Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 K1.01	
	Importance Rating	4.0	

K/A Statement: Knowledge of the physical connections and/or cause- effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: RHR/LPCI: Plant-Specific

Question: RO #1

Given:

- A small break LOCA exists in a RWCU pipe chase.
- The leak has not been isolated.
- There are no other leaks from primary containment.
- HPCI is injecting.
- RPV level is 30 inches above the top of active fuel and continues to lower.
- Drywell pressure is at .75 psig and steady.

Which of the following describes when ADS initiates to allow LPCI injection into the RPV?

A. 105 seconds after the first two LPCI (RHR) pumps start

B. 105 seconds after ADS logic determines that there is a leak

C. 405 seconds after ADS logic determines that RPV level has been less than the ADS setpoint

D. 105 seconds after RPV level 1 (-129 inches) is confirmed with a RPV level 3 (12.5 inches) signal in

Proposed Answer: **C**

Explanation (Optional): IAW HC.OP-SO.SN-0001(Q) section 3.3.1 (see attached)

Plausibility Justification:

A: **Incorrect-**. The 105 second timer is only functional if RPV level is low coincident with **high drywell pressure,** A leak in the RWCU pipe chase is **outside of containment and does not affect drywell pressure the 5 minute timer has to time out for leaks outside the drywell.**

B: **Incorrect-** The 105 second timer is only functional if RPV level is low coincident with **high drywell pressure,** A leak in the RWCU pipe chase is **outside of containment and does not affect drywell pressure the 5 minute timer has to time out for leaks outside the drywell.**

C: **Correct-** The ADS **drywell bypass timer will timeout at 300 seconds** and initiated depressurization. **TAF is -161"** hence, the initial condition is -131" **plus 105 second timer.** The LPCI (RHR) pumps will be running from the RPV level 1 (-129 inches) start signal.

D: **Incorrect-**. The 105 second timer is only functional if RPV level is low coincident with **high drywell pressure,** A leak in the RWCU pipe chase is **outside of containment and does not affect drywell pressure the 5 minute timer has to time out for leaks outside the drywell.**

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

NUCLEAR PRESSURE RELIEF AND AUTOMATIC DEPRESSURIZATION SYSTEM OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 30912

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K1.06	
	Importance Rating	3.4	

K/A Statement: Knowledge of the physical connections and/or cause- effect relationships between RELIEF/SAFETY VALVES and the following: Drywell instrument air/ drywell pneumatics: Plant-Specific

Question: RO #2

Given:

- The plant is operating at 100% power.

Then:

- A small leak develops on the H-T210 pneumatic accumulator tank for the PSV-F013H SRV.

What effect will this have on plant operations?

A. Drywell pressure will rise steadily due to the in leakage. Containment venting will be required to maintain drywell pressure in the normal band.

B. The frequency of nitrogen makeup to the drywell will rise due to drywell oxygen concentrations rising from the leak.

C. The frequency of nitrogen makeup to the drywell will rise due to lowering drywell pressure from the accumulator leakage.

D. The Auto-Lead PCIG compressor will cycle more frequently. There will be NO significant net change in drywell pressure or oxygen concentration.

Proposed Answer: **D**

Explanation (Optional): **SRV Accumulator** ensures that the SRV's can be opened and held open following a failure of the pneumatic supply, **(PCIG)**, to the valve. The PCIG system takes suction on the drywell atmosphere, which **is primarily nitrogen** when inerted; compresses that gas and returns it to the drywell for use **by pneumatic valves (SRVs).** Normal operation is one compressor **in AUTO LEAD** mode cycling on and off between 94-106 psig and the other compressor in AUTO mode cycling on and off between 85-106 psig.

Plausibility Justification:

A: **Incorrect-** The nitrogen is **in a closed loop.** The N2 leaked into the drywell would be drawn back into the PCIG system when the compressor cycles. There would be no net change in drywell atmosphere.

B: **Incorrect-** The SRV accumulators are charged N2 from PCIG. There would be no oxygen introduced into the drywell as a result of the leak. The student has to understand the pneumatic supply to the SRV accumulators.

C: **Incorrect-** The nitrogen is **in a closed loop.** The N2 leaked into the drywell would be drawn back into the PCIG system when the compressor cycles. There would be no net change in drywell atmosphere.

D: **Correct-** PCIG compressors normally take suction on the drywell atmosphere. **PCIG supplies the SRV accumulators.** A leak on an accumulator would result is PCIG receiver pressure lowering more quickly, **which would result in more frequent PCIG compressor runs.** The nitrogen; however, is in a closed loop. Since the leaked N2 would be drawn back into the PCIG system when the compressor cycles, there would be no net change in the drywell atmosphere.

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

CONTAINMENT ATMOSPHERE
CONTROL SYSTEM OPERATION
HC.OP-SO.KL-0001(Q)
PRIMARY CONTAINMENT
INSTRUMENT GAS SYSTEM
OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a system that is physically connected to or required for support of the Main Steam System, summarize the purpose of that interrelationship

Question Source: Bank # 62165

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	203000 K2.01	
Importance Rating	3.5	

K/A Statement: Knowledge of electrical power supplies to the following: Pumps- RHR/LPCI: Injection Mode

Question: RO #3

Given:

- The plant was operating at 100% power in a normal electrical line-up.

When:

- A high drywell pressure condition of 1.68 psig occurs.

After 30 seconds, what will be the power supply to the 'C' RHR Pump running in the LPCI mode of operation?

A. Station Service Transformer 1BX501 via 10A403 Switchgear

B. 1CG400 diesel generator via 10A403 Switchgear

C. 1CG400 diesel generator via 10B430 USS

D. Station Service Transformer 1AX501 via 10A403 Switchgear

Proposed Answer: **D**

Explanation (Optional):

4.16KV buses 10A401 (Channel A) and **10A403 (Channel C)** are normally powered from section 7 of the 13.8KV ring bus **via station service transformer 1AX501.** This source (1AX501) is also the alternate power supply for 10A402 (Channel B) and 10A404 (Channel D).

4.16KV buses 10A402 and 10A404 are normally powered from section 2 of the 13.8KV ring bus via station service transformer 1BX501. This source (1BX501) is also the alternate power supply for 10A401 and 10A403.

If both the normal and alternate sources to the 4.16KV Class 1E vital **buses are lost (LOP), the EDGs will start** to restore power to their respective buses.

Each 4.16KV Class 1E vital bus supplies two 480VAC USSs via stepdown transformers.

4.16KV Class Bus 480VAC USS

10A401 10B410 and 10B450

10A402 10B420 and 10B460

10A403 (4.16KV) 10B430 and 10B470 **(480 VAC)**

10A404 10B440 and 10B480

See attached Table 1 of RHR power supplies.

Plausibility Justification:

A: **Incorrect-**. 1BX501 is an alternate power supply to the 10A403 switchgear. Since the plant is in a normal electrical line up and there is no LOP, **the 'C' RHR pump is powered from the 10A403 4.16KV Switchgear which is normally powered by the 1AX501 transformer.**

B: **Incorrect-** Due to the LOCA signal from high drywell pressure the 1CG400 ('C' EDG) will be running but not loaded. **Since there is no LOP signal, the 'C" RHR pump will be powered from the 10A403 4.16KV Switchgear which is normally powered by the 1AX501 transformer.**

C: **Incorrect-** Since there is no LOP signal, the pump will be powered from normal AC Distribution lineup. The normal power supply for the 'C' RHR pump is the **4.16 KV 10A403** Switchgear. The 10B430 Unit Substation is a 'C' Channel 1E power supply for the **480 VAC distribution** not the 4.16 KV

D: **Correct-**. Since there is no LOP signal, the pump will be powered from normal AC Distribution lineup. The 'C' RHR pump is powered from the **10A403 4.16KV Switchgear which is normally powered by the 1AX501 transformer.**

Technical Reference(s):	HC.OP-SO.BC-0001(Q) Table 1 RESIDUAL HEAT REMOVAL SYSTEM OPERATION	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Concerning the 1E AC distribution switchgear:
Given a list of electrical loads (motor/unit substations); choose which are powered from the 1E 4.16KV switchgear(s).

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

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K2.01	
	Importance Rating	2.5	

K/A Statement: Knowledge of electrical power supplies to the following: IRM channels/detectors

Question: RO #4

Given:

- The plant is conducting a startup IAW HC.OP-IO.ZZ-0003, "Startup from Cold Shutdown to Rated Power".
- The Reactor Mode Switch is in STARTUP/HOT STANDBY.

When:

- The IRM System "A" UPSCALE/INOPERATIVE (C3-C2) alarm is received.
- RPS Trip System "A" is in (half-scram).

Which one of the following distribution panels' loss of power would be the cause of the current plant status?

A. 1AD307, +24 VDC Power Distribution

B. 1AD417, 1E 125 VDC Power Distribution

C. 1AD318, Non-1E 125 VDC Power Distribution

D. 1AJ483, Non-1E 120 VAC Power Distribution

Proposed Answer: **A**

Explanation (Optional): The IRM voltage pre-regulator receives **+ 24 VDC** from plant **DC distribution (1AD307/1BD307)** and provides the required regulated output (+20 VDC) to the IRM channel voltage regulator. The voltage regulator receives the output (+20 VDC) from the pre-regulator and generates the required operating voltages (+ 15 VDC) for input to the high voltage power supply and the IRM electronic circuitry. Loss of either the +15VDC or -15VDC power supply **will cause an INOP trip of the affected channels and a resultant half scram.**

Plausibility Justification:

A: **Correct-**. With the loss of power from the **1AD307 +24 VDC distribution panel** the "A" & "C" channel IRMs would lose the electronic circuitry power and would cause a low voltage on the High Voltage Power Supply to the IRM drawers and bring in an **INOP alarm** which then would cause an RPS trip on the "A" channel (half-scram).

B: **Incorrect-** IRMs are not powered by a 1E VDC source. The 1E 125 VDC provides control power to 1E loads. The student has to decipher between a 1E source versus a Non-1E source and also the proper DC distribution 125 VDC versus 24 VDC. The IRMs are powered from 24VDC 1AD307 ("A" & "C" channel IRMs) and 1BD307 ("B" & "D" channel IRMs).

C: **Incorrect-** IRMs are not powered by a 125 VDC source. The Non-1E 125 VDC provides control power to various Non-1E loads. The student has to decipher between a 125 VDC versus 24 VDC distribution. The IRMs are powered from 24 VDC 1AD307 ("A" & "C" channel IRMs) and 1BD307 ("B" & "D" channel IRMs).

D: **Incorrect-**. The APRMs, LPRMs (APRM slaves) and RBMs are powered from two 120VAC UPS buses, **1AJ483** (thru EPA breakers 1AN413 and 1BN413) and 1BJ483 (thru EPA breakers 1AN414 and 1BN414). The student has to recognize that the PRNMs are AC powered and the IRMs are DC powered even though they are part of the Nuclear Instrumentation System.

Technical Reference(s): E-0010 (Attach if not previously provided)

Single Line Meter and Relay Drawing

24VDC System

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a loss of electrical power to the IRM Drives explain what response would be expected, IAW available Control Room Procedures.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

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

K/A Statement: 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:

- The plant is operating at 100% power.
- All Circulating Water pumps are running with their discharge valves full open.

Then, power is lost to the Circulating Water 4.16 KV Bus, 10A502.

- It has been sixty (60) seconds since the loss of power to the bus.
- NO operator actions have been performed on the circulating water system.

What is the present Circulating Water system configuration?

A. AP501 and CP501 pumps are running
BP501 and DP501 pumps are tripped
HV-2152A and HV-2152C Circ Water Pump Discharge Valves remain as-is with NO position indication
HV-2152B and HV-2152D Circ Water Pump Discharge Valves are in the CLOSED position

B. AP501 and CP501 pumps are running
BP501 and DP501 pumps are tripped
Circ Water Pump Discharge Valves HV-2152A,B,C & D are in the OPEN position

C. BP501 and DP501 pumps are running
AP501 and CP501 pumps are tripped
HV-2152B and HV-2152D Circ Water Pump Discharge Valves remain as-is with NO position indication
HV-2152A and HV-2152C Circ Water Pump Discharge Valves are in the CLOSED position

D. BP501 and DP501 pumps are running
AP501 and CP501 pumps are tripped
Circ Water Pump Discharge Valves HV-2152A,B,C & D are in the OPEN position

Proposed Answer: **A**

Explanation (Optional): HC.OP-SO.DA-0001, Section 3.3.8
In the event of a Bus Power Failure:
In the event of a 10A501 bus power failure, the "A" and "C" Pumps will trip with their respective valves closing within 30 seconds. The discharge valves for "B" and "D" Pumps will fail as is with loss of position indication.
In the event of a **10A502 bus power failure**, the **"B" and "D" Pumps will trip** with the respective valves closing within 30 seconds. **The Discharge Valves for "A" and "C" Pumps will remain as with NO position indication.**

Plausibility Justification:

A: **Correct-**. With the loss of the **10A502, the "B" and "D" Circ Pumps** will immediately trip and their associated **discharge valves will stroke close within 30 seconds**. Since the "A" and "C" Circ pumps still have power along with their discharge valves the pumps will continue to run the valves will not move, however they will **lose position indication**.

B: **Incorrect-** "A" and "C" pumps will be running and the "B" and "D" pumps will be tripped, however the "B" and "D" discharge valves will go closed to prevent spinning the tripped pumps backwards and running out the "A" and "C" pumps.

C: **Incorrect-** The 10A502 powers the "B" and "D" pumps. They would trip immediately and the respective discharge valves would go closed.

D: **Incorrect-**. The 10A502 powers the "B" and "D" pumps. They would trip immediately and the respective discharge valves would go closed.

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

Circulating Water System

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation, and/or alarms located in the Main Control Room, assess the status of the Circulating Water System by evaluation of the controls, instrumentation, and alarms.

Question Source: Bank # 34417

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	262002	K3.02	
	Importance Rating	2.9		

K/A Statement: Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Recirculation pump speed: Plant-Specific

Question: RO #6

Given:

- The ‘A’ Reactor Recirculation pump is operating at 88% speed with all cells in service.
- The ‘A’ NXG Controller was in control with the ‘B” NXG Controller available.
- The 1AD131 VFD 120V UPS was operating in NORMAL.

When:

- Due to a manipulation error, the 480VAC supply to the 1AD131 is inadvertently opened.
- ‘A’ NXG Controller failure occurs.
- Automatic NXG Controller switchover to ‘B’ NXG Controller occurs.
- (C1-D4) REACTOR RECIRC A TROUBLE OHA is received

No other operator actions have been taken.

What is the expected condition of the 'A' Recirculation pump/VFD following the NXG Controller Switchover?

A. The pump remains operating at 88% speed.

B. The pump is operating at less than 88% speed.

C. The pump is operating at greater than 88% speed.

D. The pump trips.

Proposed Answer: **B**

Explanation (Optional): Two dedicated **120 VAC UPS** units are provided to supply the control power for the VFD's. 480 VAC supply power is transformed into two 120 VAC feeds that supply the UPS. The redundant control power supplies of the VFD's are powered by separate 120 VAC sources, one from each UPS. The **1A-D-131 UPS** is the normal supply to the **A components** and the backup supply to the B components. The 1B-D-131 UPS is the normal supply to the B components and the backup supply to the A components. If there is **a loss of AC power**, **when AC input power is restored**, the Inverter **will automatically switch back to AC input and the UPS will begin recharging the batteries.**

The **‘A’ NXG controller is normally in control** with the ‘B’ in standby. If a fault occurs on the ‘A’ NXG controller, **control will swap to the ‘B’ NXG controller**. Drive output will shutdown for approximately **0.5 seconds during the swap, which will result in a small decrease in pump speed**. The main **Drive synchronizes to the new lower pump speed and initiates an Automatic Speed Hold.**

Plausibility Justification:

A: **Incorrect-**. Drive output will shutdown for approximately **0.5 seconds during the swap, which will result in a small decrease in pump speed**.

B: **Correct-** When input power is restored, the UPS will automatically swap back to NORMAL. If a fault occurs on the ‘A’ NXG controller, **control will swap to the ‘B’ NXG controller**. Drive output will shutdown for approximately **0.5 seconds during the swap, which will result in a small decrease in pump speed**. The main **Drive synchronizes to the new lower pump speed.**

C: **Incorrect-** The UPS will automatically swap back to NORMAL upon restoration of input power. The main **Drive synchronizes to the new lower pump speed.**

D: **Incorrect-**. The UPS will only swap to BYPASS operation due to an internal fault. The UPS would still be operating and able to automatically transfer back to NORMAL when input power was restored. Drive output will shutdown for approximately **0.5 seconds during the swap, which will result in a small decrease in pump speed.** The main **Drive synchronizes to the new lower pump speed.**

Technical Reference(s): HC.OP-SO.NQ-0003(Q) (Attach if not previously provided)

OPERATION OF REACTOR RECIRC VFD UNINTERRUPTIBLE POWER SUPPLIES

NOH01RECCON-16 Recirc. VFD LP

Proposed References to be provided to applicants during examination: none

Learning Objective: Identify the response of the Reactor Recirculation VFD UPS to the following:
Loss of Normal AC supply
Restoration of Normal AC Supply
Identify the response of the VFD to a Power Cell Bypass and NXG Controller failure.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K4.09	
	Importance Rating	3.1	

K/A Statement: Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Single element control

Question: RO #7

Given:

- The plant is conducting a startup IAW HC.OP-IO.ZZ-0003, "Startup from Cold Shutdown to Rated Power".
- The Digital Feedwater Control System is in Single Element Control.

Which of the following describes why RPV water level is the single controlling parameter during low power operations?

A. Steam flow/Feed flow signals are more accurate at low power.

B. RPV level changes are faster at low power than at high power.

C. Steam flow/Feed flow signals are less accurate at low power.

D. Narrow Range Rosemount Level Detectors are inaccurate at low power.

Proposed Answer: **C**

Explanation (Optional): **Single element control** is used as the controlling parameter **during low power operation** and means that vessel water level is monitored as the controlling parameter. At low power operations: **Vessel level changes are slower, Steam/feed flow signals are less accurate**, the control signal is developed by taking the median value of the three Narrow Range Rosemount Level detectors (PDT-N004 A, B &C)

Plausibility Justification:

A: **Incorrect-**. Steam flow/feed flow are parameters used by the DFCS, however at low powers theses parameters are less accurate and therefore the RPV water level will be the controlling parameter.

B: **Incorrect-** Actual RPV level changes are slower at lower powers as compared to higher power level changes.

C: **Correct-** Steam flow/feed flow are parameters used by the DFCS, however at low powers theses parameters **are less accurate** and therefore the **RPV water level will be the controlling parameter.** The Feedwater flow transmitters N002A/B- C32 indicate the Feedwater flows to the reactor vessel. The associated instrument loops are calibrated for Power Operation (OPCON 1) and thus **may indicate inaccurate values in other conditions.**

D: **Incorrect-**. Narrow Range (0" to +60") **is the most accurate level indication** during normal operations. It is referenced to instrument zero and calibrated for saturated steam-water mixture at 1000 psig, 135°F Drywell temperature, 75°F Reactor Building temperature. Provides level input to the Main Control Room, **DFCS**, ADS, RPS, and NS4

Technical Reference(s): HC.OP-IO.ZZ-0003(Q) (Attach if not previously provided)

STARTUP FROM COLD
SHUTDOWN TO RATED POWER
HC.OP-SO.AE-0001(Q)
FEEDWATER SYSTEM OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, describe the basic control scheme for single element (startup) and three element (master), and differential pressure Feedwater control modes including process variables that are used as input signals for each mode

Question Source: Bank # 30946

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K4.05	
	Importance Rating	3.6	

K/A Statement: Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: Reactor cooldown rate

Question: RO #8

Given:

- The plant is in the Shutdown Cooling mode of operation on the 'A' RHR loop.
- RHR Shutdown Cooling flow is normal.

When:

- The crew recognizes that the cooldown rate is excessive.
- The CRS directs the RO/PO to reduce the cooldown rate.

The current valve status/lineup for the 'A' RHR loop:

- BC-HV-F048A, RHR HX A SHELL SIDE BYP VLV is 100% open.
- BC-HV-F003A, RHR HX A OUTLET VLV is 80% open.
- BC-HV-F015A, RHR LOOP RET TO RECIRC VLV is 60% open.

Which one of the following valve manipulations would reduce the cooldown rate IAW plant procedures?

A. Throttling CLOSED on the BC-HV-F048A RHR HX SHELL SIDE BYP VLV.

B. Throttling OPEN on the BC-HV-F003A RHR HX A OUTLET VLV.

C. Throttling OPEN on the BC-HV-F015A RHR LOOP RET TO RECIRC VLV.

D. Throttling CLOSED on the BC-HV-F003A RHR HX A OUTLET VLV.

Proposed Answer: **D**

Explanation (Optional): See attached HC.OP-SO.BC-0002 Section 5.2.40

Plausibility Justification:

A: **Incorrect-**. Closing the F048A would send **more flow through the 'A' RHR HX which would increase the cooldown rate.** The student needs to understand the operational effects of manipulating the F048A and F003A and how the F048A being full opened requires the **F003A to be closed to decrease cooldown rate.**

B: **Incorrect-** With the F048A fully opened and then throttling the F003A further open would increase the cooldown rate. **More flow through the 'A' RHR HX.**

C: **Incorrect-** F015A is throttled to maintain Shutdown Cooling flow at required flow rates. The F015A is not throttled for temperature control IAW procedure.

D: **Correct-**. Closing the F003A would **reduce the flow through the 'A' RHR HX** which would decrease cooldown rate. The student needs to understand the operational effects of manipulating the F048A and F003A and how the F048A being full opened requires the **F003A to be closed to decrease cooldown rate.**

Technical Reference(s): HC.OP-SO.BC-0002(Q) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given procedure HC.OP-SO.BC-0002, "Decay Heat Removal Operation", explain the listed prerequisites, precautions, and/or limitations during operation.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(14)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	263000 K5.01	
Importance Rating	2.6	

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

Question: RO #9

Given:

- The plant is operating at 100% power.
- HPCI 250 VDC battery has just completed deep discharge rate surveillance testing.
- The HPCI Battery charger has been placed in service and is charging the associated HPCI battery bank.

Then:

- OHA Diesel Area HVAC Panel 1EC483 E6-C2 is received.
- The field operator reports a loss of all Battery room ventilation.
- The applicable Abnormal procedure(s) is(are) entered.

Which of the following describes an operational implication associated with the above conditions on the HPCI 250 VDC Distribution System?

A. Heat induced cracking of the battery cells

B. Buildup of hazardous lead sulfate ($PbSO_4$) dust on Battery Room components

C. Explosive hydrogen-air mixture in the Battery Room

D. Electrical fire caused when moisture condenses across battery terminals

Proposed Answer: **C**

Explanation (Optional): See attached HC.OP-AB.HVAC-0001

Plausibility Justification:

A: **Incorrect-**. The battery room temperature will not be high enough to cause cracking. However, the student could see this as a possible answer due to the higher than normal temperatures in the battery room. The room temperature will actually come down without ventilation. The battery rooms are kept between 74°F and 80 °F. The batteries are inoperable at <74°F.

B: **Incorrect-** As long as the batteries are intact there will be no lead sulfate on the batteries and certainly no dust will accumulate due to higher temperatures.

C: **Correct-** During **charging the battery produces hydrogen** which can build to explosive conditions without ventilation. Even though this would take a long time for an explosive amount of Hydrogen to build up, the HVAC abnormal specifically **directs monitoring both temperature and hydrogen concentration** during each shift (see attached).

D: **Incorrect-**. With the battery cells intact, no moisture will build up. With no ventilation the student might think the humidity would rise in the room. A rise in humidity would not cause a spark across terminals. The temporary ventilation would control battery room temperature between 74°F and 80 °F

Technical Reference(s): HC.OP-AB.HVAC-0001 (Attach if not previously provided)

HVAC

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of plant conditions/malfunctions associated with battery ventilation, evaluate whether a loss of ventilation to a battery room can result in equipment failure.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 K5.05	
	Importance Rating	3.4	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Paralleling A.C. power sources

Question: RO #10

Given:

- HC.OP-ST.KJ-0003, EMERGENCY DIESEL GENERATOR 1CG400 OPERABILITY TEST – MONTHLY is in progress.
- The PO is preparing to synchronize "C" Emergency Diesel Generator (EDG) to its vital bus.

Prior to synchronizing, the "C" EDG governor is placed in the __________ mode, to ensure the "C" EDG __________.

A. ISOCHRONOUS; exhibits proper load sharing characteristics.

B. DROOP; exhibits proper load sharing characteristics.

C. DROOP; vital bus frequency is maintained.

D. ISOCHRONOUS; vital bus frequency is maintained.

Proposed Answer: **B**

Explanation (Optional): ISOCHRONOUS MODE places the governor in isochronous **(frequency governing)**. DROOP MODE places the governor in droop **(load sharing).** SPEED DROOP control: permits load division and parallel operation of units **when controlled by the mechanical governor**. If the diesel is running in test (paralleled with an offsite source) when an **auto start signal** is received, the generator output breaker is tripped open. This is done to prevent severe power and voltage surges when the governor is shifted to **isochronous** and the voltage regulator is set for 4160 volts. After the breaker is tripped, it will reclose on the bus if required to supply power. This is the normal EDG standby line up.

Plausibility Justification:

A: **Incorrect-**. The EDG normal standby line up has the governor control in the Isochronous mode to provide 4160 KV/60 Hz. in the case of an Auto Start (LOP/LOCA). However, this mode of operation is used for frequency governing and not for manual synchronizing of the EDG.

B: **Correct-** The EDG governor control will be switched to DROOP from ISOCHRONOUS mode when manually synchronizing the EDG (see attached). The DROOP mode allows for load sharing with the mechanical governor.

C: **Incorrect-** Isochronous mode of operation is used for frequency governing. The DROOP mode allows for load sharing with the mechanical governor.

D: **Incorrect-**. The EDG governor control will be switched to DROOP from ISOCHRONOUS mode when manually synchronizing the EDG (see attached). The DROOP mode allows for load sharing with the mechanical governor.

Technical Reference(s): HC.OP-ST.KJ-0003 (Attach if not previously provided)

EMERGENCY DIESEL GENERATOR 1CG400 OPERABILITY TEST – MONTHLY

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a labeled diagram/drawing of, or access to, the Diesel Generator controls located in the main control room: Explain the effect of each control switch on the emergency diesel generators. Determine plant conditions or permissive required for the control switches to perform their intended function.

Question Source: Bank # 110708

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	212000	K6.04	
	Importance Rating	2.8		

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : D.C. electrical distribution

Question: RO #11

Given:

- The plant was operating at 100% power.

When:

- The power to both RPS Backup Scram valves was lost.

Then:

- A full actuation (de-energized) of the RPS system occurs.

The power lost to both RPS Backup Scram valves was the _____ (1) _____.

A full Reactor Scram will _____ (2) ______.

A. (1) 125 VDC distribution
(2) NOT occur

B. (1) 120 VAC distribution
(2) occur

C. (1) 125 VDC distribution
(2) occur

D. (1) 120 VAC distribution
(2) NOT occur

Proposed Answer: **C**

Explanation (Optional): The RPS Backup Scram valves consist of two **1E powered DC solenoid** operated, three-way, normally de-energized, pilot valves installed in series. **The 1E 125 VDC distributions (1AD417/1BD417)** supply the power to the normally **de-energized** Backup Scram valves. Air at 70 - 75 psig is supplied to the scram air header via the Backup Scram air header via the Backup scram valves (see attached). The Backup Scram Valves **automatically energize** to the vent position in the event of both RPS channels receiving a trip signal. The Backup Scram valves will depressurize the scram air header to vent air from the scram pilot valves and scram dump valves to allow the control rods to scram on a RPS trip signal. When the RPS system sends a trip signal to the Backup Scram valves the valves will not be able to be energized due to the loss of power and reposition to vent the Scram Air header.

Plausibility Justification:

A: **Incorrect-**. The Backup Scram valves are powered from the 1E 125 VDC distribution systems (1AD417/1BD417). However, due to the Backup Scram valves **not repositioning to vent** the **scram air header, the header will still be pressurized** and a **RPS reactor scram will not** have occurred. The student will have to decipher if the RPS scram was successful or not based on the **scram air header still being pressurized.**

B: **Incorrect-** The 120VAC distribution from RPS will power the **normally energized scram valves (scram pilot and dump valves)**. The Backup Scram valves are powered from the 1E 125 VDC distribution systems (1AD417/1BD417) and are normally de-energized. They will energize on a RPS trip signal.

C: **Correct-** The Backup Scram valves are powered from the 1E 125 VDC distribution systems (1AD417/1BD417). Due to the Backup Scram valves **not repositioning to vent** the **scram air header, the header will still be pressurized** and a **RPS reactor scram will not** have occurred.

D: **Incorrect-**. The 120VAC distribution from RPS will power the **normally energized scram valves (scram pilot and dump valves)**. Due to the Backup Scram valves **not repositioning to vent** the **scram air header, the header will still be pressurized** and a **RPS reactor scram will not** have occurred.

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

REACTOR PROTECTION SYSTEM OPERATION
M-47-1 CRDH P&ID
HC.OP-AB.ZZ-0001 Scram Hard Card

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions, evaluate the response of RPS to an electrical failure.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K6.03	
	Importance Rating	3.2	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY LIQUID CONTROL SYSTEM : A.C. power

Question: RO #12

Given:

- A loss of off-site power (LOP) has occurred.
- The "A" and "D" Diesel Generators did NOT start.
- All other systems responded as designed.

Which of the following describes the Standby Liquid Control (SLC) system injection capabilities for these conditions?

A. SLC can inject at 50% capacity until the "A" Diesel Generator is started.

B. SLC can inject at 100% capacity and both squib valves can fire.

C. SLC can inject at 50% capacity and both squib valves can fire.

D. NO SLC injection capability exists until at least the "A" or "D" Diesel Generator is started.

Proposed Answer: **A**

Explanation (Optional): 480V 1E AC Distribution - Power supplied to the below listed components originates from the **Emergency Diesel Generator Buses.** "A" SLC pump and squib valve F004A, and isolation valve F006A supplied by **MCC 10B212 (EDG"A"),** "B" SLC pump and squib valve F004B supplied by **MCC 10B222 (EDG"B"),** Isolation valve F006B supplied by **MCC 10B242 (EDG"D"),** SLC Tank Operating Heater supplied by **MCC 10B252 (EDG"A")** Non 1E from 1E SLC Tank Mixing Heater supplied by **MCC 10B282 (EDG"D")** Non 1E from 1E. Automatic or manual initiation, both SLC pumps START and both explosive valves FIRE, developing a flowpath from the storage tank to the "A" core spray sparger in the reactor vessel. The solution will pass through explosive valves F004A, B, enter a common header, flow through both outboard isolation **stop check valves** F006A, B, and then into a common header, terminating in 'A' core spray header between the nozzle penetration and the inboard manual isolation valve F007A.

Plausibility Justification:

A: **Correct-**. With the "B" EDG running and loaded (MCC10B222) the "B" SLC pump will be running along with its squib valve firing and allowing flow through the stop check valve to the Core Spray header. The "A" SLC pump and its squib valve will not have power until the "A" EDG is restored.

B: **Incorrect-** The "A" EDG needs to be restored and power the 10B242 MCC.

C: **Incorrect-** The squib valve power comes from the associated pump power (breaker), therefore only the "B" SLC pump and squib valve will have power.

D: **Incorrect-**. With the "B" EDG running and loaded (MCC10B222) the "B" SLC pump will be running along with its squib valve firing and allowing flow through the stop check valve to the Core Spray header. Restoration of "A" EDG will allow 100% capacity. Restoration of "D" EDG will not have an effect on the SLC injection flowpath due to the **F006B stop check valve in a normal open (allow flow) standby condition.**

Technical Reference(s): HC.OP-AB.ZZ-0170 (Attach if not previously provided)

LOSS OF 4.16KV BUS 10A401
A CHANNEL
HC.OP-AB.ZZ-0171
LOSS OF 4.16KV BUS 10A402
B CHANNEL

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory identify the power supply (i.e., 1E or Non-1E) to each of the following:
Standby Liquid Control Pumps.
Standby Liquid Control System Squib valves.
Standby Liquid Control System Storage Tank Heaters.

Question Source: Bank # 32602

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	271000 A1.05	
	Importance Rating	3.7	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC turbine speed

Question: RO #13

Given:

- The Reactor Core Isolation Cooling (RCIC) is operating in Full Flow Recirc Operation.
- The RCIC flow controller is in "Automatic".
- RCIC turbine speed is 4500 rpm.

Which of the following describes the response of RCIC turbine speed and system flow (after conditions stabilize), if the operator momentarily throttles the RCIC Pump Discharge Test Return to CST Valve (BD-HV-F022) in the "open" direction for the given conditions?

A. RCIC turbine speed lowers
System flow lowers

B. RCIC turbine speed lowers
System flow is unchanged

C. RCIC turbine speed rises
System flow is unchanged

D. RCIC turbine speed rises
System flow rises

Proposed Answer: **B**

Explanation (Optional): Flow Controller - The flow controller is a Bailey control station that allows the operator to select either the manual or automatic mode of operation. MANUAL - In this mode the operator sets desired RCIC turbine speed. **AUTOMATIC -** In the automatic mode **RCIC turbine speed is automatically adjusted to maintain desired RCIC pump discharge flow** established by the operator.

Plausibility Justification:

A: **Incorrect-**. In "Automatic" flow control, the RCIC flow controller will reduce turbine speed to **maintain flow at the setpoint.**

B: **Correct-** When F022 is throttled open, **system flow will rise**. In Auto flow control, controller **will reduce turbine speed** to maintain flow at the setpoint.

C: **Incorrect-** Throttling open on the F022 will cause a rise in system flow and the RCIC flow controller will actually lower the turbine speed to maintain the system flow at the setpoint.

D: **Incorrect-**. System flow will lower not rise when opening the F022. The RCIC flow controller will actually lower the turbine speed to maintain the system flow at the setpoint.

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

REACTOR CORE ISOLATION COOLING SYSTEM OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the effect of each control on the RCIC System.
Summarize plant conditions or permissives required for the control switches to perform their intended function.

Question Source: Bank # 35775

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A1.04	
	Importance Rating	2.8	

K/A Statement: Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: Surge Tank Level

Question: RO #14

Given:

- The plant is operating at 100% power.
- "A", "B", and "C" SACS pumps are operating.
- "D" SACS pump is in AUTO.

Then:

- EG-LT-2508A, "A" Expansion Tank level transmitter fails providing a low-low-low level signal.

Which of the following describes the response of the SACS system?

This low-low-low tank level _____________.

A. causes "A" and "C" SACS pump to trip, closes the associated SACS to TACS supply and return isolation valves. Valve closure produces a low SACS to TACS flow from Loop "A" and sends a start signal to the "D" SACS pump. The "D" pump start signal opens the Hx inlet valve, and the associated SACS to TACS supply and return valves.

B. sends a close signal to the associated TACS supply and return Isolation valves. Valve closure produces a low SACS to TACS flow from Loop "A". This produces a "D" SACS pump start and subsequent open signal to the associated Hx inlet valve and the "B" SACS Loop SACS to TACS supply and return isolation valves.

C. causes "A" SACS pump to trip. This produces a low SACS to TACS flow from Loop "A". This produces a "D" SACS pump start and subsequent open signal to the "B" SACS Loop SACS to TACS supply and return isolation valves. Loops are now cross-connected.

D. sends an open signal to the "B" SACS Loop SACS to TACS supply and return isolation valves and a start signal to the "D" SACS pump. The pump start opens the associated HX inlet valve. Loops are now cross-connected.

Proposed Answer: **B**

Explanation (Optional): The STACS will automatically swapover to the standby loop upon reaching a **low-low-low level condition** in the associated SACS expansion tank. SACS/TACS Supply Isolation Valves (HV-2522A,B,C,D) **AUTO CLOSE on Low-Low-Low level** in the **associated** SACS loop expansion tank **(LSLLL-2508A,B,C,D).** These valves AUTO OPEN when a **low flow condition exists in the opposite SACS loop** (FSL-2544A,B,C,D) and the control switch for the respective **SACS pump is in AUTO**. SACS/TACS Return Isolation Valves (HV-2496A, - D) will **AUTO CLOSE on Low-low-low level** in the **associated** SACS loop expansion tank (LSLLL-2508A,B,C,D). These valves will AUTO OPEN when a **low flow condition exists in the opposite** SACS loop (FSL-2544A,B,C,D) and the respective **SACS pump control switch is in the AUTO position.** Any SACS pump start (manual or automatic) opens its associated Hx inlet valve (HV-2491A(B), HV-2494A (B)).

Plausibility Justification:

A: **Incorrect-**. No Pump trip on LO-LO-LO tank level, Low TACS loop flow in the opposite loop sends a signal to auto start the "D" pump and all supply and return valve opening. The student has to understand that the "A" and "C" pumps will not trip on the low low low tank level but on a running loop low flow condition.

B: **Correct-** Low-low-low tank level closes the associated loop valves, causing a low flow in the operating TACS loop. Low TACS loop flow is in on the opposite loop. Auto pump start and all supply and return valves opening will occur.

C: **Incorrect-** No Pump trip on LO-LO-LO tank level. Loop Low flow condition (< 9,900 gpm). The loops will NOT be cross connected due to the closure of the associated loop SACS to TACS valves.

D: **Incorrect-**. The "D" pump auto start comes from the opposite loop low flow condition not from the LO LO LO expansion tank level signal. The SACS to TACS valves also receive their open signal from the opposite loop low flow condition. The loops will NOT be cross connected due to the closure of the associated loop SACS to TACS valves.

Technical Reference(s): HC.OP-SO.EG-0001(Q) SAFETY AND TURBINE AUXILIARIES COOLING WATER SYSTEM OPERATION (Attach if not previously provided)
HC.OP-AB.COOL-0002 SAFETY/TURBINE AUXILIARIES COOLING SYSTEM

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and/or a drawing of access to the controls, instrumentation and/or alarms located in the Main Control Room; assess the status of the SACS or its components by evaluation of the controls/instrumentation/alarms.

Question Source: Bank # 34022

Modified Bank # (Note changes or attach parent)

New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 A2.11	
	Importance Rating	3.2	

K/A Statement: Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High containment pressure

Question: RO #15

Given:

- The plant is operating at 100% power.
- RBVS and RBVE fans are running in a normal lineup.
- FRVS is in a normal standby configuration.

When:

- A high drywell pressure of 1.68 psig occurs.
- RBVS and RBVE system isolates.
- FRVS auto initiates.

Then:

- Two minutes later the operators complete the required actions for FRVS Auto Initiation Observation IAW HC.OP-SO.GU-0001, Filtration, Recirculation and Ventilation System Operation.

How many FRVS Recirculation Fans and Vent Fans will be in service following the actions taken IAW HC.OP-SO.GU-0001, Filtration, Recirculation and Ventilation System Operation?

	FRVS Recirculation Fans	FRVS Vent Fans
A.	6	2
B.	4	1
C.	6	1
D.	4	2

Proposed Answer: **B**

Explanation (Optional): FRVS Recirculation Fans **AV213 through FV213** in AUTO and **FRVS Vent Fan in AUTO LEAD** will automatically start under any of the following conditions:
- **High Drywell Pressure (1.68 psig).**
- Low RPV Water Level (Level 2, - 38").
- Refueling Floor Exhaust Duct High Radiation (≥ 2x10 -3 micro Ci/cc).
- Reactor Building Exhaust Air High Radiation (≥ 1x10 -3 micro Ci/cc).

There are 2 FRVS Vent Fans. One in Auto Lead the other in Auto. **The Auto Lead will start on the 1.68 psig signal** while the FRVS Vent Fan in AUTO will automatically start upon failure of the operating FRVS Vent Fan otherwise the Auto Vent fan is in a standby condition not running.
So, an auto initiation of the FRVS system will have all **six Recirculation** Fans running and **the Auto Lead Vent Fan running.** The operators will then IAW plant procedures **(HC.OP-SO.GU-0001) secure the "E" and "F" Recirculation fans** and place them in a standby condition looking for a low flow condition from the other four Recirculation fans that auto started. Post LOCA lineup for the FRVS system will have **four Recirculation Fans** in service with the **one Auto Lead Vent Fan** in service.

Plausibility Justification:

A: **Incorrect-**. All six Recirculation fans will auto start on the 1.68 high drywell pressure signal, however the operators will secure two Recirculation fans ("E" and "F"). There are two Vent fans, however only the Auto Lead will start on the high drywell pressure signal. The other Vent fan is in a standby condition to start on a failure of the Auto Lead Vent fan.

B: **Correct- Four Recirculation fans** will be running after the actions of HC.OP-SO.GU-0001 are completed ("E" and "F" fans are in standby for low flow condition on the running Recirculation fans). **The Auto Lead Vent Fan will auto start** and will continue to run with the Auto Vent fan in a standby condition (the Auto Vent fan will NOT receive a start on the 1.68 high drywell pressure signal).

C: **Incorrect-** All six Recirculation fans will auto start on the 1.68 high drywell pressure signal, however the operators will secure two Recirculation fans ("E" and "F") IAW HC.OP-SO.GU-0001.

D: **Incorrect-**. **Four Recirculation fans** will be running after the actions of HC.OP-SO.GU-0001 are completed ("E" and "F" fans are in standby for low flow condition). The Auto Vent fan will NOT receive a start on the 1.68 high drywell pressure.

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

FILTRATION, RECIRCULATION AND VENTILATION SYSTEM OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective: Concerning the Filtration Recirculation Ventilation System (FRVS): Distinguish between the automatic starts and stops associated with the FRVS Vent and Recirc Fans.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments

2021 NRC Written Examination

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A2.01	
	Importance Rating	2.7	

K/A Statement: Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR 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:

- The plant is operating at 100% power.
- All Neutron monitoring systems are OPERABLE.

When:

- The Neutron Monitoring Electrical Protection Assembly (EPA) breakers 1AN413 AND 1BN413 from 1AD483 trip open.

The following Abnormal procedures have been entered:

- HC.OP-AB.IC-0003, Reactor Protection System
- HC.OP-AB.IC-0004, Neutron Monitoring
- HC.OP-AB.ZZ-0136,Loss of 120 VAC Inverter

- The cause of the trip is unknown at this time.

(1) What is the current status of the Power Range Neutron Monitoring System?
(2) What additional action is required IAW abnormal plant procedures?

A. (1) "A" AND "C" APRM's are DE-ENERGIZED; "A" AND "C" 2/4 voters are DE-ENERGIZED.
(2) Bypass "A" AND "C" APRMs.

B. (1) "A" AND "C" APRM's remain energized; "A" AND "C" 2/4 voters are ENERGIZED.
(2) Reset the EPA breakers 1AN/BN413 once the cause is known.

C. (1) "A" AND "C" APRM's are DE-ENERGIZED; "A" AND "C" 2/4 voters are ENERGIZED.
(2) Bypass "A" AND "C" APRMs.

D. (1) "A" AND "C" APRM's remain energized; "A" AND "C" 2/4 voters are DE-ENERGIZED.
(2) Reset the EPA breakers 1AN/BN413 once the cause is known.

Proposed Answer: **D**

Explanation (Optional): The **APRMs,** LPRMs(APRM slaves) and RBMs are powered from two 120VAC UPS buses, **1AJ483 (thru EPA breakers 1AN413 and 1BN413**) and 1BJ483 (thru EPA breakers 1AN414 and 1BN414). Each **APRM**, LPRM, and RBM have **two auctioneered power supplies from their respective Quad Low Voltage Power Supply (QLVPS) providing +5VDC, and ±15VDC. A loss of one 120VAC bus will not result in a loss of power to the PRNM instruments. The 2/4 Voter Modules are not powered from QLVPS** but directly **from their associated EPA breakers. Voter Modules A & C are powered from 1AJ483,** Voter Modules B & D are powered from 1BJ483. A loss of power from **one of the**

120VAC UPS's will result in two (2) voters to change state, fail-safe and all their indications OFF with no lights. This will result in a "half scram" as ½ of RPS will be tripped. The **other two (2) voters will remain energized and the logic is still "any 2 out of 4" APRMs on the remaining two voters.**

Plausibility Justification:

A: **Incorrect-**. Due to the fact that only one of the 120VAC UPS power supplies was lost (1AJ483), **the "A" & "C" APRMs (PRNMs) will still have power through 1BJ483 and the auctioneered power supply from the respective QLVPS.** However, the 2/4 Voters Modules A & C supplied by the 120VAC 1AJ483 **will be de-energized** and therefore a RPS trip on A1/A2 RPS trip logic (half scram). Since the APRMs are still powered and indicating they DO NOT need to be bypassed.

B: **Incorrect-** The 2/4 Voters Modules A & C supplied by the 120VAC 1AJ483 **will be de-energized** and therefore a **RPS trip on A1/A2 RPS trip logic (half scram).** The source of the problem would have to be corrected (**Reset the EPA breakers**) before resetting the RPS trip.

C: **Incorrect-** The "A" & "C" APRMs (PRNMs) **will still have power** through 1BJ483 and the auctioneered power supply from the respective QLVPS. The 2/4 Voters Modules A & C supplied by the 120VAC 1AJ483 **will be de-energized** and therefore a **RPS trip on A1/A2 RPS trip logic (half scram).** Since the APRMs are still powered and indicating they DO NOT need to be bypassed.

D: **Correct-**. The "A" & "C" APRMs (PRNMs) will still have power from 1BJ483 and the auctioneered power supply from the respective QLVPS. The 2/4 Voters Modules A & C supplied by the 120VAC 1AJ483 **will be de-energized** and therefore a RPS trip on A1/A2 RPS trip logic (half scram). The source of the problem would have to be corrected (**Reset the EPA breakers**) before resetting the RPS trip.

Technical Reference(s): PN1-C51-1080-0026, Sheet 6 (Attach if not previously provided)

120VAC Power Supply to PRNMs

HC.OP-AB.IC-0003, RPS
HC.OP-AB.IC-0004, Nis
HC.OP-AB.ZZ-0136, Loss of 120VAC

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, determine the rod blocks and/or scrams initiated by the PRNM System.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7,10)

Comments:

2021 NRC Written Examination

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A3.02	
	Importance Rating	3.5	

K/A Statement: Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: Valve closures

Question: RO #17

Given:

- Drywell pressure is at 1.3 psig
- Reactor water level is at -56 inches
- Main condenser pressure is at 22 inHgA
- Reactor pressure is at 75 psig

Which one of the following describes the system valves that are isolated based on exceeding isolations signals?

- BB-SV4310/SV4311- Reactor Recirculation Sample Line isolation valves.
- BG-HVF001/F004 - RWCU isolation valves.
- AB-HVF022/F028 - MSIVs.
- AB-HVF016/F019 - MSL Drain isolation valves.
- GB-HV9531 – Drywell Chilled Water isolation valve.

A. Reactor Recirculation Sample Line isolation valves, Drywell Chilled Water Isolation valves, MSIVs ONLY

B. RWCU isolation valves, MSIVs and MSL Drain isolation valves, Drywell Chilled Water isolation valves ONLY.

C. Reactor Recirculation Sample Line isolation valves, MSIVs and MSL Drain isolation valves, RWCU isolation valves ONLY.

D. RWCU isolation valves, MSL Drain isolation valves, Drywell Chilled Water isolation ONLY.

Proposed Answer: **C**

Explanation (Optional): See attached HC.OP-SO.SM-0001 Tables of valve isolations
Plausibility Justification:

A: **Incorrect-. RX** Sample valves close on RPV level **(-38 inches).** Drywell Chilled Water valves close on DW press **(1.68 psig)** or RPV level **(-129 inches),** MSIVs and drains are closed on low vacuum **(21.5 inHgA).**

B: **Incorrect-** Drywell Chilled Water valves close on DW press **(1.68 psig)** or RPV level **(-129 inches),** RWCU isolated on RPV level **(-38 inches**), MSIVs and drains are closed on low vacuum **(21.5 inHgA).**

C: **Correct-** MSIVs and drains are closed on low vacuum **(21.5 inHgA)**, RWCU isolated on RPV level **(-38 inches**), RX Sample valves close on RPV level **(-38 inches).**

D: **Incorrect-**. MSIVs and drains are closed on low vacuum **(21.5 inHgA)**, RWCU isolated on RPV level **(-38 inches**), Drywell Chilled Water valves close on DW press **(1.68 psig)** or RPV level **(-129 inches)**.

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

ISOLATION SYSTEMS OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective: Interpret and apply charts, graphs and tables contained within Primary Containment.

Question Source: Bank # 35757

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 A3.01	
	Importance Rating	3.6	

K/A Statement: Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: Turbine speed: BWR-2,3,4

Question: RO #18

Given:

- HPCI has initiated from a valid Hi Drywell Pressure signal of 1.68 psig
- RPV Water level is at 35"
- Reactor pressure is at 980 psig

Then:

- The operator observes that the HPCI turbine control valve (FV-4879) is in the throttled position AND that turbine speed is lowering.

Additionally, the operator observes the following valves going closed:

- FV-4880, HPCI turbine stop valve
- HV-F006, HPCI pump discharge to Core Spray
- HV-8278, HPCI pump discharge to Feedwater
- HV-F012, HPCI minimum flow

Later, with the Hi Drywell Pressure signal still in, the operator notices that HPCI turbine speed is again rising and HPCI realigns for injection without taking any actions.

Which of the following could have caused this HPCI response?

A. A mechanical overspeed HPCI trip

B. An automatic reset of a HPCI isolation

C. An automatic swap of the HPCI pump suction source

D. A HPCI flow controller incorrectly set

Proposed Answer: **A**

Explanation (Optional): Upon reaching an overspeed condition (approximately 5200 rpm), hydraulic oil will be removed from the HPCI turbine stop valve actuator, allowing spring tension to close it. **This causes a loss of steam flow to the turbine. When the turbine has slowed sufficiently**, hydraulic oil will be re-applied to the stop valve actuator and **the HPCI System will restart if an initiation signal is still present.** A mechanical overspeed trip will cause the **turbine stop valve (FV-4880) to close**. This causes the following valves to close: HPCI pump discharge valve to Core Spray **(HV-F006)** and HPCI pump minimum flow valve **(HV-F012)**. **The HV-8278** will automatically close upon receipt of FV-4880 fully closed. The turbine control valves **(FV-4879)** will be throttled by the turbine governor control system to limit overspeed. HPCI trips, isolations, and interlocks **(see attached HC.OP-SO.BJ-0001).**

Plausibility Justification:

A: **Correct-**. Upon reaching an overspeed condition (approximately 5200 rpm), hydraulic oil will be removed from the HPCI turbine stop valve actuator, allowing spring tension to close it. **This causes a loss of steam flow to the turbine. When the turbine has slowed sufficiently**, hydraulic oil will be re-applied to the stop valve actuator and **the HPCI System will restart if an initiation signal is still present.**

B: **Incorrect-** There is no HPCI isolation signal present (see attached). The student will have to determine that the valves that are isolating are due to a mechanical overspeed trip not a isolation.

C: **Incorrect-** BJ-HV-F042, PMP SUCT FROM SUPP CHB Auto closes on HPCI Div 1 Isolation signal. Auto opens on CST low level OR Suppression Chamber high level. **There is no change in CST or Suppression pool level.** The BJ-HV-F042 has no interlock with the governor control or the other valves mentioned.

D: **Incorrect-**. Placing HPCI FIC-R600 in MAN results in an open loop control with the flow controller output becoming **a fixed speed demand signal to the turbine governor**. This will provide stable, **constant turbine speed control**, but will require operator action to maintain the desired vessel injection flow rate.

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

HIGH PRESSURE COOLANT INJECTION SYSTEM OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective: For HPCI System trips and isolations: Given plant conditions, determine the sequence of events following receipt of a HPCI turbine trip signal

Question Source: Bank # 30478

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000	A4.01
	Importance Rating	2.6	

K/A Statement: Ability to manually operate and/or monitor in the control room: Pressure gauges- Instrument Air.

Question: RO #19

Given:

- The 10K107 Service Air compressor is running.
- Both the 00F-104 and 10F-104 Air Dryers are in-service.
- The 00K107 Service Air compressor is in standby
- The 1AF-104 Air Dryer is in standby.
- The 10K100 Emergency Instrument Air Compressor is in standby.
- The KAHV-7595, Service Air Supply Header Isolation Valve, is open.

Then:

- The following annunciators are received:
  - A2-A1, INST AIR HEADER A PRESSURE LO
  - A2-A2, INST AIR HEADER B PRESSURE LO
  - A2-B1, COMPRESSED AIR SYSTEM TROUBLE
  - A2-B2, COMPRESSED AIR PANEL 00C188
- Current air pressures are:
  - Service Air pressure is 90 psig.
  - Instrument air pressure at the Emergency Instrument Air Receiver is 82 psig.
  - Instrument air pressure is 77 psig.

After one minute with the same air pressures above,

What is the configuration of the Service and Instrument Air System?

A. The 00K107 Service Air Compressor is running
Instrument Air Dryer 1AF104 is in-service.
The Emergency Instrument Air Compressor is running.
The KAHV-7595 is open.

B. The 00K107 Service Air Compressor is in standby not running
Instrument Air Dryer 1AF104 is in standby.
The Emergency Instrument Air Compressor is running.
The KAHV-7595 is closed.

C. The 00K107 Service Air Compressor is in standby not running.
Instrument Air Dryer 1AF104 is in-service.
The Emergency Instrument Air Compressor is in standby not running.
The KAHV-7595 is closed.

D. The 00K107 Service Air Compressor is running.
Instrument Air Dryer 1AF104 is in standby.
The Emergency Instrument Air Compressor is in standby not running.
The KAHV-7595 is open.

Proposed Answer: **A**

Explanation (Optional): Normal Service/Instrument Air System line up will have the LEAD air compressor in-service with the LAG compressor in standby along with the EIAC in standby. With the two of the three Air Dryers in-service the third air dryer (1AF-104) will be in standby. The service air header isolation valve (7595) will be normally opened. As the air pressures lower there are specific setpoints for the above equipment to Auto Start or isolate along with specific air pressure gauge readings (see attached table).

Plausibility Justification:

A: **Correct-**. With the **Service Air Pressure <92 psig** and the pre-lube timer timed out (the timer is set for 1 minute), the standby air compressor 00K107 will auto start. With the **Instrument Air Pressure ≤ 85 psig, t**he standby Air Dryer 1AF-104 will go into service. With the **Emergency Instrument Air Compressor Receiver Pressure ≤ 85 psig** the EIAC will auto start. For the KAHV-7595 to isolate (close) the **Instrument Air Pressure has to reach ≤ 70 psig.**

B: **Incorrect-** With the **Service Air Pressure <92 psig** and the pre-lube timer timed out (the timer is set for 1 minute), the standby air compressor 00K107 will auto start. With the **Instrument Air Pressure ≤ 85 psig, t**he standby Air Dryer 1AF-104 will go into service. For the KAHV-7595 to isolate (close) the **Instrument Air Pressure has to reach ≤ 70 psig.**

C: **Incorrect-** With the **Service Air Pressure <92 psig** and the pre-lube timer timed out (the timer is set for 1 minute), the standby air compressor 00K107 will auto start. With the **Emergency Instrument Air Compressor Receiver Pressure ≤ 85 psig** the EIAC will auto start. For the KAHV-7595 to isolate (close) the **Instrument Air Pressure has to reach ≤ 70 psig.**

D: **Incorrect-**. With the **Instrument Air Pressure ≤ 85 psig, t**he standby Air Dryer 1AF-104 will go into service. With the **Emergency Instrument Air Compressor Receiver Pressure ≤ 85 psig** the EIAC will auto start.

Technical Reference(s): HC.OP-AB.COMP-0001(Q) (Attach if not previously provided)

INSTRUMENT AND/OR SERVICE AIR

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a list of possible trips, determine which are valid compressor trips. Summarize/identify the operation of the emergency instrument air compressor.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	2150004 A4.06	
	Importance Rating	3.2	

K/A Statement: Ability to manually operate and/or monitor in the control room: Alarms and lights- Source Range Monitor.

Question: RO #20

Given:

- The plant is performing control rod withdrawals with the mode switch in STARTUP.

When:

- The "ROD OUT MOTION BLOCK" overhead window illuminates.
- All IRM range switches are on range 3 or 4.
- "A" SRM has failed downscale to 1 cps and is being withdrawn.
- "A" SRM "DNSC" light is illuminated.
- "B" SRM is fully withdrawn and reading 95 cps.
- "C" SRM is partially withdrawn and reading 98,000 cps.
- "D" SRM is partially withdrawn and reading 103,000 cps.

Which SRM is the cause of the rod block?

A. "A" SRM

B. "B" SRM

C. "C" SRM

D. "D" SRM

Proposed Answer: **D**

Explanation (Optional): SRM Upscale rod block: **SRM reading 10E5 cps or greater**. Bypasses: Joystick selected for the associated SRM channel or Range 8 or greater on associated IRM's or Reactor Mode Switch in RUN (See attached). SRM downscale rod block: **Less than 3 cps**. Bypasses: Joystick selected for the associated SRM channel or associated IRM range switches positioned to range 3 or greater or Reactor Mode Switch in RUN (See attached).

Plausibility Justification:

A: **Incorrect-**. With the IRMs on range 3 and 4 the "A" SRM downscale rod block is bypassed. The student has to know the bypass interlocks for the SRMs.

B: **Incorrect-** "B" SRM is not at a level that would either be upscale or downscale for a Rod Block. The detector not fully inserted and counts less than 100 cps is a Rod Block, however this rod block is bypassed with the IRMs on range 3 or greater (See attached).

C: **Incorrect-** "C" SRM is below the upscale setpoint and greater than 100 cps with the detector not fully inserted along with the bypass of the IRMs on range 3 or greater.

D: **Correct-**.."D" SRM is above the Rod Block setpoint for cps at > 10E5 cps. Due to the fact that the "A" SRM Rod Block is bypassed with the IRMs on range 3 or greater the only SRM that is causing the Rod Block is the "D" SRM.

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

Nuclear Instrumentation System Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, choose the parameter, setpoint, and bypass conditions for each SRM signal which will initiate a rod block and/or reactor scram.

Question Source: Bank # 30618

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A4.11	
	Importance Rating	3.7	

K/A Statement: Conduct of Operations: Ability to manually operate and/or monitor in the control room: System flow

Question: RO #21

A Caution/Step in HC.OP-SO.BE-0001, Core Spray System Operation states that:

If INIT AND SEALED IN A (B, C, D) is on and Core Spray auto initiation has not occurred, BOTH pumps in a loop must be manually started.

WHICH ONE (1) of the following describes the basis for this Caution/Step?

A. Both pump manual start pushbuttons must be depressed to start the associated diesel generator in case of Loss of offsite Power.

B. Both pumps must be started to satisfy the interlock for opening the Outboard Injection Valve.

C. Both pumps must be operating to provide sufficient driving head to open check valve HV-F006A(B).

D. Both pumps must be running to prevent pump runout

Proposed Answer: **D**

Explanation (Optional): A Caution in HC.OP-SO.BE-0001, Core Spray System Operation states that if INIT AND SEALED IN A(B,C,D) is on and Core Spray auto initiation has not occurred, BOTH pumps in a loop must be manually started. (See attached)

HC.OP-SO.BE-0001 - 3.0 Precaution and Limitations

3.1.3. Arming AND pressing A(B) MAN INIT PB will start the associated Core Spray Pump AND open the Outboard Injection Valve (WHEN the low Reactor pressure interlock is satisfied). **Arming AND pressing C(D) MAN INIT PB will start the associated pump. <u>To ensure full Core Spray injection and prevent pump runout, BOTH pumps in a loop MUST be operated.</u>**

Plausibility Justification:

A: **Incorrect-**. The EDG's start on the CS initiation logic **not the pump start logic**

B: **Incorrect-** The injection valve needs to see either the "A" or "B" logic and the < 461 pressure permissive

C: **Incorrect-** When pump discharge pressure is > RPV pressure the check valve will open ≈ 380 psi.

D: **Correct-**. Both pumps must be running to prevent pump runout

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

Core Spray System

Proposed References to be provided to applicants during examination: none

Learning Objective: Summarize/identify the sequence of events following receipt of an automatic or manual Core Spray System initiation signal

Question Source: Bank # 29633

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 2.1.30	
	Importance Rating	4.4	

K/A Statement: Conduct of Operations: Ability to locate and operate components, including local controls- SLC

Question: RO #22

Given:

- The plant is operating at 90% power.
- HC.OP-IS.BH-0003, STANDBY LIQUID CONTROL PUMP - AP208 - INSERVICE TEST, to check flow rates during power operations is to be performed.

How is the firing of the squib valve(s) avoided, when starting the pump(s) for this surveillance?

A. The SLC squib valve firing circuitry comes directly from RRCS and is unaffected during this test.

B. The squib valve(s) must be physically removed from the system before running the SLC pump(s).

C. The breaker(s) for the appropriate squib valve(s) must be opened prior to running the SLC pump(s).

D. Starting the Standby Liquid Control pump with the local control switch bypasses the squib firing circuit.

Proposed Answer: **D**

Explanation (Optional): Local START-STOP of each SLC pump can be affected from local control panel 10C011. Operation of either local switch **will START only the associated pump and will not FIRE** either of the squib valves, or cause a RWCU system isolation. The surveillance requires the local switches to be manipulated so that the squib valves do not fire and introduce sodium pentaborate into the reactor.(see attached). Primer circuits for the squib valves are powered from a Class 1E Power Supply which originates at the SLC A/B pump breakers. The actuation circuitry of the squib valves prevents the firing of the explosive charges when pump operation is initiated from the TEST switches located on panel 10C011

Plausibility Justification:

A: **Incorrect-**. An RRCS signal would automatically start both pumps and fire the squib valves, however the squib valve circuitry comes directly from the pump breaker and is bypassed when using the local panel pump control switches.

B: **Incorrect-** The squib valves IAW the surveillance are not removed due to the fact that using the local control switches will bypass the firing circuit for the squib valves and still start the pumps.

C: **Incorrect-** The power for the squib valves come from the associated pump breaker and the firing circuitry will be bypassed by the manipulation of the local control switch for the associated pump.

D: **Correct-**. Operating the local control switch for the "A" SLC pump will bypass the firing circuitry for the associated squib valve and therefore the "A" pump will start for testing of the flowrate of the pump without firing the squib valve and not introducing sodium pentaborate into the reactor.

Technical Reference(s): HC.OP-IS.BH-0003(Q) (Attach if not previously provided)

STANDBY LIQUID CONTROL PUMP - AP208 - INSERVICE TEST

Proposed References to be provided to applicants during examination: (none)

Learning Objective: From memory, summarize/identify the locations from which the Standby Liquid Control System pumps may be manually started/stopped and summarize the effect that operating the pumps from each location will have on Standby Liquid Control System response.

Question Source: Bank # 72736

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 K4.01	
	Importance Rating	3.4	

K/A Statement: Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump

Question: RO #23

Given:

- All service water pump controls are in AUTO.
- "A", "C", and "D" SSW pumps are running.
- The 'B' PCIS MAN Initiation Pushbutton (10C651C) is armed and depressed.

Select the response of "B" and "D" Service Water pumps.

A. The D SW pump will trip, and then both B and D SW pumps will start in approximately 55 seconds as controlled by the LOCA Sequencer.

B. The D SW pump will trip, and then only the B SW pump will start immediately.

C. The D SW pump will continue running, and the B SW pump will start immediately.

D. The D SW pump will continue running, and the B SW pump will start in approximately 55 seconds as controlled by the LOCA Sequencer.

Proposed Answer: **C**

Explanation (Optional): Each PCIS channel can be manually initiated by ARMING (rotating respective collar fully clockwise) and DEPRESSING the respective CNTMT ISLN MAN INITIATION pushbutton. Manually tripping **a "PCIS channel" will initiate the RBE/RFE functions associated with that respective channel.** The Service water systems receive a start signal from: LOCA Level 1 and LOCA Level 2/ **Reactor Building-Refuel Floor Exhaust Hi-Hi Radiation**. (see attached table).

Plausibility Justification:

A: **Incorrect-**. The "D" SSW pump will continue to run since it was already running. "B" channel PCIS does not send a trip signal to the "D" SSW pump. PCIS also does not send a signal to start the LOCA sequencer. The "D" SSW pump will receive an AUTO start from the "D" PCIS channel not the "B" PCIS channel. The student needs to know that the PCIS system is single channelized.

B: **Incorrect-** There would be no trip signal to the "D" SSW pump from the "B" PCIS channel. The "B" SSW pump would start immediately (see attached).

C: **Correct-** Since the PCIS system is channelized the manual initiation of the "B" channel would start the "B" SSW pump immediately and since there is no effect on the "D" SSW pump the pump would continue to run since it was already running.

D: **Incorrect-**. PCIS does not send a signal to start the LOCA sequencer. The manual initiation of "B" PCIS channel will start the "B" SSW pump immediately without any time delays (see attached).

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

ISOLATION SYSTEMS OPERATION

Proposed References to be provided to applicants during examination: none

Learning Objective: Given an Automatic or Manual Initiation of the Primary Containment Isolation System (PCIS), evaluate the effect on the individual system and determine the overall plant effect on plant operations

Question Source: Bank # 30000

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	259002 A4.01	
Importance Rating	3.8	

K/A Statement: Ability to manually operate and/or monitor in the control room: All individual component controllers in the manual mode. Reactor Water Level Control.

Question: RO #24

Given:

- The plant was operating at 100% power.

When, a valid Reactor Vessel High Dome Pressure signal is received:

- The reactor failed to scram.
- APRMs are NOT DOWNSCALE.
- FEEDWATER RUNBACK INITIATED is on.
- All three RFP controller speeds indicate 2500 RPMs.
- All applicable emergency operating procedures have been entered.

RFP speed controllers are reduced to limit feedwater flow to 0% _____________.

A. until reference APRMs are downscale, then automatic control of the RFP controllers is available.

B. for 30 seconds, then manual control of the RFP controllers is available.

C. until reference APRMs are downscale, then manual control of the RFP controllers is available.

D. for 30 seconds, then automatic control of the RFP controllers is available.

Proposed Answer: **B**

Explanation (Optional): The RRCS runback limits RFPT speed to **2500 rpm** if: 25 seconds after Reactor pressure ≥1071 psig and APRM power is not downscale (below 4%) or is INOP. **RFP control is returned at the DFCS controller in Manual at the minimum setting after 30 seconds** as indicated by the **MAN CONTROL AVAIL** light on 10C651B being illuminated. This runback lowers inlet subcooling which provides a negative reactivity effect (See attached).

Plausibility Justification:

A: **Incorrect-**. The APRMs > 4% along with the ≥1071 psig reactor pressure will cause the RFPT runback. The RFP speed controllers will have **MANUAL control** available after 30 seconds of the RRCS runback regardless of what the APRMs are reading. Due to the fact that the APRMs caused the runback, the student will have to decipher if the APRM Downscales allows automatic control of the RFPs or not.

B: **Correct-** RFP control is returned at the **DFCS controller in Manual** at the minimum setting after 30 seconds.

C: **Incorrect-** The APRMs > 4% along with the ≥1071 psig reactor pressure will cause the RFPT runback. The RFP speed controllers will have **MANUAL control** available after 30 seconds of the RRCS runback regardless of what the APRMs are reading.

D: **Incorrect-**. RFP control is returned at the **DFCS controller in Manual** at the minimum setting after 30 seconds. The student could decipher that the controllers would be back in automatic.

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

REDUNDANT REACTIVITY
CONTROL SYSTEM OPERATION
HC.OP-AR.ZZ-0013

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, predict the sequence of events which occur within the Redundant Reactivity Control System upon: Automatic initiation in response to a high reactor vessel pressure condition with or without the APRM permissive.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 2.1.20	
	Importance Rating	4.6	

K/A Statement: Conduct of Operations: Ability to interpret and execute procedure steps.- A.C Electrical Distribution.

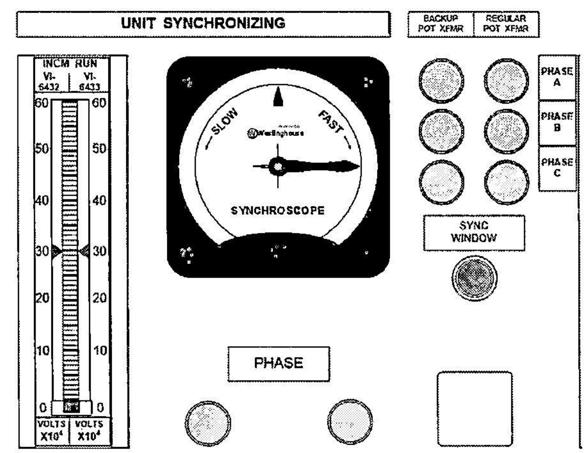
Question: RO #25

Given:

- A unit startup is in progress.
- The main generator is being synchronized to the grid IAW HC.OP-SO.MA-0001, MAIN GENERATOR & EXCITER OPERATION & SWITCHING Section 5.2.

The following indications are observed:

- Keylocked SYNC SCOPE switch is in the ON position.
- The SYNCH WINDOW light is NOT illuminated.
- Sync Scope is rotating slowly in the Clockwise direction.
- Generator and Grid voltages are set IAW the SOP.



With the SYNC WINDOW light NOT illuminated __________.

A. the SYNC CHECK OFF pushbutton must be depressed before depressing the breaker BS6-5 (BS2-6) CLOSE pushbutton.

B. the main generator exciter field breaker is open.

C. the Generator Voltage must be lowered to less than the grid voltage before depressing the breaker BS6-5 (BS2-6) CLOSE pushbutton.

D. the 52x60 Generator Disconnect is open.

Proposed Answer: **A**

Explanation (Optional): See attached section 5.2 of the HC.OP-SO.MA-0001, MAIN GENERATOR & EXCITER OPERATION & SWITCHING for Synchronizing and Loading the Main Generator.

Plausibility Justification:

A: **Correct-**. SYNCH CHECK OFF (momentary pb) - De-energizes the synch check relay and removes the requirement for synch check relay input to enable breaker closure. When the synch check relay is not functioning properly, **this pb must be depressed while closing its associated main generator output breaker. IAW Section 5.2 (see attached) Steps 5.2.13.A and B** allows the operator to bypass the synch check relay and close the main generator breaker to synchronize the unit to the grid.

B: **Incorrect-** For the sync scope and voltages to indicate that the machine is properly synchronized the exciter field breaker and 52x60 must both be closed. The exciter field breaker is closed **IAW step 5.2.5.** (See attached)

C: **Incorrect-** Generator voltage must be greater than grid voltage **IAW step 5.2.11**. (See attached).

D: **Incorrect-**. The 52x60 Main Generator Disconnect is closed **IAW Step 5.4.15** (See attached) as a prerequisite to Section 5.2 being performed.

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

MAIN GENERATOR & EXCITER OPERATION & SWITCHING

Proposed References to be provided to applicants during examination: Unit Synchronization Figure in stem

Learning Objective: Given a labeled diagram/drawing of, or access to, the Main Power System controls/indication bezel:
Explain the function of each indicator.
Predict the conditions that will cause the indicators to light or extinguish.
Determine the effect of each control switch on the Main Power System.
Explain the conditions or permissives required for the control switches to perform their intended functions.

Question Source: Bank # 32750

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 A3.02	
	Importance Rating	3.2	

K/A Statement: Ability to monitor automatic operations of the STANDBYAS TREATMENT SYSTEM including: Fan start

Question: RO #26

Given:

- The plant is operating at 100% power.
- OHA E6-A5 RB EXH RADIATION ALARM/TRBL has annunciated.
- The Plant Operator reports that RB Ventilation Exhaust is reading 1.5 x 10-3 µCi/cc on all three channels.

Based on this, the Plant Operator IAW HC.OP-AB.CONT-0004, Radioactive Gaseous Release is required to ____________________.

A. manually place FRVS in service and monitor Off Site Release Rates.

B. ensure reactor building ventilation is in service and building dP is > -0.30 inches water gauge.

C. ensure reactor building ventilation is isolated and FRVS started.

D. ensure reactor building ventilation is isolated and FRVS started, if RB Ventilation Exhaust reaches 2.0 x 10-3 µCi/cc.

Proposed Answer: **C**

Explanation (Optional): Air exhausted from the reactor building is **monitored for radiation,** prior to passing through the secondary containment isolation dampers, by three rad monitors. If 2 out of 3 monitors sense a **rad level of 1x10-3 µCi /cc**; then, the 1E breakers for the RBVE and RBVS fans trip, the secondary containment supply and exhaust dampers close, **FRVS starts.**

Plausibility Justification:

A: **Incorrect-**. The FRVS system **will auto start** on the RB Exhaust radiation level of ≥ 1x10 -3 micro Ci/cc. on 2 out of 3 channels. The student will have to understand that the setpoint is reached for hi rad level start of the FRVS system. Manually starting the FRVS system is an option when rad level are rising, however the auto start setpoint has been reached.

B: **Incorrect-** The RBVS system **will automatically isolate** on the RB Exhaust radiation level of ≥ 1x10 -3 micro Ci/cc. on 2 out of 3 channels. If the RBVS system was still in service then the procedure would guide the operator to maintain secondary containment dP.

C: **Correct-** The FRVS system **will auto start and t**he RBVS system **will automatically isolate** on the RB Exhaust radiation level of ≥ 1x10 -3 micro Ci/cc. on 2 out of 3 channels. (See attached).

D: **Incorrect-**. Air exhausted from the refuel floor to the RBVE system is monitored for radiation, prior to passing through secondary containment isolation dampers, by three rad monitors. If 2 out of 3 sense a rad level **of 2x10-3 micro ci/cc** the following occurs: the 1E breakers for the RBVE and RBVS fans trip, the secondary containment supply and exhaust dampers close, FRVS starts. **The rad levels are at 1.5 x 10-3 µCi/cc on the RB Exhaust so all of the automatic actions above would already have happened.**

Technical Reference(s): HC.OP-AB.CONT-0004(Q) (Attach if not previously provided)

RADIOACTIVE GASEOUS RELEASE

HC.OP-SO.GU-0001(Q)
FRVS System Operation

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with Radioactive Gaseous Release.

Question Source: Bank # 113146

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	256000 K1.07	
	Importance Rating	2.9	

K/A Statement: Knowledge of the physical connections and/or cause- effect relationships between REACTOR CONDENSATE SYSTEM and the following: SJAE condenser.

Question: RO #27

Given:

- A plant startup is in progress.
- The operators were placing the first PCP (Primary Condensate Pump) in-service IAW HC.OP-SO.AD-0001, Condensate System Operations, the 'A' PCP pump.

When:

- The PO (Plant Operator) noticed that the PRI CNDS PUMP 'A' START ENABLE status light was NOT illuminated.

Which of the following would be the cause of the above condition?

A. HV-1680A, PCP 'A' DISCH VLV - 100% CLOSED.

B. PDV-1719, SJAE/SPE BYPASS VLV - 90% OPEN.

C. Condenser level at 9 inches for 2 out of 3 shells.

D. HV-1639A, PCP 'A' SUCT VLV - 100% OPEN.

Proposed Answer: **B**

Explanation (Optional): A (B, C) P102 START-STOP pb control and indication on 10C651A. As a permissive for a pump START, **the START ENABLE light must be illuminated**, identifying the following conditions are satisfied: Condenser **level greater than 6"** for 2/3 shells, HV-1639A (B, C) pump **suction valve 100% OPEN**, HV-1680A (B, C) **pump discharge valve 100% CLOSED**, and **SJAE/SPE Bypass valve PDV-1719 100% OPEN.** SJAE/SPE Condenser Bypass Valve (PDV-1719) maintains a differential pressure across the **SJAE/SPE condensers** to provide design flow through the tube side of the heat exchangers. **PDV-1719 must be 100% OPEN to enable START of the first primary condensate pump. This can only be identified by the presence of the START ENABLE status light(s) at the primary condensate pump control bezel(s).**

Plausibility Justification:

A: **Incorrect-**. The discharge valve 100% closed is a permissive for the START ENABLE of the PCP.

B: **Correct-** The SJAE/SPE bypass valve will be normally throttled to control the d/p and therefore flow across the SJAE condensers, however, for the first PCP going into service this valve **has to be 100% open** to provide the permissive input into the **START ENABLE** status light and also provide a flowpath for the PCP.

C: **Incorrect-** The condenser level has to be greater than 6 inches in 2 of the 3 shells is also a permissive for the START ENABLE status light. The student will have to realize that the level even though low is still above the Low Low setpoint.

D: **Incorrect-**. The suction valve 100% open is a permissive for the START ENABLE of the PCP.

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

CONDENSATE SYSTEM OPERATION

Proposed References to be provided to applicants during examination: Pump controls in stem

Learning Objective: Concerning the primary condensate pumps: Select the four conditions that, when satisfied, illuminate the Start Enable light at the pump control bezel. Concerning the SJAE/SPE bypass valve, explain the functions of the BYPASS ON and BYPASS OPEN pushbuttons used to control the SJAE/SPE bypass valve

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	223001 K2.10	
Importance Rating	2.7	

K/A Statement: Knowledge of electrical power supplies to the following: Drywell chillers: Plant-Specific

Question: RO #28

Given:

- The plant is operating at 100% power.
- The AK111, BK111, and DK111 Turbine Building Chilled Water Units are in-service.
- Drywell parameters are reading normal.

Then:

- A loss of power transient occurs.
- Drywell pressure and temperature start to rise.
- 'A' Primary Condensate Pump trips.
- 'A' Secondary Condensate Pump trips.
- Only one Turbine Building Chiller remains running.

Which of the following loss of power caused this transient?

A. 10A101, 4.16KV Switchgear

B. 10A120, 7.2KV Switchgear

C. 10A102, 4.16KV Switchgear

D. 10A110, 7.2KV Switchgear

Proposed Answer: **D**

Explanation (Optional): K/A is matched because TBCW units supply Drywell Cooling. Power supplies (Turbine Building Chilled Water compressors): **1AK111 - 10A110***; 1BK111 - 10A120*; 1CK111 - 10A101; **1DK111 - 10A110***. * 7.2KV to a step-down transformer to 4.16KV for the AK111 and DK111 chillers. Along with powering the AK111 and DK111, the 10A110 switchgear also powers the 'A' PCP and 'A' SCP. (See attached).

Plausibility Justification:

A: **Incorrect-**. The 10A101 is the power supply to the CK111. Since the Chillers are 4.16KV power the student might select 4.16KV switchgear even though they should realize the trip of the PCP and SCP which are 7.2 KV powered.

B: **Incorrect-** The 10A120 powers the BK111 along with the 'B' PCP and 'B' SCP. The BK111 would be the remaining chiller running.

C: **Incorrect-** The 10A102 does not power any of the chillers, however the student could decipher the BK111 and DK111 because of the 'B' channel status of 10A102 and the fact that it is at the right voltage to supply a Turbine Chilled Water chiller.

D: **Correct-**. The 10A110 supplies both the AK111 and DK111 with a 7.2KV transformer to 4.16KV to supply the chillers. The 'A' PCP and 'A' SCP also come off of the 10A110 7.2KV switchgear.

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

7.2 KV System

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	243000 K3.04	
	Importance Rating	2.9	

K/A Statement: Knowledge of the effect that a loss or malfunction of the FUEL HANDLING EQUIPMENT will have on following: Reactor manual control system: Plant-Specific

Question: RO #29

Given:

- The Mode Switch is in REFUEL.
- All Control Rods are full in.
- The Refuel Platform is over the core.
- The Main Fuel Hoist/Fuel Grapple is NOT loaded.

Then:

- The Refuel Bridge System Rod Out Relay (ROR) contact fails indicating all rods are NOT full in.

What is the effect, if any, on the Reactor Manual Control System (RMCS) due to this malfunction?

A. REACTOR CONTROL SYSTEM INOP

B. ROD SELECTION BLOCK

C. No effect

D. ACTIVITY CONTROLS DISAGREE

Proposed Answer: **B**

Explanation (Optional): The **"one rod out" interlock** required by Tech Spec 3.9.1 **does not actually produce a rod block.** Instead, the interlock **is enforced by logic within RMCS that prevents the selection of a second control rod for movement with any other rod not fully inserted while in the**

refuel mode. The main fuel hoist motion will be inhibited and the Fuel Hoist Interlock light will be illuminated when the following conditions exist. The **Rod Out Relay (ROR) contact is opened** indicating **at least one control rod is withdrawn in the reactor**. AND The **refueling platform is over the reactor vessel**. i.e. (RS1 contact is open). **Refueling Rod Blocks:** Fuel Grapple Loaded AND Refuel Platform Over Reactor Cavity or Refuel Platform Over Reactor Cavity AND Reactor Mode Switch in STARTUP (see attached). Activity Control Cards compare control rod selection data, plant status, and RWM data to determine if control rod movement is allowed or should be blocked. They receive input from the Fuel bundle loaded on service platform, Hoist loaded or grapple down or loaded, Refueling platform over core. The Reactor Controls System INOP, this informs the operator that the RDCS (Rod Drive Control System) has shutdown due to a fault (Acknowledge and Command signals disagree). No control rod motion is possible. The ROR relay does not produce an INOP condition.

Plausibility Justification:

A: **Incorrect-**. The Reactor Controls System INOP, this informs the operator that the RDCS (Rod Drive Control System) has shutdown due to a fault (Acknowledge and Command signals disagree). No control rod motion is possible. The ROR relay does not produce an INOP condition. The RDCS is still available.

B: **Correct-** With the ROR contact failed open and indicating to the RMCS system that all rods are NOT full in; a ROD SELECT BLOCK will be enforced.

C: **Incorrect-** The main fuel hoist interlock will be in, however the RMCS is affected through the ROD SELECT BLOCK (one-rod-out interlock).

D: **Incorrect-**. With inputs from the Refuel System, the Activity Control Cards will process the information and will either allow or block rod motion, but only if they disagree. In this scenario there is no mention of a malfunction of the signals going to the RMCS Activity Control Cards.

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

Refuel Platform Operation

HC.OP-ST.KE-0001(Q)
Refuel Interlock Functional Test

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, explain the interrelationships between the Reactor Manual Control System and the following: Refueling System

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	271000 K4.01	
	Importance Rating	2.9	

K/A Statement: Knowledge of OFFGAS SYSTEM design feature(s) and/or interlocks which provide for the following: Dilution of hydrogen gas concentration

Question: RO #30

Which ONE the following statements correctly explains the consequences if the Off Gas Recombiner catalyst were to deteriorate or become wet.

The design rate of (dilution) recombination of hydrogen and oxygen will ____________.

A. remain the same, however there will be moisture carryover to the adsorber beds.

B. rise, because of the rise in temperature in the Recombiner vessel.

C. lower, because the diatomic oxygen will not come in contact with the platinum.

D. remain the same, however Recombiner vessel temperature will increase.

Proposed Answer: **C**

Explanation (Optional): The Off Gas Recombiners cause free H2 and O2 to recombine to water vapor. This will control the Hydrogen concentration in the off gas train. The recombiners utilize a homogeneous **platinum** and **palladium** mixture as a catalyst. As the O2 comes into contact with the platinum it causes the diatomic O2 to separate making it easier for it to combine with the H2 and form water. The palladium acts to raise surface area and thereby help control the recombination reaction. This is actually a form of combustion and gives off heat. The recombiner temperature is expected to rise 135°F per 1% H2. **If water coats the platinum, splitting of the diatomic O2 cannot take place and recombiner efficiency lowers.** When the recombiner is suspected of being wet (high moisture content), then the recombiner will be purged with nitrogen IAW radwaste operating procedures.

Plausibility Justification:

A: **Incorrect-**. If there is excessive moisture in the offgas flow, this could make it to the adsorber beds downstream of the feedgas recombiner. The guard bed for the adsorber beds would alarm on hi d/p to warn of excessive moisture on the adsorber beds. However, due to the moisture content, splitting of the diatomic O2 cannot take place and recombiner efficiency lowers.

B: **Incorrect-** The recombiner temperature is expected to rise 135°F per 1% H2. However, due to the moisture content, splitting of the diatomic O2 cannot take place and recombiner efficiency lowers.

C: **Correct-** If water coats the platinum, splitting of the diatomic O2 cannot take place and recombiner efficiency lowers.

D: **Incorrect-**. The recombiner temperature is expected to rise 135°F per 1% H2. If water coats the platinum, splitting of the diatomic O2 cannot take place and recombiner efficiency lowers.

Technical Reference(s): HC.RW-OP.HA-0001(R) (Attach if not previously provided)

Gaseous Radwaste Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain/identify the effect of moisture in the process gas stream on the following components: Recombiner

Question Source: Bank # 33172

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	230000 K5.04	
	Importance Rating	2.5	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE : Evaporative cooling

Question: RO #31

Given:

- A LOCA has occurred in the Drywell.
- All plant equipment functioned as designed.
- The applicable EOPs have been entered.
- "A" RHR is placed in Torus Cooling and Spray IAW HC.OP-AB.ZZ-0001, Transient Plant Conditions (Hard Card).

Which of the following describes the status of Torus parameters?

Suppression Pool temperature and pressure are lowered by ________________.

A. evaporative cooling, convective cooling, and heat rejection to the RHR Heat Exchanger.

B. evaporative cooling and heat rejection to the RHR Heat Exchanger ONLY.

C. convective cooling and heat rejection to the RHR Heat Exchanger ONLY.

D. heat rejection to the RHR Heat Exchanger ONLY.

Proposed Answer: **A**

Explanation (Optional): **Suppression Pool Cooling-** RHR loops A and/or B can be aligned to support Suppression Pool cooling. Pump suction is received from the Suppression Pool through F004A(B). Pump discharge is directed **through the RHR HX** via the shell side inlet HV-F047A(B) and HV-F024A(B) test return to the Suppression Pool.

Suppression Chamber Spray- RHR loops A and/or B can be aligned to support Suppression Pool chamber spray. Pump suction is from Suppression Pool through HV-F004A(B) through RHR pump A(B). Discharge is directed **through RHR HX** and to **Suppression Pool air space** via HV-F027A(B).

Plausibility Justification:

A: **Correct-**. Torus Spray and Torus cooling flowpaths are aligned simultaneously (see attached) so evaporative cooling and convective cooling of spray flow and heat rejection to the RHR Hx will occur simultaneously.

B: **Incorrect-** Convective cooling will be a significant cooling mechanism along with Evaporative cooling and the flow through the RHR heat exchanger (see attached bases).

C: **Incorrect-** Evaporative cooling will contribute to the reduction of pressure and temperature of the suppression chamber.

D: **Incorrect-**. With both flowpaths (Spray and Cooling) lined up through the RHR heat exchanger, the RHR heat exchanger will act as a heat sink, however, the spraying of the suppression chamber air space will provide both evaporative and convective cooling mechanism which will reduce pressure (condenses) and temperature.

Technical Reference(s): HC.OP-EO.ZZ-0102-BASES (Attach if not previously provided)

HC.OP-AB.ZZ-0001 Att. 2

Transient Plant Conditions

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing a given Abnormal Operating Procedure.

Given any step of the emergency operating procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations

Question Source: Bank # 32645

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(14)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000 K6.01	
	Importance Rating	3.1	

K/A Statement: Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION : A.C. electrical distribution

Question: RO #32

How will the Nuclear Boiler Instrumentation system be affected by the loss of the 1BD481?

A. ECCS Rosemount Trip Units will NOT provide initiation/permissive signals and RPV level indications will fail upscale.

B. ECCS Rosemount Trip Units will still provide initiation/permissive signals; however, RPV level indications will fail downscale.

C. ECCS Rosemount Trip Units will NOT provide initiation/permissive signals and RPV level indications will fail downscale.

D. ECCS Rosemount Trip Units will still provide initiation/permissive signals; however, RPV level indications will fail upscale.

Proposed Answer: **C**

Explanation (Optional): AD-DD481 Inverters supply **120 VAC power** to the instruments utilized by ECCS. The 1(A-D)D481 Inverters power the ECCS Analog Trip Units. **Loss of 1BD481 inverter causes the loss** of DIV II "B" Channel ECCS/RCIC Auto Trip Units and Start Relays - in general, **Process Signal Transmitter failures affecting initiation signals, Min. Flow Valves, pressure permissives,** etc. Channels "B" and "F" affected along with loss of instrument power affecting various systems. **Instrument indications will fail low** (see attached).

Plausibility Justification:

A: **Incorrect-**. The ECCS transmitters will lose power and therefore the ability to provide initiation/permissive signals for ECCS systems. The loss of the inverter will cause the RPV level indications to fail downscale.

B: **Incorrect-** The ECCS transmitters will lose power and therefore the ability to provide initiation/permissive signals for ECCS systems.

C: **Correct-** The student will have to determine if the 481 inverter provides power to the ECCS transmitters and from a control room awareness of indications they will have to determine that the loss of the 481 inverter will cause the level indications to fail downscale.

D: **Incorrect-**. The ECCS transmitters will lose power and therefore the ability to provide initiation/permissive signals for ECCS systems. The loss of the inverter will cause the RPV level indications to fail downscale.

Technical Reference(s): HC.OP-AB.ZZ-0136(Q) (Attach if not previously provided)

Loss of 120VAC Inverter

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a system that interrelates with the Nuclear Boiler Instrumentation System, evaluate the effects on that system due to loss of or malfunctions with the Nuclear Boiler Instrumentation System and/or associated components.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001 A1.13	
	Importance Rating	3.1	

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION SYSTEM controls including: Recirculation loop temperatures: Plant-Specific

Question: RO #33

Given:

- The plant is in OPCON 3.
- NO Reactor Recirculation Pumps are running.
- RWCU is in-service.
- RPV pressure is at 985 psig.
- "A" Recirculation Loop Temperature is at 499°F.
- "B" Recirculation Loop Temperature is at 483°F.
- RPV Bottom Head Drain Coolant Temperature is at 391°F.

(Assume the readings are taken within 15 minutes prior to attempting a pump start)

Which one of the following describes whether starting the "A" and/or "B" Reactor Recirculation Pumps is permitted?

	START "A" RECIRC	START "B" RECIRC
A.	YES	NO
B.	YES	YES
C.	NO	YES
D.	NO	NO

Proposed Answer: **D**

Explanation (Optional): To start an idle Reactor Recirculation Pump the following have to be satisfied:
Within 15 minutes prior to starting pump, **VERIFY** temperature differential between Reactor coolant within idle loop AND coolant in pressure vessel is **≤ 50°F**.
Within 15 minutes prior to starting pump, **VERIFY** temperature differential between Reactor coolant within dome AND bottom head drain is **≤ 145°F.**
RPV Pressure is 985 psig (1000 psia) using steam tables the corresponding temperature is **544.6°F** for Rx Pressure Vessel Steam Space Coolant Saturation Temperature. **D/T for Idle loop to steam space temp** is "A" Pump= **45.6°F** and "B" Pump= **61.6°F**. **Limit is ≤50°F**. "B" Pump is **UNSAT** for start. "A" Pump is **SAT**.
D/T Steam dome (steam space temp) to bottom head is 544.6°F - 391°F = **153.6°F**. **Limit is ≤ 145°F**. **Therefore, neither pump may be started**. "B" Pump is outside D/T limit for start. "A" Pump is within limits in loop to RPV temp, however outside limits for steam dome to bottom head. **Both D/T temp criteria have to be met.**

Plausibility Justification:

A: **Incorrect-**. "B" Pump is outside D/T limit for start. "A" Pump is within limits in loop to RPV temp, however outside limits for steam dome to bottom head. **Both D/T temp criteria have to be met.**

B: **Incorrect-** "B" Pump is outside D/T limit for start. "A" Pump is within limits in loop to RPV temp, however outside limits for steam dome to bottom head.

C: **Incorrect-** "B" Pump is outside D/T limit for start. "A" Pump is within limits in loop to RPV temp, however outside limits for steam dome to bottom head.

D: **Correct-**. See above calculations and explanation.

Technical Reference(s): HC.OP-SO.BB-0002(Q) (Attach if not previously provided)

Reactor Recirculation Operations

HC.OP-AB.RPV-0003(Q)
Recirc System/Power Oscillations

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with the Recirculation System/Power Oscillations.

Question Source: Bank # 68133

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

2021 NRC Written Examination

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	201003	A2.05	
	Importance Rating	4.1		

K/A Statement: Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reactor Scram

Question: RO #34

Given:

- A reactor startup is in progress.
- Reactor pressure is at 900 psig.

When:

- A reactor scram occurs.
- The scram inlet valves for two control rods fail to open.

(1) Which of the following describe the effect of this failure?
(2) Abnormal/emergency operating procedures are entered due to ___________.

A. (1) The control rods insert and their blue scram lights on the full core display illuminate.
(2) the full core reactor scram.

B. (1) The control rods insert and their blue scram lights on the full core display do NOT illuminate.
(2) the full core reactor scram.

C. (1) The control rods fail to insert and their blue scram lights on the full core display do NOT illuminate.
(2) the ATWS.

D. (1) The control rods fails to insert and their blue scram lights on the full core display illuminate.
(2) the ATWS.

Proposed Answer: **B**

Explanation (Optional): Reactor pressure will insert the rod at 900 psig. The blue lights are actuated by both scram inlet and outlet valves picking up there limit switches. The control rods are scram time tested at >800 psig and the scram inlet and outlet blue lights on the full core display are observed (see attached surveillance). Pressure from the scram accumulators cannot drive the rods due to the Scram Inlet valves being closed. However, the Scram Outlet valves opening cause **a significant DP** between the reactor and scram discharge volume (vented to atmosphere) to allow **the rods to scram**. Since the two control rods will scram, there will be a full core reactor scram. With no ATWS condition (Reactor shutdown under all conditions without Boron), the crew will enter EOP-101 or AB.ZZ-0000 for the successful reactor scram.

Plausibility Justification:

A: **Incorrect-**. The scram inlet valves did not move, therefore the limit switches for the blue lights on the full core display for the two control rods will NOT illuminate.

B: **Correct-** The scram inlet valves did not move, therefore the limit switches for the blue lights on the full core display for the two control rods will **NOT illuminate**. Due to the large DP across the CRDM the two control rods will fully insert and the reactor will be shutdown under all conditions without Boron. This is a full core reactor scram which would require entry into AB.ZZ-0000 Reactor Scram or EOP-101 RPV Control. There is no ATWS condition.

C: **Incorrect-** Due to the large DP across the CRDM the two control rods will fully insert and the reactor will be shutdown under all conditions without Boron. This is a full core reactor scram which would require entry into AB.ZZ-0000 Reactor Scram or EOP-101 RPV Control. **There is no ATWS condition.**

D: **Incorrect-**. The scram inlet valves did not move, therefore the limit switches for the blue lights on the full core display for the two control rods will NOT illuminate. This is a full core reactor scram which would require entry into AB.ZZ-0000 Reactor Scram or EOP-101 RPV Control. **There is no ATWS condition.**

Technical Reference(s): HC.RE.ST-BF-0001 (Attach if not previously provided)

Control Rod Scram Time

HC.OP-AB.ZZ-0000
Reactor Scram

Proposed References to be provided to applicants during examination: none

Learning Objective: Given P&ID M-47-1, select the various riser isolation valves, scram pilot valve assemblies, scram valves, directional flow control valves, accumulators and instrumentation assemblies.

Question Source: Bank #

Modified Bank # 30889 (Added part (2) to match the K/A)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	219000	A3.01	
	Importance Rating	3.3		

K/A Statement: Ability to monitor automatic operations of the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE including: Valve operation.

Question: RO #35

Given:

- The plant is operating at 89% power.
- "B" RHR Pump is running in Suppression Pool Cooling mode.

Then:

- A complete loss of offsite power occurs.
- All Emergency Diesel Generators have automatically started and aligned to their respective busses

Which one of the following describes the response of the BC-HV-F024B, "B" RHR Test Return Valve?

A. Receives a close signal immediately after the bus is reenergized.

B. Remains open until AUTO CLOSE OVERRIDE PB is pressed.

C. Remains open until CLOSE PB is pressed.

D. Receives a close signal 5 seconds after the bus is reenergized.

Proposed Answer: **C**

Explanation (Optional): HV-F024A(B) will automatically CLOSE in response to a manual or automatic LPCI initiation signal in the respective loop logic. To reopen F024A(B) following the Auto closure, the following permissive/ operator actions are required: A LPCI initiation signal is present (in the respective loop logic) and LPCI injection valve for the respective loop (HV-F017A(B) is 100% CLOSED. Depressing the **AUTO CL OVRD pushbutton** on 10C650 will then enable the valve(s) to be operated as desired. The stem of the question has **no reference to a LOCA condition** and therefore the **F024B will not isolate and can be CLOSED by the normal CLOSE PB.** The "B" RHR pump will start immediately after the bus is energized by the "B" EDG.

Plausibility Justification:

A: **Incorrect-**. With a LPCI initiation the F024B would isolate immediately after the bus was energized, however the LPCI initiation signal is not present only a LOP, therefore the F024B will NOT isolate.

B: **Incorrect-** With a LPCI initiation signal present and the F017B, injection valve closed, the F024B can be operated with the AUTO CLOSE OVERRIDE PB pressed, however there is no LPCI signal present.

C: **Correct-** With no LPCI initiation signal present and the fact that the valve is an MOV the valve will fail as is and when power is restored to the bus, the valve can be operated with the normal CLOSE PB.

D: **Incorrect-**. With no LPCI initiation signal the F024B will NOT isolate. If the LPCI initiation signal was in then the valve would close immediately after power was restored to the bus. The student could decipher that there is a time delay due to the pump having to start first then the valve opening. There is a 5 second time delay on the "C" and "D" RHR pumps starting off of normal power.

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

RHR System Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System or its components by evaluation of the controls/instrumentation/alarms

Question Source: Bank # 36238

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006 A4.03	
	Importance Rating	3.0	

K/A Statement: Ability to monitor automatic operations of the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) including: Latched group indication

Question: RO #36

Given:

- A startup is in progress.
- The Rod Worth Minimizer (RWM) indicates STEP 04.
- All Rod Pull Listing – RWM STEP 04 rods are at their withdraw limit – position 08.

Which statement describes when the RWM will latch to STEP 05?

A. When a control rod in the next RWM Group (STEP 05) is selected.

B. When a control rod in the next RWM Group (STEP 05) is selected and withdrawn at least two notches.

C. The RWM will remain latched at STEP 04 since the Banked Position Withdrawal Sequence Group is the same.

D. When a control rod in the next RWM Group (STEP 05) is selected and withdrawn one notch.

Proposed Answer: **D**

Explanation (Optional): Each RWM STEP consists of 1 or more Control Rods that are to be moved within the specified notch limits. NOTE: RWM STEPs are listed as RWM GROUPS on the Rod Pull Listings. The RWM will allow any order of control rod movement within an RWM STEP, but RWM STEPs must be withdrawn in sequence. The RWM determines the Current STEP by a process called "Latching". Once the RWM "Latches" the Current STEP, it will follow control rod movements, and will change the Current STEP when a Step Boundary **is reached and crossed.** At a step boundary, **if a control rod is selected from the next STEP** such as during a startup or shutdown and **once the control rod is moved, the STEP associated with that control rod will be displayed.** Selection of a control rod which if moved could cause deviation from the sequence, will result in INSERT and WITHDRAW Blocks.

Plausibility Justification:

A: **Incorrect-**. Since the rods are the same in both groups, the RWM will not latch up to STEP 05 **until it sees a STEP 05 rod withdrawn** at least one notch.

B: **Incorrect-** Since the rods are the same in both groups, the RWM will not latch up to STEP 05 **until it sees a STEP-05 rod withdrawn** at least **one notch**.

C: **Incorrect-** The RWM STEPS are not the same as the BPWS Groups.

D: **Correct-**. The RWM will latch up to STEP 05 **when it sees a STEP 05 rod withdrawn** at least one notch since all of STEP 04 have been withdrawn to their limit,

Technical Reference(s): HC.OP-SO.SF-0003(Q) (Attach if not previously provided)

RWM Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions, summarize the criteria used by the RWM to select the rod group to be latched.

Question Source: Bank # 36040

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 2.4.9	
	Importance Rating	3.8	

K/A Statement: Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. RWCU.

Question: RO #37

Given:

- The reactor has been in COLD SHUTDOWN for two (2) days following power operations.
- Reactor vessel water level is +30 inches.
- Neither Reactor Recirculation Pump is available.

When:

- A loss of Shutdown Cooling occurs.
- The shutdown cooling suction valves have isolated and cannot be re-opened.
- HC.OP-AB.RPV-0009, Shutdown Cooling has been entered.

Which of the following operator actions will aid in preventing reactor vessel thermal stratification and provide alternate decay heat removal?

A. Place RWCU System in service, maximizing RACS to the NRHX and bypass the regenerative heat exchanger if necessary.

B. Line up the Condensate Transfer system through the Core Spray injection line to feed the vessel.

C. Start a second CRD pump and maximize CRD flow by opening the drive water pressure control valve and adjusting both flow control valves.

D. Raise reactor vessel water level ONLY until the HIGH REACTOR LEVEL annunciator is received.

Proposed Answer: **A**

Explanation (Optional): RWCU bottom head drain suction maximizes vessel **circulation to prevent/minimize thermal stratification (cold CRD water).** RWCU Heat Exchanger Bypass diverts a portion of, or all of the flow from going through the heat exchangers to the return line to the reactor. This helps **limit thermal stratification and cool down of the reactor vessel during periods of no recirculation flow.**

Plausibility Justification:

A: **Correct-**. Per subsequent action "F" of AB.RPV-0009 (see attached) with the Reactor Recirculation pumps not available along with the normal SDC flowpath due to the isolation valves failed closed, RWCU is an alternate decay heat removal system and can be lined up in a timely manner along with the reduction in vessel thermal stratification. The RWCU pumps take suction off the bottom of the vessel and return through the feedwater lines.

B: **Incorrect-** Per AB.RPV-0009, the Condensate Transfer system lined up through either Core Spray or RHR is also a viable alternate decay heat removal of the core. However, the suction temperature (CST) for the Condensate Transfer system is much cooler then the vessel bottom suction and therefore would contribute to thermal stratification.

C: **Incorrect-** The CRD pumps will provide increase flow through the core, however the temperature of the suction of the CRD system (Condensate/ Hotwell Reject) will be much cooler then the vessel bottom suction and therefore would contribute to thermal stratification.

D: **Incorrect-**. Per AB.RPV-0009, raising RPV level when there is no core circulation will provide natural circulation (cooling) and also help with reducing thermal stratification. However, the level needs to be raised to >= 80 inches and < 90 inches (see attached). The HIGH REACTOR LEVEL alarm is at +39 inches.

Technical Reference(s): HC.OP-AB.RPV-0009 (Attach if not previously provided)

Shutdown Cooling

HC.OP-AB.ZZ-0001
Transient Plant Conditions

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing Shutdown Cooling.

Question Source: Bank # 33592

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201002 K3.01	
	Importance Rating	3.4	

K/A Statement: Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: Ability to move control rods

Question: RO #38

Given:

- A reactor startup is in progress.
- Reactor power is currently on Range 3 of the Intermediate Range Monitoring system.
- Control rod 30-31 is the next control rod to be moved.
- Control rod 30-31 is currently at notch position 12.

Following the selection of control rod 30-31:

- The "ACTIVITY CONTROLS DISAGREE" light on the Rod Select Module is illuminated.

Based on the above conditions, which one of the following describes the ability to move control rod 30-31?

Using the Reactor Manual Control System, control rod 30-31 can be _______________.

A. neither inserted nor withdrawn

B. inserted ONLY

C. withdrawn ONLY

D. inserted or withdrawn

Proposed Answer: **A**

Explanation (Optional): When each transmitter card has coded the selected control rod ID, each transmitter card will illuminate one half of the Rod Select Matrix pushbutton. If a transmitter card is faulty, it is possible to have only one half of two Rod Select Matrix pushbuttons illuminated. More importantly, if a transmitter card is faulty, a comparator trip is generated in the Analyzer section of the RMCS. **An ACTIVITY CONTROLS DISAGREE light will be illuminated and control rod motion will not be allowed.**

Plausibility Justification:

A: **Correct-**. neither inserted nor withdrawn. Activities Controls Disagree prevents rod movement in either direction.

B: **Incorrect-** Insert error prevents rod insertion. Would be correct if Withdraw error only.

C: **Incorrect-** Withdrawn only. Would be correct if Insert error only

D: **Incorrect-** Activities Controls Disagree prevents rod movement in either direction.

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

RMCS Operation

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and a drawing of the controls, instrumentation and/or alarms located in the Control Room, assess the status of the Reactor Manual Control System.

Question Source: Bank # 66301

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EK1.01	
	Importance Rating	4.1	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE Drywell integrity: Plant-Specific

Question: RO #39

The Drywell Spray Initiation Limit (DWSIL) is defined to be the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below the ____________________.

A. drywell-below-suppression pool differential pressure capability or the high drywell pressure scram setpoint.

B. drywell-below-suppression pool differential pressure capability or the suppression chamber spray initiation pressure.

C. suppression pool below reactor building differential pressure capability or the high drywell pressure scram setpoint.

D. suppression pool below reactor building differential pressure capability or the suppression chamber spray initiation pressure.

Proposed Answer: **A**

Explanation (Optional): The purpose of EOP-102 is to maintain primary containment integrity, and protect equipment in the primary containment. Changes in drywell temperature can directly effect changes in **primary containment pressure**. Prior to spraying the drywell, the drywell temperature must be below curve DWT-P (see attached), which represents the Drywell Spray Initiation Limit (DWSIL). The DWSIL is the highest drywell temperature at which initiation of drywell sprays will not result in an **evaporative cooling pressure drop to below the high drywell pressure scram setpoint.** The final pressure following evaporative cooling is limited to the scram setpoint to ensure that the operator has time to terminate sprays before convective cooling reduces pressure below 0 psig. **This prevents developing and/or exceeding the negative design pressure of the primary containment (-3 PSID drywell below suppression pool D/P) and ensures the integrity of the primary containment.** Suppression chamber spray is the initial mitigation strategy employed by EOP-102 in preference to drywell spray as it does not affect electrical components in the drywell and it can be used prior to **reaching 9.5 psig**, which is the Suppression Chamber Spray Initiation Pressure (SCSIP). **See attached EOP-102 Drywell Pressure Leg.**

Plausibility Justification:

A: **Correct-**. The DWSIL is the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below **the high drywell pressure scram** setpoint. This prevents developing and/or exceeding the negative design pressure of the primary containment **(-3 PSID drywell below suppression pool D/P)** and **ensures the integrity of the primary containment.**

B: **Incorrect-** Even though Suppression Chamber Spray will be the initial mitigation strategy, The DWSIL is the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below the **the high drywell pressure scram** setpoint.

C: **Incorrect- Suppression chamber sprays** precludes air from the Reactor Building being drawn if the suppression chamber was allowed to go negative. However, **Drywell Sprays** will preclude a pressure drop to below the primary containment **D/P of -3 PSID between drywell below suppression pool.**

D: **Incorrect-**. Even though Suppression Chamber Spray will be the initial mitigation strategy, Drywell Sprays will preclude a pressure drop to below the primary containment **D/P of -3 PSID between drywell below suppression pool.**

Technical Reference(s): HC.OP-EO.ZZ-0102BASES (Attach if not previously provided)

EOP-102 BASES

Proposed References to be provided to applicants during examination: none

Learning Objective: Define the term "Drywell Spray Initiation Limit"

Question Source: Bank # 110262

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EK1.03	
	Importance Rating	4.2	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Boron effects on reactor power (SBLC)

Question: RO #40

Given:

- A failure to scram has occurred.
- The crew is taking actions per HC.OP-EO.ZZ-0101A, ATWS RPV Control.

Current plant conditions:

- RPV pressure is being maintained 800-1000 psig with SRVs.
- RPV level is being maintained -100 to -50 inches with reactor feedpumps.
- 50% of the SLC Tank contents have been injected into the RPV.
- Rods are being inserted manually.
- 3 rods at 48 will NOT move.

Which of the following statements correctly describe the current plant status?

A. The reactor is shutdown and cooldown may now commence.

B. The reactor is NOT shutdown but cooldown is permitted because SLC is injecting.

C. The reactor will NOT be shutdown until the Cold Shutdown Boron Weight has been injected.

D. SLC may be secured if pressure is maintained within current pressure band.

Proposed Answer: **C**

Explanation (OptionalInjection of the Cold Shutdown Boron Weight (CSBW) of boron into the RPV also provides adequate assurance that the reactor is and will remain shutdown. The CSBW is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. CSBW has been injected into the core when there is <1100 gallons left in the SLC tank. (See attached EOP-101A BASES).

Plausibility Justification:

A: **Incorrect-**. Cooldown is not permitted until Cold Shutdown Boron Weight (CSBW) is injected or only one rod not at 00. 50% of the tank injected into the RPV would equate to more than **2000 gallons** remaining in the SLC tank. If any amount of boron less than the CSBW has been injected into the RPV, the core reactivity response from cooldown in a partially borated core is unpredictable and subsequent EPG steps may not prescribe the correct actions for such conditions if criticality were to occur.

B: **Incorrect-** CSBW must be injected, which is **<1100 gallons** remaining in the SLC tank. 50% of the tank injected into the RPV would equate to more than **2000 gallons** remaining in the SLC tank. The cooldown is permitted when CSBW is injected or if it will remain shutdown without boron. Due to 3 rods full out it will take the injection of CSBW for the reactor to remain shutdown.

C: **Correct-** CSBW must be injected, which is **<1100 gallons** remaining in the SLC tank because more than one rod (3 rods stuck at 48 not shutdown under all conditions) will remain full out. Tank level is normally between 4880 gal (Hi ALARM) and 4640 gal (Lo ALARM). 50% of the tank injected into the RPV would equate to more than **2000 gallons** remaining in the SLC tank. If any amount of boron less than the CSBW has been injected into the RPV, the core reactivity response from cooldown in a partially borated core is unpredictable and subsequent EPG steps may not prescribe the correct actions for such conditions if criticality were to occur.

D: **Incorrect-**. SBLC is not permitted to be secured until CSBW has been injected or the reactor will remain SD under all conditions without boron. Due to 3 rods full out it will take the injection of CSBW for the reactor to remain shutdown.

Technical Reference(s): HC.OP-EO.ZZ-0101ABASES (Attach if not previously provided)

ATWS RPV Control

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, explain the basis for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step.

Question Source: Bank # 35685

Modified Bank # (Note changes or attach parent)

New

Question History: NRC2016

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EK1.03	
	Importance Rating	3.8	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Heat capacity

Question: RO #41

Which ONE (1) of the following is the bases for the Suppression Pool level at which the primary containment pressure allowable limits could be exceeded and steam may not be adequately condensed?

A. HPCI exhaust line becomes uncovered.

B. Vent header drain lines become uncovered.

C. Downcomers become uncovered.

D. Suppression Pool Technical Specification minimum water level value.

Proposed Answer: **C**

Explanation (Optional): See attached HC.OP-EO.ZZ-0102 BASES

Plausibility Justification:

A: **Incorrect-**. If suppression pool level cannot be maintained **above 26 inches**, and adequate core cooling is assured the operator is directed to secure HPCI. Operation of the HPCI turbine with its exhaust **unsubmerged will tend to directly pressurize the suppression chamber.** Action is already taken **at the 38.5 inch level based on downcomers becoming uncovered** and loosing suppression capabilities of the Suppression pool.

B: **Incorrect-** The threshold of **55 inches** was selected as there is a 1¼ inch drain pipe attached to the low point of each of the eight vent pipes located in the torus. These drain pipes open into the torus at an **indicated level of 50 IN**; this level is between the low level LCO and the level at which the downcomers become uncovered. It is prudent to take the anticipatory actions to shutdown the reactor prior to the uncovering of these drain pipes.

C: **Correct-** Suppression pool water level **must be maintained above the elevation of the downcomer vent openings to ensure that steam discharged from the drywell into the suppression pool following a primary system break will be adequately condensed.** If suppression pool water level cannot be maintained above the specified minimum value, **steam may not be adequately condensed and primary containment pressure could exceed allowable limits.**

D: **Incorrect-**. When suppression pool level lowers to below the Technical Specification lower limit, EOP-102 provides direction to use ECCS and safety-related service water systems and alignments not normally used to maintain suppression pool water level in general plant procedures.

Technical Reference(s): HC.OP-EO.ZZ-0102-BASES (Attach if not previously provided)

EOP-102 BASES

Proposed References to be provided to applicants during examination: none

Learning Objective: 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.

Question Source: Bank # 2019 NRC Exam (#15)

Modified Bank # (Note changes or attach parent)

New

Question History: 2019 NRC Exam

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 EK2.06	
	Importance Rating	3.4	

K/A Statement: Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Process liquid radiation monitoring system

Question: RO #42

Given:

- A discharge of the Radwaste Floor Drain Sample Tank is in progress to the Delaware River.

When:

- The Liquid Radwaste Discharge Isolation Valve (HV-5377A) to the Cooling Tower Blowdown line automatically closes (isolates).
- The RM-11 is in alarm.

(Assume NO operator action)

Which condition below would cause this termination (isolation) when the listed setpoint was reached?

(1) Liquid Radwaste Effluent HIGH radiation
(2) Cooling Tower Blowdown dilution flow LOW flow
(3) Liquid Radwaste Effluent sample flow rate HIGH
(4) Cooling Tower Blowdown RMS HIGH radiation
(5) Liquid Radwaste Effluent HIGH discharge flow

A. (1) and (3) ONLY

B. (2) and (5) ONLY

C. (2), (3) and (4) ONLY

D. (1), (2) and (5) ONLY

Proposed Answer: **D**

Explanation (Optional): Waste discharge from the liquid radwaste system shall be sampled before discharge, shall be monitored during discharge, and **shall be automatically terminated when the instantaneous radioactivity concentration would reach 10CFR20 limits** for an unrestricted area after dilution. IAW HC.OP-AR.SP-0001 RM-11 alarm response, isolation of HV-5377A&B is due to any one of the following:
- High radiation (HIGH LED on 0SP-RI-4861)
- High Disch Flow (setpoint determined by Liquid Effluent Permit)
- Low Dilution Flow (setpoint determined by Liquid Effluent Permit)
- Low Sample Flow (0HBFIS-4861)
- Monitor Failure

Plausibility Justification:

A: **Incorrect-. Low sample flow rate** would be an isolation setpoint. The high sample flow rate would be sufficient for the RMS to accurately measure the discharge radiation levels.

B: **Incorrect-** The effluent high radiation which is upstream of the cooling tower blowdown RMS will isolate the discharge before it can reach the downstream RMS. The cooling tower blowdown RMS is NOT an isolation signal to the HV-5377A&B.

C: **Incorrect-** The high sample flow rate would be sufficient for the RMS to accurately measure the discharge radiation levels. The effluent high radiation which is upstream of the cooling tower blowdown RMS will isolate the discharge before it can reach the cooling tower blowdown RMS. The cooling tower blowdown RMS is NOT an isolation signal to the HV-5377A&B.

D: **Correct-**. With low dilution flow and high effluent discharge flow, the RMS will not be able to accurately sample the discharge and also dilute the discharge sufficiently to reduce the radiation levels of the discharge and therefore reach the setpoint of high radiation of the discharge. Any one of these parameter/setpoints will terminate (isolate) the discharge.

Technical Reference(s): HC.OP-AR.SP-0001(Q) (Attach if not previously provided)

RM-11 Alarm Response

Proposed References to be provided to applicants during examination: none

Learning Objective: Summarize/identify the conditions that will cause an Automatic Isolation of the Radwaste Release Line (HV-5377A&B).

Question Source: Bank # 120363

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AK2.02	
	Importance Rating	4.1	

K/A Statement: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: Emergency generators.

Question: RO #43

Given:

- The plant is in a normal AC electrical line up.
- The 10A403 bus trips due to a BUS DIFFERENTIAL OVERCURRENT.

Which of the following describes how this affects the "C" Emergency Diesel Generator and its output breaker?

A. The diesel can be manually started and the output breaker must be manually closed.

B. The diesel is locked out and the output breaker is locked out.

C. The diesel will automatically start and the output breaker must be manually closed.

D. The diesel will automatically start and the output breaker will automatically close.

Proposed Answer: **B**

Explanation (Optional): The EDG lockout circuitry is comprised of **regular**, backup, and test lockouts. When energized, the regular lockout relay will initiate the following actions: **Trip and lockout the Diesel engine. Trip and lockout the Generator breaker**. Enable Generator breaker failure protection. Regular lockout relay (86R) actuation results from the following signals: Generator differential overcurrent, **Bus differential overcurrent**, Engine overspeed, and Low lube oil pressure Emergency stop PB. Loss of the 10A403 1E Switchgear abnormal HC.OP-AB.ZZ-0172 will be entered along with other various abnormal procedures for the affected systems (see attached).

Plausibility Justification:

A: **Incorrect-**. Without the 86R reset (will not reset with the BUS Differential condition still in), the EDG and its output breaker will NOT be able to start and close either Automatically or manually. The student could decipher that the EDG is not affected only the 10A403 switchgear (bus).

B: **Correct-** With the 86R energized the EDG will be locked out along with the output breaker. The 10A403 bus will be de-energized and the appropriate affected systems abnormal procedures will be entered.

C: **Incorrect-** The EDG will be locked out from the 86R energized due to the BUS DIFFERENTIAL. The student could decipher that the EDG is not affected only the 10A403 switchgear (bus).

D: **Incorrect-**. Without the 86R reset (will not reset with the BUS Differential condition still in), the EDG and its output breaker will NOT be able to start and close Automatically. The student could decipher that the EDG is not affected only the 10A403 switchgear (bus).

Technical Reference(s): HC.OP-SO.PB-0001(Q) 4.16 KV (Attach if not previously provided)

HC.OP-SO.KJ-0001(Q) EDG

HC.OP-AB.ZZ-0172(Q) 10A403

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, summarize/identify the response of the 4.16KV busses to each of the following conditions:
Bus Differential Overcurrent
Given plant conditions, determine the automatic actions which result from the following:
Diesel Generator regular or backup lockout energized.

Question Source: Bank # 33973

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AK2.03	
	Importance Rating	2.5	

K/A Statement: Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Motors.

Question: RO #44

Given:

- The plant is operating at 100% power.

When :

- OHA A2-A5 FIRE PROT PANEL 10C671 is received.
- The Fire Computer screen shows a fire in room 4303.
- Control board walk down reveals that 'D' RHR pump has spuriously started.
- NO other control room OHA alarms have been received.
- NO other equipment has spuriously started.

- The operators have secured the 'D' RHR pump.

Which one of the following action(s) is (are) required for this condition IAW HC.OP-AB.FIRE-0001, FIRE- Spurious Operations?

A. Place the 'D' RHR pump breaker control switch in pull-to-lock AND lockout the 'D' EDG by pressing both emergency stop push buttons.

B. Place the 'D' RHR pump breaker control switch in pull-to-lock

C. Place the 'D' Core Spray and 'D' RHR pump breaker control switches in pull-to-lock.

D. Place the 'D' Core Spray and 'D' RHR pump breaker control switches in pull-to-lock AND lockout the 'D' EDG by pressing both emergency stop push buttons.

Proposed Answer: **B**

Explanation (Optional): See attached HC.OP-AB.FIRE-0001 Condition E and Attachment 2. Spurious Activation of D Channel equipment.

Plausibility Justification:

A: **Incorrect-**. Once the pump is secured to prevent it from restarting because of the fire damage the pump breaker has to be taken to PTL, the procedure would require that the associated EDG locked out only if the 'D' Core Spray pump spuriously started.

B: **Correct-** With the spurious start attributed to the fire damage the AB.FIRE-0001 Condition E has the operators put the 'D' RHR pump breaker into PTL.

C: **Incorrect-** IAW AB-Fire-0001 Condition E the required actions only pertain to the equipment that spuriously started. Even though the 'D' Core Spray pump is on the same channel (same room affected attachment 2) the Core Spray pump did NOT start, therefore the pump breaker does not need to be in PTL.

D: **Incorrect-**. IAW AB-Fire-0001 Condition E the required actions only pertain to the equipment that spuriously started. The Core Spray pump did NOT start; therefore, the pump breaker does not need to be in PTL. In addition, the 'D' EDG will NOT have to be locked out (see condition E).

Technical Reference(s): HC.OP-AB.FIRE-0001(Q) (Attach if not previously provided)

FIRE- Spurious Operation

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with the Fire - Spurious Operations.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295004 AK3.02	
Importance Rating	2.9	

K/A Statement: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Ground isolation/fault determination.

Question: RO #45

Which of the following correctly describes the indication of a direct negative short to ground on one of the 125VDC class 1E power supplies (10D410)?

A. A negative ground current will be only indicated on panel 1AD417.

B. Both white lights on panel 10D410 will be brighter than normal.

C. A negative ground current will be indicated on Control Room panel 10C650D.

D. Both white lights on panel 10D410 will be dimmer than normal.

Proposed Answer: **C**

Explanation (Optional): Switchgear ground detection lights: Normally both **lights are dim**. If a ground exists, **one light will dim and the other will be brighter**. Brightness is determined by magnitude of ground. LOCAL/REMOTE ground detection ammeter selector switch. Located on the associated distribution panel. **In LOCAL**, indication of a ground (positive or negative) **on the respective switchgear/distribution panel will be provided on the distribution panel DC ground detection ammeter**. **In REMOTE**, indication of a ground (positive or negative) **on the respective switchgear/distribution panel will be provided on control room panel 10C650D.**

Plausibility Justification:

A: **Incorrect-**. In REMOTE, indication of a ground (positive or negative) on the respective switchgear/distribution panel will be provided on control room panel 10C650D.

B: **Incorrect-** One light will be dim and the other will be brighter on a ground (positive or negative).

C: **Correct-** LOCAL/REMOTE ground detection ammeter selector switch. Located on the associated distribution panel. In REMOTE, indication of a ground (positive or negative) on the respective switchgear/distribution panel will be provided on control room panel 10C650D.

D: **Incorrect-**. Normally both lights are dim. If a ground exists, one light will dim and the other will be brighter.

Technical Reference(s): HC.OP-AB.ZZ-0147(Q) (Attach if not previously provided)

D.C. System Grounds

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing, DC System Grounds, Abnormal Operating Procedure.
Given a set of plant conditions evaluate those conditions and determine if a D.C. ground exists.

Question Source: Bank # **X**

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295001	AK3.03	
	Importance Rating	2.8		

K/A Statement: Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Idle loop flow

Question: RO #46

Given:

- The plant is operating at 100% power.

When:

- The “B” Reactor Recirculation Pump trips due to a Variable Frequency Drive (VFD) fault.

Immediately following the transient, the plant stabilizes with the following parameters:

- Reactor Power is at 50% rated.
- "B" Reactor Recirculation Pump is tripped.
- "A" Reactor Recirculation Pump speed is at 45%.
- Jet Pump Loop “B” Flow (FI-R611B-B21) is at 4 Mlbm/lhr.
- Jet Pump Loop "A" Flow (FI-R611A-B21) is at 38.5 Mlbm/hr.
- Jet Pump Flow Recorder (FR-R613-B21) is at 33.1 Mlbm/hr.

What is actual core flow (WT)?

A. 42.5 Mlbm/hr., because flow in the idle loop is negative (reverse) flow.

B. 35.1 Mlbm/hr., because flow in the idle loop is negative (reverse) flow.

C. 35.1 Mlbm/hr., because flow in the idle loop is positive (forward) flow.

D. 42.5 Mlbm/hr., because flow in the idle loop is positive (forward) flow.

Proposed Answer: **D**

Explanation (Optional): During single loop operations, the total core flow indication may be incorrect. This is due to the subtraction network in the core flow instrument subtracting out flow **from the secured loop** when this flow may be FORWARD FLOW not reverse flow. If operating loop recirc drive flow (loop flow) is <23,000 gpm **[speed is <48%], the flow in the idle loop is positive (forward) flow. The idle loop jet pump flow and the operating loop jet pump flow should be added to obtain actual core flow.** IF Operating Recirc Loop flow ≤ 23 Kgpm. **DETERMINE** Actual Core Flow by **ADDING** Idle Loop Jet Pump Flow AND Operating Loop Jet Pump Flow. **(FI-R611A-B21 and FI-R611B-B21)**

IF Operating Recirc Loop flow **>** 23 Kgpm. THEN: **DETERMINE** Actual Core Flow by **SUBTRACTING 85% of Idle Loop Jet Pump Flow FROM Operating loop Jet Pump Flow**
[FI-R611A(B)-B21 – (0.85 x FI-R611B(A)-B21)]
VERIFY proper function of the subtraction circuit by checking that calculated core flow (step A6) is the same as Total Jet Pump Flow (FR-R13-B21 OR A190).

Plausibility Justification:

A: **Incorrect-**. If operating loop recirc drive flow (loop flow) is <23,000 gpm [speed is <48%], the flow in the idle loop is positive (forward) flow. The idle loop jet pump flow and the operating loop jet pump flow **should be added** to obtain actual core flow of **42.5 Mlbm/hr.**

B: **Incorrect-** IF Operating Recirc Loop flow **>** 23 Kgpm. THEN: **DETERMINE** Actual Core Flow by **SUBTRACTING 85% of Idle Loop Jet Pump Flow FROM Operating loop Jet Pump Flow** [FI-R611A (B)-B21 – (0.85 x FI-R611B (A)-B21)] **[38.5 – (.85) 4] = 35.1 Mlbm/hr.**
The operating loop recirc drive flow (loop flow) is <23,000 gpm [speed is <48%], the flow in the idle loop is positive (forward) flow. The idle loop jet pump flow and the operating loop jet pump flow **should be added** to obtain actual core flow of **42.5 Mlbm/hr.**

C: **Incorrect-** The idle loop jet pump flow and the operating loop jet pump flow should be added to obtain actual core flow.

D: **Correct-**. The operating loop recirc drive flow (loop flow) is <23,000 gpm **[speed is <48%],** the flow in the idle loop is positive (forward) flow. **The idle loop jet pump flow** and the operating loop jet pump flow **should be added** to obtain actual core flow of **42.5 Mlbm/hr.**

Technical Reference(s): HC.OP-AB.RPV-0003(Q) (Attach if not previously provided)

Recirc System/Power Oscillations

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the reasons for how plant/system parameters respond when implementing Recirculation System/Power Oscillations.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AK3.02	
	Importance Rating	3.6	

K/A Statement: Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Actions contained in abnormal operating procedure for voltage and grid disturbances.

Question: RO #47

Given:

- The Reactor is at 16% power.
- A plant startup is in progress.

When:

- An SMD (Solar Magnetic Disturbance) Alert of K7 occurs.
- ESOC Excess MVARS is in alarm.
- DC Neutral Ground Current is in alarm.
- HC.OP-AB.BOP-0004, Grid Disturbances, is entered.

Then:

- BX500 Main Power Transformer Oil Temperature exceeds the Max Peak Setpoint.

Which of the following action(s) is(are) required IAW HC.OP-AB.BOP-0004, Grid Disturbance?

A. LOCK the Mode Switch in SHUTDOWN ONLY due to exceeding Main Power Transformer Oil Temperature.

B. REDUCE Recirc. Pump speed to minimum, LOCK the Mode Switch in SHUTDOWN, and TRIP the Main Turbine due to Excess MVARS in alarm.

C. REDUCE Recirc. Pump speed to minimum and LOCK the Mode Switch in SHUTDOWN ONLY due to Excess MVARS in alarm.

D. TRIP the Main Turbine ONLY due to exceeding Main Power Transformer Oil Temperature.

Proposed Answer: **D**

Explanation (Optional): See attached HC.OP-AB.BOP-0004 Condition C

Plausibility Justification:

A: **Incorrect-**.With Reactor power <18% and the Max peak temperature setpoint reached IAW AB.BOP-0004 Condition C, Trip of the Main Turbine is the only action needed. <18% locking the mode switch in shutdown is NOT required at this power level. The reactor will still be at power.

B: **Incorrect-** With Reactor power >18% and the oil temperature at Max Peak Setpoint, then IAW AB.BOP-0004 Condition C run recirc to minimum, lock the M.S. in shutdown and then trip the main turbine.

C: **Incorrect-** With Reactor power >18% and the oil temperature at Max Peak Setpoint, then IAW AB.BOP-0004 Condition C run recirc to minimum, lock the M.S. in shutdown and then trip the main turbine. With Reactor power <18% and the Max peak temperature setpoint reached IAW AB.BOP-0004 Condition C, Trip of the Main Turbine is the only action needed.

D: **Correct-**. With Reactor power <18% and the Max peak temperature setpoint reached IAW AB.BOP-0004 Condition C, Trip of the Main Turbine is required removing the main generator from the grid and the main power transformers.

Technical Reference(s):	HC.OP-AB.BOP-0004(Q) Grid Disturbances	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with Grid Disturbances.

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

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EA1.03	
	Importance Rating	3.9	

K/A Statement: Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring

Question: RO #48

Given:

- The plant is operating at 70% power.
- Reactor Core Isolation Cooling (RCIC) is operating in the CST to CST mode of operation IAW HC.OP-IS.BD-0001(Q) - Reactor Core Isolation Cooling Pump - OP203 - Inservice Test.
- Suppression pool temperature is 89°F and rising.
- Suppression Pool cooling is in service.

HC.OP-EO.ZZ-0102, Primary Containment Control, will be entered ONLY if Suppression Pool Average Water Temperature reaches _____ and continues to rise as monitored on panel section (temperature recorder) _____________.

A. 95°F; 10C650E CAS (TR-4967 A1/B1- Suppression Chamber Atmospheric Temperature)

B. 105°F; 10C650C PAM (TR-3881 A1/B1- Suppression Pool Temperature)

C. 105°F; 10C650E CAS (TR-4967 A1/B1- Suppression Chamber Atmospheric Temperature)

D. 95°F; 10C650C PAM (TR-3881 A1/B1- Suppression Pool Temperature)

Proposed Answer: **B**

Explanation (Optional): EOPs are entered whenever an entry condition is satisfied, **unless the entry condition is the result of a planned maintenance activity or approved procedure.** The planned maintenance or approved procedure must evaluate the impact of NOT performing the EOP actions at the entry condition setpoint. If suppression pool temperature is exceeded during the performance of testing for two specific system testing procedures: HC.OP-IS.BJ-0001(Q) and **HC.OP-IS.BD-0001(Q)** and no other heat input to the suppression pool exists, **then entry into EOP-102 is not required. Specific guidance for EOP-102 entry at a suppression pool temperature of 105°F** under these conditions is provided in these procedures. **The specific TR recorders that the operators use to determine the Average Suppression Pool temperature are on the 10C650C PAM section** (see attached table) and this is also **IAW the DL-26 Attachment 3m** (see attached). Due to the variations in temperatures of the Suppression pool because of the large volume, Suppression Pool Average Water Temperature needs to be monitored.

Plausibility Justification:

A: **Incorrect-. 95°F** is the normal EOP-102 entry temperature; however IAW T.S, EOP Bases, and the In-service test procedure (IS.BD-0001), **105°F** and continuing to rise is the entry into EOP-102 with the given conditions. IAW the DL-26 Attachment 3m for Suppression Chamber Average Water temperature check, the **TR-3881 A1/B1 on 10C650C PAM section** is used.

B: **Correct- 105°F** and continuing to rise is the entry into EOP-102 with the given conditions. IAW the DL-26 Attachment 3m for Suppression Chamber Average Water temperature check, the **TR-3881 A1/B1 on 10C650C PAM section** is used.

C: **Incorrect-** IAW the DL-26 Attachment 3m for Suppression Chamber Average Water temperature check, the **TR-3881 A1/B1 on 10C650C PAM section** is used.

D: **Incorrect-. 95°F** is the normal EOP-102 entry temperature; however IAW T.S, EOP Bases, and the In-service test procedure (IS.BD-0001), **105°F** and continuing to rise is the entry into EOP-102 with the given conditions.

Technical Reference(s): HC.OP-EO.ZZ-0102BASES (Attach if not previously provided)

HC.OP-IS.BD-0001- RCIC

HC.OP-DL.ZZ-0026 Att. 3m

T.S. 3.6.2.1 Suppression Chamber

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions, recognize the five (5) entry conditions for the Primary Containment Control Emergency Operating Procedure IAW HC.OP-EO.ZZ-0102

Question Source: Bank #

Modified Bank # #13 on 2019 *(Modified to fit the K/A)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295023 AA1.07	
Importance Rating	3.6	

K/A Statement: Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Fuel pool cooling and cleanup system

Question: RO #49

Given:

- The plant is in a refueling outage performing fuel moves in the spent fuel pool.
- 'A' Fuel Pool Cooling (FPCC) pump is in service cooling the fuel pool.
- 'B' FPCC pump is in standby.

Then:

- A pipe break occurs, which results in a trip of the 'A' FPCC pump and a significant loss of fuel pool inventory.

The IMMEDIATE operator action IAW HC.OP-AB.COOL-0004(Q), Fuel Pool Cooling is to ______________.

A. evacuate the Refuel Floor and return the irradiated fuel assembly to the vessel or pool.

B. add water to the fuel pool from Condensate Transfer, Suppression Pool via RHR, Fire Water, or Service Water.

C. place 'B' FPCC pump in service, and verify actual fuel pool temperature remains bounded within projected heat-up curves.

D. check liner drains to locate the leakage path.

Proposed Answer: **A**

Explanation (Optional): See attached HC.OP-AB.COOL-0004, Fuel Pool Cooling for I.O.A and subsequent operator actions.

Plausibility Justification:

A: **Correct-**. Due to the concerns of ALARA (radiation exposure) of personnel on the refuel floor, the IMMEDIATE action is to evacuate those personnel and return any fuel back to its original position in either the vessel or pool for shielding purposes. The other actions are appropriate to help mitigate the issue; however they are not the IMMEDIATE concern.

B: **Incorrect-** Subsequent operator action for lowering fuel pool level (see attached). These are all sources that can be used to make up to the skimmer surge tank; however this is not the IMMEDIATE concern.

C: **Incorrect-** This is a subsequent operator action in HC.OP-AB.COOL-0004(Q) for loss of fuel pool heat removal capability. This action would certainly take place; however this is not the IMMEDIATE concern.

D: **Incorrect-**. This is a subsequent operator action in HC.OP-AB.COOL-0004(Q) for loss of fuel pool inventory. This is not the IMMEDIATE concern.

Technical Reference(s): HC.OP-AB.COOL-0004(Q) (Attach if not previously provided)

Fuel Pool Cooling

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, recall the Immediate Operator Actions for a given Abnormal Operating Procedure.

Question Source: Bank # #50 on NRC 2016

Modified Bank # (Note changes or attach parent)

New

Question History: NRC2016

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EA1.05	
	Importance Rating	3.7	

K/A Statement: Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: RCIC

Question: RO #50

Given:

- The plant was operating at 100% power.

When:

- A reactor scram occurred.

Ten minutes after the scram:

- An MSIV Isolation occurred.
- With the high reactor pressure condition, RCIC is placed in pressure control augmented by SRVs IAW HC.OP-AB.ZZ-0001, Transient Plant Conditions.
- With the RCIC flow controller in AUTO, the plant operator (PO) observes RCIC speed oscillations.

Which of the following explains the RCIC speed oscillations?

A. Swings in RPV pressure are occurring due to the methods being used for pressure control. This causes the RCIC speed to change as the controller maintains a constant flow.

B. In pressure control, the RCIC controller attempts to maintain a constant speed, but CANNOT respond fast enough to maintain speed as the RPV pressure changes.

C. Using RCIC for pressure control is inherently less stable than using it for level control due to the lower pressure in the CST compared to the RPV. The greater instability is seen as an increase in oscillations.

D. The comparatively small CST volume results in the RCIC suction and discharge points in the CST being close together, and at high flow the turbulence causes oscillations.

Proposed Answer: **A**

Explanation (Optional): MANUAL - In this mode the operator sets desired RCIC turbine speed. Placing RCIC flow control in MAN results in an open loop control with the flow controller output becoming a fixed speed demand. Although this will provide stable, constant turbine speed, an operator will have to maintain desired vessel injection flow rate.

AUTOMATIC - In the automatic mode **RCIC turbine speed is automatically adjusted to maintain desired RCIC pump discharge flow** established by the operator. So, as the steam pressure changes (SRV cycling) to the RCIC system, the speed has to change to maintain the discharge flow. The operators would observe this operation of the RCIC system with a high reactor pressure condition that was being controlled by RCIC along with the SRVs. The flow controller will normally be in AUTO.

Plausibility Justification:

A: **Correct-**. As the SRVs cycle the reactor pressure will change, this changes the steam pressure to operate the RCIC pump. **To maintain the constant flow, the speed must change. This would be a normal condition and observation by the plant operator while monitoring the RCIC system with the flow controller in AUTO.**

B: **Incorrect-** In **AUTO** the controller **maintains flow** and in manual it **maintains speed**. The candidate may reverse the methods of RCIC control.

C: **Incorrect-** The controller is equally stable in the pressure and level control modes. The candidate may believe that the operation of RCIC in other than its design function of injecting to the core is less stable.

D: **Incorrect-**. The CST has a relatively small volume compared to the Suppression Pool but it does not result in oscillations. The operator may accept that the smaller flow volume results in suction/discharge interaction.

Technical Reference(s): HC.OP-AB.ZZ-0001(Q) Transient Conditions (Attach if not previously provided)
HC.OP-SO.BD-0001(Q) RCIC

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a labeled diagram/drawing of the RCIC System controls/indication bezel:. Explain the effect of each control on the RCIC System. Summarize plant conditions or permissives required for the control switches to perform their intended function.

Question Source: Bank # 119950

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 AA2.04	
	Importance Rating	4.1	

K/A Statement: Ability to determine and/or interpret the following as they apply to SCRAM : Reactor pressure

Question: RO #51

Given:

- The plant is operating at 100% power.

Then:

- 2 of the 3 DEHC Steam Header Pressure transmitters (PT-1001A & PT-1001B) slowly drift UPSCALE.

With NO operator action, the plant will scram as reactor pressure __ (1) __ AND the Components/Systems available for reactor pressure control following the scram include ___ (2) ___.

A. (1) lowers
(2) Bypass Valves, RCIC

B. (1) lowers
(2) SRVs, HPCI

C. (1) rises
(2) SRVs, HPCI

D. (1) rises
(2) Bypass Valves, RCIC

Proposed Answer: **B**

Explanation (Optional): Two of the three DEHC pressure transmitters **drifting upscale** will cause the Turbine Control/Bypass valves **to open resulting in an Uncontrolled Lowering of RPV pressure**. With the Mode switch remaining in RUN and NO operator action, as reactor pressure **lowers to 756 psig**, the MSIVs will close, the reactor will scram (MSIVs 8% closed RPS setpoint), and the bypass valves will be unavailable for pressure control. SRVs, RCIC and HPCI will be used for pressure control in this situation.

Plausibility Justification:

A: **Incorrect-**. With the two DEHC pressure transmitters drifting UPSCALE, the high reactor pressure signal input to DEHC will cause the Turbine Control and Bypass Valves to open. This will cause an actual uncontrolled lowering of actual reactor pressure. With no operator action, the reactor pressure will reach the setpoint of 756# for the MSIVs to isolate and the reactor scrams. With the MSIVs isolated, the bypass valves will no longer control pressure.

B: **Correct-** With the two DEHC pressure transmitters drifting UPSCALE, the high reactor pressure signal input to DEHC will cause the Turbine Control and Bypass Valves to open. This will cause an actual uncontrolled lowering of actual reactor pressure. With no operator action, the reactor pressure will reach the setpoint of 756# for the MSIVs to isolate and the reactor scrams. Turbine Control and Bypass valves will no longer control reactor pressure. Reactor pressure will be controlled with SRVs, HPCI, and RCIC.

C: **Incorrect-** The student could interpret the transmitter failure as a high pressure condition, which will scram the reactor at 1037#. Since reactor pressure actually lowers and with no operator action, the reactor pressure will reach the setpoint of 756# for the MSIVs to isolate and the reactor scrams. Turbine Control and Bypass valves will no longer control reactor pressure. Reactor pressure will be controlled with SRVs, HPCI, and RCIC.

D: **Incorrect-**. The student could interpret the transmitter failure as a high pressure condition, which will scram the reactor at 1037#. Since reactor pressure actually lowers and with no operator action, the reactor pressure will reach the setpoint of 756# for the MSIVs to isolate and the reactor scrams. With the MSIVs isolated, the bypass valves will no longer control pressure.

Technical Reference(s): HC.OP-AB.RPV-0005(Q) (Attach if not previously provided)

Reactor Pressure

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions and access to control room references (EHC Logic simplified drawing), determine system response to the following:
Loss of pressure or speed signal inputs.

Question Source: Bank # 151506

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AA2.03	
	Importance Rating	3.1	

K/A Statement: Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Turbine valve position

Question: RO #52

Given:

- The plant is operating at 50% power.
- The Main Turbine First Stage Pressure is at 243 psig.

When:

- A main generator load reject occurs.
- The crew enters the applicable plant procedures.

Which of the following are the immediate responses of the Turbine Control Valves (TCVs), Intercept Valves (IVs) and the Reactor Protection System (RPS)?

A. The TCVs and IVs Fast Close.
RPS will trip.

B. The TCVs and IVs Fast Close.
RPS will NOT trip.

C. The TCVs and IVs Throttle Close.
RPS will trip.

D. The TCVs and IVs Throttle Close.
RPS will NOT trip.

Proposed Answer: **A**

Explanation (Optional): If a power-to-load unbalance (PLU) (also called a load reject) occurs: The load signal is immediately set to minimum (Digital EHC). **The control valve and intercept valve fast acting solenoids are actuated**. A direct **turbine trip is generated**. This is done to prevent the turbine overspeed condition that could result from a sudden loss of significant generator load. With power >30%, the reactor will scram, RPS will trip with first stage pressure >98.1 psig with a TCV fast closure.

Plausibility Justification:

A: **Correct-**. With a power-to-load unbalance (PLU) condition, the fast acting solenoids for BOTH the TCVs and IVs will actuate causing a fast closure of the turbine valves. Also, through the digital EHC system this will be a direct turbine trip. With power >30%, this will be a reactor scram. RPS will trip with first stage pressure >98.1 psig with the TCV fast closure.

B: **Incorrect-** With a power-to-load unbalance (PLU) condition, the fast acting solenoids for BOTH the TCVs and IVs will actuate causing a fast closure of the turbine valves. RPS WILL trip due to the TCV fast closure and power level/first stage pressure.

C: **Incorrect-** With a normal turbine runback, the turbine valves would throttle, however, the PLU will input a minimum value (0) into the DEHC causing a direct turbine trip with activation of the fast acting solenoids for the TCVs and IVs.

D: **Incorrect-**. With a normal turbine runback, the turbine valves would throttle, however, the PLU will input a minimum value (0) into the DEHC causing a direct turbine trip with activation of the fast acting solenoids for the TCVs and IVs. With the power level and first stage pressure, RPS WILL trip.

Technical Reference(s): HC.OP-AR.ZZ-0014/0020 (Attach if not previously provided)

HC.OP-BOP-0002 Main Turbine

HC.OP-SO.SB-0001 RPS

Proposed References to be provided to applicants during examination: none

Learning Objective: Regarding a power-to-load unbalance signal:
Determine when the power-to-load unbalance circuit is enabled/disabled.
Choose the parameters monitored to initiate the signal.
Explain why this signal is generated.

Question Source: Bank # 119986

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 AA2.01	
	Importance Rating	3.5	

K/A Statement: Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system pressure

Question: RO #53

Given:

- The plant is operating at 100% power.

When:

- A large leak on the Instrument Air header occurs.
- The Instrument Air header pressure is lowering at 10 psig/minute.

When is locking the Mode Switch in Shutdown required and why?

A. More than one control rod DRIFTS due to a Low Accumulator Pressure.

B. One control rod DRIFTS due to its Scram Inlet Valve opening.

C. More than one control rod DRIFTS due to their Scram Outlet Valves opening.

D. One control rod DRIFTS due to a Low Accumulator Pressure.

Proposed Answer: **C**

Explanation (Optional): The scram inlet and outlet valves for each control rod HCU are normally held closed by the scram air header pressure (70 psig) from the instrument air system. As the instrument air header lowers the scram air header will lower which will cause the scram inlets and outlets to fail open which will cause the control rods to start to drift. The low accumulator pressure condition is an indication of either a low nitrogen gas pressure on the accumulator (possible leak at the accumulator) without control rod movement or if the control rod is scrammed. The accumulator will discharge on a scrammed control rod to assist the control rod to the full in position. With the loss of air the Scram Inlet and Outlet valves will start to fail open allowing the control rod to drift. A rod can drift without a loss of air due to other control rod movement operations.

Plausibility Justification:

A: **Incorrect-**. IAW AB.IC-0001 multiple rods drifting (not scramming) requires the M.S. to be locked in Shutdown. However, the low accumulator pressure is an indication of either a local accumulator trouble or a scrammed rod.

B: **Incorrect-** With the scram air header depleting due to the instrument air header lowering, the Scram Inlet and Outlet valves will fail open which will allow the control rod to drift close. The I.O.A for AB.IC-0001 has multiple rods drifting not just one before locking the mode switch in Shutdown. A rod can drift without a loss of air due to other control rod movement operations.

C: **Correct-** With the scram air header depleting due to the instrument air header lowering, the Scram Inlet and Outlet valves will fail open which will allow the control rod to drift close. IAW AB.IC-0001 multiple rods drifting (not scramming) requires the M.S. to be locked in Shutdown.

D: **Incorrect-**. The low accumulator pressure is an indication of either a local accumulator trouble or a scrammed rod. IAW AB.IC-0001 multiple rods drifting (not scramming) requires the M.S. to be locked in Shutdown.

Technical Reference(s): HC.OP-AB.IC-0001(Q) (Attach if not previously provided)

Control Rod

HC.OP-AR.ZZ-0011 Rod Drift/Accumulator Alarms

Proposed References to be provided to applicants during examination: none

Learning Objective: Regarding HC.OP-AB.IC-0001(Q), Control Rod: From memory, state the immediate operator actions.

Question Source: Bank # 118766

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 AK2.02	
	Importance Rating	3.4	

K/A Statement: Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: Plant operations

Question: RO #54

Given:

- The plant is operating at 100% power.
- "B","C", and "D" Station Service Water Pumps are in-service.
- "A" Station Service Water Pump is in standby.

When:

- The "C" Station Service Water Pump trips on a low flow condition.
- The "A" Station Service Water Pump auto starts.
- HC.OP-AB.COOL-0001, Station Service Water is entered.

75 seconds after the start of the "A" Station Service Water Pump:

- The "A" Service Water loop flow has risen by approximately 2,000 GPM.

Based on this, the "A" Station Service Water Pump ____________.

A. flow is NOT responding as designed. Recommend restarting the "C" Station Service Water Pump.

B. flow is responding as designed. Ensure full pump flow in approximately 85 more seconds.

C. flow is NOT responding as designed. Recommend securing the pump to investigate the low flow condition.

D. flow is responding as designed. Ensure full pump flow in approximately 25 more seconds.

Proposed Answer: **B**

Explanation (Optional): Station Service Water Pump Start - **If in AUTO**, SSW Pump Discharge Valve Opens in the following sequence:
Forty seconds after open signal, valve opens to 4%, valve should be open to **4% by <u>43 seconds</u> (from initial start).**
Thirty seconds later the valve opens to 8%, valve should be open to **8% by <u>75 seconds</u> (from initial start)**
Thirty seconds later valve opens to 100%, valve should be open to **100% by <u>160 seconds</u> (from initial start)**
The Service Water Pumps are **rated for 16,500 gpm.** The Service Water discharge valves are 28" motor operated butterfly valves.

Plausibility Justification:

A: **Incorrect-**. With the discharge valve open to 8% (75 seconds) and with approximately 2000 gpm of flow added to the loop flow (a fraction of the 16,500 gpm), the pump and discharge valve operation would be responding properly. The cause of the "C" SSW pump would be known, so theoretically the crew could restart the "C" SSW pump, however there would be no procedural guidance in this situation.

B: **Correct-** With the discharge valve open to 8% (75 seconds) and with approximately 2000 gpm of flow added to the loop flow (a fraction of the 16,500 gpm), the pump and discharge valve operation would be responding properly and the operators would continue to monitor for the discharge valve to indicate full open with full flow in another 85 seconds.

C: **Incorrect-** With the discharge valve open to 8% (75 seconds) and with approximately 2000 gpm of flow added to the loop flow (a fraction of the 16,500 gpm), the pump and discharge valve operation would be responding properly. The recommendation to secure the pump would be correct, however the pump and discharge valve are responding correctly with the given conditions. This is not a low flow condition.

D: **Incorrect-**. With the discharge valve open to 8% (75 seconds), adding 25 seconds would be the start of the discharge valve opening to 100%. The valve still has to travel to full open at 160 seconds, so full flow has not yet been established.

Technical Reference(s): HC.OP-AB.COOL-0001(Q) (Attach if not previously provided)

Station Service Water

Proposed References to be provided to applicants during examination: none

Learning Objective: Recognize abnormal indications/alarms and/or procedural requirements for implementing Station Service Water.

Question Source: Bank # 110944

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AA1.06	
	Importance Rating	4.0	

K/A Statement: Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Reactor water level

Question: RO #55

Given:

- The plant was operating at 100% power.

When:

- A Control Room fire causes a Main Turbine Trip and MSIV closure.
- The Control Room has been abandoned.
- Control has been established IAW HC.OP-AB.HVAC-0002, Control Room Environment.
- HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room has been entered.

Current plant conditions:

- A plant cooldown is in progress.
- Preparations are being made to place "B" RHR in (SDC) Shutdown Cooling.
- Current Reactor Coolant temperature is at 350°F.
- RCIC is in-service and maintaining Reactor Water Level.
- Indicated RPV water level is at 30" on Wide Range level instrumentation.

As RPV pressure is lowered, which of the following actions will be required IAW HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room to continue the cooldown of the plant? **[Reference attached]**

A. Maintain greater than +80" indicated on Shutdown Range. Place "B" RHR in SDC when Reactor Coolant temperature is less than 324°F.

B. Maintain -38" to +54" indicated on Wide Range with RCIC. Place "B" RHR in SDC immediately.

C. Maintain greater than +80" indicated on Shutdown Range. Place "B" RHR in SDC immediately.

D. Maintain -38" to +54" indicated on Wide Range with RCIC. Place "B" RHR in SDC when Reactor Coolant temperature is less than 324°F.

Proposed Answer: **D**

Explanation (Optional): IO-0008 step 5.2.2 direction states; **maintain -38" to +54" Wide Range (see att. 10 for actual RPV level)** and RPV pressure 800-1000 psig, SDC interlocks are not cleared **until temperature is < 324°F** which corresponds with **80 psig (see att 6)** , the interlock will not close the valve once it is opened, **but will prevent initial valve opening until the pressure interlock is cleared (82 psig).** Either AB.HVAC-0002 or IO-0008 has the crew place RCIC in-service to maintain level and will also reduce pressure (plant cooldown). With HPCI or RCIC not maintaining level and no recirc pumps available, IO-0008 has the crew raising RPV level to +80" Shutdown Range (see att.10) to promote natural circulation to maintain cooling. The **Attachment 10** will give the operator a more accurate actual RPV level. **Attachment 6** will allow the student to determine what pressure the plant is at and the permissives for placing "B" RHR in SDC (< 82 psig).

Plausibility Justification:

A: **Incorrect-**. With RCIC in-service, IAW IO-0008 the RPV level band is **-38" to +54"**. IO-0008 does reference +80" for natural circulation if nothing else is available. RCIC is providing level and pressure control along with a cooldown of the plant.

B: **Incorrect-** With the reactor coolant temperature at 350°F (see att. 6) the RPV pressure is still greater than the SDC valves interlock of 82 psig, therefore “B” RHR cannot be placed in SDC until the RPV pressure is lowered below 82 psig which corresponds to 324°F (80 psig).

C: **Incorrect-** With RCIC in-service, IAW IO-0008 the RPV level band is **-38” to +54”**. IO-0008 does reference +80” for natural circulation if nothing else is available. “B” RHR cannot be placed in SDC until the RPV pressure is lowered below 82 psig which corresponds to 324°F (80 psig).

D: **Correct-**. With RCIC in-service, IAW IO-0008 the RPV level band is **-38” to +54”**. With the reactor coolant temperature at 350°F (see att. 6) the RPV pressure is still greater than the SDC valves **interlock of 82 psig**, therefore “B” RHR cannot be placed in SDC until the RPV pressure is lowered below 82 psig which corresponds to **324°F (80 psig).**

Technical Reference(s): HC.OP-AB.HVAC-0002 Control Room Environment (Attach if not previously provided)
HC.OP-IO.ZZ-0008 Shutdown from Outside the Control Room

Proposed References to be provided to applicants during examination: **HC.OP-IO.ZZ-0008 Att. 6/ Att. 10**

Learning Objective: Interpret charts, graphs and tables contained within the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM Integrated Operating Procedure to maintain plant operations within specified limits.

Question Source: Bank # 31125

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EK2.12	
	Importance Rating	4.5	

K/A Statement: Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Primary containment isolation system/NS4.

Question: RO #56

Given:

- The plant is operating at 100% power.
- Reactor Engineering is running a TIP trace.

When:

- A reactor scram occurs.
- The reactor operator reports that reactor water level reached -50".

Current plant conditions:

- Level has recovered and is being maintained between +12.5" to +54".
- The reactor engineer reports that the TIPs failed to retract.
- There is NO evidence of containment leakage.

IAW HC.OP-AB.CONT-0002, PRIMARY CONTAINMENT, what is the next action, if any?

A. No action required since TIPS does not receive an isolation signal until -129".

B. Fire the shear valve at the discretion of the SM/CRS.

C. Manually retract the TIPs and ensure the TIP valve closes.

D. No action required since reactor water level has been restored to above +12.5".

Proposed Answer: **C**

Explanation (Optional): TIP system response to a Nuclear Steam Supply Shutoff System containment isolation signal: **Low reactor vessel level (-38 inches, LEVEL 2),** or High drywell pressure (1.68 psig), or Actuation of the NSSSS Channel A manual isolation switch. All TIP detectors not in the "in-shield" position will automatically be withdrawn. All TIP ball valves will automatically close once their respective detectors have reached the "in-shield" position. See attached actions of HC.OP-AB.CONT-0002 Primary Containment and the NS4 isolations IAW HC.OP-SO.SM-0001, Isolation System Operations.

Plausibility Justification:

A: **Incorrect-**. TIPs RPV level isolation is at -38". On the scram the level reached -50", this would cause an NS4 isolation signal, and therefore TIPs would retract and isolate. The student would have to know what level the system isolates. Level 1 (-129") is also a primary containment level isolation setpoint (see attached SM-0001 table for the TIPs system).

B: **Incorrect-** Shear valves when fired, a chisel-type plunger (guillotine) is driven into the TIP guide tube with enough force to shear the drive/signal cable and seal the reactor end of the guide tube. This is action is necessary when the TIP cannot be manually isolated and is the source of the leak (no evidence of leakage) (See attached subsequent action F of AB.CONT-0002).

C: **Correct-** With the -50" initial level after the scram, the TIPs should have retracted and isolated. IAW the I.O.A of AB.CONT-0002 and subsequent action G, the TIPs need to be manually retracted which will isolate the system.

D: **Incorrect-**. The Level 2 (-38") will seal in for the NS4 system to allow the TIPs to isolate immediately. Even though level was recovered from -50" to a normal level band of +12.5" to 54" the isolation that did not occur has to be manually isolated.

Technical Reference(s): HC.OP-AB.CONT-0002 Primary Containment (Attach if not previously provided)
HC.OP-SO.SM-0001

Isolation Systems

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory explain the response of the TIP System following the receipt of an isolation signal from the Nuclear Steam Supply Shutoff System.

Question Source: Bank # 110478

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 AA2.02	
	Importance Rating	3.4	

K/A Statement: Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : RHR/shutdown cooling system flow

Question: RO #57

Given:

- A reactor cooldown is being performed.
- All valves have been aligned for placing 'B' RHR into Shutdown Cooling.

T= 0:

- The 'B' RHR pump is started.
- The operator opens BC-HV-F015B, RHR LOOP B RET TO RECIRC.
- BC-HV-F015B lights indicate the valve is stroking open.

T= 30 seconds:

- The Plant Operator reports that the 'B' RHR loop flow is indicating 1200 gpm.

Continuing to operate in this condition will cause ________________.

A. lowering of RPV level

B. overheating of the 'B' RHR pump

C. lowering of suppression pool level

D. tripping of the 'B' RHR pump on overcurrent

Proposed Answer: **A**

Explanation (Optional): The minimum flow valve is normally open when the RHR System is in standby. The valve will automatically close when the pump is running for >4 seconds AND RHR pump flow exceeds 1270 GPM. If system flow lowers below 1250 GPM for ten seconds or more (with the pump breaker closed) **the minimum flow valve will open. The minimum flow valve sends flow to the suppression pool.** Exiting the RHR HX, flow is returned to the RPV via return valve HV-F015A (B) and testable check HV-F050A (B). Flow enters the Rx Recirc Loop A (B) discharge piping where it reenters the vessel through the jet pumps.

Plausibility Justification:

A: **Correct-**. With 1200 gpm flow indication, the minimum flow valve (F007) will be open and therefore returning flow back to the suppression pool. This system alignment would drain the RPV to the suppression pool. The forced circulation is not making it through the core (establishing proper SDC flow) and AB.RPV-0009 would be entered for a loss of shutdown cooling.

B: **Incorrect-** With the minimum flow valve opened, the RHR pump will not overheat. The student might interpret the no flow as the condition for overheating the pump. If the pump is overheating it would have to be secured and AB.RPV-0009 would be entered for a loss of shutdown cooling.

C: **Incorrect-** The minimum flow valve (F007) will be open and therefore returning flow back to the suppression pool. This system alignment would drain the RPV to the suppression pool. The suppression pool level would be rising not lowering. With the suction from the suppression pool the RHR pump would have a reduced NPSH with a actual lowering suppression pool and therefore a potential securing of the pump. AB.RPV-0009 would be entered for a loss of shutdown cooling.

D: **Incorrect-**. With the amps rising the student might interpret a high current condition and therefore a trip of the "B" RHR pump at the breaker due to overcurrent. AB.RPV-0009 would be entered for a loss of shutdown cooling.

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

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the Residual Heat Removal System.

Question Source: Bank # 34143

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 2.4.18	
	Importance Rating	3.3	

K/A Statement: Emergency Procedures / Plan: Knowledge of the specific bases for EOPs. High Drywell Temperature

Question: RO #58

Which of the following describes the negative impact of Drywell temperature in excess of 340°F?

A. All RPV water level instrumentation is invalidated.

B. Emergency depressurization capabilities with ADS could become impaired.

C. The operation and effectiveness of drywell sprays will be adversely affected.

D. Containment venting will be required to get within the safe area of the Drywell Spray Initiation Limit Curve.

Proposed Answer: **B**

Explanation (Optional): EOP-102 Bases (see attached): If drywell temperature cannot be controlled by operation of all available drywell cooling, direction is provided to run back the recirculation pumps to minimum speed, initiate a manual scram. This is performed in anticipation of shutting down the reactor recirculation pumps as the motors are not qualified for continuous operation in a spray environment; drywell spray is required before both **the maximum temperature at which ADS is qualified (UFSAR Table 5.2-6) and the drywell design temperature (UFSAR Table 1.3-4) limits are reached at 340°F.**

Plausibility Justification:

A: **Incorrect-**. Inaccurate level indication may occur when drywell temperature exceeds the saturation temperature for the existing RPV pressure. But this is not the bases for the high drywell temperature of **340°F.**

B: **Correct-** 340°F is the maximum drywell temperature at which ADS is qualified to operate.

C: **Incorrect-** Sprays are more effective with higher temperatures due to the increased evaporative cooling.

D: **Incorrect-**. The DSIL Curve permits Spray for a wide range of pressures with Drywell temperature above 340°F.

Technical Reference(s): HC.OP-EO.ZZ-0102-BASES (Attach if not previously provided)

Primary Containment Control

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 34099

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295008 AK1.03	
Importance Rating	3.2	

K/A Statement: Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL : Feed flow/steam flow mismatch

Question: RO #59

Given:

- All 3 RFPT's are in manual speed control.
- RPV level is presently @ 35 inches.

Feedwater pump flows:

- 'A' 3.2 Mlbm/hr
- 'B' 3.6 Mlbm/hr
- 'C' 3.5 Mlbm/hr

Main Steam flows:

- 'A' 2.6 Mlbm/hr
- 'B' 2.5 Mlbm/hr
- 'C' 2.6 Mlbm/hr
- 'D' 2.4 Mlbm/hr

Based on these conditions, RFPT speed demand must be adjusted to FIRST prevent an RPV _____________ IAW HC.OP-AB.RPV-0004, Reactor Level Control.

A. high level alarm

B. low level reactor scram

C. high level main turbine trip

D. low level alarm

Proposed Answer: **A**

Explanation (Optional): Total Feed Flow is 10.3 Mlbm/hr. Total Steam flow is 10.1 Mlbm/hr. **This mismatch will result in a rising RPV water level.** RPV speed demand must lower to prevent the Level 7 alarm. The Level 7 high level alarm will occur if no action is taken. IAW HC.OP-RPV-0004, Reactor Level Control, the operators would have manual control controlling level between Level 4 and Level 7 (see attached I.O.A of AB.RPV-0004). These two level alarms preclude the RPS setpoint of +12.5" (Level 3 Reactor Scram) and +54" (Level 8) Main Turbine Trip.

Plausibility Justification:

A: **Correct-**. With the mismatch between Total Feed Flow and Total Steam Flow at .2 Mlbm/hr for feedwater flow, the RPV water level would start to rise. The operators have manual control of feedwater, therefore the speed demand on the RFPTs would be lowered to compensate for the RPV level rising. IAW AB.RPV-0004 the operators would maintain level between Level 4 (low level alarm) and Level 7 (high level alarm).

B: **Incorrect-** With the mismatch between Total Feed Flow and Total Steam Flow at .2 Mlbm/hr for feedwater flow, the RPV water level would start to rise. The RFPTs would be lowered to compensate for the RPV level rising. IAW AB.RPV-0004 the operators would maintain level between Level 4 (low level alarm) and Level 7 (high level alarm).

C: **Incorrect-** IAW AB.RPV-0004 the operators would maintain level between Level 4 (low level alarm) and Level 7 (high level alarm). The operators would prevent the level rising to a Main Turbine Trip setpoint.

D: **Incorrect-**. With the mismatch between Total Feed Flow and Total Steam Flow at .2 Mlbm/hr for feedwater flow, the RPV water level would start to rise. The RFPTs would be lowered to compensate for the RPV level rising.

Technical Reference(s): HC.OP-AB.RPV-0004 (Attach if not previously provided)

Reactor Level Control

Proposed References to be provided to applicants during examination: none

Learning Objective: 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 Feedwater Control System

Question Source: Bank # 35529

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022 AK2.04	
	Importance Rating	2.5	

K/A Statement: Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Reactor water level

Question: RO #60

Given:

- A plant startup is in progress IAW HC.OP-IO.ZZ-0003, Startup from Cold Shutdown to Rated Power.
- Reactor Power is on range 4 of the IRMs and slowly rising.
- Reactor Level is at + 36 inches and stable.
- Reactor Pressure is at 0 psig and stable.
- Reactor Temperature is at 180°F and stable.
- The plant is NOT at the point of adding heat (POAH).

When:

- The 'A' Control Rod Drive Pump (CRD) trips.
- The standby, 'B' CRD Pump trips on the attempted start by the Reactor Operator (RO).
- Control Rod movement has been suspended.

Reactor water level will _______ and the operators will have to ____________.

A. rise; increase the RWCU blowdown flowrate to the Main Condenser.

B. lower; increase the Feedwater flowrate to the vessel.

C. rise; reduce the Feedwater flowrate to the vessel.

D. lower; reduce the RWCU blowdown flowrate to the Main Condenser.

Proposed Answer: **D**

Explanation (Optional): RWCU blowdown operations is normally balanced to reject the makeup rate from CRD. Without the CRD pump running, RWCU is rejecting at approximately the same rate. **RPV level will lower**. The CRD pumps will provide a continuous make-up rate due to the low power and the plant being below the POAH. Once the plant reaches the POAH and starts to generate steam, the steam generation will be greater than the CRD pump makeup rate. At this point in IO-0003, the crew is to place the Feedwater system in-service feeding the vessel to provide the necessary make-up due to the steam generation.

Plausibility Justification:

A: **Incorrect-**. With the reactor recirc pumps in-service (initial heatup and pressurization of the RPV), the pumps would be adding heat to the vessel inventory, however the RWCU blowdown flowrate would exceed any heat expansion by the reactor recirc. pumps. Level will lower due to RWCU blowdown flowrate and the loss of the CRD system.

B: **Incorrect-** Level will lower due to RWCU blowdown flowrate and the loss of the CRD system. The plant is below the POAH, so the Feedwater system would not be feeding the vessel at this time. If the student does not recognize the significance of the plant being below the POAH, they might select the Feedwater system feeding the vessel.

C: **Incorrect-** With the reactor recirc pumps in-service (initial heatup and pressurization of the RPV), the pumps would be adding heat to the vessel inventory, however the RWCU blowdown flowrate would exceed any heat expansion by the reactor recirc. pumps. If the student does not recognize the significance of the plant being below the POAH, they might select the Feedwater system feeding the vessel.

D: **Correct-**. Level will lower due to RWCU blowdown flowrate and the loss of the CRD system. The plant is below the POAH, so the CRD system will provide vessel inventory and the RWCU blowdown flowrate will control the CRD make-up rate and therefore RPV level.

Technical Reference(s): HC.OP-IO.ZZ-0003 (Attach if not previously provided)

S/U from Cold S/D

HC.OP-SO.BG-0001 RWCU

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, determine why a method of reactor water level control must be available prior to placing the CRDH System in-service including the preferred method of level control.

Question Source: Bank # 36244

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295009 AA2.01	
	Importance Rating	4.2	

K/A Statement: Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water level

Question: RO #61

Given:

- Reactor Power is at 94% power.
- All three Reactor Feedpumps are in Auto.
    - ➢ Narrow Range "A" (PDT-N004A) = 36 inches.
    - ➢ Narrow Range "B" (PDT-N004B) = 35 inches.
    - ➢ Narrow Range "C" (PDT-N004C) = 34 inches

Then:

- A large leak develops in the "A" Steam flow detector pressure diaphragm causing a gross fail alarm on the instrument.
    - ➢ OHA B3-F1 –"DFCS ALARM/TRBL" alarm is in.
    - ➢ CRIDS display D5921- "DFCS Trouble" alarm is in.

In response to these conditions, the Digital Feed Control System will ___(1)___ and RPV level will ___(2)___.

A. (1) remain in three element
(2) will remain constant

B. (1) transfer to single element
(2) remain constant

C. (1) remain in three element
(2) will raise slowly due to the lower total steam flow input to the Master Controller

D. (1) transfer to single element
(2) rapidly rise due to feed flow steam flow mismatch response

Proposed Answer: **B**

Explanation (Optional HC.OP-SO.AE-0001 - 2.3.21. The Master Level Controller will automatically switch from single element to three element level control at > 31.4% total steam flow after a 1 minute time delay (assuming all feed flow and steam flow inputs are good) AND will switch from three element to single element level control instantaneously at < 27.8% total steam flow. See attached.
HC.OP-AR.ZZ-0007 - D5921 - DFCS Trouble - RFPTs may swap from 3-element control to single element control. On a loss of any Steam Flow signal, the control circuit **will transfer to single element control.** The operator can detect this failure by noting that the individual Steam Flow signal on the FI-R603 recorder has failed on panel 10C650C, and by the DFCS TROUBLE overhead alarm. Additionally, the Operator Display screen(s) will digitally display the failed detector(s). See attached.

Plausibility Justification:

A: **Incorrect-**. With the loss of the steam flow signal and with the given alarms, the Digital Feedwater System will transfer from 3-element to single element control with no change in reactor water level due to Narrow Range detectors matched to Master controller of the Digital Feedwater System..

B: **Correct-** Transfer to single element, there will be no level perturbation based on Narrow Range detectors matched to Master controller.

C: **Incorrect-** -. With the loss of the steam flow signal and with the given alarms, the Digital Feedwater System will transfer from 3-element to single element control

D: **Incorrect-**. There will be no level perturbation based on Narrow Range detectors matched to Master controller.

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

Feedwater System Operations

HC.OP-AR.ZZ-0007 (B3-F1, D5921)

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, describe the response of the FWLC system if a system transmitter were to fail.

Question Source: Bank # 124703

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	295034	EA1.03	
	Importance Rating	3.8		

K/A Statement: Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Secondary Containment Ventilation: Plant-Specific

Question: RO #62

Given:

- The plant is operating at 100% power.
- An I&C PCIS (Primary Containment Isolation System) surveillance is in progress.

When:

- The control room receives a High Reactor Building and Refuel Floor Radiation isolation signal.
- Reactor Building Ventilation Supply (RBVS) and Reactor Building Ventilation Exhaust (RBVE) fans trip.
- All automatic actions occur for secondary containment ventilation.
- The I&C surveillance was secured.
- The PCIS high radiation isolation signals were reset.

WHAT actions are required to restore the Reactor Building Ventilation Supply (RBVS) and Reactor Building Ventilation Exhaust (RBVE) fans?

The ____ (1) _____ breakers for both Reactor Building Supply and Exhaust fans must be manually closed, then the fans will be restarted from the ____ (2) _____.

A. (1) 1-E
(2) local controls (10C382)

B. (1) 1-E
(2) main control room (MCR)

C. (1) Non 1-E
(2) main control room (MCR)

D. (1) Non 1-E
(2) local controls (10C382)

Proposed Answer: **A**

Explanation (Optional): (K/A Statement) At Hope Creek the RBVS system is the secondary containment ventilation. The RBVS fan units are supplied power thru 1E and non-1E circuit breakers.If any of the following conditions occur, the 1E breaker will trip, causing the non-1E to trip on under voltage. (The Non-1E breaker opens/closes to stop/start the fan. The breakers are in series.). -38" Rx LVL, 1.68 psig Drywell Pressure, **Reactor building ventilation exhaust High radiation 1x10-3 mCi/cc, Refuel floor ventilation exhaust High radiation 2x10-3 mCi/cc.**
The Class 1-E breakers are located in the respective channel 1-E Unit Sub Station switchgears The **Non 1-E breaker** is the breaker actuated for routine equipment operation via normal STOP/START control switches at the **local panel 10C382.** To reclose the 1E breaker, **all initiating signals must be clear and PCIS reset**; then, the breaker must be manually reclosed at the 480V 1E unit substation. Indication of 1E breaker position is provided on 10C650E (Main Control Room).

Plausibility Justification:

A: **Correct-**. With the High radiation signals in, the RBVS/RBVE fans will trip (1E breaker then the Non-1E breaker). With PCIS reset, the 1E breakers will have to be closed manually (locally). Then, the fans will be started at the local panel 10C382 to place the RBVS in a normal line up.

B: **Incorrect-** With PCIS reset, the 1E breakers will have to be closed manually (locally). The RBVS system is operated locally at the 10C382 panel. Indication of 1E breaker position is provided on 10C650E (Main Control Room).

C: **Incorrect-** The Non-1E breaker opens/closes to stop/start the fan. The 1E breaker will trip on the high radiation signal from PCIS. The RBVS system is operated locally at the 10C382 panel.

D: **Incorrect-**. The Non-1E breaker opens/closes to stop/start the fan. The 1E breaker will trip on the high radiation signal from PCIS.

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

Isolation System Operation

HC.OP-SO.GR-0001(Q) RBVS

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions associated with the Reactor Building Ventilation Exhaust (RBVE) and Supply (RBVS) system: Summarize/identify the automatic trips of the electric supply.

Question Source: Bank # 115987

Modified Bank # (Original attached)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032 EA2.01	
	Importance Rating	3.8	

K/A Statement: Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : Area temperature

Question: RO #63

Given:

- The plant was operating at 50% power.

When:

- A fire was reported in the HPCI Pump room.

Current plant conditions:

- The smoke and heat has spread to the RCIC Pump room as a result of firefighting efforts.
- HPCI and RCIC have isolated due to high temperatures.
- Temperatures in both the HPCI Pump room and RCIC Pump room are at 275°F.
- Firefighting efforts have been hampered due to previously tagged fire suppression systems.

TABLE 1		
Area Description & Room Number	**Column 1 Max Normal Op Temp**	**Column 2 Max Safe Op Temp**
CRD Pump Room (4202)	115°F	140°F
HPCI (4111)	115°F	250°F
Core Spray Pump Rooms A(4118) & C(4116)	115°F	140°F
RHR Pump Rooms A(4113) & C(4114)	115°F	140°F
SACS A & C (4309)	115°F	140°F
RCIC Pump Room (4110)	115°F	250°F
Core Spray Pump Rooms B(4104) & D(4105)	115°F	140°F
RHR Pump Rooms B(4109) & D(4107)	115°F	140°F
SACS B & D (4307)	115°F	140°F
RWCU Pipe Chase (4402)	160°F	350°F

Which of the following actions is required IAW Emergency Operating Procedures? **[Reference attached]**

A. Manually scram the reactor and emergency depressurize.

B. Bypass High Room Temperature isolations for RCIC and restore to standby lineup.

C. Shutdown the reactor and commence a normal cooldown.

D. Runback reactor recirculation and manually scram the reactor.

Proposed Answer: **C**

Explanation (Optional): See attached flowchart of HC.OP-EO.ZZ-0103/4. Student will determine from Table 1 that the HPCI and RCIC Pump rooms are at the Max Safe Op level which will lead them to the fact that the reactor will have to be shutdown with a normal cooldown (IO.ZZ-0004). But due to the fact that there is NO reactor coolant discharge in progress, the student will have to determine where to go down the RB Leg of EOP-103 (attached). There is no requirement for a manual scram or emergency depressurization of the reactor.

Plausibility Justification:

A: **Incorrect-**. A **reactor coolant system is not discharging** into the Reactor Building, IAW EOP-103 step RB-15 (see attached). Emergency depressurization is not required.

B: **Incorrect-** The bypassing of the RCIC or HPCI high temperature isolations is driven by a Station Blackout condition IAW HC.OP-AB.ZZ-0135 Attachment 10. There is no SBO in progress, therefore no requirement to bypass the RCIC high temperature isolation and place it in standby. Due to the fact that the fire is not in the RCIC pump room, the student might choose to have RCIC available and therefore select the bypassing of the high temperature isolations.

C: **Correct-** A reactor coolant system is not discharging into the Reactor Building and the Max Safe Operating Limit in 2 areas (Table 1) has been exceeded. Therefore IAW RB-15, RB-21, and RB-22, the reactor is required to be shutdown with a normal cooldown (IO.ZZ-0004).

D: **Incorrect-**. A **reactor coolant system is not discharging** into the Reactor Building, IAW EOP-103 step RB-15 (see attached). A manual reactor scram is not required.

Technical Reference(s): HC.OP-EO.ZZ-0103/4 (Attach if not previously provided)

RX Bldg and RAD Release Control

Proposed References to be provided to applicants during examination: Table 1 of EOP-103 in stem of question and the Reactor Building Control Leg attached

Learning Objective: 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.

Question Source: Bank # 66819

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295020 2.1.31	
Importance Rating	4.6	

K/A Statement: Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. Inadvertent Containment Isolation.

Question: RO #64

Given:

- The plant was operating at 75% power.
- An I&C surveillance was in progress.

When:

- An erroneous loss of main condenser vacuum signal causes an inadvertent isolation of all Main Steam Isolation Valves (MSIVs).
- All operator actions for a reactor scram have been completed.

Current plant conditions:

- The loss of main condenser vacuum signal has been cleared.
- Main condenser vacuum indications are at normal vacuum levels.

Which of the following would be REQUIRED to reset the NSSSS (NS4) MSIV isolation logic?

A. The MSIV control switches must be in "Close".

B. The Turbine Stop Valves must be closed.

C. The Reactor Mode Switch must be in "Shutdown".

D. The Main Condenser Low Vacuum Bypass Switches must be in "Bypass".

Proposed Answer: **A**

Explanation (Optional): The MSIVs will isolate on a low vacuum signal of 21.5"Hg.Abs (see attached SM-0001). To reset the MSIV isolation logic, main condenser vacuum needs to be below the setpoint of 21.5"Hg.Abs and the MSIV control room switches have to be in the "CLOSE" position (see attached section of SM-0001).

Plausibility Justification:

A: **Correct-**. With main condenser vacuum below the isolation setpoint of 21.5 "HGA due to the erroneous loss of vacuum isolation signal cleared, to RESET the MSIV logic the MSIV valve switches on 10C651 will have to be in the "CLOSE" position.

B: **Incorrect-** With the scram actions completed, the main turbine will be tripped and therefore the TSVs will be closed. However, due to the fact that the loss of main condenser vacuum signal is cleared, the Main Condenser Low Vacuum Bypass Switches are NOT needed. For the Bypass Switches to complete the bypass of a 21.5"HGA vacuum condition, the TSVs have to be <90% open. If a bypass was needed, then the TSVs being closed would be required.

C: **Incorrect-** Reactor Mode Switch **not in run** bypasses the <8% closed RPS setpoint for MSIVs. The mode switch can be in any position except "Run", therefore the MSIV isolation logic does not have to see the Mode Switch in "Shutdown" specifically.

D: **Incorrect-**. Low main condenser vacuum of 21.5"HGA can be bypassed via 4 keylock switches at 10C609, 10C611 (NS4 Panels) when turbine stop valves are ≤ 90% OPEN (main turbine tripped). Since the erroneous loss of vacuum signal is cleared, there is no need for the bypass switches to be in "Bypass".

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

Isolation System Operation

Proposed References to be provided to applicants during examination: none

Learning Objective: 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.

Question Source: Bank # 33823

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	500000 EK2.03	
Importance Rating	3.3	

K/A Statement: Knowledge of the interrelations between HIGH CONTAINMENT HYDROGEN CONCENTRATIONS and the following: Containment Atmosphere Control System

Question: RO #65

Given:

- A large break LOCA has occurred in the Drywell
- Multiple Equipment failures have occurred.
- Drywell Pressure is 16 psig and rising.
- RPV level has lowered to below top of active fuel (TAF) and continues to lower.
- The H2/O2 Analyzers must be placed in service due to the potentially high Hydrogen and Oxygen concentrations in primary containment.

ALL sample locations monitored by the H2/O2 Analyzers are _____ (1) _______.

AND

Prior to opening the Containment Isolation Valves (CIVs) for the H2/O2 Analyzers, the containment isolation signal ______ (2) ______ to be overridden.

A. (1) The upper drywell and the torus ONLY
(2) is required

B. (1) The upper drywell and the torus ONLY
(2) is NOT required

C. (1) The upper drywell, lower drywell and the torus
(2) is NOT required

D. (1) The upper drywell, lower drywell and the torus
(2) is required

Proposed Answer: **D**

Explanation (Optional): Each package (H2O2 analyzer) takes samples from three different locations; **High - Drywell head region, Low - Drywell cylindrical region, Suppression Chamber Air Space.** Each of the 3 sample suction lines and the one return line is provided with 2 MOV for containment isolation. Containment Isolation Valves (16 Total) automatically close upon: High DRYWELL pressure (+1.68 psig), Reactor Water Level 2 (-38 inches), Reactor Building Vent Exhaust high-high Radiation (1 x 10-3 mci/cc). All 16 containment isolation valves associated with HOAS have identical control bezels: OPEN, CLSD, OVLD/PWR FAIL, and OVERRIDDEN. **The CIVs can be individually opened after the associated isolation override P.B. is depressed at (10C650E).** (See attached figures of the controls and the H2/O2 drawing)

Plausibility Justification:

A: **Incorrect-**. Each package (H2O2 analyzer) takes samples from three different locations; **High - Drywell head region**, **Low - Drywell cylindrical region, Suppression Chamber Air Space.** Due to the given conditions all the CIVs would isolate on either the Drywell pressure or RPV level. Therefore, they would all have to be overridden to bypass the automatic isolation.

B: **Incorrect-** Each package (H2O2 analyzer) takes samples from three different locations; **High - Drywell head region**, **Low - Drywell cylindrical region, Suppression Chamber Air Space.** The CIVs have the capabilities of being overridden on the containment isolation signals from drywell high pressure and RPV low level. The H2/O2 analyzers can be placed in-service to monitor the high Hydrogen and Oxygen concentrations in the primary containment.

C: **Incorrect-** Each package (H2O2 analyzer) takes samples from three different locations; **High - Drywell head region, Low - Drywell cylindrical region, Suppression Chamber Air Space.** ALL the CIVs for theses sample points would be able to be lined up for sampling.

D: **Correct-**. Each package (H2O2 analyzer) takes samples from three different locations; **High - Drywell head region, Low - Drywell cylindrical region, Suppression Chamber Air Space. The CIVs can be individually opened after the associated isolation override P.B. is depressed at (10C650E)**

Technical Reference(s): HC.OP-SO.GS-0002(Q) (Attach if not previously provided)

H2/O2 Analyzer Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Select the three parameters, including setpoints, which will automatically isolate the HOAS and predict the required operator action to: Reset the isolation signal and restore the HOAS to service. Manually override the isolation signal and restore the HOAS to service.

Question Source: Bank # 120390

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.1.7	
Importance Rating	4.4	

K/A Statement: Conduct of Operations: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Question: RO #66

Given:

- The plant is in Operational Condition 4, preparing for plant startup.
- 'B' RHR Loop is in Shutdown Cooling in accordance with HC.OP-SO.BC-0002, Decay Heat Removal Operations.

When:

- The Reactor Operator (RO) reports Total Core Flow has lowered significantly.
- RPV level is slowly rising and is currently at +85 inches.
- Reactor Head Vent temperature readings are also rising.

Which of the following is the cause of the given plant conditions?

A. RHR Pump 'B' RHR loop test return MOV valve BC-HV-F024B is open.

B. RHR Pump 'B' Min Flow valve BC-HV-F007B is open.

C. RHR Pump 'B' Suppression Pool Spray Header isolation valve BC-HV-F027B is open.

D. Reactor Recirculation Pump 'B' Discharge valve BB-HV-F031B is open.

Proposed Answer: **D**

Explanation (Optional): See attached Precautions and Limitations of HC.OP-SO.BC-0002, Decay Heat Removal Operations.

Plausibility Justification:

A: **Incorrect-**. Opening this valve while in shutdown cooling would cause a lowering of RPV level. BC-HV-F024A(B), RHR LOOP TEST RET MOV will drain the Reactor Vessel to the Suppression Pool if opened in Shutdown Cooling (Precaution 3.1.1).

B: **Incorrect-** Opening the F007 valve while in shutdown cooling would cause a lowering of RPV level. BC-HV-F007A(B), RHR PUMP A(B) MIN FLOW MOV will drain the Reactor Vessel to the Suppression Pool if opened in Shutdown Cooling, due to flow below the low-flow setpoint precluding automatic valve closure. To prevent this from occurring, the BC-HV-F007A (B) is CLOSED and tagged while in Shutdown Cooling (Precaution 3.1.3).

C: **Incorrect-** Opening this valve while in shutdown cooling would cause a lowering of RPV level. BC-HV-F027A (B), RHR LOOP A (B) SUPP POOL SPRAY HDR ISLN MOV will drain the Reactor Vessel to the Suppression Pool if opened while the associated RHR Pump is in Shutdown Cooling. To prevent this from occurring, the BC-HV-F027A (B) is CLOSED and tagged while in Shutdown Cooling (Precaution 3.1.2).

D: **Correct-**. By opening BB-HV-F031B while in shutdown cooling on the 'B' RHR loop, a core bypass is initiated, which causes head vent temperatures to increase. The cooled shutdown cooling flow is not returned to the vessel via the jet pumps (cause of total core flow lowering), but instead is sent through the 'B' recirc pump due to its discharge valve being open. This creates a bypass loop where hot reactor water is removed from the vessel, but the cooled water is not returned directly to the vessel, thereby causing heatup and swell (cause of rising level) (Limitations 3.2.4 and 3.2.5).

Technical Reference(s): HC.OP-SO.BC-0002 (Attach if not previously provided)

Decay Heat Removal Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions involving a Loss of Shutdown Cooling, summarize required actions to mitigate the condition.

Question Source: Bank # 30780

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2016

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.1.32	
Importance Rating	3.8	

K/A Statement: Conduct of Operations: Ability to explain and apply all system limits and precautions.

Question: RO #67

Given:

- A reactor cooldown is in progress IAW HC.OP-IO.ZZ-0004, Shutdown from Rated Power to Cold Shutdown.
- Both reactor recirculation pumps are in service.
- Preparations are being made to place the 'B' RHR system into shutdown cooling.

During the transition from normal Reactor Recirculation System operations to establishing Shutdown Cooling flow with the 'B' RHR pump,

Which of the following is the preferred Reactor Recirculation Pump lineup?

A. Both Reactor Recirculation Pumps are in service.

B. ONLY 'A' Reactor Recirculation Pump is in service.

C. Both Reactor Recirculation Pumps are secured.

D. ONLY 'B' Reactor Recirculation Pump is in service.

Proposed Answer: **B**

Explanation (Optional): See attached Precautions and Limitations of HC.OP-IO.ZZ-0004, Shutdown from Rated Power to Cold Shutdown along with the Caution for minimizes the time at which there is no forced flow through the core from either the Reactor Recirculation System or the RHR System. The Reactor Recirc Pump associated with the RHR Loop to be placed in Shutdown Cooling must be secured with its discharge valve shut. The discharge valve of any Reactor Recirculation Pump which is **NOT** in operation should remain closed throughout Shutdown Cooling operations.

Plausibility Justification:

A: **Incorrect-**. The 'B' Recirc pump must be secured prior to placing the 'B' RHR pump in service and the loop into SDC. The Reactor Recirc Pump associated with the RHR Loop to be placed in Shutdown Cooling must be secured with its discharge valve shut.

B: **Correct-** During transition from normal Reactor Recirculation System operations to establishment of Shutdown Cooling, only the AP201 Reactor Recirc Pump may be left in operation until the BP202 (only) RHR Pump is operating satisfactorily, and then only until the required B RHR Loop flow of approximately 10,000 gpm is achieved.(See attached IO-0004).

C: **Incorrect-** Minimizes the time at which there is no forced flow through the core from either the Reactor Recirculation System or the RHR System. The AP201 Reactor Recirc Pump may be left in operation until the BP202 (only) RHR Pump is operating satisfactorily.

D: **Incorrect-**. The Reactor Recirc Pump associated with the RHR Loop to be placed in Shutdown Cooling must be secured with its discharge valve shut.

Technical Reference(s): HC.OP-IO.ZZ-0004(Q) (Attach if not previously provided)

S/D from rated Power to Cold S/D

Proposed References to be provided to applicants during examination: none

Learning Objective: Analyze plant conditions and parameters to determine if plant operation is in accordance with the SHUTDOWN FROM RATED POWER TO COLD SHUTDOWN Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications.

Question Source: Bank # 36162

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.12	
	Importance Rating	3.7	

K/A Statement: Equipment Control: Knowledge of surveillance procedures.

Question: RO #68

Given:

- Preparation for plant Startup is in progress IAW HC.OP-IO.ZZ-0002, Preparation for Plant Startup.
- The CRD system has just been placed in service.

Once the CRD system is in service, the crew is required to exercise __________ to ensure proper rod motion to satisfy surveillance requirements for plant start up.

A. at normal pressure ONLY for those control rods which had maintenance performed on them

B. at elevated pressure ONLY for those control rods which had maintenance performed on them

C. all control rods at both elevated AND normal pressure

D. at elevated pressure ONLY for those control rods which historically have had operational problems

Proposed Answer: **C**

Explanation (Optional): With preparation for plant start up, the CRD system will be placed into a normal system lineup to allow the required control rod exercising surveillance prior to reactor criticality and to satisfy Tech Spec surveillance requirements. See attached sections of HC.OP-IO.ZZ-0002 and the surveillance requirements (purpose) of HC.OP-ST.BF-0001.
Plausibility Justification:

A: **Incorrect-**. There are requirements for retesting any control rods that have had maintenance performed on them; however, this requirement for operability pertains to every control rod.

B: **Incorrect-** To satisfy the requirements for prestart up to criticality, all control rods will have to be exercised at **both** normal and el elevated pressures IAW HC.OP-IO.ZZ-0002.

C: **Correct-** IAW HC.OP-IO.ZZ-0002 and Tech Spec requirements all control rods will be exercised to both normal and elevated pressures along with the surveillance required coupling check. The control rods are exercised IAW HC.OP-ST.BF-0001.

D: **Incorrect-**. To satisfy the requirements for prestart up to criticality, all control rods will have to be exercised.

Technical Reference(s): HC.OP-IO.ZZ-0002(Q) (Attach if not previously provided)

Preparation for Plant S/U

HC.OP-ST.BF-0001(Q)
Control Rod Drive Exercise

Proposed References to be provided to applicants during examination: none

Learning Objective: Analyze plant conditions and parameters to determine if plant operation is in accordance with the PREPARATION FOR PLANT STARTUP Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications.

Question Source: Bank # 33038

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek
Vendor: GE
Exam Date: 2021
Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.15	
	Importance Rating	3.9	

K/A Statement: Equipment Control: Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.

Question: RO #69

IAW OP-AA-108-101, CONTROL OF EQUIPMENT AND SYSTEM STATUS:

An Abnormal Component Position Sheet (ACPS) is required to be filled out if ____ (1) _____.
eSOMS (Work Clearance Module) is updated by the NCO ____(2)____ .

A. (1) aligning equipment outside of routine operations.
(2) at the end of each shift.

B. (1) a change in component position is performed per an approved maintenance activity.
(2) at the end of each shift.

C. (1) aligning equipment outside of routine operations.
(2) once daily.

D. (1) a change in component position is performed per an approved maintenance activity.
(2) once daily.

Proposed Answer: **A**

Explanation (Optional): Situations will occur when it is desired to reposition a component and no approved documentation exists. The ACPS (Abnormal Component Position Sheet) will be the approved configuration control method. There are some limitations, however to using an ACPS, for example an approved maintenance activity that controls the component position. At the end of each shift the NCO shall update the ACPS (See attached OP-AA-108-101).

Plausibility Justification:

A: **Correct-**. With no formal document to control the component configuration (position), an ACPS will be the controlling document. Due to the importance of configuration control from one shift to the next, the off going NCO will update the eSOMs (Work Clearance Module) and the ACPS will be filed away with no changes (See attached section on updating the ACPS).

B: **Incorrect-** An approved maintenance activity that controls component position is one of the limitations of using a ACPS document (See attached section on the ACPS Limitations).

C: **Incorrect-** Due to the importance of configuration control from one shift to the next, the off going NCO will update the eSOMs (Work Clearance Module) and the ACPS will be filed away with no changes (See attached section on updating the ACPS).

D: **Incorrect-**. An approved maintenance activity that controls component position is one of the limitations of using a ACPS document. Due to the importance of configuration control from one shift to the next, the off going NCO will update the eSOMs (Work Clearance Module).

Technical Reference(s): OP-AA-108-101 (Attach if not previously provided)

CONTROL OF EQUIPMENT AND SYSTEM STATUS

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory State the responsibilities of the following personnel with regard to Component Configuration Control:
All personnel
CRS
Duty Operator[NCO/NEO Building Watch]

Question Source: Bank # 111253

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.7	
Importance Rating	3.5	

K/A Statement: Radiation Control: Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Question: RO #70

Given:

- A RWCU system valve independent verification is being completed in the field.

When:

- The on-duty NCO discovers that two valves on the verification list are in the "A" RWCU pump room (High Radiation Area).
- The shift radiation protection technician and the independent verifier reviewed the Radiation Work Permit (RWP) survey for the "A" RWCU pump room.
- The general area dose rate at the valves is 110 mRem/hr.
- The job is estimated to take six minutes.

What is the estimated cumulative dose the verifier will receive and is the "Hands On" independent verification required IAW OP-AA-108-101-1002, Component Configuration Control?

A. 18 mRem; independent verification is required.

B. 11 mRem; independent verification is NOT required.

C. 11 mRem; independent verification is required.

D. 18 mRem; independent verification is NOT required.

Proposed Answer: **B**

Explanation (Optional): Dose rate calculation- **[6 minutes/ 60, then 0.1 hr x 110mRem/hr = 11 mRem]** Calculation of 110mRem/hr/ 6 minutes = 18.3 mRem, estimate **18 mRem**, if the candidate did not carry the units correctly. **IAW OP-AA-108-101-1002 Attachment 5** (see attached) General Rules for Verification, If significant cumulative radiation exposure **(10 mRem)** would be received by the person performing the Independent Verification or by persons assisting the performance of the Independent Verification. **"Hands On" Independent Verification is NOT required.**

Plausibility Justification:

A: **Incorrect-**. With a calculation of 110/6 (carrying the wrong units), the student would get 18 mRem. The student would have to know the General Rule of >10mRem for the "Hands On" independent verification NOT required.

B: **Correct-** With a calculation of 0.1 x 110, the student would get 11 mRem. This calculation is correct and the fact that it is >10mRem, the "Hands On" independent verification is NOT required.

C: **Incorrect-** The calculation of 11mRem accumulative dose is correct, however the independent verification is NOT required due to the dose being >10mRem. The student would have to know the General Rule of >10mRem for the "Hands On" independent verification NOT required.

D: **Incorrect-**. With a calculation of 110/6 (carrying the wrong units), the student would get 18 mRem. With the total >10mRem, the student could pick this distractor as correct.

Technical Reference(s): OP-AA-108-101-1002 (Attach if not previously provided)

Component Configuration Control

Attachment 5 (1.5 and 1.5.1)

Proposed References to be provided to applicants during examination: none

Learning Objective: Describe what the worker is acknowledging when signing a RWP prior to use.

Question Source: Bank # NRC2019 Q#72 (New)

Modified Bank # (Note changes or attach parent)

New

Question History: NRC2019

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.13	
Importance Rating	3.4	

K/A Statement: Radiation Control: Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Question: RO #71

Which of the following is the required action if a Locked High Radiation Area key is lost by the responsible individual who checked it out from the Radiation Protection Department?

The individual shall immediately notify ___________________.

A. Security and establish positive control of access to the area.

B. the Radiation Protection Manager and verify the area locked after checking for unauthorized personnel.

C. the Shift Manager, re-lock the area and have Radiation Protection check for exposures in excess of those expected.

D. Shift Radiation Protection Technician and Radiation Protection Supervisor and control all access to the area.

Proposed Answer: **D**

Explanation (Optional): See attached section of RP-AA-463, High Radiation Area Key Controls
Plausibility Justification:

A: **Incorrect-**. Security is plausible due to the fact that the security watch is always on duty in all sections of the plant. Positive control of access to the area is a responsibility of the key holder IAW RP-AA-463.

B: **Incorrect-** This would be the responsibility of the Shift Rad Pro Technician and Supervisor after the key holder has informed them of the lost key. IAW RP-AA-463, the Shift Rad Pro Tech would normally verify the LHRA is locked once the operator returned the key.

C: **Incorrect-** The key holder would have to **immediately contact** the Shift Rad Pro Tech and Rad Pro Supervisor before contacting the Shift Manager. The Shift Manager would ensure that the Rad Pro department completed the proper investigation and also had the LHRA re-locked.

D: **Correct-**. IAW RP-AA-463, responsibilities of the key holder includes controlling access to the area and if the key is lost to immediately inform the SRPT and RPS.

Technical Reference(s): RP-AA-463 (Attach if not previously provided)

High Radiation Area Key Controls

Proposed References to be provided to applicants during examination: none

Learning Objective: State the responsibilities of the following personnel for issuance of keys to Locked High Radiation Areas:
Key Holder
SM
Shift Radiation Protection Technician (SRPT) IAW RP-AA-463, High Radiation Area Key Controls

Question Source: Bank # 30961

Modified Bank # (Note changes or attach parent)

New

Question History: NRC2016

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.35	
	Importance Rating	3.8	

K/A Statement: Emergency Procedures/Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.

Question: RO #72

Given:

- The plant was at 100% power.

When:

- A Loss of Offsite Power occurred.
- The 'A' EDG failed to automatically start.
- Operators were unable to start the 'A' EDG from the Control Room.
- HC.OP-AB.ZZ-0135, Station Blackout/Loss of Offsite Power/Emergency Diesel Generator Malfunction is being implemented.

Which of the following sets of actions accurately describes how to start AND load the 'A' EDG under these conditions IAW HC.OP-AB.ZZ-0135?

Local and Remote EDG Panels:

1A-C-421: Local Engine Control Panel (102' elev.)
1A-C-422: Remote Generator Control Panel (130' elev.)
1A-C-423: Remote Engine Control Panel (130' elev.)

After verifying that the READY FOR AUTO START is ON, Panel 1A-C-423, then ______________.

A. PLACE EMERGENCY TAKE-OVER Switch in EMERG position on Panel 1A-C-422, PLACE the REMOTE ENGINE CONTROL in START on 1A-C-423, and then 'A' EDG output breaker will auto-close.

B. Press the DIESEL ENG REMOTE pushbutton for the 'A' EDG on 10C651(MCR) and ensure REMOTE light is on 10C651, PLACE LOCAL ENGINE CONTROL Switch on 1A-C-421 in START, and then the 'A' EDG output breaker will be closed from the Main Control Room.

C. PLACE EMERGENCY TAKE-OVER Switch in EMERG position on Panel 1A-C-422, PLACE REMOTE ENGINE CONTROL in START at Panel 1-A-C423, and then the 'A' EDG output breaker will be closed at Panel 1-A-C-422.

D. PLACE the REM/LOC/MAINT CONTROL SELECT switch on 1A-C-421 in REMOTE, Place LOCAL ENGINE CONTROL Switch on 1A-C-421 in START, and then the 'A' EDG output breaker will be closed from the Main Control Room.

Proposed Answer: **C**

Explanation (Optional): Placing the Emergency Take over switch in EMERG enables the **local start**. The auto-closure circuit for the EDG output breaker also remains enabled. When the EDG frequency and voltage satisfy the Diesel Ready for Load permissive (>95% Frequency and Voltage), the breaker can be closed manually. In REMOTE on the REM/LOC/MAINT switch at the 1A-C-421 panel, diesel control is established **either in the control room or at the remote engine/generator control panels (422 and 423).** IAW AB-135 Subsequent action B, if the EDG does not start and load from the Main Control Room, then a local start is requires at the Remote Panels 422 and 423 (see attached). AB-135 allows for either an EDG start from local panels or remote panels. This particular scenario is for a remote panel start; however there are distractors for local panel start. In any case (remote or local panel start) the EDG output breaker will be closed from the remote panel (422).

Plausibility Justification:

A: **Incorrect-**. Placing the Emergency Take over switch in EMERG enables the local start. However, placing the EMERGENCY TAKE-OVER switch on 1A-C-422 in EMERG **defeats the automatic closure of the EDG output breaker. It has to be manually closed from panel 1A-C-422.**

B: **Incorrect-** Selecting REMOTE for the 'A' EDG engine on 10C651E does NOT enable the LOCAL ENGINE CONTROL switch on 1A-C-421. It does enable control at the Remote panels 1A-C-422 and 423.

C: **Correct-** Placing the Emergency Take over swith in EMERG enables the local start of the EDG from the remote panels (422 and 423). When the EDG frequency and voltage satisfy the Diesel Ready for Load permissive (>95% Frequency and Voltage), the breaker can be closed manually.

D: **Incorrect-**. In REMOTE on the REM/LOC/MAINT switch at the 1A-C-421 panel, diesel control is established **either in the control room or at the remote engine/generator control panels** (422 and 423).

Technical Reference(s): HC.OP-AB.ZZ-0135(Q) (Attach if not previously provided)

Station Blackout/Loss of Offsite Power/EDG Malfunction
HC.OP-AR.KJ-0001 1A-C-423 Panel

Proposed References to be provided to applicants during examination: none

Learning Objective: Discuss the operational implications of the abnormal indications/alarms for system operating parameters related to Station Blackout/Loss Of Offsite Power Diesel Generator Malfunction, Abnormal Operating Procedure.

Question Source: Bank # 62474

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.28	
	Importance Rating	3.2	

K/A Statement: Emergency Procedures/Plan: Knowledge of procedures relating to a security event.

Question: RO #73

Given:

- There was an Unusual Event declared due to a fire in the Auxiliary Building caused by an explosion.
- The Site Protection is responding to the fire.
- The Security Department is responding to the Security Event.
- It is determined that there is an on-going security threat at Hope Creek.

IAW Emergency Preparedness procedures relating to the Security Event,

The activation of the Operations Support Center (OSC) _____ (1) _____.

All ONSITE personnel will _____ (2) _____.

A. (1) is required
(2) report to Assembly/Accountability Stations

B. (1) is NOT required
(2) take cover and Shelter-in-Place

C. (1) is NOT required
(2) report to Assembly/Accountability Stations

D. (1) is required
(2) take cover and Shelter-in-Place

Proposed Answer: **D**

Explanation (Optional): The Operations Support Center (OSC) will be activated **if the UE classification was the result of a Security event**. Otherwise, OSC activation is optional at a UE classification. Assembly is required at an Alert and optional at a UE. Accountability is implemented at an Alert or higher level.

Plausibility Justification:

A: **Incorrect-**. The activation of the OSC is required due to the Security Event. With the Security Event on-going, the reporting to an Assembly/Accountability Station would not be prudent with the possible terrorist activities still in progress.

B: **Incorrect-** Normally during an Unusual Event classification, it would be at the discretion of the Emergency Coordinator to activate the OSC (optional). However, due to the Security Event the OSC activation would be required. Due to the shift personnel manning the OSC, the OSC personnel would man the OSC and shelter-in-place.

C: **Incorrect-** Normally during an Unusual Event classification, it would be at the discretion of the Emergency Coordinator to activate the OSC (optional). However, due to the Security Event the OSC activation would be required. With the Security Event on-going, the reporting to an Assembly/Accountability Station would not be prudent with the possible terrorist activities still in progress.

D: **Correct-**. The activation of the OSC is required due to the Security Event. All ONSITE personnel will "Take Cover and Shelter-in-Place" (see attached emergency procedure relating to the security event).

Technical Reference(s): NC.EP-EP.ZZ-0102(Q) (Attach if not previously provided)

Emergency Coordinator Response

Proposed References to be provided to applicants during examination: none

Learning Objective: List the four emergency classifications and what happens during the classification including:

- Facilities activated
- Facilities staffed

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.42	
	Importance Rating	3.9	

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

Question: RO #74

Given:

- The plant is operating at 100% power.
- Delaware River temperature is at 82°F and rising slowly.
- HC.OP-DL.ZZ-0026 Attachment 3h, Plant Systems (River Water Temperature) is implemented.

No additional Technical Specification actions are required until River Water Temperature reaches ___________.

A. 83.1°F

B. 84.1°F

C. 85.1°F

D. 88.1°F

Proposed Answer: **C**

Explanation (Optional): See attached procedures, HC.OP-SO.EA-0001, HC.OP-DL.ZZ-0026, and T.S. 3.7.1.3

Plausibility Justification:

A: **Incorrect-**. With river temperature > 82°F, the Attachment 3h is implemented. Once temperature is >84°F, then the SOP section for elevated river temperature is implemented, but only after river temperature exceeds 85°F. This is also the entry condition for T.S. 3.7.1.3 Ultimate Heat sink specification. At 84.5°F, the crew would continue to monitor river temperature for >85°F. IAW T.S. 3.7.1.3 at 88°F all SSWS, SACS, EDGs and SACS cross ties not cross connected would allow continued power operations.

B: **Incorrect-** Once temperature is >84°F, then the SOP section for elevated river temperature is implemented, but only after river temperature exceeds 85°F.

C: **Correct-** With river temperature >85°F, the crew would enter T.S. 3.7.1.3 and implement the actions IAW SSW SOP section for elevated river temperature which would require opening the yard dump valves to establish an alternate discharge path for the SSW system

D: **Incorrect-**. IAW T.S. 3.7.1.3 at 88°F all SSWS, SACS, EDGs and SACS cross ties not cross connected would allow continued power operations.

Technical Reference(s): HC.OP-DL.ZZ-00026(Q) Att. 3h (Attach if not previously provided)

HC.OP-SO.EA-0001(Q) Sect. 5.9

T.S. 3.7.1.3 Ultimate Heat Sink

Proposed References to be provided to applicants during examination: none

Learning Objective: Given specific plant operating conditions which require operator actions within 1 hour From Memory select the correct Technical Specification action(s) for the following:
T.S 3.7.1.3 Ultimate Heat Sink

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	2.3.14	
Importance Rating	3.4	

K/A Statement: Radiation Control: Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question: RO #75

IAW HC.OP-EO.ZZ-0318, Containment Venting:

With high radioactivity conditions in primary containment, venting the primary containment via either the Suppression Chamber 2" or 24" Exhaust lines would be the preferred vent paths.

However, what is the concern for using either of these vent paths?

A. High radiation conditions in the reactor building.

B. Unmonitored radioactive release.

C. Releasing of radioactivity without the scrubbing effect.

D. Inability to reduce pressure to prevent containment damage.

Proposed Answer: **A**

Explanation (Optional): If the containment atmosphere is believed to be contaminated, vent lineups should be selected so as to minimize the amount of radioactivity released while still achieving the objective of the venting requirement. The suppression pool is the preferred primary containment vent path under accident conditions. Venting from the suppression chamber is generally preferred, to obtain the benefits of suppression pool scrubbing. However, the vent path passes through ductwork that contains Back Draft Dampers and Blow-out Panels causing elevated radiological conditions in the Reactor Building.

Plausibility Justification:

A: **Correct-**. Using the preferred vent path through the suppression chamber for scrubbing and then through the Reactor Building, this will cause elevated radiological conditions in the personnel spaces of the Reactor Building. Good engineering radiation protection controls will have to be in place.

B: **Incorrect-** The vent path is scrubbed by the suppression pool, treated by FRVS and monitored prior to being released.

C: **Incorrect-** By utilizing the vent path through the suppression chamber (2" or 24" Exhaust line see attached), the scrubbing effect will help reduce the radioactive release.

D: **Incorrect-**. Venting the containment is still accomplished through the preferred vent path of either suppression chamber exhaust line, however there is an added benefit through the suppression chamber for scrubbing (reducing the radioactivity of the release).

Technical Reference(s): HC.OP-EO.ZZ-0318(Q) (Attach if not previously provided)

Containment Venting

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the basis/reason for all prerequisites, precautions, and limitations of each of the 300 series Emergency Operating Procedures.

Question Source: Bank # 113307

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295026 EA2.01
Importance Rating		4.2

K/A Statement: Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature

Question: SRO #76

Given:

- The plant was operating at 75% power.

When:

- One relief valve stuck open and could not be closed.
- The reactor has been manually shut down.

Current plant conditions:

- Suppression pool temperature is at 96°F and slowly rising.
- Suppression Pool level is at 79 inches and slowly rising.
- Drywell pressure is at 1.1 psig and stable.
- Drywell temperature is at 100°F and stable.

What actions IAW emergency operating procedures is required?

I. Place all available Drywell Cooling in service.

II. Place all available Suppression Pool Cooling in service.

III. Initiate Suppression Chamber Sprays.

IV. Reject water from the suppression pool through 'B' RHR to Radwaste as necessary.

A. II & IV ONLY

B. I, & IV ONLY

C. II, III, & IV ONLY

D. I, II, & III ONLY

Proposed Answer: **A**

Explanation (Optional): See attached EOP-102 legs for suppression pool parameters. If it is determined that suppression pool temperature cannot be maintained below 95°F using normal methods, subsequent instructions provide guidance on controlling suppression pool temperature **using all available suppression pool cooling.** When suppression pool level rises above the Technical Specification upper limit, EOP-102 provides direction to use ECCS and / or alignments not normally used to maintain suppression pool water level in general plant procedures. 'B' RHR to Radwaste is used in an effort to maintain primary containment in its normal configuration and to prevent level from rising to the point where the more severe actions of reactor scram, termination of drywell sprays, termination of external injection sources, and emergency RPV depressurization will be required.

Plausibility Justification:

A: **Correct-**. See attached SP/T-3. With Suppression Pool temperature >95°F and rising, the action IAW EOP-102 is to place all available Suppression Pool Cooling in service. See attached SP/L-12. With Suppression Pool level High >78.5 inches and rising, the action IAW EOP-102 is to lower the Suppression Pool level using 'B' RHR to Radwaste line up.

B: **Incorrect-** Maximizing drywell cooling would have been directed from the HC.OP-AB.CONT-0001 for the elevated drywell pressure, however due to the plant shutdown and the current primary containment parameters, drywell cooling is not ordered from the EOPs specifically EOP-102. Drywell pressure is currently stable.

C: **Incorrect-** Suppression Chamber Sprays are not directed until suppression chamber pressure exceeds 9.5 psig, and are not required at this time.

D: **Incorrect-**. See attached SP/T-3, SP/L-12, DW/T-2, and DW/P-2. Suppression Chamber Sprays are not directed until suppression chamber pressure exceeds 9.5 psig, and are not required at this time

Technical Reference(s): HC.OP-EO.ZZ-0102-FC (Attach if not previously provided)

Primary Containment Control
HC.OP-AB.CONT-0001 Drywell Pressure

Proposed References to be provided to applicants during examination: none

Learning Objective: 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

Question Source: Bank # 152245

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		295016 AA2.03
	Importance Rating		4.4

K/A Statement: Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : Reactor pressure

Question: SRO #77

Given:

- The control room was evacuated.
- The reactor is shutdown.
- Alternate Shutdown Cooling from the Remote Shutdown Panel (RSP) following a Loss of Offsite Power is being performed.
- All RSP controls are functional.

During the approach to Cold Shutdown, IAW HC.OP-IO.ZZ-0008, Shutdown From Outside the Control Room,

If Reactor Pressure does NOT stabilize below 160 psig above the Suppression Pool, SRVs ___(1)___ can be opened as required from their RSP controls.

If the cooldown rate exceeds 90°F/hr., then direct the crew to reduce LPCI injection into the RPV UNTIL Reactor Pressure lowers to within ___(2)___ psig of Suppression Chamber Pressure.

A. (1) A and E
(2) 100

B. (1) F, H, and M
(2) 50

C. (1) A and E
(2) 50

D. (1) F, H, and M
(2) 100

Proposed Answer: **B**

Explanation (Optional): See attached IO-0008 Att. 11. The following SRV's can be operated from the remote shutdown panel (10C399): **F013F (non-ADS),F013H (LLS SRV, non ADS),and F013M (non-ADS)** If the RSP-10C399 Channel B transfer switch is placed in EMERGENCY, the F, H & M SRV's become inoperable from the main control room. The **A and E SRV's** have AUTO-OPEN key lock switches in the lower relay room on panel 10C631, (Div. 4 Relay Vertical Panel). Controlling reactor pressure with the given conditions, SRVs will be used as required. In the case of operating SRVs from the RSP local controls, only the **F, H, or M SRVs can be operated**. A and E SRVs are available through a key lock switch, however their controls are at the lower relay panel 10C631. If the cooldown rate is being exceeded (90°F/hr) with the above conditions, IAW IO-0008 Att. 11, the direction to reduce LPCI injection until Reactor Pressure lowers to within 50 psig of Suppression Chamber Pressure or the cooldown decreases below 100°F /hr. is the mitigation strategy at this point.

Plausibility Justification:

A: **Incorrect-**. A and E SRVs are available through a key lock switch, however their controls are at the lower relay panel 10C631. IO-0008 gives the option to use the A and E SRVs, however the stem of the question asks for the controls at the RSP. When exceeding the cooldown rate, the direction to reduce LPCI injection until Reactor Pressure lowers to within 50 psig of Suppression Chamber Pressure or the cooldown decreases below 100°F /hr. is the mitigation strategy at this point.

B: **Correct- F013F (non-ADS), F013H (LLS SRV, non ADS), and F013M (non-ADS) can** be operated from the remote shutdown panel (10C399). When exceeding the cooldown rate, the direction to reduce LPCI injection until Reactor Pressure lowers to **within 50 psig** of Suppression Chamber Pressure or the cooldown decreases below 100°F /hr. is the mitigation strategy at this point.

C: **Incorrect-** A and E SRVs are available through a key lock switch, however their controls are at the lower relay panel 10C631. IO-0008 gives the option to use the A and E SRVs, however the stem of the question asks for the controls at the RSP.

D: **Incorrect-**. When exceeding the cooldown rate, the direction to reduce LPCI injection until Reactor Pressure lowers to **within 50 psig** of Suppression Chamber Pressure or the cooldown decreases below 100°F /hr. is the mitigation strategy at this point.

Technical Reference(s): HC.OP-IO.ZZ-0008(Q) (Attach if not previously provided)

Shutdown From Outside Control Room

Proposed References to be provided to applicants during examination: none

Learning Objective: 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.

Question Source: Bank # 125070

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295037 EA2.07
Importance Rating		4.2

K/A Statement: Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Containment conditions/isolations

Question: SRO #78

Given:

- While operating at 60% reactor power, a reactor scram on low reactor water level occurs.
- The reactor fails to scram.
- All control rods remain at their pre-trip conditions.

T=30 minutes after the transient:

- SLC tank level is at 2600 gallons.
- Rx power is <4%.
- RPV pressure is at 900 psig.
- RPV level was intentionally lowered and maintained within -50 inches to -120 inches.
- Suppression pool level is at 79 inches and steady.
- Suppression pool temperature is at 155°F and slowly lowering.
- Drywell pressure is at 4.5 psig and steady.
- Main condenser vacuum is at 6 inches Hg abs and slowly degrading.
- NO indications of a fuel failure or steam line break exist.

Which of the following action(s) is (are) required IAW emergency operating procedures?

A. Lower reactor pressure to stay below the Heat Capacity Temperature Limit Curve.

B. Emergency depressurize the reactor.

C. Bypass interlocks as required and maintain the MSIVs open.

D. Depressurize the reactor and maintain cooldown rate below 90°F/hr.

Proposed Answer: **C**

Explanation (Optional): Under failure-to-scram conditions, certain isolations may be defeated to prevent closure of the MSIVs and permit continued use of the main condenser as a heat sink. If the main steam lines were allowed to close, the energy addition to the containment would likely increase and the Heat Capacity Temperature Limit could be reached in a relatively short time. Defeating the main steam line isolations may thus be a principal contributor to successful mitigation of a failure-to-scram event. Defeating the low level isolations anticipates the possible level reduction and prevents unnecessary loss of the main condenser. See attached EOP-101A FC.

Plausibility Justification:

A: **Incorrect-**. The combination of RPV pressure and Suppression pool temperature for HCTL do not challenge the Action required area of HCTL. See attached **HCTL curve.**

B: **Incorrect-** RPV water level can be maintained above - 185 inches, emergency depressurization is not required. See attached **LP-18.**

C: **Correct-** Isolations may be defeated to prevent closure of the MSIVs and permit continued use of the main condenser as a heat sink. Low RPV water level isolations are defeated since subsequent steps may lower RPV water level to below the low RPV water level MSIV isolation setpoint. See attached **LP-8 and RC/P-17.**

D: **Incorrect-**. Cooldown is not permitted until the Cold Shutdown Boron weight has been added. (1100 gallons in the SLC tank). See attached **RC/P-19.**

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

Proposed References to be provided to applicants during examination: none

Learning Objective: Given any step of the procedure, explain the basis for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step.

Question Source: Bank # 33931

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		295021 2.2.40
	Importance Rating		4.7

K/A Statement: Equipment Control: Ability to apply Technical Specifications for a system- Loss of Shutdown Cooling (RHR).

Question: SRO #79

Given:

- The plant is shutdown.
- Detensioning of the Reactor Head is in progress.
- 'A' RHR is in shutdown cooling.
- 'B' RHR is in standby.
- Reactor coolant temperature is at 135°F and steady.
- RHR flow is at 10,000 gpm.
- The Plant Operator (PO) determines that the 'A' RHR Heat Exchanger has failed due to leakage into the Station Auxiliary Cooling System (SACS).
- The PO removes 'A' RHR Loop from service and isolates the 'A' RHR Heat Exchanger.

With the above plant conditions and ____ (1) ____ RPV level as defined in Technical Specifications, the CRS determines that ________ (2) ________.

A. (1) low
(2) at least one shutdown cooling mode loop of the residual heat removal (RHR) system SHALL be OPERABLE with one RHR pump and one heat exchanger in operation.

B. (1) high
(2) two shutdown cooling mode loops of the residual heat removal (RHR) system SHALL be OPERABLE and at least one loop is in operation with one RHR pump and one heat exchanger OPERABLE in each loop.

C. (1) high
(2) at least one shutdown cooling mode loop of the residual heat removal (RHR) system SHALL be OPERABLE with one RHR pump and one heat exchanger in operation.

D. (1) low
(2) two shutdown cooling mode loops of the residual heat removal (RHR) system SHALL be OPERABLE and at least one loop is in operation with one RHR pump and one heat exchanger OPERABLE in each loop.

Proposed Answer: **D**

Explanation (Optional): With a loss of SDC on the 'A' RHR loop and the 'A' RHR heat exchanger isolated due to the leak, the 'A' RHR loop of SDC is INOPERABLE. The current status of the plant with RPV level <22 feet 2 inches above the top of the reactor pressure vessel flange **(low level)** and heat losses to ambient are not sufficient to maintain OPERATIONAL CONDITION 5, the CRS would have to determine that **T.S. 3.9.11.2** is applicable which would require two shutdown cooling mode loops of the residual heat removal (RHR) system OPERABLE and at least one loop is in operation with one RHR pump and one heat exchanger OPERABLE in each loop. This is the **entry conditions for T.S. 3.9.11.2.** The entry conditions for **T.S. 3.9.11.1** with the **high level** as defined as >22 feet 2 inches above the top of the reactor pressure vessel flange and heat losses to ambient are not sufficient to maintain OPERATIONAL CONDITION 5 would be at least one shutdown cooling mode loop of the residual heat removal (RHR) system SHALL be OPERABLE with one RHR pump and one heat exchanger OPERABLE. (See attached entry conditions and applicability's for T.S. 3.9.11.1 High Level and 3.9.11.2 Low Level).

Plausibility Justification:

A: **Incorrect-**. As defined in T.S. 3.9.11, <22 feet 2 inches above the top of the reactor pressure vessel flange is applicable to T.S. 3.9.11.2 LOW WATER LEVEL specification. The entry condition for T.S 3.9.11.2 is two shutdown cooling mode loops of the residual heat removal (RHR) system SHALL be OPERABLE and at least one loop is in operation with one RHR pump and one heat exchanger OPERABLE in each loop. The student will have to determine either T.S. 3.9.11.1 HIGH WATER LEVEL or T.S. 3.9.11.2 LOW WATER LEVEL and then the entry condition for each Tech Spec.

B: **Incorrect-** T.S. 3.9.11.2 LOW WATER LEVEL is applicable for the given conditions. Detensioning of the head is OPCON 5, however the RPV level is <22 feet 2 inches above the top of the reactor pressure vessel flange (the refuel cavity above of the vessel head has not been filled yet).

C: **Incorrect-** T.S. 3.9.11.2 LOW WATER LEVEL is applicable for the given conditions. Detensioning of the head is OPCON 5, however the RPV level is <22 feet 2 inches above the top of the reactor pressure vessel flange (the refuel cavity above of the vessel head has not been filled yet). The entry condition for T.S 3.9.11.2 is two shutdown cooling mode loops of the residual heat removal (RHR) system SHALL be OPERABLE and at least one loop is in operation with one RHR pump and one heat exchanger OPERABLE in each loop.

D: **Correct-**. T.S. 3.9.11.2 LOW WATER LEVEL is applicable for the given conditions. The entry condition for T.S 3.9.11.2 is two shutdown cooling mode loops of the residual heat removal (RHR) system SHALL be OPERABLE and at least one loop is in operation with one RHR pump and one heat exchanger OPERABLE in each loop.

Technical Reference(s): T.S. 3.9.11.1 and 3.9.11.2 (Attach if not previously provided)

Refueling Operations

Proposed References to be provided to applicants during examination: none

Learning Objective: Assess Residual Heat Removal System operability and determine required actions associated with Residual Heat Removal System inoperability.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295001 2.4.50
Importance Rating		4.0

K/A Statement: Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. Partial or Complete Loss of Forced Core Flow Circulation

Question: SRO #80

Given:

- The plant was operating at 100% power.

When:

- Reactor flow and power began to trend downward.
- RPT breakers CN205 and DN205 are tripped open.

(1) What is the required action IAW HC.OP-AB.RPV-0003, Recirculation System abnormal?
(2) Based on the above conditions and IAW 10 CFR 50.72, what is the earliest reporting requirement?

A. (1) lock the Reactor Mode Switch in the Shutdown position.
(2) eight hours.

B. (1) check power-to-flow relationship for 'A' Reactor Recirculation Pump in single loop operation.
(2) four hours.

C. (1) check power-to-flow relationship for 'B' Reactor Recirculation Pump in single loop operation.
(2) eight hours.

D. (1) lock the Reactor Mode Switch in the Shutdown position.
(2) four hours.

Proposed Answer: **D**

Explanation (Optional): The RPT breakers function to interrupt power from the recirc Variable Frequency Drive unit to the reactor recirc pumps, thus providing a rapid cessation of core flow and subsequent rise in core voiding to reduce reactor power. Each recirc pump is supplied power through **two breakers arranged in series**. "A" Recirc Pump: RPT Breakers AN205 & CN205, "B" Recirc Pump: RPT Breakers BN205 & DN205. With CN205 and DN205 tripped, both "A" and "B" reactor recirculation pumps are tripped, therefore, there are no recirc pumps running with the reactor critical. The immediate operator actions IAW HC.OP.AB.RPV-0003 is to **Lock The Mode Switch in Shutdown** (see attached RPV-0003). A **4-hour report** is required for an event involving a critical scram unless it resulted from and was part of a pre-planned sequence. Manual RPS actuation in anticipation of receiving an automatic RPS actuation is reportable (RAL 11.3.2).

Plausibility Justification:

A: **Incorrect-**. An 8 hour report is correct for any other actuation (for example PCIS and ECCS). Due to the manual RPS actuation, this is reportable to the NRC within 4 hours IAW RAL 11.3.2.

B: **Incorrect-** Due to the RPT breaker arranged in series the 'A' Recirc pump RPT breakers are AN205 and CN205. With the trip of the CN205 the 'A' Recirc pump will be tripped, along with the 'B' Recirc pump due to the DN205 breaker trip. Therefore; IAW AB.RPV-0003 immediate operator actions is to Lock the Mode Switch in Shutdown (RPS trip). The student will have to determine the RPT arrangement and understand that the plant is not in single loop operation and the I.O.A will have to be taken. Due to the manual RPS actuation, this is reportable to the NRC within 4 hours IAW RAL 11.3.2.

C: **Incorrect-** Due to the RPT breaker arranged in series the 'B' Recirc pump RPT breakers are BN205 and DN205. With the trip of the DN205 the 'B' Recirc pump will be tripped, along with the 'A' Recirc pump due to the CN205 breaker trip. Therefore; IAW AB.RPV-0003 immediate operator actions is to Lock the Mode Switch in Shutdown (RPS trip). The student will have to determine the RPT arrangement and understand that the plant is not in single loop operation and the I.O.A will have to be taken. Due to the manual RPS actuation, this is reportable to the NRC within 4 hours IAW RAL 11.3.2.

D: **Correct-**. The immediate operator actions IAW HC.OP.AB.RPV-0003 is to **Lock The Mode Switch in Shutdown.** Due to the manual RPS actuation, this is reportable to the NRC within 4 hours IAW RAL 11.3.2.

Technical Reference(s): RAL 11.3.2 System Actuation (Attach if not previously provided)

HC.OP-AB.RPV-0003 Recirculation

HC.OP-SO.BB-0002 Recirc. Ops.

Proposed References to be provided to applicants during examination: EALs and RALs without the attachments

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with the Recirculation System/Power Oscillations.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #		295018 2.4.4
Importance Rating		4.7

K/A Statement: Emergency Procedure/Plan: Ability to recognize abnormal indications for system operating parameters that are entry level conditions for emergency and abnormal operating procedures: Partial or Total Loss of CCW

Question: SRO #81

Given:

- The plant is at 100% rated power.

Then:

- The plant experiences a grassing event.
- HC.OP-AB.COOL-0001, "Station Service Water" has been entered.
- All actions to clear the SSWS Strainer Hi Hi differential pressure alarm have failed.
- 'A' SSWS Strainer dP is 6 psid.
- The 'A' SSW Pump is in service at 3500 gpm on the loop supplying TACS.
- SSW Temperature is 53°F.

'A' SSWS Strainer operation is considered ___ (1) ___. Additional actions include ___ (2) ___.
[Reference attached]

A. (1) inoperable
(2) placing the standby SSW pump in service.

B. (1) degraded
(2) ensuring the standby SSW pump is in Manual.

C. (1) inoperable
(2) ensuring the standby SSW pump is in Manual.

D. (1) degraded
(2) placing the standby SSW pump in service.

Proposed Answer: **B**

Explanation (Optional): With the given parameters for SSW, the student will have to determine that the entry condition into subsequent action C of AB.COOL-0001 is applicable (see attached). With strainer dP above 5 psid, the attachment 4 graph will have to be used to determine the operability of the 'A' SSW traveling screen. Placing the standby SSW pump in the "A" SSW loop in MANUAL is also required IAW subsequent action C of AB.COOL-0001 to prevent a possible automatic start of the standby pump from a low flow condition in the associated loop. Normal alignment would have one SSW pump in service per loop of SSW and the standby pumps in AUTO.

Plausibility Justification:

A: **Incorrect-**. With the given SSW parameters of flow and temperature, the point on the graph of attachment 4, places the 'A' SSW strainer in the acceptable region for operations. However, the 'A' SSW traveling screen is considered degraded. Due to a potential low flow condition in the associated loop the standby SSW pump will be placed in MANUAL. The student will have to both determine from the graph the operability of the strainer and the fact that the standby pump could auto start due to the conditions on the "A" SSW loop.

B: **Correct-** The point on the graph of attachment 4, places the 'A' SSW strainer in the acceptable region for operations. However, IAW the instruction of attachment 4, the 'A' SSW strainer is considered degraded. With the hi strainer dP on the loop of SSW supplying TACS and a potential of a low flow automatic start (condition C) the standby pump is placed in MANUAL.

C: **Incorrect**- With the 'A' SSW pump the only pump in service, the standby pump will be placed in MANUAL. The point on the graph of attachment 4, places the 'A' SSW strainer in the acceptable region for operations (degraded).

D: **Incorrect**-. With the 'A' SSW pump the only pump in service, the standby pump will be placed in MANUAL to prevent the auto start on associated loop low flow condition.

Technical Reference(s): HC.OP-AB.COOL-0001 Station Service Water Condition C and Attachment 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: HC.OP-AB.COOL-0001 Attachment 4

Learning Objective: Given plant conditions and plant procedures, determine required actions of the retainment override(s) and subsequent operator actions in accordance with Station Service Water.

Question Source: Bank # 73176

Modified Bank # (Modified stem with SSW flow and temperature to change the answer from degraded to inoperable)

New

Question History: #78 on NRC2018

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		600000 AA2.03
	Importance Rating		3.2

K/A Statement: Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Fire alarm

Question: SRO #82

Given:

- The plant has just started up after a refueling outage.

When:

- A fire alarm for the Service/Radwaste Building has alarmed on the MCR fire computer.
- A fire has broken out in the Radwaste Trash Compacter Area.
- All applicable procedures have been entered.

Current plant conditions:

- The Fire Brigade has been fighting the fire for 18 minutes.
- Levels of airborne contamination have been rising in the Radwaste Building.
- South Plant Vent Radiation effluent levels are 9.12 E+2 uCi/sec NG.
- NO safety systems have been affected by the fire at this time.

With the above conditions, declare an ____________________.

A. Unusual Event for a plant fire.

B. Unusual Event for a radiological release.

C. Alert for a plant fire.

D. Alert for a radiological release.

Proposed Answer: **A**

Explanation (Optional): HU4.1 – UNUSUAL EVENT: This EAL addresses the magnitude and extent of **FIRES** that may be indicative of a potential degradation of the level of safety of the plant. The Table H-1 (see attached EAL) Fire Areas include those plant structures identified as Seismic Category I. Upon receipt, operators will take prompt actions to confirm the validity of an **initial fire alarm**, indication, or report. For EAL HU4.1 the intent of the 15-minute duration is to size the **FIRE** and to discriminate against small **FIRES** that are readily extinguished. With the duration of the firefighting efforts at 18 minutes and in an area on Table H-1, the CRS would declare an U.E IAW HU4.1 (see attached). The radiation levels of the SPV do NOT exceed the requirement of an U.E or Alert (RU1.1 or RA1.1). Escalation of the U.E due to the fire would occur if the fire had affected a train of a Safety System (SA8.1). The students will be provided the EAL and RALs without attachments. The student will have to determine both the extent of the fire (Radwaste is on the Table H-1) and the offsite release of the SPV.

Plausibility Justification:

A: **Correct-**. With the fire (alarm) in the Radwaste area (Table H-1) and firefighting efforts have been in progress for greater than 15 minutes, the CRS would declare an Unusual Event with HU4.1 due to the fire. The SPV release is below the setpoint for either the U.E or Alert for offsite release.

B: **Incorrect-** The given value for the SPV effluent level is below the threshold for declaring an U.E for offsite release (RU1.1).

C: **Incorrect-** With the fire (alarm) in the Radwaste area (Table H-1) and firefighting efforts have been in progress for greater than 15 minutes, the CRS would declare an Unusual Event with HU4.1 due to the fire. Escalation of the U.E due to the fire would occur if the fire had affected a train of a Safety System (SA8.1).

D: **Incorrect-**. The given value for the SPV effluent level is below the threshold for declaring an U.E for offsite release (RU1.1) and therefore not for an Alert level (RA1.1).

Technical Reference(s): EAL HU4.1, SA8.1, RU1.1, & RA1.1 (Attach if not previously provided)

Fire and Offsite Release

Proposed References to be provided to applicants during examination: EALs and RALs without the attachments

Learning Objective: ECG/E-Plan/Fire & Medical Questions: Knowledge of the reasons for the following responses as they apply to the implementation of site emergency plan.

Question Source: Bank # 35661

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #		295015 AA2.01
Importance Rating		4.3

K/A Statement: Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Reactor power

Question: SRO #83

Given:

- The plant was operating at 100% power.

When:

- A Main Steam Line break occurred in the Steam Tunnel causing an MSIV isolation and a reactor scram.

Current plant conditions:

- All control rods did NOT insert.
- HC.OP-EO.ZZ-0101A, ATWS-RPV Control was entered.
- All SCRAM Hard Card actions are completed.
- Reactor Power is at 5% and slowly lowering.
- Suppression Pool Temperature is at 115 °F and slowly rising.
- SRVs are cycling.
- RPV level was intentionally lowered to -129 inches and is now being maintained between -129 inches and -185 inches.
- Secondary Containment is being maintained.
- Secondary Containment area temperatures and radiation monitors are stable.
- Drywell Pressure is at 1.8 psig and slowly rising.
- Reactor Coolant Sample Activity is 250 µCi/gm Dose equivalent I-131.

What describes the classification level for this event and the cause of the classification?

A. Alert due to the Main Steam Line Break.

B. Site Area Emergency due to High Drywell Pressure.

C. Alert due to the Failure to Scram.

D. Site Area Emergency due to High RCS Iodine Concentration.

Proposed Answer: **C**

Explanation (Optional): SA6.1 – ALERT- An automatic or manual scram fails to shut down the reactor as indicated by **reactor power >4%.** This EAL addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken (Hard Card actions of initiating ARI) at the reactor control consoles to shutdown the reactor are also **unsuccessful.** If the failure to shut down the reactor is prolonged enough to cause a challenge to RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a **SITE AREA EMERGENCY** via EAL SS6. (see attached EALs for Failure to Scram). The student will have to determine from the current reactor power and Hard Card actions that they are in an ALERT due to failure to Scram. The student will have to reference the ECG barrier table to realize that the Main Steam Line Break has been isolated (no classification due to main steam line break). The high drywell pressure will be a 5-point ALERT not an SAE and the dose equivalent iodine is below the threshold for declaration (See attached barrier tables).

Plausibility Justification:

A: **Incorrect-**. Due to the MSIVs isolating the main steam line break outside of primary containment would be isolated; therefore there would be no declaration on the main steam line break. IAW the barrier table (RB3.L), if the MSIV did not isolate than a declaration for a 5-point ALERT would be correct.

B: **Incorrect-** With secondary containment maintained and rad monitors and area temps stable, the break would not cause an SAE to be declared. Barrier Table would give only 5 points ALERT (RB2.L).

C: **Correct-** An automatic or manual scram fails to shut down the reactor as indicated by **reactor power >4%** and subsequent operator manual actions taken (Hard Card actions of initiating ARI) at the reactor control consoles to shutdown the reactor are also **unsuccessful. Then declare an ALERT (SA6.1).**

D: **Incorrect-**. The RCS iodine did not reach the limit specified in Fission Product Barrier Table (FB3.L).

Technical Reference(s): EAL S6- RPS Failure (Attach if not previously provided)

Fission Product Barrier Table

Proposed References to be provided to applicants during examination: EALs and RALs without the attachments

Learning Objective: ECG/E-Plan/Fire & Medical Questions: Knowledge of the reasons for the following responses as they apply to the implementation of site emergency plan.

Question Source: Bank # 119580

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #		295012 2.1.25
	Importance Rating		4.2

K/A Statement: Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.- High Drywell temperature

Question: SRO #84

Given:

- The plant was operating at 100% power.

When:

- A large reactor coolant leak occurs.
- Suppression chamber pressure rapidly rises.

Current plant conditions:

- Drywell sprays have been initiated.
- Drywell pressure and temperature are lowering.

If the drywell pressure and temperature lowering results in entering the UNSAFE region of the Drywell Spray Initiation Limit (DWT-P) curve, what action is required IAW emergency operating procedures?

A. Secure drywell sprays at 9.5 psig drywell pressure and lowering IAW HC.OP-EO.ZZ-0102, Primary Containment Control.

B. Secure all drywell sprays when the Drywell Spray Initiation Limit curve is reached IAW HC.OP-EO.ZZ-0102, Primary Containment Control.

C. Emergency Depressurize the reactor IAW HC.OP-EO.ZZ-0202, Emergency RPV Depressurization.

D. Continue drywell sprays until drywell pressure approaches 0 psig IAW HC.OP-EO.ZZ-0102, Primary Containment Control.

Proposed Answer: **D**

Explanation (Optional): Drywell spray operation must be terminated by the time drywell pressure decreases to 0 psig to ensure that primary containment pressure is not reduced below atmospheric. Maintaining a positive pressure precludes air intake through the vacuum relief system to de-inert the primary containment and also provides a positive margin to the negative design pressure of the primary containment. (See attached retainment override PCC-1 of EOP-0102). Note that while *operation* of drywell sprays is permitted down to pressures approaching 0 psig, the Drywell Spray Initiation Limit curve prohibits spray *initiation* at low pressures. The curve is for iniating Drywell sprays but not for securing Drywell sprays. This is directed from the retainment override (PCC-1).

Plausibility Justification:

A: **Incorrect-**. RHR is not needed to assure adequate core cooling with the above conditions. Drywell sprays will continue until reaching 0 psig IAW the retainment override (BEFORE Drywell press reaches **0 psig**, **TERMINATE** Drywell sprays) of EOP-102 PCC-1. The 9.5 psig is the threshold for suppression chamber pressure to allow Drywell sprays.

B: **Incorrect-** The DWT-P Drywell Spray Initiation curve is for determining the parameters to allow Drywell sprays, however, the curve does not define when to secure from the Drywell sprays. The retainment override (BEFORE Drywell press reaches **0 psig**, **TERMINATE** Drywell sprays) of EOP-102 PCC-1 directs this action.

C: **Incorrect-** With the Drywell temperature below 340°F and lowering, IAW EOP-102 maintaining drywell temperature below 340°F will preclude entry into EOP-202 and Emergency Depressurizing the reactor.

D: **Correct-**. Once initiated, DW sprays need only be secured BEFORE DW pressure reaches 0 psig. IAW EOP-102, PCC-1, Retainment Override (BEFORE Drywell press reaches **0 psig**, **TERMINATE** Drywell sprays).

Technical Reference(s): HC.OP-EO.ZZ-0102 FC (Attach if not previously provided)

Primary Containment Control

Proposed References to be provided to applicants during examination: DWT-P curve in stem

Learning Objective: Given plant conditions and access to the following curves determine the region of acceptable operation and explain the bases for the curve
Drywell Spray Initiation Limit

Question Source: Bank # 33995

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #		295010 AA2.02
	Importance Rating		3.9

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

Question: SRO #85

Given:

- The plant was operating at 100% power.

When:

- An earthquake causes a Loss of Offsite Power (LOP) coincident with a Loss of Coolant Accident (LOCA).

At T= 0 minutes:

- All rods are fully inserted.
- All EDGs failed to start.
- RPV Level is being maintained between -30" and +30" with RCIC.
- RPV Pressure is being maintained between 500 and 600 psig by cycling SRVs as necessary.
- Suppression Chamber water level is 65 inches, lowering 0.1 inches/minute.
- The cause of the lowering level is being investigated.
- Drywell and Suppression Chamber pressures are 15 psig and rising 1 psig / 3 minutes.
- Drywell temperature is currently 200°F and rising 1°F/ 3 minutes.
- Suppression Pool Temperature is currently 165°F and rising 1°F/ 3 minutes.

At T=10 minutes:

- 'A' and 'B' EDGS are running and loaded.
- The Reactor Operator (RO) reports that the "A" RHR loop and "B" Core Spray Loops are running and available.
- The Operators in the field report that the FLEX Diesel Pump hoses are being run.

Which of the following actions is required for these conditions?

A. Exit HC.OP-EO.ZZ-0105, RPV Control- HPCI/RCIC Only & HC.OP-EO.ZZ-0106, Primary Containment Control-HPCI/RCIC Only. Enter HC.OP-EO.ZZ-0209, Rapid RPV Depressurization. Rapid RPV Depressurization is Required.

B. Enter HC.OP-EO.ZZ-0101, RPV Control & HC.OP-EO.ZZ-0102, Primary Containment Control and take containment control actions.

C. Make up to the Suppression Pool with Service Water from the FLEX Diesel pump.

D. Vent the Drywell IAW HC.OP-EO.ZZ-0318, Containment Venting.

Proposed Answer: **B**

Explanation (Optional): EOP-106, Primary Containment Control-HPCI/RCIC Only PCC-10 and EOP-105, Reactor Pressure Vessel Control- HPCI/RCIC Only RC-11 Retainment Overrides (see attached) are designed to work in conjunction to optimize the use of steam-driven injection systems, and manage containment parameters without the use of motor-driven systems, the exit criteria contained in the override step are identical. This override sends the operator to **EOP 102** which directs the mitigating strategy for events where motor-driven RPV injection systems are available. In this circumstance it is also likely that forced suppression pool cooling and **containment spray systems (drywell spray to control drywell pressure)** will also be available; EOP-102 provides strategies that use these mitigation tools to maintain primary containment viability, Drywell pressure and temperature along with Suppression pool parameters.

Plausibility Justification:

A: **Incorrect-**. With RPV level above -185 inches, Drywell temperature below 340°F, and Suppression Chamber pressure below PSP, rapid depressurization is NOT required.

B: **Correct-** Override step PCC-10 sends the operator to EOP 102 which directs the mitigating strategy for events where motor-driven RPV injection systems are available. In this circumstance it is also likely that forced suppression pool cooling and containment spray systems will also be available; EOP-102 provides strategies that use these mitigation tools to maintain primary containment viability.

C: **Incorrect-** The information given in the stem is the hoses have been run for the FLEX Diesel Pump, so this option is not yet ready to implement.

D: **Incorrect-**. IAW EOP-106, DW/P-33 Retainment Override step states **IF** Drywell pressure reduction is required to restore and maintain adequate core cooling or reduce the total offsite radiation dose, **THEN** VENT Primary Containment. This is NOT required with the given and current conditions.

Technical Reference(s): HC.OP-EO.ZZ-0106 FC (Attach if not previously provided)

HC.OP-EO.ZZ-0105 FC

Retainment Overrides

Proposed References to be provided to applicants during examination: none

Learning Objective: Given plant conditions, recognize the entry conditions for the Primary Containment Control HPCI / RCIC ONLY Emergency Operating Procedure IAW HC.OP-EO.ZZ-0106.

Question Source: Bank # 152011

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		215005 A2.08
	Importance Rating		3.4

K/A Statement: Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions: Faulty or erratic operation of detectors/systems.

Question: SRO #86

Given:

- The plant is operating at 100% rated power.
- With no rods selected on the rod display, the following alarm is received:
    - ➢ LPRM UPSCALE (C3-D5)
- The operator confirms that one LPRM is upscale as shown on the PPC OD-8 (Plant Process Computer).

What subsequent action will have to be taken IAW HC.OP-AB.IC-0004, Neutron Monitoring, after bypassing the failed LPRM and what is the requirement for APRM operability?

A. Direct the reactor engineer to evaluate the failed LPRM.
APRM operability requires a minimum of 4 LPRMs per level.

B. Reset the tripped RPS channel.
APRM operability requires a minimum of 3 LPRMs per level.

C. Direct the reactor engineer to evaluate the failed LPRM.
APRM operability requires a minimum of 3 LPRMs per level.

D. Reset the tripped RPS channel.
APRM operability requires a minimum of 4 LPRMs per level.

Proposed Answer: **C**

Explanation (Optional): See attached HC.OP-AB.IC-0004 and T.S. 3.3.1, Reactor Protection System Instrumentation.

Plausibility Justification:

A: **Incorrect-**. IAW T.S. 3.3.1, an APRM channel is inoperable if there are less than **3 LPRM** inputs per level or less than 20 LPRM inputs to an APRM channel.

B: **Incorrect-** There is no RPS trip for LPRM inputs (administrative operability concern, T.S. 3.3.1).

C: **Correct-** IAW AB.IC-0004 subsequent action C.3 (See attached), the RE will have to evaluate the failed LPRM. There is no RPS trip for LPRM inputs (administrative operability concern). IAW T.S. 3.3.1, an APRM channel is inoperable if there are less than **3 LPRM** inputs per level or less than 20 LPRM inputs to an APRM channel.

D: **Incorrect-**. There is no RPS trip for LPRM inputs (administrative operability concern. IAW T.S. 3.3.1, an APRM channel is inoperable if there are less than **3 LPRM** inputs per level or less than 20 LPRM inputs to an APRM channel.

Technical Reference(s): HC.OP-AB.IC-0004 Neutron Monitoring (Attach if not previously provided)
T.S. 3.3.1 Table/Notes

Reactor Protection System Instrumentation

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a scenario of applicable conditions and access to Technical Specifications:
Select those sections which are applicable to the LPRMS/APRMS
Evaluate LPRM/APRM operability and determine required actions applicable to the APRMS (SRO Only)
Explain the bases for those Technical Specifications associated with the APRMS IAW HCGS Technical Specifications.
Explain the reasons for how plant/system parameters respond when implementing Neutron Monitoring.

Question Source: Bank # #88 2019NRC

Modified Bank # (Note changes or attach parent)

New

Question History: NRC2019

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2,5)

Comments:

2021 NRC Written Examination

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		263000 A2.01
Importance Rating		3.2

K/A Statement: Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Grounds

Question: SRO #87

Given:

- The plant is operating at 85% power.

When:

- A loss of the 10D410, 1E 125VDC bus occurs.
- The DC supply breaker (72-41022) to the AD481 Inverter trips open.
- The applicable abnormal procedures have been entered.
- The cause of the 10D410, 1E 125VDC bus loss is a ground confirmed by the Class 1E Ground Ammeters in the main control room and local indications.
- Repairs are in progress for the 10D410, 1E 125VDC bus.

What is the most limiting, if any, Technical Specification action required?
[Reference attached]

A. Enter a tracking LCO. No Technical Specification LCO entry is required.

B. Re-energize the 10D410 within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

C. Re-energize the AJ481 within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

D. The AD481 must be made OPERABLE within 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: **B**

Explanation (Optional): "Energized" 120 VAC distribution panels [A-D]J48[1/2] require the panels to be energized to their proper voltage from the associated inverter **via inverted DC voltage**, inverter using the normal AC source, or Class 1E backup AC source via voltage regulator (See attached schematic and table). OPERABLE inverters require the associated 120 VAC distribution panels ([A-D]J48[1/2]) to be powered by the inverter with output voltage within tolerances, and power input to the inverter from the associated station battery. Due to the loss of the 'A' channel 1E 125VDC distribution, TS 3.8.3.1 action b (see attached) would be entered and would be most limiting (2 hours). The student will have to analyze the 3.8.3.1 table and determine that the DC distribution is the most limiting. The AD481 will be declared inoperable (7 days), however the AD481 is still energized with the normal and backup AC power.

Plausibility Justification:

A: **Incorrect-**. Due to the loss of the 'A' channel 1E 125VDC distribution, TS 3.8.3.1 action b (see attached) would be entered and would be most limiting (2 hours).

B: **Correct-** Due to the fact that the DC needs to be available for the AD481 inverter to be operable, the CRS would be in the action d for restoring the inverter within 7 days. However; with the loss of the 'A' channel 1E 125VDC distribution, TS 3.8.3.1 action b (see attached) would be entered and would be most limiting (2 hours).

C: **Incorrect-** . The AD481 is still energized with the normal and backup AC power..

D: **Incorrect-**. Due to the fact that the DC needs to be available for the AD481 inverter to be operable, the CRS would be in the action d for restoring the inverter within 7 days. However; with the loss of the 'A' channel 1E 125VDC distribution, TS 3.8.3.1 action b (see attached) would be entered and would be most limiting (2 hours).

Technical Reference(s): T.S. 3.8.3.1 On Site Pwr Distribution (Attach if not previously provided)

HC.OP-SO.PK-0001 DC SOP

HC.OP-AB.ZZ-0136 Loss of Inverter
HC.OP-AB.ZZ-0150 125VDC Malfunction

Proposed References to be provided to applicants during examination: T.S. 3.8.3.1

Learning Objective: Given a D.C. electrical load. a. Determine the power supply to the load. b. Evaluate the effect of a loss of D.C. power for each component

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Facility: Hope Creek
Vendor: GE
Exam Date: 2021
Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		223002 2.4.49
Importance Rating		4.4

K/A Statement: Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. PCIS/Nuclear Steam Supply Shutoff System.

Question: SRO #88

Given:

- The plant was operating at 100% power.

When:

- A valid -38" RPV Level NS4 (Nuclear Steam Supply Shutoff System) signal is received.
- All applicable procedures have been entered.
- The plant operator (PO) reports that the RWCU system did not isolate.
- When attempting to close the BG-HV-F001, RWCU INBOARD ISOLATION, the valve did not move in the closed direction.

What are the required actions IAW plant procedures?

A. BG-HV-F004, RWCU OUTBOARD ISOLATION must be closed and deactivated within 4 hours.

B. BG-HV-F100, F101, and F106 RWCU SUCTION ISOLATIONS must be closed and deactivated within 1 hour.

C. BG-HV-F100, F101, and F106 RWCU SUCTION ISOLATIONS must be closed and deactivated within 4 hours.

D. BG-HV-F004, RWCU OUTBOARD ISOLATION must be closed and deactivated within 1 hour.

Proposed Answer: **A**

Explanation (Optional): With a VALID -38" NS4 signal in, IAW the abnormal procedure for Primary Containment, HC.OP-AB.CONT-0002, the immediate operator actions (I.O.A) are to close either a redundant isolation valve or to manually close the valve that did not isolate (F001). Since the primary containment isolation is to have one inboard and one outboard, the F004 would be the isolation valve that needs to be isolated, if not already isolated. Because of the failure of the F001 to isolate on a valid isolation signal, the action to close and deactivate the redundant valve (F004) is required IAW Technical Specifications 3.6.3 (see attached).

Plausibility Justification:

A: **Correct-**. With the failure of the INBOARD containment valve F001, the I.O.A is to CLOSE the redundant OUTBOARD valve F004. The suction valves F100, 101.and 106 are redundant valves (see attached drawing) to the F001, however all three valves are inboard of containment along with the F001. Therefore containment is NOT isolated. The failure of the F001 will have the CRS enter T.S. 3.6.3 and take the action to isolate the penetration (F001 and F004) and deactivated the isolated valve (F004) within 4 hours.

B: **Incorrect-** The suction valves F100, 101.and 106 are redundant valves (see attached drawing) to the F001, however all three valves are inboard of containment along with the F001. Therefore containment is NOT isolated. The failure of the F001 will have the CRS enter T.S. 3.6.3 and take the action to isolate the penetration (F001 and F004) and deactivated the isolated valve (F004) within 4 hours.

C: **Incorrect-** The suction valves F100, 101.and 106 are redundant valves (see attached drawing) to the F001, however all three valves are inboard of containment along with the F001. Therefore containment is NOT isolated.

D: **Incorrect-**. The failure of the F001 will have the CRS enter T.S. 3.6.3 and take the action to isolate the penetration (F001 and F004) and deactivated the isolated valve (F004) within 4 hours. (see attached T.S.3.6.3).

Technical Reference(s): HC.OP-AB.CONT-0002 Primary Cont. (Attach if not previously provided)

HC.OP-SO.SM-0001 Isolations Sys.

T.S. 3.6.3 PCIVs

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, recall the Immediate Operator Actions for Primary Containment.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		300000 2.4.11
Importance Rating		4.2

K/A Statement: Emergency Procedures / Plan: Knowledge of abnormal condition procedures. Instrument Air

Question: SRO #89

Given:

- The plant is in OPCON 4 having just completed Refueling Operations.
- The Refueling Cavity is being decontaminated.

When:

- The Control Room reports the FUEL POOL COOLING SYS LEAKAGE HI alarm has been received.
- The Rad Pro Supervisor reports a small amount of water coming from the Fuel Pool Shield Blocks.

Which of the following is the cause of this condition ____(1)____ AND
What is the bases behind the procedural requirements for this condition ____(2)____?

A. (1) The normal and emergency air supplies to the seals of the inner and outer Fuel Pool gates have failed.
(2) To ensure removal of 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

B. (1) The Fuel Pool Gate and Shield Block seals normal air supply has failed. The emergency bottles must be manually aligned to supply the seal.
(2) Ensure a large heat sink is available and adequate time is provided to initiate alternate methods capable of heat removal.

C. (1) The normal air supplies to the seals of the inner and outer Fuel Pool Gates have failed. The emergency bottles must be manually aligned to supply the seals.
(2) To ensure removal of 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

D. (1) The normal and emergency air supplies to the seals of the Fuel Pool gate and Shield Block have failed.
(2) Ensure a large heat sink is available and adequate time is provided to initiate alternate methods capable of heat removal.

Proposed Answer: **A**

Explanation (Optional): See attached AB.COOL-0004 subsequent operator action A.3 and T.S. Bases 3.9.9 Water Level- Spent Fuel Pool.

Plausibility Justification:

A: **Correct-**. Due to the fact that the emergency bottles will automatically provide air to the seals and the seals are still leaking both the normal and emergency air supply has failed. Sufficient water depth is needed to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

B: **Incorrect-** Emergency bottles are normally aligned to automatically provide air if the pressure of the normal supply lowers below a set value (Approx. 35#). Sufficient water depth is needed to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

C: **Incorrect-** Emergency bottles are normally aligned to automatically provide air if the pressure of the normal supply lowers below a set value (Approx. 35#).

D: **Incorrect-**. The shield blocks do not have pressurized seals. The Emergency bottles are normally aligned to automatically provide air if the pressure of the normal supply lowers below a set value (Approx.35#). Sufficient water depth is needed to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly.

Technical Reference(s): HC.OP-AB.COOL-0004 Fuel Pool Cooling (Attach if not previously provided)
T.S. bases 3.9.9 Water Level- Spent Fuel pool

Proposed References to be provided to applicants during examination: none

Learning Objective: Given a system or component that is either physically connected to or required for support of the Instrument Air System or emergency instrument air compressor, assess the interrelationship

Question Source: Bank # 72069

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(2,5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #		218000 2.2.44
	Importance Rating		4.4

K/A Statement: Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives effect plant and system conditions: ADS

Question: SRO #90

Given:

- The reactor scrammed from 100% power due to a LOP coincident with a LOCA.
- All EDGs started; however busses 10A401, 10A402, 10A403, and 10A404 all have bus lockouts (due to ground faults).

Current plant conditions:

- Reactor pressure is 400 psig and lowering.
- Reactor water level has reached -130 inches and is stabilizing.
- RCIC is injecting at rated flow.
- HPCI is injecting and reaching rated flow.
- Drywell pressure is 7.5 psig and slowly rising.

Which one of the following is the status of the ADS (Automatic Depressurization System) valves and what actions are required to be directed?

A. The ADS valves are NOT available, inhibit ADS and continue to evaluate the level leg of HC.OP-EO.ZZ-0105, RPV Control - HPCI/RCIC Only.

B. The ADS valves are available, enter HC.OP-EO.ZZ-0209, Rapid RPV Depressurization, and open up to five SRVs to reduce RPV pressure to 150 - 250 psig.

C. The ADS valves are NOT available, enter HC-OP-EO.ZZ-0209, Rapid RPV Depressurization, and use Alternate Depressurization Systems to rapidly depressurize the RPV.

D. The ADS valves are available, inhibit ADS and continue to evaluate the level leg of HC.OP-EO.ZZ-0105, RPV Control - HPCI/RCIC Only.

Proposed Answer: **D**

Explanation (Optional): Automatic ADS initiation is prevented by placing the ADS INHIBIT switches in the INHIBIT position. ADS actuation can impose a severe thermal transient on the RPV and may complicate efforts to control RPV water level. If only steam-driven systems (HPCI/RCIC) are available for injection, ADS actuation may directly lead to loss of adequate core cooling and subsequent core damage. Automatic initiation of ADS is therefore prevented. However; the five SRVs that are part of the ADS system are still available for manual operation.

Plausibility Justification:

A: **Incorrect-**. The ADS valves are available (the auto ADS function is NOT available but the SRVs can be manually opened). Due to the current RPV level stabilizing above -185 inches, entry into EOP-209 is not required at this time (see attached flow charts).

B: **Incorrect-** Due to the current RPV level stabilizing above -185 inches, entry into EOP-209 is not required at this time (see attached flow charts).

C: **Incorrect-** The ADS valves are available (the auto ADS function is NOT available but the SRVs can be manually opened).

D: **Correct-**. The ADS valves are available (the auto ADS function is NOT available but the SRVs can be manually opened). Although no RHR or Core Spray systems are available, HPCI and RCIC are both preferred systems and operating. With level just below -129 inches, ADS must be inhibited and then the level situation evaluated (see attached flow charts).

Technical Reference(s): HC.OP-EO.ZZ-0105FC RPV Control (Attach if not previously provided)

HC.OP-EO.ZZ-0209FC Rapid Depressurization

Proposed References to be provided to applicants during examination: none

Learning Objective: 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.

Question Source: Bank # 120331

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #		215002 A2.03
Importance Rating		3.3

K/A Statement: Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of associated reference APRM channel: BWR-3,4,5

Question: SRO #91

Given:

- The plant is operating at 45% power during a reactor startup.
- Control Rod 30-31 is selected, which is located in the center of the core.
- MCPR is at 1.7.

Then:

- APRM Channel 'B' fails full downscale.

(1) Determine the status of RBM Channel 'B', AND
(2) Which of the following is a required action, if any, IAW Technical Specifications?
[Reference attached]

A. (1) Automatically bypasses rod block inputs.
(2) Verify the reactor is not operating on a Limiting Control Rod Pattern AND restore RBM 'B' to OPERABLE within 24 hours.

B. (1) Generates a downscale trip and rod block.
(2) No action required due to RBM Channel 'B' is still OPERABLE.

C. (1) Generates an INOP trip and rod block.
(2) Verify the reactor is not operating on a Limiting Control Rod Pattern AND restore RBM 'B' to OPERABLE within 24 hours.

D. (1) Automatically shifts to APRM Channel 'D' for its reference APRM input.
(2) No action required due to RBM Channel 'B' is still OPERABLE.

Proposed Answer: **A**

Explanation (Optional): The APRM failing downscale will cause the reference signal to be less than 30%, thereby bypassing the RBM. No RBM trips or Rod Blocks can occur. The APRM must be bypassed using the Joy Stick in order for the Reference APRM transfer to occur. With the given conditions the 'B' APRM will have to be bypassed for the RBM Channel 'B' to see the 'D' APRM. The 'B' RBM will be INOPERABLE until the reference 'D' APRM is the reference APRM for the 'B' RBM. (see attached table of APRM references). See attached T.S. 3.1.4.3 and the COLR (MCPR) for RBM operability.

Plausibility Justification:

A: **Correct-**. 'B' RBM will automatically bypass all rod blocks and trips due to the primary reference APRM 'B' failing downscale with a signal of <30%. The 'B' RBM will be INOPERABLE until the reference 'D' APRM is the reference APRM for the 'B' RBM. The actions of 3.1.4.3a will be applicable with the given MCPR value of < 1.75 (see attached).

B: **Incorrect-** The APRM failing downscale will cause the reference signal to be less than 30%, thereby bypassing the RBM. No RBM trips or Rod Blocks can occur. The 'B' RBM is inoperable.

C: **Incorrect-** The APRM failing downscale will cause the reference signal to be less than 30%, thereby bypassing the RBM. No RBM trips or Rod Blocks can occur. There is also no INOP trip generated from the bypassed RBM

D: **Incorrect-**. The APRM failing downscale will cause the reference signal to be less than 30%, thereby bypassing the RBM. The APRM must be bypassed in order for the Reference APRM transfer to occur. The 'B' RBM will be INOPERABLE until the reference 'D' APRM is the reference APRM for the 'B' RBM. With MCPR < 1.75, the actions of 3.1.4.3a will be applicable.

Technical Reference(s): T.S. 3.1.4.3 RBM/ COLR Sect. 5.2-5.5 (Attach if not previously provided)

HC.OP-SO.SF-0002 RBM Operations

Proposed References to be provided to applicants during examination: T.S. 3.1.4.3 Rod Block Monitor and Sections 5.2-5.5 pages 10-15 of the COLR

Learning Objective: Given a scenario of applicable operating conditions and access to the Technical Specifications:
Select those sections which are applicable to the Rod Block Monitor (RBM) System
Evaluate Rod Block Monitor (RBM) System operability and determine required actions based upon system inoperability.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2,5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	290003	2.4.31
	Importance Rating		4.1

K/A Statement: Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures. Control Room HVAC

Question: SRO #92

Given:

- The plant was operating at 100% rated power.

When:

- A pipe break causes a valid high drywell pressure of 8 psig.

Then:

- Two (2) hours after the LOCA signal is received, smoke is detected in the outside air supply to the Control Room.
- HC.OP-AB.HVAC-0002, Control Room Environment was entered.

Current plant conditions:

- The control room has been evacuated and control is established at the Remote Shutdown Panel (10C399).
- The reactor has been depressurized to less than 80 psig with SRV's and RCIC.
- 'A' RHR is in Suppression Pool Cooling.

(1) Which of the following describes the Control Area Ventilation System response and any required operator actions IAW HC.OP-AB.HVAC-0002, Control Room Environment?

(2) What loop of the RHR System will be used to achieve Cold Shutdown, and what is the maximum cooldown rate IAW HC.OP-IO.ZZ-0008, Shutdown from Outside the Control Room?

A. (1) CAVS remains in the Outside Air Mode. Operator action is required to place CREF in RECIRC.
(2) 'A' RHR will be secured from Suppression Pool Cooling and placed in Shutdown Cooling with a cooldown rate that will NOT exceed 100°F/Hour.

B. (1) CAVS remains in the Outside Air Mode. The smoke will be removed by the CREF Units. Operator action is required to isolate the CAVS.
(2) 'A' RHR will be secured from Suppression Pool Cooling and placed in Shutdown Cooling with a cooldown rate that will NOT exceed 90°F/Hour.

C. (1) CAVS remains in the Outside Air Mode. The smoke will be removed by the CREF Units. Operator action is required to isolate the CAVS.
(2) 'B' RHR will be placed in Shutdown Cooling with a cooldown rate that will NOT exceed 100°F/Hour.

D. (1) CAVS remains in the Outside Air Mode. Operator action is required to place CREF in RECIRC.
(2) 'B' RHR will be placed in Shutdown Cooling with a cooldown rate that will NOT exceed 90°F/Hour.

Proposed Answer: **D**

Explanation (Optional): Smoke entering the control room is a retainment override to isolate CAVS and place CREF in RECIRC (See attached HC.OP-AB.HVAC-0002). With the control room evacuated and control at the Remote Shutdown Panel, the CRS would order the 'A' RHR pump to remain in Suppression Pool Cooling mode and place the 'B' RHR pump in Shutdown Cooling with a cooldown rate of < 90°F/Hour. (See attached HC.OP-IO.ZZ-0008).

Plausibility Justification:

A: **Incorrect-**. IAW the retainment override of AB.HVAC-0002, the CREF unit would have to be taken to RECIRC. 'A' RHR would remain in Suppression Pool Cooling and 'B' RHR would be placed in SDC with a cooldown rate of <90°F/Hour.

B: **Incorrect-** CREF units are not designed to remove smoke. CAVS is automatically isolated on the LOCA signal (8 psig drywell pressure). 'A' RHR would remain in Suppression Pool Cooling and 'B' RHR would be placed in SDC.

C: **Incorrect-** CREF units are not designed to remove smoke. CAVS is automatically isolated on the LOCA signal (8 psig drywell pressure). 'B' RHR would be placed in SDC with a cooldown rate of <90°F/Hour.

D: **Correct-**. IAW the retainment override of AB.HVAC-0002, the CREF unit would have to be taken to RECIRC. 'B' RHR would be placed in SDC with a cooldown rate of <90°F/Hour.

Technical Reference(s): HC.OP-AB.HVAC-0002 Contro Room Environment (Attach if not previously provided)
HC.OP-IO.ZZ-0008 Shutdown from Outside the Control Room
HC.OP-SO.GK-0001 CAVS

Proposed References to be provided to applicants during examination: none

Learning Objective: From memory, describe the response of the Control Area Ventilation System to any of the following conditions:
LOCA
Apply Precautions, Limitations and Notes while executing the SHUTDOWN FROM OUTSIDE THE CONTROL ROOM.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #		272000 2.4.30
	Importance Rating		4.1

K/A Statement: 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: Radiation Monitoring.

Question: SRO #93

Given:

- The plant is in Operational Condition 4 for Reactor Vessel disassembly.

When, due to the mishandling of the Reactor Vessel head insulation package:

- All 3 channels of Refuel Floor Exhaust Radiation Monitoring System (RMS) alarm HIGH on the RM-11.
- Primary Containment Isolation System (PCIS) responds as designed.

With the above conditions and IAW 10 CFR 50.72, what is the earliest reporting requirement, If any?

A. 1 hour report

B. 4 hour report

C. 8 hour report

D. NOT reportable

Proposed Answer: **C**

Explanation (Optional): See attached RAL 11.3, the RAL basis for 11.3 System Actuations, which includes a list of reportable systems.
Plausibility Justification:

A: **Incorrect-**. With all three Refuel Floor Exhaust Radiation Monitors in High alarm, EAL RA2.2 Abnormal Rad Levels would be reviewed. Due to the fact that there is no irradiated fuel damage, there would be no need to make an ECG call (see attached EAL), therefore the 1 hour report to the NRC is not warranted.

B: **Incorrect-** The 4 hour report (11.3.1) would be valid if there was a ECCS actuation, however; the High Rad signal from the Refuel Floor Exhaust Radiation Monitors is not setpoint for any ECCS actuation.

C: **Correct-** Valid actuation of listed systems listed in the RAL 11.3 Basis. The RFE RMS responded to a **valid** Hi radiation condition due to the mishandled insulation package. The actuations were not part of a pre-planned test (see attached RAL). Therefore, the High Rad/PCIS actuation is reportable (11.3.3).

D: **Incorrect-**. If the student decides that the actuation is invalid then the report would not be required. 10 CFR 50.72 does not require an event report for invalid actuations of any of the systems listed in the RAL Basis 11.3. 10 CFR 50.73 allows telephone notification to the NRC operations center within sixty days (see attached 11.3.4).

Technical Reference(s): RAL 11.3.3 (Attach if not previously provided)

RAL Basis 11.3 System Actuations

EAL RA2.2

Proposed References to be provided to applicants during examination: EALs and RALs without the attachments

Learning Objective: ECG/E-Plan/Fire & Medical Questions: Knowledge of the reasons for the following responses as they apply to the implementation of site emergency plan.

Question Source: Bank # 34118

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		2.1.45
	Importance Rating		4.3

K/A Statement: Conduct of Operations: Ability to identify and interpret diverse indications to validate the response of another indicator.

Question: SRO #94

Given:

- The plant was operating at 100% power.

When:

- A loss of 10A110 7.2KV Bus due to a bus differential overcurrent occurred during the previous shift.

Current plant conditions:

- The plant has subsequently stabilized at 55% power.
- The Reactor Operator performed panel walkdowns to assess plant status after the transient.
- When performing the next hourly panel walkdown the Reactor Operator notices that Core Plate dP indication and steam flow indication are lower than they were last hour.
- HC.OP-AB.RPV-0003, Recirculation System/Power Oscillations is entered.

As the control room supervisor what actions do you take IAW HC.OP-AB.RPV-0003?

A. Declare the jet pumps inoperable, and commence a unit shutdown per HC.OP-IO.ZZ-0004, be in hot shutdown within 12 hours.

B. Direct the reactor operator to perform HC.OP-ST.BB-0007, Recirculation Jet Pump Operability – Single Loop Daily.

C. Direct the reactor operator to perform HC.OP-ST.BB-0001, Recirculation Jet Pump Operability.

D. Declare the jet pumps inoperable, and immediately insert a full scram, be in hot shutdown within one hour.

Proposed Answer: **B**

Explanation (Optional): The plant is in single loop operation due to loss of the 'A' Recirc pump on the loss of power (10A110 7.2Kv bus). With the loss of the recirc. pump, the core plate dP indication would lower to due to the total core flow lowering. Reactor power would lower due to the lowering of total core flow. Steam flow would lower due to Reactor power lowering. The multiple indications that the crew would observe due to the transient would be normal for the given transient. The abnormal procedure would have the CRS direct the jet pump operability surveillance for single loop operations.

Plausibility Justification:

A: **Incorrect-**. IAW RPV-0003 (see attached), if the jet pumps are inoperable then a normal plant shutdown IAW IO-0004 would be directed and IAW tech specs (see attached) the plant would be in Hot Shutdown within 12 hours. Due to the fact that these are normal multiple indications to support the given reactor power transient, the CRS would only direct a Jet Pump operability surveillance IAW RPV-0003.

B: **Correct-** Due to the fact that these are normal multiple indications to support the given reactor power transient, the CRS would only direct a Jet Pump operability surveillance IAW RPV-0003. With the loss of the 'A' Recirc pump (10A110 7.2 Kv supply to the VFD), the single loop jet pump operability surveillance would be directed.

C: **Incorrect-** With the loss of the 'A' Recirc pump (10A110 7.2 Kv supply to the VFD), the single loop jet pump operability surveillance would be directed not the ST.BB-0001 for normal two loop operations.

D: **Incorrect-**. If the jet pumps are inoperable then a normal plant shutdown IAW IO-0004 would be directed and IAW tech specs (see attached) the plant would be in Hot Shutdown within 12 hours.

Technical Reference(s): HC.OP-AB.RPV-0003 Recirculation (Attach if not previously provided)

HC.OP-ST.BB-0007 Single Loop (Jet)

T.S. 3.4.1.2 Jet Pumps

Proposed References to be provided to applicants during examination: none

Learning Objective: Explain the parameters that are compared to determine jet pump operability and identify symptoms of a jet pump failure

Question Source: Bank # 2016 NRC Exam

Modified Bank # (Note changes or attach parent)

New

Question History: 2016 NRC Exam

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2021
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.2.11	
	Importance Rating		3.3

K/A Statement: Equipment Control: Knowledge of the process for controlling temporary design changes.

Question: SRO #95

Which of the following modifications would be installed IAW CC-AA-112, Temporary Configuration Changes?

A. Installation of a pressure gauge on an instrument taps during the conduct of a system pressure test.

B. Installation of a jumper to perform a periodic surveillance test.

C. Jumpering of a battery cell.

D. Hookup of an air supply hose to a station air manifold during maintenance.

Proposed Answer: **C**

Explanation (Optional): All distracters are not considered Temporary Modifications as described in CC-AA-112 (See attached CC-AA-112).

Plausibility Justification:

A: **Incorrect-**. Covered as part of a test and is not a temporary modification.

B: **Incorrect-** Covered as part of surveillance testing and is not a temporary modification.

C: **Correct-** IAW CC-AA-112 jumpering of a battery cell is NOT consider an exclusion from a temporary change.

D: **Incorrect-**. Covered as maintenance and is not a temporary modification.

Technical Reference(s): CC-AA-112 (Attach if not previously provided)

Temporary Configuration Changes

Proposed References to be provided to applicants during examination: none

Learning Objective: Provided access to control room references. Determine the approved review and extension requirements for installed Temporary Configuration change packages IAW CC-AA-112

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #		2.3.6
Importance Rating		3.8

K/A Statement: Radiation Control: Ability to approve release permits.

Question: SRO #96

Given:

- The plant is in Operational Condition 3 - Hot Shutdown, going to Cold Shutdown.
- The reason for shutdown was excessive unidentified RCS leakage.

Current plant conditions:

- Reactor pressure is at 920 psig.
- Drywell Oxygen concentration is at 2.5%.
- Primary Containment Gaseous Effluent Release permit has been obtained.
- De-inerting will begin at 0700 on day shift 02/21/21.

As the CRS, you are given the attached Valve Permit Form 2 from the NCO for review.

Based on the review of the Valve Permit Form 2 and IAW OP-HC-103-105, Administrative Control of Containment Atmosphere Control (GS) Valve Open Time, you should __________________. **[Reference attached]**

A. not approve it. The hours authorized this date should be 24.

B. approve it as written since only 17 hours remain in the day.

C. not approve it. The hours available this date should be 410.

D. approve it as written since 410 hours will not exceed any limits.

Proposed Answer: **A**

FORM 2
CONTAINMENT PREPURGE CLEANUP, INERTING, OR PRESSURE CONTROL VALVE PERMIT

SECTION A

Date: 02/21/21 **NOTE**: This permit is valid only until 2400 of this date

Effluent Permit #: 21-001

SECTION B

HOURS VALVES/LINES OPEN PREVIOUS YEAR (Note 1)

Calculate Total Hours Open During Previous Year **(NOTE 1)**

DATE	NUMBER OF HOURS
10/24/20	14.0
10/25/20	11.0
12/14/20	8.0
12/15/20	9.0

(1) Max. allowed for 365 days (Admin Limit)	**452 hrs**
(2) Total previous year (**NOTE 1**)	(-) 42.0
Hours available this date (line 1 minus line 2)	(=) 410.0
Hours authorized this date (24 hours or the hours available this date whichever is less)	17.0

NCO performing calculation	Date/Time
J. Smith	02/21/21/ 0600
SM/CRS verification and authorization	**Date/Time**

SECTION C

VALVE/LINE OPEN TIME (Note 2)

START TIME Time at which valve/line was open or Condition 1, 2, or 3 was entered with valve/line open	**STOP TIME** Time at which valve/line was closed or Condition 4 or 5 was entered with valve/line opened	**TOTAL HOURS** Total number of hours valve/line opened this cycle **(NOTE 3)**

Total number of hours valves/line open this permit: ______

NCO performing calculations ______ Date/Time ______

SM/CRS Closing permit ______ Date/Time ______

NOTE 1: The previous year includes the period from 2400 on today's date back to 0001 on the same date one year earlier.

NOTE 2: Completed Form 2 should be filed in the AP-104 binder in the Control Room.

NOTE 3: When computing the total hours (round up to the nearest 0.5 hr or to the nearest 1.0 hr)

Explanation (Optional): See attached section of OP-HC-103-105 and Form 2.

Plausibility Justification:

A: **Correct-**. The hours authorized this date should be 24. - Per direction in OP-HC-103-105. (See attached Form 2 Section B)

B: **Incorrect-** The hours authorized this date should be 24. - Per direction in OP-HC-103-105.

C: **Incorrect-** The hours authorized this date may exceed the actual hours remaining in the day for which the permit was prepared. However; the hours should be 24 hours or hours available on this date, whichever is less. (See attached Form 2 Section B).

D: **Incorrect-**.The hours should be 24 hours or hours available, whichever is less.

Technical Reference(s): OP-HC-103-105 (Attach if not previously provided)

GS Valve Open Time

Proposed References to be provided to applicants during examination: OP-HC-103-105 Form 2 Valve Permit

Learning Objective: Evaluate Containment Inerting and Purge System operability and determine required actions based upon system inoperability.

Question Source: Bank # 111231

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.40	
	Importance Rating		4.5

K/A Statement: Emergency Procedures/Plan: Knowledge of SRO responsibilities in emergency plan implementation.

Question: SRO #97

An Alert has just been declared. IAW, NC.EP-EP.ZZ-0102 , Emergency Coordinator Response, when is personnel accountability performed?

A. Always during an Alert.

B. At the Shift Manager/ Emergency Duty Officer (EDO) discretion during the Alert.

C. During the Alert ONLY if fuel damage has occurred or high radiation levels are identified.

D. During the Alert ONLY on a loss of one or more fuel barriers.

Proposed Answer: **B**

Explanation (Optional): See attached NC.EP-EP.ZZ-0102, Emergency Coordinator Response.
Plausibility Justification:

A: **Incorrect-**. It is at the discretion of the Emergency Coordinator to implement accountability when in an Alert level but it is required at the SAE or GE level.

B: **Correct-** At the Alert level the SM or EDO (depending on who is the EC at the time), can implement accountability IAW EP-EP-102.

C: **Incorrect-** The student could decipher that the accountability is optional until the Alert is due to a high radiation condition or if fuel damage occurs.

D: **Incorrect-**. The student could decipher that the accountability is optional until the level of the Alert elevates to a higher point Alert from the fuel barrier table.

Technical Reference(s): NC.EP-EP.ZZ-0102(Q) Emergency Coordinator Response (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ECG/E-Plan/Fire & Medical Questions: Knowledge of the reasons for the following responses as they apply to the implementation of site emergency plan.

Question Source: Bank # 34190

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.23	
	Importance Rating		4.4

K/A Statement: Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question: SRO #98

Given:

- The plant is in Operational Condition 4 with preparations to enter Operational Condition 5.
- All systems and equipment required to enter Operational Condition 5 are operable.
- Reactor Pressure Vessel (RPV) Metal Temperatures are being recorded on a 30 minute interval.
- All departments have signed for the system requirements to enter Operational Condition 5 IAW HC.OP-IO.ZZ-0005, Cold Shutdown to Refueling.

Which one of the following personnel grants permission to begin de-tensioning of the first RPV Head Stud IAW HC.OP-IO.ZZ-0005, Cold Shutdown to Refueling?

A. Refueling Outage Manager

B. Shift Manager (SM)

C. Work Control SRO

D. Shift Operations Manager (SOM)

Proposed Answer: **B**

Explanation (Optional): See attached Caution of HC.OP-IO.ZZ-0005, Cold Shutdown to Refueling.
Plausibility Justification:

A: **Incorrect-**. Even though the refueling outage manager has responsibilities for the entire outage execution, the refueling outage manager is not an active NRC license holder. The Shift Manager has the ultimate responsibility of the plant and the configuration of the plant systems and operations and therefore any mode changes.

B: **Correct-** The Shift Manager is an active NRC license holder, and IAW HC.OP-IO.ZZ-0005 is the authorizing manager to de-tension the first head bolt and enter OPCON 5. This authorization can also come from the on duty CRS; however that is not one of the choices. By procedure the sign off is SM/CRS.

C: **Incorrect-** The Work Control SRO is also an NRC licensed holder and on shift, however the Work Control SRO would not be in the position of CRS, therefore the Work Control SRO would not authorize the mode change.

D: **Incorrect-**. The SOM is also an NRC license holder, however, the SOM will not be part of the shift compliment and the responsibility would fall on the Shift (SM/CRS) IAW HC.OP-IO.ZZ-0005.

Technical Reference(s): HC.OP-IO.ZZ-0005(Q) (Attach if not previously provided)

Cold Shutdown to Refueling

Proposed References to be provided to applicants during examination: none

Learning Objective: Apply Precautions, Limitations and Notes while executing the COLD SHUTDOWN TO REFUELING Integrated Operating Procedure.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(6)

Comments:

Facility: Hope Creek
Vendor: GE
Exam Date: 2021
Exam Type: SRO

Examination Outline Cross-reference:

	RO	SRO
Level		
Tier #		3
Group #		
K/A #	2.2.7	
Importance Rating		3.6

K/A Statement: Equipment Control: Knowledge of the process for conducting special or infrequent tests.

Question: SRO #99

Given:

- The plant is operating at 100% power.
- It has been determined that work must be performed on the 6A Feedwater Heater level controller.
- The work could result in an unplanned load reduction of 50 MWe.

The evolution is ____ (1) ____ IAW WC-AA-105, Work Activity Risk Management.

IAW OP-AA-108-110, Evaluation of Special Tests or Evolutions, the ____ (2) ____ shall conduct the Heightened Level of Awareness/Infrequent Plant Activity (HLA/IPA) briefing prior to performing a special test or evolution.

A. (1) a production risk activity and an HLA/IPA brief is required
(2) Control Room Supervisor (CRS), or designee

B. (1) NOT a production risk activity but an HLA/IPA brief is required.
(2) Responsible Senior Line Manager (SLM), or designee

C. (1) NOT a production risk activity but an HLA/IPA brief is required.
(2) Control Room Supervisor (CRS), or designee

D. (1) a production risk activity and a HLA/IPA brief is required
(2) Responsible Senior Line Manager (SLM), or designee

Proposed Answer: **D**

Explanation (Optional): IAW WC-AA-105 – Step 2.7 defines production risk activity as >20 MWe (see attached). Then an HLA/IPA will be performed and conducted by the SLM or designee IAW OP-AA-108-110 step 4.3.3 (see attached).

Plausibility Justification:

A: **Incorrect-**. The 30MWe change in load would constitute a production risk activity which would require an HLA/IPA review and briefing; however the pre-job brief will be conducted by the responsible SLM or designee. The operations management (SM/CRS) shall **ENSURE** required plant conditions are maintained as required for the special test or evolution.

B: **Incorrect-** The 30MWe change in load would constitute a production risk activity (see attached WC-AA-105). The SLM or his designee shall **CONDUCT** a HLA/IPA briefing prior to performing the special test or evolution (see attached OP-AA-108-110).

C: **Incorrect-** The 30MWe change in load would constitute a production risk activity. The operations management (SM/CRS) shall **ENSURE** required plant conditions are maintained as required for the special test or evolution.

D: **Correct-**. The 30MWe change in load would constitute a production risk activity which would require an HLA/IPA review and briefing. The SLM or his designee shall **CONDUCT** a HLA/IPA briefing prior to performing the special test or evolution.

Technical Reference(s): WC-AA-105, Work Activity Risk Management (Attach if not previously provided)
OP-AA-108-110, Evaluation of Special Tests or Evolutions

Proposed References to be provided to applicants during examination: none

Learning Objective: Determine if an activity meets the criteria for a Special Test or Evolution. IAW OP-AA-108-110.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History:

Question Cognitive Level: Memory or Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

2021 NRC Written Examination

Facility: Hope Creek

Vendor: GE

Exam Date: 2021

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.6	
	Importance Rating		4.7

K/A Statement: Emergency Procedures/ Plan: Knowledge of EOP mitigation strategies.

Question: SRO #100

Given:

- The plant was operating at 100% power.

When:

- A Large Break LOCA occurred in the Drywell concurrent with a LOP.

Current plant conditions:

- Only 'C' EDG is running.
- All control rods are fully inserted.
- Wide Range RPV level indicator LR-623A is reading +20" and lowering.
- Wide Range RPV level indicator LR-623B is reading -55" and lowering.
- Drywell pressure is 29 psig and rising.
- Drywell temperature is 350°F and rising.
- Reactor pressure is 10 psig and steady.
- Suppression Pool Level is 80" and rising.
- Suppression Chamber pressure is 30 psig and rising.
- 'C RHR Pump has been injecting Low Pressure Coolant Injection (LPCI) flow for 3 minutes.

Based on the above conditions, which one of the following actions is required IAW emergency operating procedures? **[Reference attached]**

A. Continue LPCI injection and continue in all control legs of HC.OP-EO.ZZ-0101, RPV Control.

B. Stop LPCI injection, emergency depressurize the reactor pressure vessel IAW HC.OP-EO.ZZ-0202, Emergency RPV Depressurization and then resume LPCI injection.

C. Continue LPCI injection and enter HC.OP-EO.ZZ-0206, RPV Flooding.

D. Continue LPCI injection, emergency depressurize the reactor pressure vessel IAW HC.OP-EO.ZZ-0202, Emergency RPV Depressurization and then enter HC.OP-EO.ZZ-0206, RPV Flooding.

Proposed Answer: **C**

Explanation (Optional): Channels A AND B of the Wide, Narrow and Upset RPV water level instruments provide the most reliable indications. RPV level instrumentation channels A and B have been designed with minimal, vertical-run drops in the Drywell. Therefore, these channels are least affected by elevated Drywell temperature. EOP Caution 1 defines conditions under which neither the displayed value nor the indicated trend of RPV water level can be relied upon. With High drywell temperature and low RPV pressure, per EOP caution 1 level is unreliable. Therefore RPV level it is not known and RPV flooding is required IAW EOP-206 RPV Flooding as the mitigation strategy. To assure adequate core cooling the 'C' RHR pump would continue in the LPCI mode of operation.

Plausibility Justification:

A: **Incorrect-**. RPV flooding is required. Pressure Control Leg of EOP-101 is exited when EOP-206 is required. This override sends the operator to EOP-206 which directs the mitigation strategy for RPV flooding (See attached EOP-101 RC/L-2 Retainment Override).

B: **Incorrect-** With the current conditions, the retainment override would be taken from EOP-101 to enter EOP-206 and flood the vessel (open 5 SRVs). The emergency depressurization would occur if level is known and cannot be maintained above -185" IAW EOP-101. LPCI injection would not be secured to assure adequate core cooling.

C: **Correct-** RPV flooding is required. Pressure Control Leg of EOP-101 is exited when EOP-206 is required. This override sends the operator to EOP-206 which directs the mitigation strategy for RPV flooding. LPCI injection would not be secured to assure adequate core cooling.

D: **Incorrect-**. With the current conditions, the retainment override would be taken from EOP-101 to enter EOP-206 and flood the vessel (open 5 SRVs).

Technical Reference(s): HC.OP-EO.ZZ-0101 RC/L-2 (Attach if not previously provided)

HC.OP-EO.ZZ-0206 RPV Flooding

EOP-CAUTION #1

Proposed References to be provided to applicants during examination: EOP-CAUTION #1

Learning Objective: 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.

Question Source: Bank # 120175

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: