as xenon.

D: suspended. The center of the core is more reactive due to starting up at a higher temperature. **INCORRECT,** Since the production of xenon-135 and its precursor, iodine-135, is dependent on the thermal neutron flux, the production at any local point in the reactor core depends on the thermal neutron flux at that core location. Since the neutron flux is not uniform over the core volume, the production and removal rates, and the isotopic concentrations, xenon-135 and iodine-135 are also not uniform over the core volume. Thus, xenon-135 has a local reactivity effect, which tends to change the thermal neutron flux shape. Where local xenon concentration is relatively high, the flux is depressed, and where local xenon concentration is relatively low, flux tends to be greater. These effects are due to the poisoning effect of Xe-135 on the neutron life cycle. As normal flux distribution tends to be highest at the center and bottom or the core, Xe concentration will also be highest there following a shutdown. Moderator temperature s an overall effect on the core not localized such as xenon.

Technical Reference(s): OP-AB-300-1003 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NOH01OPSSTDE001, Following classroom/simulator training, and provided (As available)

access to the Operations Standards, the student will determine actions needed to meet standards in the following areas: Reactivity Management, Industrial Safety Practices, Radiation Worker Practices Self-Assessment/ Corrective Action, Housekeeping/Cleanliness Control/Fme Briefs, Accessing Equipment Radiological Contamination Of Clean Systems, Post Accident Use Of Valve And Breaker Overrides, Electromagnetic Interference (Emi) & Radio Frequency Interference (Rfi), Operator Appearance

Question Source: Bank #

Modified Bank # 111245 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 6

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #	2.2.17	
Importance Rating		3.8

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, coordination with the transmission system operator.

Question: SRO #95

The plant is operating at 100% power.

When:

The Electric System Operator (ESO) contacts the Operations Shift Manager (SM) and has informed him that T & D (Transmission and Distribution) work has found a substantial SF_6 gas leak on 500KV breaker, BS 2-6. The 500KV switchyard is in normal alignment. A notification from the Maintenance Department has been generated to address the leaking breaker.

In accordance with LS-AA-120, Issue Identification and Screening, which of the following are responsibilities of the Operations Shift Management Reviewer concerning the notification written by the Maintenance Department?

A. Determine if there are any operability concerns AND ensure the area is quarantined.

B. Quarantine the area AND ensure Licensing has initiated required NRC reports.

C. Determine reportability requirements AND perform a Quick Human Performance Investigation.

D. Initiate required field work AND ensure all responsible personnel are documented by name in the notification.

Proposed Answer: A

Explanation (Optional):

A: Determine if there are any operability concerns and ensure areas, materials and procedures are quarantined. **CORRECT**, LS-AA-120, sect 3.10, "Operations Shift Management Reviewer:

Ensure immediate actions are taken including: Quarantine areas, equipment, or records to preserve physical evidence, Initiate a Prompt Investigation per OP-AA-106-101-1001, Event Response Guidelines, Initiate required field work, Determine operability and reportability."

B: Quarantine areas, materials and procedures and ensure licensing has initiated required NRC reports. **INCORRECT**, Operations is required to initiate reportability.

C: Determine reportability requirements and Quick Human Performance Investigation has been initiated. **INCORRECT,** Operations is required ensure a prompt investigation has been initiated, NOT perform a QHPI.

D: Initiate required field work and ensure all responsible personnel are documented by name in the notification. **INCORRECT,** Supervisor is required to ensure personnel responsibilities are documented but names are not used. Per LS-AA-120 sect 3.9, "Document issues with individual personnel performance with statements such as "performance management should be implemented by the appropriate manager/supervisor". Specific information, such as individual names, disciplinary actions take, etc., should not be included in notifications."

Technical Reference(s): LS-AA-120 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NOH04ADA120E003, From memory describe the actions for Operations Shift Management Review of a issue to determine the following: a. If operability is required, b. Determine Operability c. Reportability requirements IAW LS-AA-120. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.3.14	
	Importance Rating		3.8

Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question: SRO #96

Select the containment vent path that, if used to control primary containment pressure, would result in an unscrubbed, unmonitored and untreated radioactive release to the environment.

A. IAW EOP-101, RPV Control, vent the containment via the drywell supply and ILRT piping.

B. IAW EOP-101, RPV Control, vent the containment via the suppression chamber supply and ILRT piping.

C. IAW EOP-102, Primary Containment Control, vent the containment via the drywell supply and ILRT piping.

D. IAW EOP-102, Primary Containment Control, vent the containment via the suppression chamber supply and ILRT piping.

Proposed Answer: C

Explanation (Optional):

A: IAW EOP-101, RPV Control, vent the containment via the drywell supply and ILRT piping. **INCORRECT**, EOP-101 does have an entry condition for high drywell pressure, however there is no guidance for containment venting contained in the flowchart. IAW HC.OP-EO.ZZ-0318 step 5.1.3, Caution: "Section 5.1.3 provides a hard-piped vent path that does not communicate with blowout panels or back draft dampers. However, the use of this vent path will result in an unmonitored, unscrubbed and untreated radioactive release to the environment and requires local operations to be performed under potentially adverse radiological and/or environmental conditions."

B: IAW EOP-101, RPV Control, vent the containment via the suppression chamber supply and ILRT piping. **INCORRECT**, EOP-101 does have an entry condition for high drywell pressure, however there is no guidance for containment venting contained in the flowchart. IAW HC.OP-EO.ZZ-0318 step 5.1.2, "Section 5.1.2 provides a hard-piped vent path that does not communicate with blowout panels or back draft dampers and is scrubbed by the Suppression Pool. However, the use of this vent path will result in an unmonitored release to the environment and requires local operations to be performed under potentially adverse radiological and/or environmental conditions."

C: IAW EOP-102, Primary Containment Control, vent the containment via the drywell supply and ILRT piping. **CORRECT**, Per EOP-102 step(s) DW/P-17 and DW/P-20, when drywell pressure can not be maintained below 65 psig vent containment per OP-EO.ZZ-0318, step 5.1.3, Caution: "Section 5.1.3 provides a hard-piped vent path that does not communicate with blowout panels or back draft dampers. However, the use of this vent path will result in an unmonitored, unscrubbed and untreated radioactive release to the environment and requires local operations to be performed under potentially adverse radiological and/or environmental conditions."

D: IAW EOP-102, Primary Containment Control, vent the containment via the suppression chamber supply and ILRT piping. **INCORRECT**, Per EOP-102 step(s) DW/P-17 and DW/P-20, when drywell pressure can not be maintained below 65 psig vent containment per OP-EO.ZZ-0318, step 5.1.2, "Section 5.1.2 provides a hard-piped vent path that does not communicate with blowout panels or back draft dampers and is scrubbed by the Suppression Pool. However, the use of this vent path will result in an unmonitored release to the environment and requires local operations to be performed under potentially adverse radiological and/or environmental conditions."

Technical Reference(s): EOP-102, EOP-318 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP300E004, From memory, describe any/all flow paths established by the performance of each of the 300 series Emergency Operating procedures. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4, 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		
K/A #	2.4.34	
Importance Rating		4.1

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Question: SRO #97

Due to a fire in the Control Room console (10C651C), you ordered the Control Room evacuated immediately.

The reactor was scrammed remotely from the RPS distribution panels.

How will you direct the Reactor Operator to verify the scram?

A. IAW OP-SO.SA-001, Redundant Reactivity Control, verify ARI valves are open on the 10C601 and 10C602 Panels.

B. IAW OP-SO.SB-001, Reactor Protection, verify with SPDS/CRIDS display terminal in the TSC.

C. IAW OP-SO.BF-001, CRD Hydraulic, verify RPS Backup Scram Air Solenoids are de-energized.

D. IAW OP-SO.BF-002, Individual CRD Operation, verify HCU accumulator pressure is 950 - 1000 psig at each HCU.

Proposed Answer: B

Explanation (Optional):

A: IAW OP-SO.SA-001, Redundant Reactivity Control, verify ARI valves are open on the 10C601 and 10C602 Panels. **INCORRECT,** The RO would not normally go to the 10C601/602. The status of the ARI valves is displayed on the 10C601 and 10C602 panels, however actual rod positions are not available here

B: IAW OP-SO.SB-001, Reactor Protection, verify with SPDS/CRIDS display terminal in the TSC. **CORRECT,** 5.2.2 ENSURE that all 185 Control Rods have fully inserted by one (OR more) of the following: SELECT CONTROL ROD POSITIONS on CRIDS AND OBSERVE Rod positions. SPDS ALL RODS INSERTED reads "YES". The TSC is physically "next" to the RSP and is used for indications displayed on either SPDS/CRIDS

C: IAW OP-SO.BF-001, CRD Hydraulic, verify RPS Backup Scram Air Solenoids are de-energized. **INCORRECT,** Back-up scram valves are energized to function and only an indication that the scram air header was depressurized and not an indication of control rod position.

D: IAW OP-SO.BF-002, Individual CRD Operation, verify HCU accumulator pressure is 950 - 1000 psig at each HCU. **INCORRECT,** while lowered HCU accumulator pressure would be an indication of the control rod being scrammed does not indicate control rod final position.

Technical Reference(s): OP-IO.ZZ-008, OP-SO.SA-001, OP-SO.SB-001, OP-SO.BF-001 OP-SO.BF-002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: IOP008E002, Determine if all Prerequisites have been met prior to implementation of the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: NRC 2003 (modified)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.35	
	Importance Rating		3.9

Knowledge of the fuel-handling responsibilities of SRO's.

Question: SRO #98

Plant conditions:

- A Spiral Fuel offload is in progress per HC.RE-FR.ZZ-0001.
- 12 Control Rod blades and a drive mechanism have been removed.
- While a fuel bundle is being moved, a problem with the refuel bridge occurs and the fuel bundle must immediately be lowered back into the core.

Then:

The Refueling SRO reports that the fuel bundle was mistakenly placed into a fuel cell with a removed control rod blade.

The Refuel Bridge has been relocated over the Fuel Pool and declared INOPERABLE

What actions are required by the Refueling SRO?

A. Stop fuel handling in the fuel pool and reinstall a Control Rod Blade in the now fueled cell.

B. Stop fuel handling in the fuel pool and reinstall a Control Rod Mechanism in the now fueled cell.

C. Stop Control Rod Blade removal from the reactor vessel. All fuel handling must stop. Once the refuel bridge is operable the misplaced fuel bundle must be removed and placed into its correct storage location.

D. Stop Control Rod Blade removal from the reactor vessel. Fuel handling in the fuel pool may continue. Once the refuel bridge is operable the misplaced fuel bundle must be removed and placed into its correct storage location.

Proposed Answer: C

Explanation (Optional):

A: Stop fuel handling in the fuel pool and reinstall a Control Rod Blade in the now fueled cell. **INCORRECT**, It is physically impossible to reinstall a control rod in a fuel cell with a fuel bundle in place. The fuel bundle must be removed prior to reinstalling the control rod. With the refuel bridge INOP, no fuel can be moved.

B: Stop fuel handling in the fuel pool and reinstall a Control Rod Mechanism in the now fueled cell. **INCORRECT,** The control rod mechanism can be reinstalled, however for the Tech Spec 3.9.10.2 to be satisfied, the control rod associated with the mechanism must be relatched and with a fuel bundle into the cell a control rod can not be reinstalled. With the refuel bridge INOP, no fuel can be moved.

C: Stop Control Rod Blade removal from the reactor vessel. All fuel loading must stop. The misplaced fuel bundle must be removed and placed into its correct storage location immediately upon the refuel bridge being operable. **CORRECT,** With the refuel bridge INOP, no fuel can be moved. Per SO.KE-001, step 2.2.3 and 2.3.3 - FSAR 9.1.4.2.12.3 As soon as the refuel bridge has been repaired and declared operable, the misplaced fuel bundle must be removed and placed into its proper storage location.

D: Stop Control Rod Blade removal from the reactor vessel. Fuel handling in the fuel pool may continue. The misplaced fuel bundle must be removed and placed into its correct storage location immediately upon the refuel bridge being operable. **INCORRECT,** With the refuel bridge INOP, no fuel can be moved. Per SO.KE-001, step 2.2.3 and 2.3.3 - FSAR 9.1.4.2.12.3 As soon as the refuel bridge has been repaired and declared operable, the misplaced fuel bundle must be removed and placed into its proper storage location.

Technical Reference(s): Tech Spec 3.9.10.1, 3.9.10.2 (Attach if not previously provided)

OP-SO.KE-001, FSAR 9.1.4.2.12.3

Proposed References to be provided to applicants during examination: Tech Spec 3.9.10.1 and 3.9.10.2 only

Learning Objective: IOP009E006, Analyze plant conditions and parameters to determine if plant operation is in accordance with the REFUELING OPERATIONS Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2, 6, 7

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.38	
	Importance Rating		4.4

Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Question: SRO #99

Given:

- A Site Area Emergency was just declared 20 minutes ago due to a leak.
- The leak is discharging to the environment and can NOT be isolated from the Control Room.
- The TSC and EOF are being staffed and are NOT activated at this time.
- The Radiological Assessment Coordinator (RAC) is on station in the TSC.
- The OSC has a plan to manually isolate the leak to terminate the release.
- An Emergency Dose Authorization is required to isolate the line.
- The Emergency Duty Officer (EDO) is NOT in the TSC, and can NOT be reached.

Who can authorize the Emergency Exposure in the absence of the EDO and what is the Planned Emergency Exposure Limit (PEEL)?

A. The RAC can authorize the dose extension exposure up to 25 REM.

B. The RAC can authorize the dose extension exposure up to 75 REM.

C. The Shift Manager can authorize the dose extension exposure up to 25 REM.

D. The Shift Manager can authorize the dose extension exposure up to 75 REM.

Proposed Answer: C

Explanation (Optional):

A: The RAC can authorize the dose extension exposure up to 25 REM. **INCORRECT**, EP-EP.ZZ-0304 states that the shift manager has the responsibility to authorize emergency exposes until the EDO assumes his or her responsibilities. The planned emergency exposure limit is 25 REM for accident mitigation and isolating the line in this case is an accident mitigating action.

B: The RAC can authorize the dose extension exposure up to 75 REM. **INCORRECT**, EP-EP.ZZ-0304 states that the shift manager has the responsibility to authorize emergency exposes until the EDO assumes his or her responsibilities. The planned emergency exposure limit is 25 REM for accident mitigation and isolating the line in this case is an accident mitigating action.

C: The Shift Manager can authorize the dose extension exposure up to 25 REM. **CORRECT**, EP-EP.ZZ-0304 states that the shift manager has the responsibility to authorize emergency exposures until the EDO assumes his or her responsibilities. The planned emergency exposure limit is 25 REM for accident mitigation and isolating the line in this case is an accident mitigating action.

D: The Shift Manager can authorize the dose extension exposure up to 75 REM. **INCORRECT**, EP-EP.ZZ-0304 states that the shift manager has the responsibility to authorize emergency exposures until the EDO assumes his or her responsibilities. The planned emergency exposure limit is 25 REM for accident mitigation and isolating the line in this case is an accident mitigating action.

Technical Reference(s): EP-EP.ZZ-0304 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: SOB200, ECG/E-Plan/Fire & Medical Questions (As available)

Question Source: Bank #

Modified Bank # 84355 (Note changes or attach parent)

New

Question History: NRC 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4, 5

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2012

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.3.15	
	Importance Rating		3.1

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question: SRO #100

Given:

- A plant startup is in progress.
- The 'A' RPS MG Set Voltage Regulator fails causing generator output voltage to drop to approximately 100VAC.
- All other plant equipment functioned as expected.

What's the status of Main Steam Line (MSL) Radiation Monitors?

A. MSL Radiation Monitors RE-N006A and RE-N006C are INOPERABLE.

B. MSL Radiation Monitors RE-N006A and RE-N006B are INOPERABLE.

C. MSL Radiation Monitors RE-N006A and RE-N006C are OPERABLE but degraded.

D. MSL Radiation Monitors RE-N006A and RE-N006B are OPERABLE but degraded.

Proposed Answer: A

Explanation (Optional):

A: MSL Radiation Monitors RE-N006A and RE-N006C are INOPERABLE. **CORRECT**, Main steam line radiation monitor drawers are power from RPS, with A and C rad monitors powered from the A side of RPS and B and D powered from B side of RPS. Loss of power to the radiation monitor causes an INOP trip of the drawer, this is due to the EPA breaker(s) having an under-voltage trip set at < 108 volts AC per T/S 3.8.4.4

B: MSL Radiation Monitors RE-N006A and RE-N006B are INOPERABLE. **INCORRECT**, Main steam line radiation monitor drawers are power from RPS, with A and C rad monitors powered

from the A side of RPS and B and D powered from B side of RPS. Loss of power to the radiation monitor causes an INOP trip of the drawer, this is due to the EPA breaker(s) having an under-voltage trip set at < 108 volts AC per T/S 3.8.4.4

C: MSL Radiation Monitors RE-N006A and RE-N006C are OPERABLE but degraded.
INCORRECT, Main steam line radiation monitor drawers are power from RPS, with A and C rad monitors powered from the A side of RPS and B and D powered from B side of RPS. Loss of power to the radiation monitor causes an INOP trip of the drawer, this is due to the EPA breaker(s) having an under-voltage trip set at < 108 volts AC per T/S 3.8.4.4

D: MSL Radiation Monitors RE-N006A and RE-N006B are OPERABLE but degraded.
INCORRECT, Main steam line radiation monitor drawers are power from RPS, with A and C rad monitors powered from the A side of RPS and B and D powered from B side of RPS. Loss of power to the radiation monitor causes an INOP trip of the drawer, this is due to the EPA breaker(s) having an under-voltage trip set at < 108 volts AC per T/S 3.8.4.4

Technical Reference(s): Tech Spec 3.8.4.4 , 3.3.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: RMSYS0E002, Regarding the main steam line Radiation Monitoring System: From memory, explain the setpoints / conditions associated with a high-high radiation or inoperative trip IAW the Radiation Monitoring System Lesson Plan. Given normal Control Room references, determine the automatic plant actuations / trips which occur as a result of a high-high radiation or inoperative trip. From memory, evaluate the effect of a loss of RPS power IAW the Radiation Monitoring System Lesson Plan. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4, 5

Comments: