Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K1.01	
	Importance Rating	3.8	

K1.01 - Knowledge of the physical connections and/or cause - effect relationships between A.C. Electrical Distribution and Emergency generators (diesel/jet)

Question: #1

Given:

- The plant is at rated power.
- The TRIP pushbutton is pressed for breaker 52-40201, Normal Feed Breaker for 10A402 on Control Room panel 10C651E.

Which choice below describes the response of the 10A402 Bus and "B" EDG?

A. Bus 10A402 will be de-energized. The "B" EDG will NOT be running.

B. Bus 10A402 will be de-energized.
The "B" EDG will be running with its output breaker open.

C. The "B" EDG will start and its output breaker will close energizing Bus 10A402.

D. The Alternate Feed Breaker, 52-40208 will close energizing Bus 10A402.
"B" EDG Lockout will prevent the EDG start and output breaker closure.

Proposed Answer: A

Explanation (Optional):

A: Correct – Bus 10A402 will be de-energized. The B EDG will NOT be running. The

automatic transfer to the alternate feed and the start of the diesel will not occur if the normal breaker is manually tripped.

B: Incorrect – The automatic start of the Diesel will not occur if the normal breaker is manually tripped.

C: Incorrect – The automatic start of the Diesel will not occur if the normal breaker is manually tripped.

D: Incorrect – The automatic transfer to the alternate feed will not occur if the normal breaker is manually tripped.

Technical Reference(s):
HC.OP-SO.PB-0001
Sect. 5.5.6.

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 35873

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K1.08	
	Importance Rating	3.0	

206000 HPCI
K1.08 - Knowledge of the physical connections and/or cause effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and AC power.

Question: #2

Given:

The plant is at rated power.

Then, MCC 10B212 de-energizes due to an electrical fault.

How is HPCI affected?

A loss of power occurs to the ____________ .

A. HPCI auxiliary oil pump

B. HPCI jockey pump

C. HPCI barometric condenser condensate pump

D. HPCI turbine steam supply isolation valve HV-F001

Proposed Answer: B

Explanation (Optional):

A: Incorrect. HPCI Auxiliary Oil Pump OP-213 is a DC electric motor driven pump which is powered from the 250VDC 1E MCC 10D251. Plausible if candidate does not recall that this is a DC pump which is not powered by an AC power supply.

B: Correct. MCC 10B212 is 480VAC. HPCI Jockey Pump AP-228 is an AC electric motor

driven pump which is powered from the 480VAC 1E MCC 10B212. A loss of keepfill capability would result if this pump lost electrical power.

C: Incorrect. HPCI barometric condenser condensate pump OP-215 is a DC electric motor driven pump which is powered from the 250VDC 1E MCC 10D251. Plausible if candidate does not recall that this is a DC pump.

D: Incorrect. HPCI turbine steam supply isolation valve HV-F001 is a DC powered motor operated valve, which is powered from the 250VDC 1E MCC 10D251. Plausible if candidate does not recall that this is a DC powered valve. This valve can be confused with HPCI outboard steam supply isolation valve HV-F003, which is powered from 10B212.

Technical Reference(s):
Lesson Plan
NOH01HPC100-12
HPCI system

Proposed References to be provided to applicants during examination: None

Learning Objective:
HPCI00E009d

Question Source: New

Question History:
None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 K2.01	
	Importance Rating	3.1	

Knowledge of electrical power supplies to ADS logic

Question: #3

Given:

- The ADS Manual Initiation Channel B and F push buttons have been armed and depressed.
- There is NO Safety Relief Valve response

Which of the following electrical failures would cause this system response?

A. 125 VDC bus 1BD417

B. 125 VDC bus 1DD417

C. 120 VAC bus 1BJ481

D. 120 VAC bus 1DJ481

Proposed Answer: A

Explanation (Optional):

A: Correct – power supply to ADS logic B&F

B: Incorrect – Power supply to ADS logic D&H

C: Incorrect – Power supply to B RHR and CS instrumentation

D: Incorrect – Power supply to D RHR and CS instrumentation

Technical Reference(s):
PN1-B21-1060-63
HC.OP-SO.SN-0001

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 30453

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K2.01	
	Importance Rating	3.1	

205000 Shutdown Cooling
K2.01 - Knowledge of electrical power supplies to pump motors.

Question: #4

With respect to the power supply and the ability to control from the Remote Shutdown Panel (RSP), which of the following is associated with the 'A' RHR Pump AP202?

	Power Supply	RSP Control
A.	4.16 kV 10A401	NO
B.	4.16 kV 10A402	NO
C.	4.16 kV 10A401	YES
D.	4.16 kV 10A402	YES

A: Correct. 'A' RHR pump AP202 is powered from 10A401 and cannot be controlled from the RSP.

B: Incorrect. While 'A' RHR pump AP202 is powered from 10A401, it cannot be controlled from the RSP.

C: Incorrect. 'A' RHR pump AP202 is not powered from 10A402 and cannot be controlled from the RSP.

D: Incorrect. . 'A' RHR pump AP202 is not powered from 10A402 and cannot be controlled from the RSP.

Answer: A

Technical Reference(s):
NOH01REMS/D-02

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Susquehanna Exam
Bank
Sig Mod'd

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K3.06	
	Importance Rating	4	

Knowledge of the effect that a loss or malfunction of the reactor protection system will have on scram air header solenoid operated valves

Question: #5

Given:

- The plant is operating at 92% power.
- An 'A' APRM surveillance is in progress.
- The RO notes that one of the RPS Group 1 logic "B" white lights located below the individual RPS manual scram keyswitches on 10C651C is not illuminated.
- An investigation reveals the fuse protecting this circuit has blown and it will take 30 minutes to replace the fuse.
- The I & C technician performing the 'A' APRM surveillance now requests permission to bring in an RPS 'A' half scram from his test.

IMMEDIATELY following the insertion of the 'A' side RPS ½ scram, what will be the condition of the control rod scram solenoids?

A. One half of all the control rods will have one of their individual scram solenoids de-energized.

B. Approximately one half of all the control rods will have both of their individual scram solenoids de-energized.

C. Approximately one quarter of all the control rods will have both of their individual scram solenoids de-energized.

D. All of the control rods will have both of their individual scram solenoids de-energized.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – 4 groups of solenoids, with each group servicing ¼ of the rods

B: Incorrect – 4 groups of solenoids, with each group servicing ¼ of the rods

C: Correct - Approximately one quarter of all the control rods will have both of their individual scram solenoids de-energized (see below)

D: Incorrect – 4 groups of solenoids, with each group servicing ¼ of the rods

Scram Pilot Solenoid Valves

There are 185 Scram Pilot Solenoid Valves divided into 4 groups:

1) Group 1 Solenoid A powered from CB2A (B from CB3B) (45 valves)
2) Group 2 Solenoid A powered from CB3A (B from CB8B) (45 valves)
3) Group 3 Solenoid A powered from CB8A (B from CB7B) (47 valves)
4) Group 4 Solenoid A powered from CB7A (B from CB2B) (48 valves)

Eight lights (2 per group). Each light corresponds to a group's solenoid valve (A or B). One 'B' light extinguished would indicate the 'B' solenoid valve in one group (~a quarter of all the rods) is deenergized. The 'A' APRM half scram would deenergize all 'A' solenoids in all 4 rod groups. The end result is a quarter of the rods having both solenoid valves deenergized.

Technical Reference(s): NOH01RPS00C-09
HC.OP-SO.SB-0001

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 33955

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K3.02	
	Importance Rating	3.3	

300000 Instrument Air

K3.02 - Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following: Systems having pneumatic valves and controls

Question: #6

Which of the following describes the reason control rods insert during a loss of instrument air?

A. A flowpath is ONLY opened to the bottom of the drive mechanism operating piston allowing reactor pressure to insert the rod.

B. The scram flowpath to and from the drive mechanism operating piston is opened, allowing accumulator and reactor pressure to insert the rod.

C. Drive water flow and pressure both rise enough to insert the rod.

D. A flowpath is ONLY opened from the top of the drive mechanism operating piston allowing accumulator pressure to insert the rod.

Proposed Answer: B

Explanation:

A: Incorrect. Both scram valves drift open, not just one.

B: Correct. Loss of air causes the scram inlet and outlet to slowly open, normal scram flowpath eventually so scram source is the accumulator then the reactor.

C: Incorrect. CRD FCVs fail closed on a loss of Instrument Air.

D: Incorrect. Both scram valves drift open, not just one.

Technical Reference(s):

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 29830

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

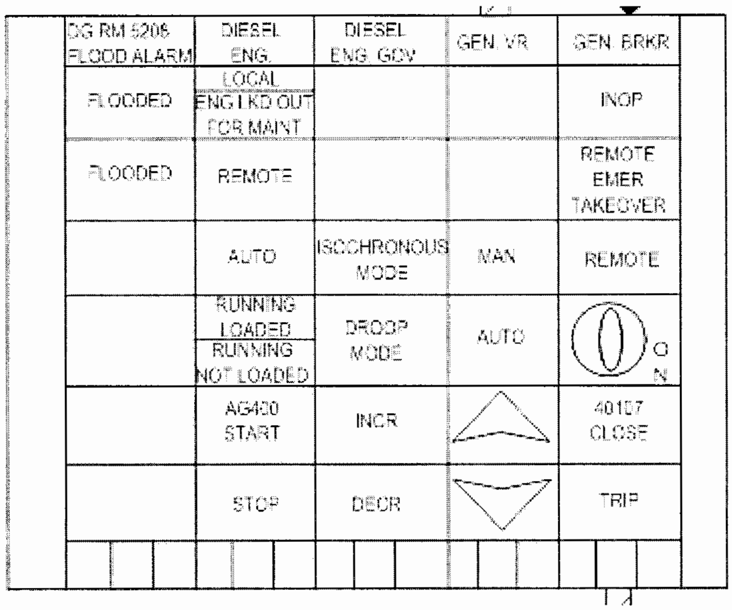
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 K4.07	
	Importance Rating	3.3	

Knowledge of emergency Generators (diesel/jet) design feature(s) and/or interlocks which provide for local operation and control

Question: #7

The Control Room operator is preparing the "A" Emergency Diesel Generator for a maintenance retest manual start IAW HC.OP-SO.KJ-0001(Q).

Which of the following Control Room indications would identify that a manual start from the control room was inhibited?



A. AUTO light is lit.

B. LOCAL light is lit.

C. REMOTE light is extinguished.

D. ENG LKD OUT FOR MAINT is extinguished.

Proposed Answer: B

Explanation (Optional):

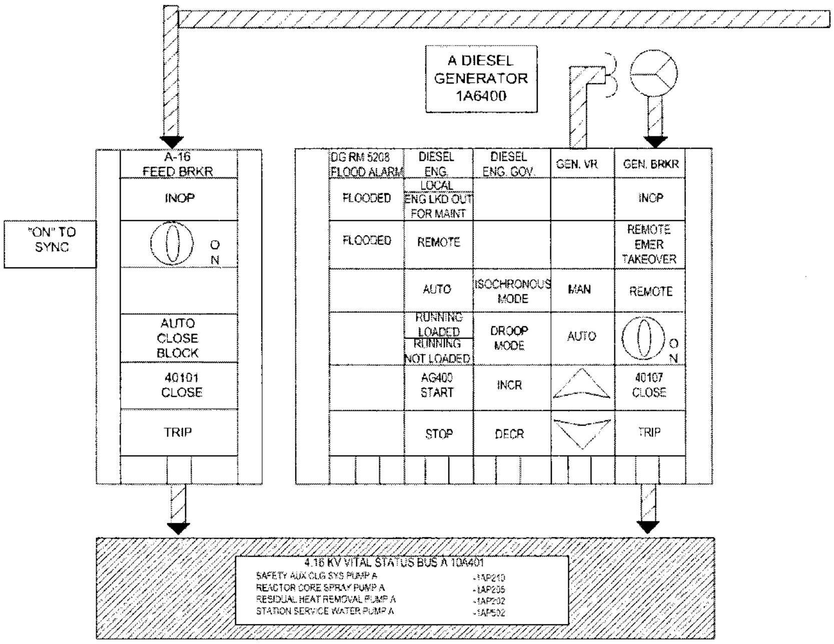
A: Incorrect – this indicates the diesel is in AUTO with manual start from control room available.

B: Correct – this indicates the diesel is in LOCAL control, which would prevent a manual control room start.

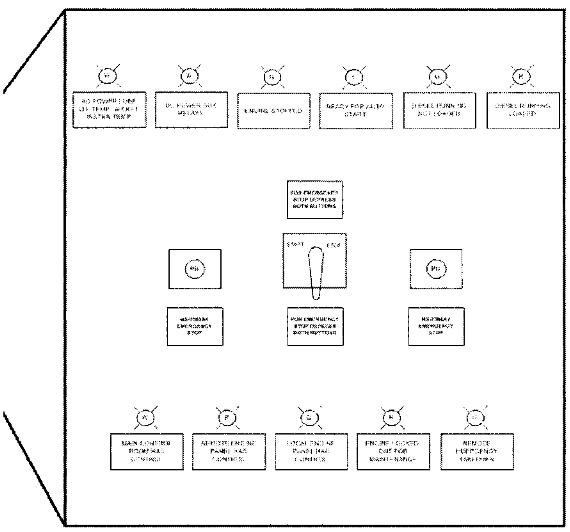
C: Incorrect – this indicates the diesel is not in REMOTE control. A manual control room start is available.

D: Incorrect – this indicates the diesel is not in maintenance. A manual control room start is available.

CONTROL ROOM CONTROLS:



LOCAL CONTROLS:



AV6039.VSD **REMOTE ENGINE CONTROL PANEL130'EL (C423)**

Technical Reference(s): HC.OP-SO.KJ-0001

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Modified Bank # 30848

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K4.01	
	Importance Rating	3.7	

261000 SGTS
K4.01 - Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Automatic system initiation

Question: #8

Given:
- The plant is in OPCON 5.
- ALL RBVS and RBVE fans are running.
- FRVS is in a normal standby configuration.

A radiological incident on the Refuel Floor causes Refuel Floor Exhaust Radiation to exceed $2x10^{-3}$ uci/cc.

Select total FRVS recirculation flow, if any, one minute after this event. (Assume NO operator actions)

A. 0 cfm

B. 60,000 cfm

C. 120,000 cfm

D. 180,000 cfm

Proposed Answer: D

Explanation:

A: Incorrect. FRVS recirculation fans will start

B: Incorrect. Plausible if candidate believes only 2 fans start or does not recall total flow for 6 fans

C: Incorrect. Plausible if candidate does not recall total flow for 6 fans. 4 fans is normal lineup.

D: Correct. All six FRVS fans will start on the high refuel floor exhaust radiation signal.

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

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 34792

Question History: 2009 Audit

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K5.04	
	Importance Rating	3.1	

211000 Standby Liquid Control System
K5.04 - Knowledge of the operational implications of the following concepts as they apply to SLC: Explosive Valve operation

Question: #9

Given:

- An ATWS is in progress.
- Reactor Power is 54 percent.
- Reactor Pressure is 900 psig.
- 'B' SLC Pump breaker is tagged out in the OPEN position for maintenance.

The 'A' SLC pump keylock switch is placed in ON and the START pushbutton is pressed. The following indications are observed:

- 'A' and 'B' Squib Valve Continuity light - "OFF"
- 'A' and 'B' SLC discharge pressure indicators (PT-N004A/B) - 950 psig
- "SLC Tank Trouble" OHA has annunciated.

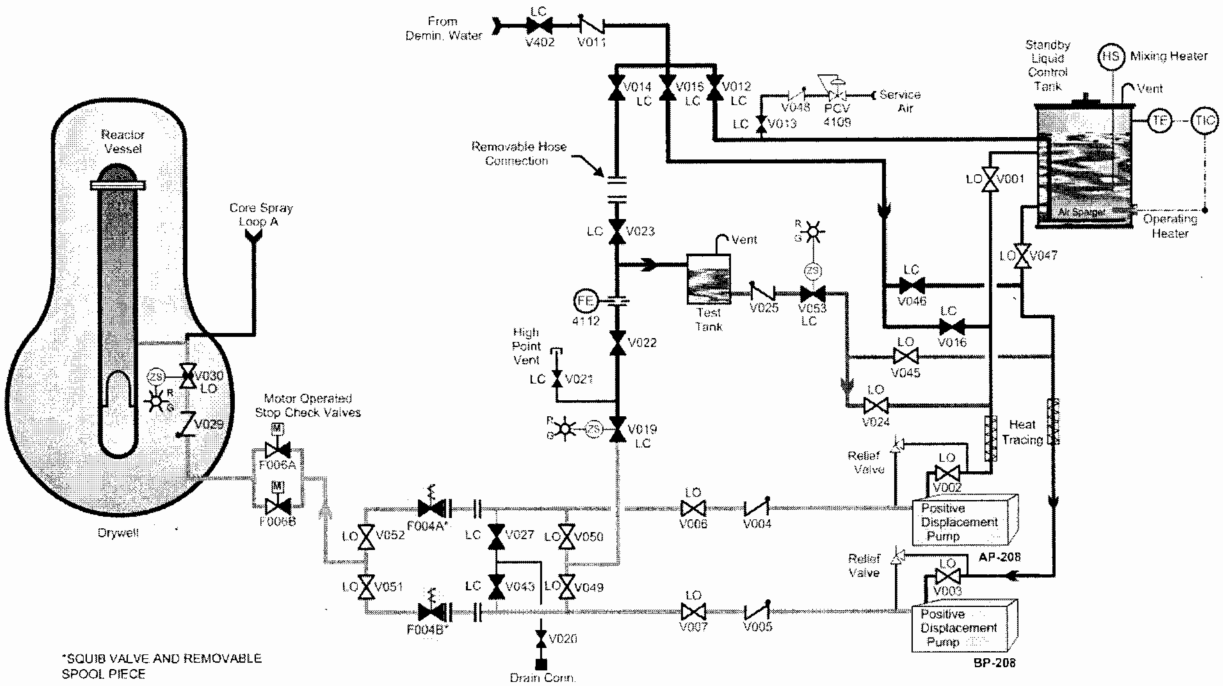
Which of the following describes the status of the 'A' SLC Pump and 'A' and 'B' Squib Valves based on the above conditions?

A. 'A' SLC Pump Running
'A' Squib Valve Fired
'B' Squib Valve NOT Fired

B. 'A' SLC Pump Running
'A' Squib Valve NOT Fired
'B' Squib Valve Fired

C. 'A' SLC Pump NOT Running
'A' Squib Valve Fired
'B' Squib Valve NOT Fired

D. 'A' SLC Pump NOT Running
'A' Squib Valve NOT Fired
'B' Squib Valve Fired



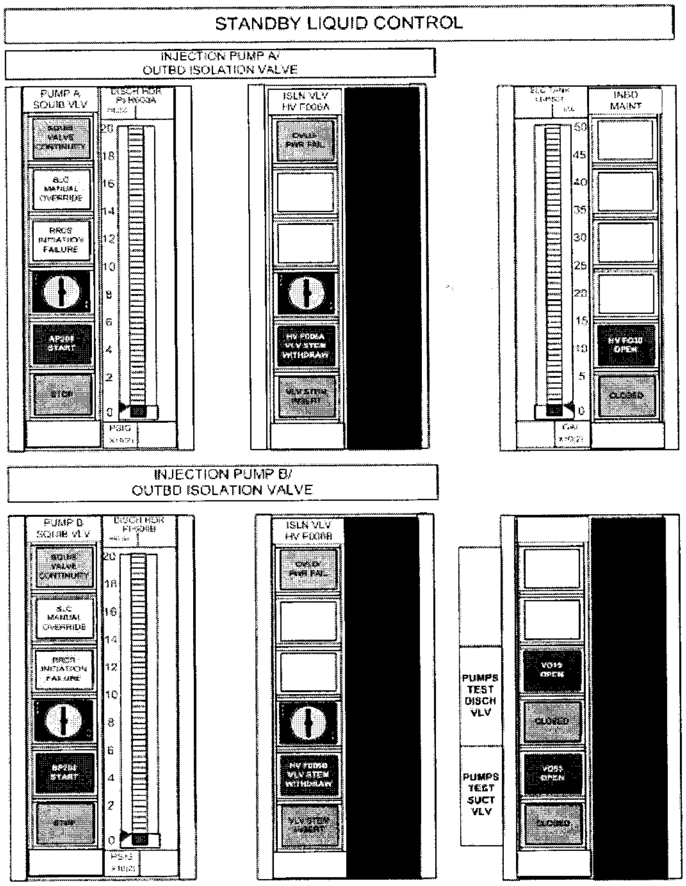
Answer: A

Explanation:

A. Correct. The 'A' SLC pump is running as evidenced by the discharge pressure and the lowering tank level (SLC Tank Trouble overhead alarm). The 'A' Squib valve fired because the continuity light is OFF. Although the 'B' Squib valve continuity light is OFF, it did not fire because its firing circuit is powered by the 'B' SLC pump breaker, which is tagged out.
B. Incorrect. The 'B' Squib did not fire because although its continuity light is OFF, its firing circuit is powered by the 'B' SLC pump breaker, which is tagged out.

C. Incorrect. The 'A' SLC pump is running as evidenced by the discharge pressure and the lowering tank level (SLC Tank Trouble overhead alarm).

D. Incorrect. The 'A' SLC pump is running as evidenced by the discharge pressure and the lowering tank level (SLC Tank Trouble overhead alarm). Additionally, the 'B' Squib did not fire because although its continuity light is OFF, its firing circuit is powered by the 'B' SLC pump breaker, which is tagged out.


STANDBY LIQUID CONTROL
INJECTION PUMP A/
OUTBD ISOLATION VALVE
PUMP A
SQUIB VLV
S.L.C
MANUAL
OVERRIDE
RRCS
INITIATION
FAILURE
STOP
20
18
16
14
12
10
8
6
4
2
0
PSIG
ISLN VLV
HV F006A
INBD
MAINT
50
45
40
35
30
25
20
15
10
5
0
HV F030
OPEN
CLOSED
INJECTION PUMP B/
OUTBD ISOLATION VALVE
PUMP B
SQUIB VLV
S.L.C
MANUAL
OVERRIDE
ISLN VLV
HV F006B
VLV STEM
INSERT
PUMPS
TEST
DISCH
VLV
PUMPS
TEST
SUCT
VLV
CLOSED
CLOSED

1. SLC Explosive-Actuated Injection Valves (Squib Valves)

   a. Purpose – provide a reliable injection pathway during manual/automatic initiation; and a positive barrier preventing sodium pentaborate from entering the reactor vessel or interconnecting piping during standby operation.

   b. Characteristics

      1) Two 150% capacity, stainless steel explosive actuated valves.

      2) Each valve consists of:

         a) A grooved plug threaded into the valve body, thereby inhibiting process flow.

         b) A shearing plunger-actuated by an explosive charge, which severs the plug upon firing.

         c) An explosive cartridge contains the explosive charge and two embedded primer circuits.

            - Either primer can ignite the explosive charge.

            - Primer circuits are powered from a Class 1E Power Supply which originates at the SLC A/B pump breakers and is jumpered at logic cabinet AC/BC652 (control power) for 120VAC output.

   c. Instrumentation and Control

      1) Instrumentation

         a) Squib valve continuity is displayed on the individual pump control bezel.

         b) A 3-5 milliamp current is passed through the primers, which illuminates the SQUIB VALVE CONTINUITY (amber) status lamp if the circuit is intact. The milliamp value can be monitored in back of 10C650, Section C.

         c) An open circuit is identified by a loss of status indication concurrent with the receipt of the SLC SQUIB VLV LOSS OF CONTINUITY (amber) annunciator.

      2) Control

         a) Igniting the explosive charge, which opens the valve, is accomplished by passing a 2-amp minimum current through the primers.

            - Source originates from pump control circuitry.

b) Valve will only "FIRE" in response to pump START commands initiated from

- Manual START pushbutton on 10C651
- Automatic or manual START commands from either division of the Redundant Reactivity Control System.

c) Only the valve associated with the pump receiving the START command will FIRE.

d. Automatic Actuation and Interlocks

1) Automatic Actuations - Upon the START of a SLC pump either manually (from 10C651) or automatically, the associated squib valve will FIRE.

2) Interlocks - 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.

SLC TANK TROUBLE (white)

Setpoint/Condition

1) A high or low temperature in any of the following:

a) SLC storage tank (110/70°F)

b) SLC pump A suction piping (110/75°F)

c) SLC pump B suction piping (110/75°F)

2) A high (4880 gallons net) or low (4640 gallons net) SLC storage tank level

Technical Reference(s):
NOH01SLCSYSC-07
M-48-1
HC.OP-SO.BH-0001(Q)

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank #35770

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	239002 K5.05	
Importance Rating	2.6	

239002 SRVs

K5.05 - Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES : Discharge line quencher operation

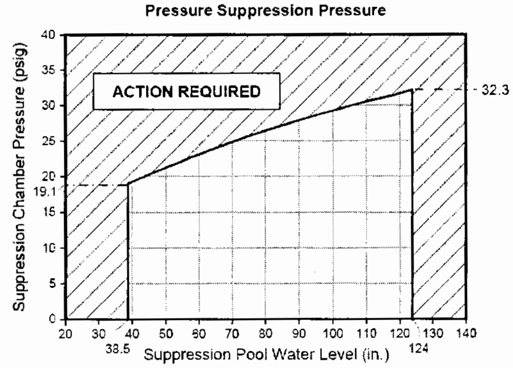
Question: #10

Given:

- The plant has experienced a small break LOCA.
- A recirculation pump runback and manual scram have been initiated.
- Reactor pressure is steady at 500 psig.
- Reactor water level is being maintained between +12.5 in. and +54 in.
- Suppression pool water level is 120 inches and rising approximately 1 in. every 3 minutes.
- Suppression chamber pressure is being maintained below 9.5 psig using suppression chamber spray.
- The SRO has ordered an Emergency Depressurization.

Curve SCP-L

Pressure Suppression Pressure



The basis for the decision to Emergency Depressurize is to __________ .

A. Prevent suppression chamber spray nozzles from becoming submerged.

B. Prevent exceeding the code allowable stresses in the SRV tailpipe, tailpipe supports and quencher.

C. Lower pressure in the reactor vessel as necessary to allow low pressure injection systems to inject.

D. Prevent possible suppression pool structural damage due to suppression pool water level.

Proposed Answer: B

Explanation:

A: Incorrect. Suppression chamber nozzles become submerged at 180 in.

B: Correct. Code allowable stresses on the tailpipe, tailpipe supports and quencher are in danger of being exceeded when suppression pool level approaches 124 in. An emergency RPV depressurization is required.

C: Incorrect. RPV level is being maintained adequately using preferred injection systems.

D: Incorrect. Structural damage of the suppression pool is not an immediate concern with the current condition of high suppression pool level.

Technical Reference(s):
HC.OP-EO.ZZ-0102 bases document

Proposed References to be provided to applicants during examination: SCP-L curve (as shown in stem)

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K6.08	
	Importance Rating	2.9	

209001 - LPCS

K6.08 - Knowledge of the effect that a loss or malfunction of the following will have on low pressure core spray system: keep fill system

Question: #11

Given:

- The plant is at rated power.
- ECCS Jockey Pump 1DP228 tripped.
- B3-C3 CORE SPRAY LOOP B TROUBLE alarm was received due to low injection line pressure.

Which system can be lined up to clear the alarm condition, and where does this system inject into the Core Spray loop?

A. Demin Water, which injects downstream of BE-HV-F005B, "Core Spray Loop Inboard Isolation Valve".

B. Demin Water, which injects upstream of BE-HV-F005B, "Core Spray Loop Inboard Isolation Valve".

C. Condensate Transfer, which injects downstream of BE-HV-F005B, "Core Spray Loop Inboard Isolation Valve".

D. Condensate Transfer, which injects upstream of BE-HV-F005B, "Core Spray Loop Inboard Isolation Valve".

Proposed Answer: D

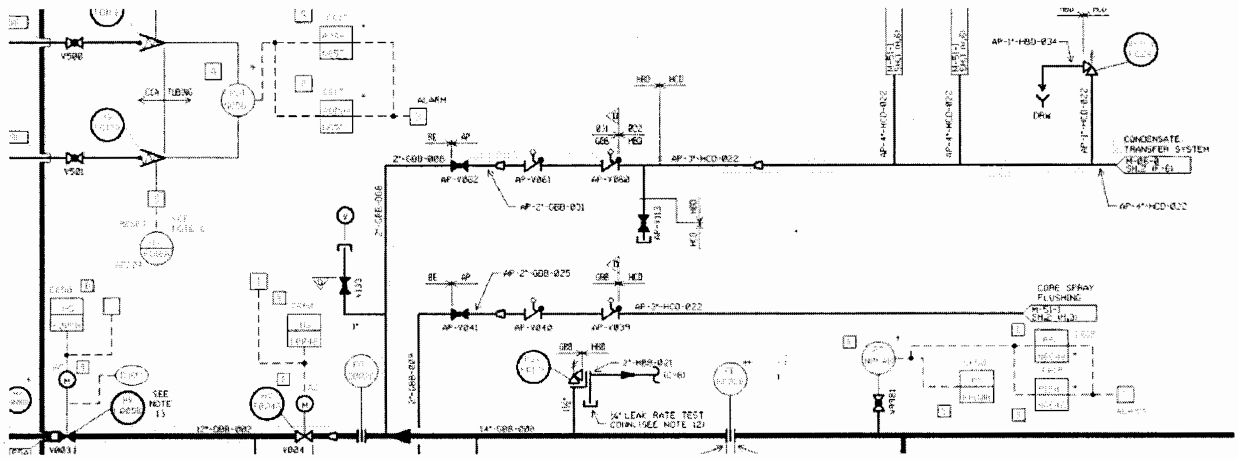
Explanation (Optional):

A: Incorrect: Demin water is not correct. Piping is installed from condensate transfer to each core spray loop and a normally closed manual valve can be opened to provide keep fill to core spray from condensate transfer.

B: Incorrect: Demin water is not correct. Piping is installed from condensate transfer to each core spray loop and a normally closed manual valve can be opened to provide keep fill to core spray from condensate transfer.

C: Incorrect: This is the correct system; however the injection point is upstream of HV-F005B

D: Correct: See P&ID M-52-1:



DIGITAL ALARM POINT D3159

NOMENCLATURE	CS LOOP B INJECTION LINE PRESS	SETPOINT	LO - 33 psig HI - 475 psig
DESCRIPTION	Low or high pressure in injection line	ORIGIN	PSL-N658B PSH-N654B

AUTOMATIC ACTION:

Alarm only

OPERATOR ACTION:

1. **DETERMINE** if indicated pressure is high or low
   AND **RESPOND** accordingly.
2. IF indicated pressure is high,
   DETERMINE leakage IAW HC.OP-GP.ZZ-0004(Q).
3. **ENSURE** compliance with the operability requirements of T/S 3.5.1, 3.6.3, & 3.4.3.2.

CAUSE	CORRECTIVE ACTION
LOW PRESSURE 1. Jockey running pump "D" failure or not running	1A. Alarm ECCS JOCKEY PUMP 1DP228 TROUBLE will indicate this problem.
2. Improper valve line-up from condensate transfer system	2A. **ENSURE** Core Spray valve line-up is complete.
3. Pipe or instrument line break	3A. **REQUEST** plant operator to inspect pipe AND instrument lines.
	3B. **REQUEST** SM/CRS to initiate corrective action.
HIGH PRESSURE 4. Improper valve line-up in injection line	4A. **ENSURE** line-up is complete.
	4B. **REQUEST** SM/CRS to initiate corrective action.

Associated Annunciator B3 C3

REFERENCES: N1-A41-46-(1)-2 J-52-0, Sht. 7, Sht. 8
M-52-1 PR 960324069

Technical Reference(s):

M-52-1 Core Spray
M-51-1 RHR
HC.OP-AR.ZZ-0007

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 33939

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 K6.12	
	Importance Rating	2.7	

203000 RHR/LPCI Injection Mode
K6.12 - Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) : ECCS room integrity

Question: #12

A seismic event has caused an unisolable leak, which has resulted in 1.5" of water to flood the 'A' and 'C' RHR pump rooms. Additionally, the temperature in the 'A' and 'C' RHR pump rooms has risen to 105°F.

Entry into EOP-0103/4, "Reactor Building and Rad Release Control," _______________.

A. is required ONLY on Reactor Building room floor level.

B. is required ONLY on Reactor Building room temperature.

C. is required BOTH on Reactor Building room floor level and Reactor Building room temperature.

D. is NOT required.

Proposed Answer: A

Explanation:

A. Correct. 1.5 in. of water exceeds the EOP 103 Reactor Building Room Floor Level Table 2, Column 1, Max Normal Op Floor Level of 1 in. The 105°F does NOT exceed the Reactor Building Room Temperature Table 1, Column 1, Max Normal Operating Temperature of 115°F. Therefore, entry into EOP 103 is required only on Reactor Building Room Floor Level. See EOP 103 entry criteria and tables below:

REACTOR BLDG <u>ROOM</u> TEMP ABOVE TABLE 1 COLUMN 1

REACTOR BLDG LOCAL RAD MONITORING ALARM

REFUEL FLOOR HVAC EXHAUST RAD LEVELS ABOVE 1E-3 UCI/ML

REACTOR BLDG HVAC EXHAUST RAD MONITOR LEVELS ABOVE 5E-4 UCI/ML

REACTOR BLDG <u>ROOM</u> FLOOR LEVEL ABOVE TABLE 2 COLUMN 1

SPENT FUEL POOL TEMPERATURE > 135 DEGREES

SPENT FUEL POOL LOW LEVEL ALARM WITH CONFIRMATION OF LOWERING LEVEL IN SPENT FUEL POOL

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	160°F

TABLE 2		
Area Description & Room Number	**Column 1 Max Normal Op Floor Level**	**Column 2 Max Safe Op Floor Level**
CRD Pump Room (4202)	1 in	4 1/2 in (25 min continuous running)
HPCI (4111)	1 in	4 1/2 in (30 min continuous running)
Core Spray Pump Rooms A(4118) & C(4116)	1 in	4 1/2 in (15 min continuous running)
RHR Pump Rooms A(4113) & C(4114)	1 in	4 1/2 in (20 min continuous running)
SACS A & C (4309)	1 in	4 1/2 in (INVESTIGATE)
RCIC Pump Room (4110)	1 in	4 1/2 in (17 min continuous running)
Core Spray Pump Rooms B(4104) & D(4105)	1 in	4 1/2 in (15 min continuous running)
RHR Pump Rooms B(4109) & D(4107)	1 in	4 1/2 in (20 min continuous running)
SACS B & D (4307)	1 in	4 1/2 in (INVESTIGATE)

B. Incorrect. Plausible if the candidate does not recall that the entry criteria for Reactor Building room temperature is above 115°F. The given temperature of 105°F does not exceed this threshold, so it does not warrant entry into EOP 103 based only on Reactor Building room temperature.

C. Incorrect. Plausible if the candidate does not recall that the entry criteria for Reactor Building room temperature is above 115°F. The given temperature of 105°F does not exceed this threshold, so it does not warrant entry into EOP 103 based on Reactor Building room temperature. Only the Reactor Building room floor level entry criteria of 1 in. is exceeded.

D. Incorrect. The Reactor Building room floor level entry criteria of 1 in. is exceeded. Plausible if the candidate does not recall entry criteria for either Reactor Building room temperature (above 115°F), or Reactor Building Room Floor Level (above 1 in.).

Technical Reference(s):
EOP 103

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 A1.04	
	Importance Rating	3.5	

215004 – Source Range Monitor
A1.04 - Ability to predict and/or monitor changes in parameters associated with operating SRM system controls including rod block status

Question: #13

Given:

- A Startup is in progress.
- ALL IRMs are on Range 5.
- SRM shorting links are INSTALLED.
- The RO has just selected and fully withdrawn all four SRMs, and all SRM count rates are now less than 3 cps.

The SRM Rod Withdraw Block is ____________.

A. enforced due to SRMs downscale.

B. NOT enforced due to IRMs above Range 2.

C. NOT enforced due to shorting links installed.

D. enforced due to detectors fully withdrawn when not permitted.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - At less than 3 cps, an SRM Downscale Rod Withdrawal block would be received; however, this would be bypassed because the associated IRMs are currently on Range 5.

B: Correct – At less than 3 cps, an SRM Downscale Rod Withdrawal block would be received; however, this would be bypassed because the associated IRMs are currently on Range 5.

SRM ROD BLOCKS:

Parameter	Setpoint	Bypassed
SRM Detector Wrong Position	<100 CPS and detectors not fully inserted	Associated IRMs ≥ Range 3 or Mode switch in Run or SRM Bypassed
SRM Downscale	< 3 CPS	Associated IRMs ≥ Range 3 or Mode switch in Run or SRM Bypassed
SRM Upscale	> 1 x 10^5 CPS	Associated IRMs ≥ Range 8 or Mode switch in Run or SRM Bypassed
SRM Inoperable	a) Module unplugged b) HVPS Low Voltage c) Mode switch out of operate	Associated IRMs ≥ Range 8 or Mode switch in Run or SRM Bypassed

C: Incorrect - Shorting link installation affect whether SRM scram functions are affected. Plausible if scram and rod block functions are confused.

SRM SCRAMS:

Parameter	Setpoint	Bypassed
SRM Upscale	> 2 x 10^5 CPS	Shorting links installed or SRM Bypassed

D: Incorrect – At less than 3 cps, an SRM Detector Wrong Position Rod Withdrawal block would be received; however, this would be bypassed because the associated IRMs are currently on Range 5.

Technical Reference(s):

NOH04SRMSYSC-01

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 A1.01	
	Importance Rating	2.5	

263000 DC Electrical Distribution
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate

Question: #14

Given:

- The 1AD413 125 Volt 1E Battery Charger is in service and providing a normal "Float" charge on its battery.
- The 1AD414 125 Volt 1E Battery Charger is tagged for maintenance.

At T= 0 minutes, AC power to the 1AD413 charger is lost.
At T= 20 minutes, the bus supplying the charger is re-energized by its associated EDG.

Which of the following describes the response of this battery charger?

The 1AD413 Battery Charger will _______________.

A. return to the "FLOAT" mode to recharge the battery.

B. trip and is interlocked "OFF" with the EDG powering the bus.

C. reset to the "EQUALIZE" mode to recharge the battery.

D. trip and must be manually restored as permitted by EDG loading.

Proposed Answer: A

Explanation:

A: Correct. Although the charging rate will be higher than prior to the charger loss, the charger will remain in the Float mode.

B: Incorrect. The charger does not trip. All chargers trip on high voltage and must be locally reset. The charger is restored when the bus power is restored.

C: Incorrect. Equalize mode must be manually initiated using the timer control on the charger.

D: Incorrect. The charger does not trip. All chargers trip on high voltage and must be locally reset. The charger returns when the AC bus is repowered.

Technical Reference(s):
HC.OP-SO.PK-0001
NOH01DCELEC-05

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 35474

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A2.03	
	Importance Rating	3.0	

223002 - PCIS/Nuclear Steam Supply Shutoff
A2.03 - Ability to (a) predict the impacts of the following on the PCIS/NS[4] and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: system logic failures

Question: #15

Given:

- The plant is shutting down for a refueling outage.
- 'A' RHR is in shutdown cooling.
- RPV temperature is 225°F and lowering.
- RPV pressure is 19 psig and lowering.
- RHR A(B) FPC MODE SUCT VLV CLOSURE TRIP OVERRIDE keylock switch is in the OVERRIDE position.

Then, a loss of RPS Bus 'B' occurs.

What is the status of shutdown cooling and what actions should be taken?

A. Isolated, pump tripped, enter HC.OP-AB.RPV-0009, "Shutdown Cooling," and HC.OP-AB.IC-0003, "Reactor Protection System."

B. Isolated, pump running, secure pump, enter HC.OP-AB.RPV-0009, "Shutdown Cooling," and HC.OP-AB.IC-0003, "Reactor Protection System."

C. Lined up, pump tripped, secure line up, enter HC.OP-AB.RPV-0009, "Shutdown Cooling," and HC.OP-AB.IC-0003, "Reactor Protection System."

D. 'A' RHR shutdown cooling remains in service, enter HC.OP-AB.IC-0003, "Reactor Protection System."

Answer: B

Explanation:

RHR Shutdown Cooling Isolation Valves (HV F008, F009):

HV F008 and F009 receive isolation signals from the Nuclear Steam Supply Shutoff System (NS4). Manual initiation of the A (D) channel of NS4 will initiate closure of F009 (F008).

HV F009(F008) will automatically CLOSE in response to:

- Low RPV LEVEL 3 (+12.5") sensed in NS4 Channels A and B (C and D).
  OR
- High reactor vessel pressure (82 psig, increasing) sensed in NS4 Channels A or B, (C or D) (Single coincidence).

NOTE: The pressure transmitters are aligned such that a loss of power to either RPS bus A or B [transmitter power supply] will cause both the HV F008 and HV F009 valves to close.

Any RHR pump will be prohibited from starting (motor bkr will close, then immediately trip) or will be tripped if running, if ANY of the following valves are not 100% OPEN:

- RHR pump suction from Suppression Pool F004A(B) OR shutdown cooling suction valves F006A(B), F008, AND F009. (A/B RHR pump)

A: Incorrect – system isolates on the loss of the B bus but the pump will not trip.

B: Correct – system will isolate, but the pump will not trip due to bypasses.

C: Incorrect – plausible if believed the loss of power doesn't cause an isolation

D: Incorrect – plausible if believed the loss of power doesn't cause an isolation or trip.

Technical Reference(s): HC.OP-AB.RPV-0005
HC.OP-SO.BB-0002
HC.OP-AB.IC-0003

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Modified Bank # 30845

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

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

262002 UPS (AC/DC)
K4.01 - Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies.

Question: #16

Given:

- A refueling outage is in progress.
- 10A401 4kV 1E bus is de-energized for pre-planned maintenance.

The operator places the 1AD481 Cyberex Inverter Manual Bypass Control Switch from "NORM" to the right "ISOLATE TO ALTERNATE" position.

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

A. NO power will be supplied to the load.

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

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

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

Proposed Answer: A

Explanation:

A: Correct. Exhibit 2 of SO-PN-0001 shows that in "ISOLATE" after Alt, that the static switch output will be disconnected from the load (contact 3 open), and the supply from Backup 480 will be connected to the load (contact 5 closed and 4 is open) However with 10A401>10B411-33, Backup 480 for AJ481 is de-energized.

B: Incorrect. Incorrect. Normal 480 VAC is not available because 10A401 is de-energized and contacts 3 and 4 are open.

C: Incorrect. Contacts 1, 2, 3, and 4 are open. No possible way to provide output from the inverter.

D: Incorrect. 10A401 bus is de-energized which is the source of back-up AC.

Technical Reference(s):
HC.OP-SO.PN-0001

Proposed References to be provided to applicants during examination:
None

Learning Objective:

Question Source: Bank # 134954

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 A3.01	
	Importance Rating	3.3	

215003 - IRM
A3.01 - Ability to monitor automatic operations of IRM system including meters and recorders

Question: #17

Given:

- A reactor startup is in progress, just past the Point of Adding Heat.
- A control rod then begins to drop and then stops after moving 8 notches.
- 'E' and 'F' IRMs were nearest the control rod and read as follows:

	'E' IRM	'F' IRM
BEFORE Drop	20 on Range 7 Stable	26 on Range 7 Stable
AFTER Drop	39 on Range 7 Stable	32 on Range 7 Stable

Which of the following describes the effects on IRM 'E" & 'F' indications, RPS and Reactor Manual Control?

A. Both IRM 'CHANGE RANGE' lamps will be illuminated,
An 'A' RPS ½ scram will be generated,
ONLY 'E' IRM will have generated a Rod Block.

B. ONLY 'E' IRM 'CHANGE RANGE' lamp will be illuminated,
An 'A' RPS ½ scram will be generated,
BOTH 'E' and 'F' IRMs will generate Rod Blocks.

C. ONLY 'E' IRM 'CHANGE RANGE' lamp will be illuminated,
An "A" RPS ½ scram will be generated,
ONLY 'E' IRM will generate a Rod Block.

D. BOTH IRM' 'CHANGE RANGE' lamps will be illuminated,
NO effect will have occurred with RPS,
BOTH 'E' and 'F' IRMs will generate Rod Blocks.

Proposed Answer: A

5) Even-numbered ranges (R-2, 4, 6, 8, 10) have a scale of 0-125.

a) Each even scale is a factor of 1 decade (10) of power from the next even scale.

b) Example:

100 on R-2 = 10 on R-4

10 on R-4 = 1 on R-6

1 on R-6 = 0.1 on R-8

0.1 on R-8 = 0.01 on R-10

However, due to the design of the indication R-6, 8, 10, would be indicating down scale based on the above example.

6) The odd-numbered ranges (R-1, 3, 5, 7, 9) are merely expanded scales of the lower 1/3 of the next higher even range and have a scale of 0-40.

Example:
If the IRM were indicating 30 on R-1 (0-40 scale) and the range switch was upscaled to range 2, the IRM would still indicate 30 (0-125 scale).

IRMS INTERLOCK SUMMARY

PARAMETER	SETPOINT		BYPASSED	TRIP ACTION
IRM Detector Wrong Position	Detector not fully inserted and reactor mode switch not in RUN	*	Reactor Mode Switch in RUN	Rod Block
IRM Downscale	5/125 of full Scale	*	Range 1 Selected or Reactor Mode Switch in RUN	Rod Block
IRM Upscale	108/125 of full Scale	*	Reactor Mode Switch in RUN	Rod Block
IRM Inoperative	• IRM Channel Mode Switch not in operate • IRM Channel Drawer Module Unplugged • Detector HVPS low output	*	Reactor Mode Switch in RUN	Rod Block and Reactor Scram
IRM Upscale	120/125 of full Scale	*	Reactor Mode Switch in RUN	Reactor Scram

* This trip is also bypassed if the channel is bypassed with the Joystick.

Explanation (Optional):

A: Both IRMs change range lamps will illuminate. A RPS ½ scram E IRM will generate a rod block; no other indications will change

B: Incorrect – F will not generate a rod block input

C: Incorrect – Both IRM change range lamps will illuminate and F will not generate a rod block input

D: Incorrect – F will not generate a rod block input.

Technical Reference(s):
NOH01IRMSYS-03

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 56305

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A3.08	
	Importance Rating	3.7	

215005 APRM/LPRM
A3.08 - Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Control rod block status

Question: #18

Given:

- The reactor is operating at 9% of rated power.
- The Reactor Mode Switch is in STARTUP/HOT STANDBY.
- No APRMs are bypassed.

The APRM channels have the following LPRM detector inputs:

	APRM A	APRM B	APRM C	APRM D	APRM E	APRM F
LPRM Inputs	17	14	15	13	14	14
Indicated Power	13%	10%	9%	9%	9%	13%

Given these conditions, ________________.

A. ONLY a full scram signal will be present.

B. a half scram signal and a rod withdrawal block signal will be present.

C. ONLY a rod withdrawal block will be present.

D. NO scram signals or rod withdrawal block signals will be present.

Proposed Answer: B

Explanation:

A. Incorrect - The scram setpoint in STARTUP is 15%. No APRMs are exceeding this setpoint. APRM "D" has an INOP trip from <14 LPRM inputs and should be generating a half-scram.
B. Correct -. With only 13 LPRM inputs, APRM "D" should have an INOP Trip causing a half-scram. The rod block setpoint in STARTUP is 12%. APRM "A" & "F" are reading 13% and should be generating rod withdrawal blocks.
C. Incorrect - APRM "D" has an INOP trip from <14 LPRM inputs and should be generating a half-scram
D. Incorrect - No scram signals or rod withdrawal block signals should be present. The rod block setpoint in STARTUP is 12%. APRM "A" & "F" are reading 13% and should be generating rod withdrawal blocks. APRM "D" has an INOP trip from <14 LPRM inputs and should be generating a half-scram.

Technical Reference(s):
HC.OP-SO.SE-0001, Section
3.2, Table SE-001 and SE-
002

Proposed References to be provided to applicants during examination:
None

Learning Objective:

Question Source: Bank # 33950

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	295002 A4.02	
	Importance Rating	3.7	

295002 Reactor Water Level control
A4.02 - Ability to manually operate and/or monitor in the control room: All individual component controllers in the automatic mode

Question: #19

Given:

- Reactor power 60 percent.
- Reactor level is 35".
- Feedwater Master Level Controller is in three element control.
- ‘A’, ‘B’, and ‘C’ RFP controllers are in AUTO.
- ‘A’, ‘B’, and ‘C’ RFP min flow controllers are in AUTO.
- ‘A’, ‘B’, and ‘C’ RFP discharge valves are full OPEN.

The discharge valve for the ‘A’ RFP fails fully closed.

Which of the following describes the speed of the ‘A’ RFP and position of the ‘A’ RFP min flow valve five minutes later?

A. Pump speed will rise, ‘A’ min flow valve will open.

B. Pump speed will rise, ‘A’ min flow valve remains closed.

C. Pump speed will lower to 2500 rpm, ‘A’ min flow valve remains closed.

D. Pump speed will lower to 2500 rpm, ‘A’ min flow valve will open.

Proposed Answer: A

Explanation (Optional):

A. Correct. With digital feedwater (DFW) in full automatic, if the handswitch for a RFP discharge valve is placed in CLOSE, the valve will close. The DFW system will see a drop in feed flow, but will not recognize the valve is closed since the command signal did not originate from the DFW system. As a result, pump speed will increase and the min flow valve will open as a result of the low flow condition.

B. Incorrect. The min flow valve will open as a result of the low flow condition.

C. Incorrect. The min flow valve will open as a result of the low flow condition.

D. Incorrect. With digital feedwater (DFW) in full automatic, if the handswitch for a RFP discharge valve is placed in CLOSE, the valve will close. The DFW system will see a drop in feed flow, but will not recognize the valve is closed since the command signal did not originate from the DFW system. As a result, pump speed will increase and the min flow valve will open as a result of the low flow condition.

Technical Reference(s): HC.OP-SO.AE-0001

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A4.01	
	Importance Rating	3.7	

217000 RCIC
A4.01 - Ability to manually operate and/or monitor in the control room: RCIC turbine speed

Question: #20

The plant has experienced a trip of all feed pumps from rated power.

- HPCI is inoperable.
- RCIC and CRD are in operation.
- Reactor water level is -40 inches and is slowly lowering.

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

• Pump Suction Pressure	15 psig
• Pump Discharge Pressure	225 psig
• Turbine Inlet Pressure	910 psig
• Turbine Exhaust Pressure	10 psig
• Turbine Speed	1900 rpm
• Flow controller setting	600 gpm

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

A. Continue to run RCIC. Raise turbine speed >2200 rpm by placing the flow controller in MANUAL and then raise flow using the increase button (arrow) to prevent bearing damage or exhaust check valve chatter.

B. Continue to run RCIC. Raise turbine speed >2200 rpm by lowering the flow controller automatic setpoint to prevent exhaust check valve damage.

C. IMMEDIATELY secure RCIC.
The High Pressure Exhaust Trip has failed.

D. IMMEDIATELY secure RCIC.
The Low Suction Pressure Trip failed.

Proposed Answer: A

Explanation:
Justification
From HC.OP-BD.0001, 3.1.6. To prevent possible bearing damage or excessive Turbine Exhaust Check Valve "chatter" RCIC Turbine speed should be maintained ≥ 2150 rpm.

From HC.OP-BD.0001, 3.3.1. During operation of the RCIC System, a solenoid operated trip mechanism will energize and trip the turbine as a result of any of the following: RCIC Pump suction pressure < 20" Hg VAC (PI-R604) after a 2 second time delay. Turbine exhaust pressure ≥ 50 psig.

A. Correct. Raising the flow setpoint will cause rpm to rise and discharge pressure to increase. System procedure HC.OP-BD.0001 warns that continued operation at turbine speeds <2,200 rpm can cause bearing or check valve problems.

B. Incorrect. The pump discharge pressure should be equivalent to the turbine steam inlet pressure (reactor pressure). Also incorrect because lowering the flow setpoint will cause rpm to lower even further. Plausible since the system procedure warns that continued operation at turbine speeds <2000 rpm can cause check valve problems. Plausible if applicant remembers that throttling closed on discharge flowpath will raise rpm in auto (in pressure control mode).
C. Incorrect. The hi exhaust pressure trip is set at >50 psig. (vs 10 psig pressure listed in the stem).
D. Incorrect. The RCIC low suction pressure trip occurs at 20"Hg vacuum with a 2 sec time delay (vs 15 psig positive pressure listed in the stem).

Technical Reference(s):
RCIC00E008
HC.OP-BD.0001

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 84452 (direct)

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	2.4.11 40000	
	Importance Rating	2.7	

Knowledge of abnormal condition procedures
Question: #21

Given:

- The plant is at rated power.
- At T=0 minutes, a complete loss of RACS occurs.
- At T=5 minutes, maintenance reports at least one RACS pump will be available to start in 15 minutes.

Which of the following identifies the effect on the Reactor Recirculation Pumps and required operator action?

A. Seal cavity temperature rises.
Reduce pump speed to maintain cavity temperature below 210°F.

B. Seal cavity temperature rises.
Trip the Recirculation pumps by T=10 minutes.

C. Pump motor winding temperature rises.
Reduce pump speed to maintain motor winding temperature below 210°F.

D. Pump motor winding temperature rises.
Trip the Recirculation pumps by T=10 minutes.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – on high seal cavity temperature, HC.OP-AB.RPV-0003(Q) directs reducing speed to maintain seal cavity temperature below 200°F:

RETAINMENT OVERRIDE	
CONDITION	**ACTION**
I. Affected Recirc Pump seal cavity temperature greater than 200 degrees F Date/Time: ____________	___ **I.a** **REDUCE** the affected Recirc pump speed to minimum, **TRIP** the affected Recirc pump and enter Condition A

RACS cools the recirculation pump motor oil coolers (upper and lower) and seals. RACS will not be restored to the recirculation pumps within 10 minutes. Therefore, HC.OP-AB.COOL-0003 subsequent action B is applicable (trip recirculation pumps within 10 minutes).

B: Correct – loss of RACS will result in increased seal cavity temperature. HC.OP-AB.COOL-0003 directs tripping the recirculation pumps within 10 mins of complete loss of RACS to both recirculation pumps:

B. RACS is lost to BOTH Reactor Recirculation Pumps. Date/Time: ____________	B.1 IF Cooling CANNOT be restored within 10 minutes OR Evidence of Recirculation Pump seal damage THEN **PERFORM** the following: ___ a. **REDUCE** Recirc. Pump Speed to MINIMUM. ___ b. **LOCK** the Mode Switch in Shutdown. ___ c. **TRIP** BOTH Reactor Recirculation Pumps.

C: Incorrect – Recirculation pump motor winding temperature is unaffected by loss of RACS. The Drywell Chilled Water System provides air coolers with 170 gpm cooling flow. The cooled air provides cooling for the pump motor windings.

D: Incorrect – Recirculation pump motor winding temperature is unaffected by loss of RACS. The Drywell Chilled Water System provides air coolers with 170 gpm cooling flow. The cooled air provides cooling for the pump motor windings.

Technical Reference(s):

HC.OP-AB.COOL-0003
NOH01RECIRC-11

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 2.2.39	
	Importance Rating	3.9	

206000 HPCI
2.2.39 - Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Question: #22

Following a reactor scram and loss of feedwater, the plant is being cooled down using HPCI in full flow recirculation.

A review of the operating logs indicates that reactor pressure for the past two hours is as follows:

Time	Reactor Pressure (psig)
0000	948
0015	916
0030	885
0045	855
0100	690
0115	641
0130	551
0145	307
0200	236

Based on these conditions, the cool down rate is ______(1)_________ administrative limits and ______(2)_________ Technical Specification limits.

A. (1) within
(2) within

B. (1) outside
(2) outside

C. (1) within

(2) outside

D. (1) outside
(2) within

Proposed Answer: B

Explanation: TS Limit is 100°F in any one hour and Admin limit is 60°F in any one hour

A. Incorrect. Exceeds both TS (max cooldown of 100°F in any one hour) and Admin limits (60°F in any one hour).
B. Correct. Between 0045 and 0145, cooldown reached 104 degrees within a one hour period.
C. Incorrect. Exceeds both TS and Admin limits.
D. Incorrect. Exceeds both TS and Admin limits.

Technical Reference(s):
HC.OP-IO.ZZ-0004
HCGS TS 3.4.6.1

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

Learning Objective:

Question Source: Bank # 35684

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K4.02	
	Importance Rating	3	

300000 Instrument Air
K4.02 - Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Cross-over to other air systems

Question: #23

Given:

- The plant is operating at rated power.

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:

- Service Air receiver pressure is 89 psig.
- Emergency Instrument Air Receiver is 87 psig.
- Instrument Air Receivers are currently 88 psig.

What is the status of the Air System?

A. Emergency Instrument Air Compressor should be running.

B. Instrument Air Dryer 1AF104 Isolation Valve, HV-11416, should be open.

C. Service Air Supply Header Isolation Valve, HV-7595, should be closed.

D. Standby Service Air Compressor should be running.

Proposed Answer: D

Explanation:

A: Incorrect – the Emergency Instrument Air Compressor starts at 85 psig.

B: Incorrect – HV-11416 opens at 85 psig.

C: Incorrect – HV-7595 closes at 70 psig.

D: Correct – The standby service air compressor starts at 92 psig.

Technical Reference(s): HC.OP-AB.COMP-0001

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 72594

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

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

400000 Component Cooling Water
A3.01 - Ability to monitor automatic operations of the component cooling water system (CCWS) including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

Question: #24

Given:

- Rx power 83%.
- Normal electrical lineup.
- TACS is being supplied from the "A" SACS loop.

Which of the below describes the response of the SACS pumps to depressing the TRIP PUSH BUTTON for breaker 40108 (10A401 Normal Feeder Breaker)?

A. The 'A' SACS Pump AP210 will trip, TACS swaps to initially supply system loads.

B. The 'A' SACS Pump AP210 will trip, loop A will remain on line supplying both SACS and TACS loads at a reduced flow.

C. The 'C' SACS Pump CP210 will trip, loop A will remain on line supplying both SACS and TACS loads at a reduced flow.

D. The 'A' SACS Pumps AP210 and the 'C' SACS Pump CP210 will both instantly trip, the TACS swaps to supply system loads.

Proposed Answer: A

Explanation:

A. Correct. AP210 pump trips on undervoltage, which closes the associated SACS to TACS isolation, causing a low flow condition in the "A" loop. The low flow will start the AUTO pump in the "B" loop, and open the "B" loop isolation valves.
B. Incorrect. AP210 pump trips on undervoltage, which closes the associated SACS to TACS isolation, causing a low flow condition in the "A" loop. The low flow will start the AUTO pump in the "B" loop, and open the "B" loop isolation valves.
C. Incorrect. "C" SACS pump CP210 is supplied from the 10A403 bus.
D. Incorrect. "C" SACS pump CP210 is supplied from the 10A403 bus

Technical Reference(s):
STACS0E009

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 33814

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 K6.01	
	Importance Rating	3.4	

217000 RCIC
K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Electrical power

Question: #25

Given:

- The plant is at rated power.
- A complete loss of 125 VDC Bus 'B' occurs.

As a result of this power loss, ______________.

A. HPCI can NOT auto initiate.

B. RCIC can NOT auto initiate.

C. RCIC can auto initiate, but NOT isolate.

D. HPCI can auto initiate, but NOT isolate.

Proposed Answer: B

Explanation (Optional):

A: Incorrect. HPCI is Channel 'A' 125 VDC.

B: Correct. RCIC is Channel 'B' 125 VDC.

C: Incorrect. RCIC will auto initiate but its ability to isolate is not affected.

D: Incorrect. HPCI is Channel ‘A’ 125 VDC and its ability to isolate is not affected.

Technical Reference(s):

NOH04RCIC00-11
PN1-E51-1040-059 6 & 7

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Modified Bank # From LGS changed to meet HC systems.

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

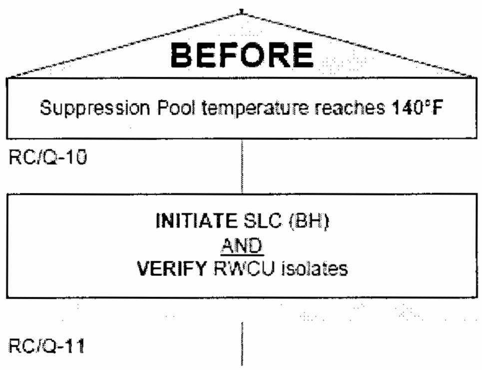
Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 2.4.20	
	Importance Rating	3.8	

211000 SLC
2.4.20 - Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.

Question: #26

In EOP 101A, "ATWS – RPV Control," step RC/Q-10, what is the primary basis for initiating SLC before suppression pool temperature reaches 140°F?



A. To initiate a reactor shutdown, which will help prevent further suppression pool temperature rise and avoid challenging or exceeding the maximum design temperature limit of the suppression pool (torus) structure.

B. To permit injection of the Cold Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.

C. To initiate a reactor shutdown and ensure cooldown rates of the reactor do not exceed Technical Specification or design requirements.

D. To permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.

Proposed Answer: D

Explanation:

A. Incorrect. The maximum design temperature limit of the suppression pool (torus) is 310°F, and is not close to being exceeded.
B. Incorrect. Plausible if candidate confuses Cold Shutdown Boron Weight with Hot Shutdown Boron Weight.
C. Incorrect. A reactor shutdown is desired however cooldown rates are not a primary concern.
D. Correct. To avoid depressurizing the RPV with the reactor at power, it is desirable to shut down the reactor prior to reaching the Heat Capacity Temperature Limit. The Boron Injection Initiation Temperature is defined so as to achieve this goal when practicable. The Boron Injection Initiation Temperature (BIIT) is the greater of:
    - The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the Heat Capacity Temperature Limit.
    - The suppression pool temperature at which a reactor scram is required by plant Technical Specifications.

The BIIT for power levels at or below 4% is approximately 144°F. The value has been conservatively rounded to 140°F.

Technical Reference(s):
EOP 101A bases, pp 19-20.
SLC lesson plan
NOH01SLCSYSC-07

Proposed References to be provided to applicants during examination:

Incorporate EOP flowchart segment as shown above in question stem.

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290001 K1.01	
	Importance Rating	3.3	

290001 – Secondary Containment
K1.01 - Knowledge of the physical connections and/or cause- effect relationships between SECONDARY CONTAINMENT and the following: Reactor building ventilation: Plant-Specific

Question: #27

Given:

- The plant is at rated power.
- All systems are operable.
- Reactor Building ΔP can NOT be maintained > - 0.30" WG.
- NO Release is in progress.

Which of the following describes the actions required and what is the effect on Secondary Containment ΔP throughout the evolution? (Actions are listed in the order performed)

A. Start an FRVS vent fan.
Remove the RBVS supply fan from service.
Remove the RBVS exhaust fan from service.
Start the FRVS recirc fans.
D/P will remain negative.

B. Remove the RBVS exhaust fan from service.
Remove the RBVS supply fan from service.
Start an FRVS vent fan.
Start the FRVS recirc fans.
D/P will remain negative.

C. Remove the RBVS exhaust fan from service.
Remove the RBVS supply fan from service.
Start an FRVS vent fan.
Start the FRVS recirc fans.
Initially D/P will go positive but then return to negative.

D. Start an FRVS vent fan.
Remove the RBVS supply fan from service.
Remove the RBVS exhaust fan from service.
Start the FRVS recirc fans.
Initially D/P will go positive but then return to negative.

Proposed Answer: A

Explanation (Optional):

A: Correct – FRVS vent fan is first action taken in the sequence to ensure negative pressure when RBVS exhaust fans are removed.

B: Incorrect – FRVS must be started first

C: Incorrect – FRVS must be started first

D: Incorrect - D/P will remain negative

Technical Reference(s):

HC.OP-SO.GU-0001,
Section 5.3.
AB.ZZ-0001, Att 20

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 117106

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	226001 A3.05	
	Importance Rating	4.0	

226001 RHR/LPCI: CTMT Spray Mode
A3.05 - Ability to monitor automatic operations of the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE including: Containment Pressure

Question: #28

In response to a steam leak in the Drywell:

- RHR Loop B was placed in Suppression Chamber Spray and Suppression Pool Cooling.
- RHR Loop A was placed in Drywell Spray.

Assuming NO operator action, how will RHR respond as Drywell Pressure lowers below 1.68 psig?

A. Drywell and Suppression Chamber sprays will isolate.

B. Drywell and Suppression Chamber sprays continue.

C. Drywell spray continues and Suppression Chamber spray isolates.

D. Drywell spray isolates and Suppression Chamber spray continues.

Proposed Answer: B

Explanation:

A. Incorrect. Drywell spray valves need 1.68 psig permissive to open however once started open the valves will stay open. There is no interlock to close the valves on low Drywell pressure.
B. Correct. Drywell spray valves need 1.68 psig permissive to open however once started open the valves will stay open. There is no interlock to close the valves on low Drywell pressure.
C. Incorrect. Drywell spray valves need 1.68 psig permissive to open however once started open the valves will stay open. There is no interlock to close the valves on low Drywell pressure.
D. Incorrect. Drywell spray valves need 1.68 psig permissive to open however once started open the valves will stay open. There is no interlock to close the valves on low Drywell pressure.

Technical Reference(s):
HC.OP-SO.BC-0001

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 34262

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	223001 K3.08	
	Importance Rating	2.7	

Question: #29

During normal, 100% power operations, a small leak develops on the H-T210 pneumatic accumulator tank for the PSV-F013H SRV.

What effect will this have on plant operation?

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

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

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

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

Proposed Answer: B

Explanation (Optional):

A. Drywell pressure will rise steadily due to the in-leakage. Containment venting will be required to maintain drywell pressure below the scram setpoint. Incorrect. 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. The Auto-Lead PCIG compressor will cycle more frequently. There will be no significant net change in drywell pressure or oxygen concentration. Correct. PCIG compressors normally take a 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.

C. The frequency of nitrogen makeup to the drywell will increase due to lowering drywell pressure from the accumulator leakage. Incorrect. 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. The frequency of nitrogen makeup to the drywell will increase due to rising drywell oxygen concentrations. Incorrect. The SRV accumulators are charged N2 from PCIG. There would be no oxygen introduced into the drywell as a result of the leak.

Technical Reference(s):
NOHO1PCIG00C-08
system descriptions

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 62165

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201001 K4.05	
	Importance Rating	3.8	

201001 CRD Hydraulic
K4.05 - Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following: Control rod SCRAM

Question: #30

Given:

- The plant is at rated conditions.
- The "A" and "B" SRI Rod Test switches at the HCU for control rod 42-03 have been placed in "Test."

What is the response of the Scram Pilot Valves for control rod 42-03 and the Scram Dump Valves for the given conditions?

A. The Scram Pilot Valves are de-energized and reposition to vent the Scram Inlet and Outlet Valves; the Scram Dump Valves remain in their initial positions.

B. The Scram Pilot Valves and the Scram Dump Valves remain in their initial positions as their power supplies are unaffected.

C. The Scram Pilot Valves are de-energized and reposition to vent the Scram Inlet and Outlet Valves; the Scram Dump Valves are de-energized and reposition to vent the Scram Discharge Volume Vent and Drain Valves.

D. The Scram Pilot Valves remain in their initial positions; the Scram Dump Valves are de-energized and reposition to vent the Scram Discharge Volume Vent and Drain Valves.

Proposed Answer: A

Explanation:

A. Correct. The Scram Pilot Valves reposition to vent the Scram Inlet and Outlet Valves, and the Scram Dump Valves remain in their initial positions. Each test switch de-energizes the individual HCU solenoids, has no effect on the system dump valves
B. Incorrect. The scram pilot valve solenoids de-energize.
C. Incorrect. Has no effect on the scram dump valves.
D. Incorrect. Has no effect on the scram dump valves

Technical Reference(s):
M-47-1 sheet 1

Proposed References to be provided to applicants during examination: None

Learning Objective:

CRDHYDE013

Question Source: Bank # 115983

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	271000 A1.08	
	Importance Rating	3.1	

271000 Off-gas
A1.08 - Ability to predict and/or monitor changes in parameters associated with operating the OFFGAS SYSTEM controls including System flow

Question: #31

Which of the following will cause the Ejector Main Steam Supply Shutoff Valve (AB-HV-2016A or B) to close resulting in a loss of Steam Jet Air Ejector Train?

A. A low steam flow condition for 30 seconds at the 3rd Stage Jet ONLY.

B. A low steam flow condition for 30 seconds at the 2nd Stage Jet ONLY.

C. A low steam flow condition for 30 seconds at the 3rd Stage Jet AND any single Air Ejector First Stage Jet Suction Valve not 100% closed.

D. A low steam flow condition for 30 seconds at 2nd Stage Jet AND any single Air Ejector Second Stage Jet Suction Valve not 100% closed.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – also need any single Air Ejector First Stage Jet Suction Valve not 100%

closed.

B. Incorrect – 2^{nd} stage and any single Air Ejector First Stage Jet Suction Valve not 100% closed.

C. Correct - A low steam flow condition at the 3rd Stage Jet (FSL-1971A or B) for 30 seconds combined with any single Air Ejector A(B) First Stage Jet Suction Valve HV-1968A1, A2, A3 (B1, B2, B3) not 100% closed will cause the Ejector Main Steam Supply Shutoff Valve (AB-HV-2016A or B) to close resulting in loss of Steam Jet Air Ejector Train.

D. Incorrect - 2^{nd} stage AND any single Air Ejector First Stage Jet Suction Valve not 100% closed.

Technical Reference(s): HC.OP-SO.CG-0001 step 3.3.2

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(13)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290003 K6.01	
	Importance Rating	2.7	

290003 Control Room HVAC
K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC : Electrical power

Question: #32

The plant has experienced a Loss of Offsite Power (LOP).

All running fans in the Control Area Ventilation system have tripped and the isolation dampers have failed closed as designed.

All EDGs have restored power to their respective buses.

The LOP sequencer has just begun to initiate start signals for a system restart.

Which of the following identifies the status of the Control Area Ventilation System 90 seconds later?

A. ONLY the Control Room Return Air (CRRA) and Control Room Supply (CRS) fans are running.

B. ONLY the CRRA, CRS, and Control Room Emergency Filtration (CREF) fans are running.

C. ONLY the CRRA, CRS, CREF, and Control Area Battery Exhaust (CABE) fans are running.

D. The CRRA, CRS, CREF, CABE and Control Equipment Room Supply (CERS) fans are running.

Proposed Answer: D

Explanation:

A. Incorrect. These fans are running, but after 90 seconds, so are the CREF, CABE and CERS fans.
B. Incorrect. Plausible if candidate forgets that after 90 seconds, the CABE fans and CERS fans should also have started (starts at 60s and 70s, respectively) in addition to all the fans listed.
C. Incorrect. Plausible if candidate forgets that after 90 seconds, the CERS fan should also have started (starts at 70 seconds) in addition to all the fans listed.
D. Correct. After 90 seconds, all the fans listed should be running:

    Following a LOP, all running fans have tripped and isolation dampers have failed closed. System restart response is as follows:

    7) The associated sequencer (LOP) is initiated and generates PSIS signals. When the diesel generator restores power to the switchgear, the LOP timer initiates start signals as follows:

        a) The CRRA, CRS, and CREF fans are started after 30 seconds. The CRS fan must be running to enable start of the CRRA and CREF fans.

        b) The CABE fan is started after 60 seconds.

        c) The CERS units are started after 70 seconds.

Technical Reference(s):
NOH01CAVENTC-05 pages
42-43

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000 A1.01	
	Importance Rating	3.4	

A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including: Recorders and Meters

Question: #33

Which of the following conditions are required for the Post Accident Monitoring (PAM) System chart recorders to shift to fast speed?

A. RPV level ≤ +12.5 inches <u>OR</u> RPV pressure ≥ 1037 psig.

B. RPV level ≤ +12.5 inches <u>AND</u> RPV pressure ≥ 1037 psig.

C. RPV level ≤ +30 inches <u>OR</u> RPV pressure ≥ 1020 psig.

D. RPV level ≤ +30 inches <u>AND</u> RPV pressure ≥ 1020 psig.

Proposed Answer: A

Explanation (Optional):

A: Correct - They automatically shift to the high-speed trend to provide increased resolution of the recorded traces of wide range reactor level and pressure. The recorders will shift to the fast-speed trend when level drops to 12.5" or pressure reaches 1037 psig.

B: Incorrect – Either +12.5" or 1037 psig

C: Incorrect – Wrong setpoints. These pertain to a high pressure alarm or the level 4 alarm setpoint

D: Incorrect - Wrong setpoints. These pertain to a high pressure alarm or the level 4 alarm setpoint

Technical Reference(s):
NOH04RXINSTC-04

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 34221

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	272000 A2.16	
	Importance Rating	2.7	

272000 Radiation Monitoring
A2.16 - Ability to predict the impacts of the following on the RADIATION MONITORING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Instrument malfunctions

Question: #34

Given:

- The plant is operating at 100% power.
- Main Steam Line (MSL) radiation levels have been averaging 100 mr/hr.

Then:

- Reactor operator notices RM11 main steam line radiation monitors for all four (4) main steam lines are averaging 130 mr/hr and steady.

Which of the following actions should be taken per HC.OP-AB.RPV-0008(Q), "Reactor Coolant Activity?"

A. REDUCE Reactor power as necessary to clear OHA C6-A3 MN STM LINE RADIATION HI alarm.

B. DIRECT Chemistry to perform sampling to determine the origin of the Main Steam Line high radiation alarm.

C. LOCK the Reactor Mode Switch in SHUTDOWN and CLOSE the MSIVs AND Main Steam Line drains F016/F019.

D. VERIFY the mechanical vacuum pumps automatically tripped and Recirc Sample Isolation Valves SV-4310 and SV-4311 automatically closed.

Proposed Answer: B

Explanation:

A. Incorrect. This is the proper subsequent operator action (B.1) in HC.OP-AB.RPV-0008(Q) for a VALID Main Steam Line Radiation Level >1.5x normal. While the two RM-11 channels indicate above that threshold (200 mr/hr), it has not yet been determined whether or not these indications are VALID. VALID is defined on the retainment override page as originating from fuel cladding deterioration, which Chemistry would determine thru sampling.
B. Correct. Step A in HC.OP-AB.RPV-0008(Q) states to DIRECT Chemistry to perform samples of various systems for rising main steam line radiation levels. This sampling is needed to determine if the high Main Steam Line radiation levels are VALID, which is defined as originating from fuel cladding deterioration.
C. This is the retainment override and proper operator action in HC.OP-AB.RPV-0008(Q) for a VALID Main Steam Line HI HI (3x normal). The two RM-11 channels indicate above HI radiation levels (200 mr/hr), and are not 3x normal. Additionally, it has not yet been determined whether or not these indications are VALID. VALID is defined as originating from fuel cladding deterioration, which Chemistry would determine thru sampling.

D. Incorrect. The automatic trip of the mechanical vacuum pumps and automatic closure of Recirc Sample Isolation Valves SV-4310 and SV-4311 occurs at Main Steam Line HI HI Radiation (3x normal). The two RM-11 channels indicate above HI radiation levels (200 mr/hr), and are not 3x normal. Additionally, at 100% power, the Mechanical Vacuum Pumps are not in-service.

Technical Reference(s):
HC.OP-AB.RPV-0008(Q)

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	241000 A3.10	
	Importance Rating	3.3	

A3.10 - Ability to monitor automatic operations of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM including: Main stop/throttle valve operation

Question: #35

Given:

- The plant is at rated power.
- One safety relief valve (SRV) fails to the full open position.
- Indicated steam flow on individual main steam line flow indicator (FI-R603A-D) lowers.

What is the reason for the indicated steam flow reduction?

A. The Turbine Control Valves throttle, lowering steam flow to the turbine, and the steam flow to the Suppression Chamber is not measured by the main steam flow indication.

B. A negative summer subtracts the SRV flowrate from the indicated steam flow signal.

C. The Turbine Intercept Valves and Turbine Control Valves all throttle to return reactor power to the original value.

D. Reactor vessel head pressure decreases, reducing main steam flow. No valves are expected to reposition.

Proposed Answer: A

Explanation (Optional):

A: Correct – when the SRV opens, steam which would have gone down the steam line through the flow venturies is routed to the torus, and therefore it is not measured

B: Incorrect – there is no summer circuit

C: Incorrect – the intercept valves will not throttle for pressure control and the TCV do not control reactor power

D: Incorrect – TCV will throttle to maintain pressure

Technical Reference(s):

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 33906

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215001 A4.03	
	Importance Rating	3.0	

215001 Traversing In-core Probe
A4.03 - Ability to manually operate and/or monitor in the control room: Isolation valves: Mark-I&II(Not-BWR1)

Question: #36

A Traversing Incore Probe (TIP) trace is in progress when a high drywell pressure event (> 1.68 psig) occurs due to a leak in the recirculation system.

Three minutes following the event, the following indications are present on the TIP Valve Control Monitor.

- "SQUIB MONITOR" lights - both extinguished
- "SHEAR VALVE MONITOR" lights - both extinguished
- "BALL VALVE OPEN" lights - both illuminated
- "BALL VALVE CLOSED" lights - both extinguished

Which of the following describes the status of the TIP system and the required operator actions?

A. The squib firing circuitry has been manually actuated, causing the shear valve to close. Operator action is required to manually close the ball valve.

B. The system has responded as designed. Operator action is required to manually close the ball valve.

C. The system has responded as designed. Operator action is required to fire the squib to close shear valves.

D. The TIP detectors may have failed to retract. Operator action is required to retract the detector and manually close the ball valve.

Proposed Answer: D

Explanation:

A. Incorrect. If the squib firing circuitry had been manually actuated, the SQUIB MONITOR and SHEAR VALVE MONITOR lights would be lit. Plausible of the candidate believes that these lights extinguish when the squibs are manually fired. Operator action to close the ball valve would be done <u>after</u> the detector is retracted.

B. Incorrect. The system has not responded as designed. In response to the high drywell pressure (>1.68), the detector, which is not in "in shield" position due to the TIP trace in progress, should have been automatically withdrawn to the "in-shield" position. Operators should not attempt to close the manual ball valve until after the detector has been retracted.

C. Incorrect. The system has not responded as designed. In response to the high drywell pressure (>1.68), the detector, which is not in "in shield" position due to the TIP trace in progress, should have been automatically withdrawn to the "in-shield" position. Firing of

the squib is done iaw HC.OP-AB.CONT-0002(Q) if the TIP cannot be isolated manually. The squib should not be fired at this point.

D. Correct. The ball valves never went closed because the TIP detector never withdrew to the "in-shield" position. All TIP ball valves will automatically close once their respective detectors have reached the "in-shield" position. IAW HC.OP-AB.CONT-0002(Q) subsequent operator action G, if a TIP valve fails to retract, operator required action is to retract the TIP and close the ball valve manually.

a. TIP system response to a Nuclear Steam Supply Shutoff System containment isolation signal

1) Containment isolation signals which will initiate a response from the TIP System are:

a) Low reactor vessel level (-38 inches, LEVEL 2), <u>or</u> High drywell pressure (1.68 psig), <u>or</u>

b) Actuation of the NSSSS Channel A manual isolation switch

NOTE: The TIP detectors will also receive an automatic withdrawal signal if the following conditions exist:

- If <u>both</u> Channel A RWCU valve G33-F001 ISLN Logic (switches S24A on 10C609 and S24B on 10C611) test switches are in the Test position, <u>or</u>

- If <u>both</u> Channel A trip valve ISLN logic (switches S22A on 10C609 and S22B on 10C611) test switches are in the TEST position.

2) Upon receipt of <u>any</u> of the above signals, the TIP System will respond as follows:

a) All TIP detectors <u>not</u> in the "in-shield" position will automatically be withdrawn at low speed while the detector is in the core. The TIP detectors will be withdrawn in high speed (12 inches per second) once they are below the core and will switch to low speed at indexer position 25, travel will then be slow until the TIP is in its "in-shield" position.

b) All TIP ball valves will automatically close once their respective detectors have reached the "in-shield" position.

<u>SQUIB MONITOR light</u>

The amber SQUIB MONITOR light illuminates when its respective shear valve has been fired, or when power has been lost to the firing circuit(s) (loss of squib valve continuity).

<u>SHEAR VALVE MONITOR light</u>

The amber SHEAR VALVE MONITOR light illuminates when its respective shear valve has been fired.

BALL VALVE OPEN light

The red BALL VALVE OPEN light illuminates when its respective ball valve is not fully closed.

BALL VALVE CLOSED light

The green BALL VALVE CLOSED light illuminates when its respective ball valve is fully closed.

TIME DELAY light

The white TIME DELAY light illuminates when the respective ball valve fails to open (within approximately 5 seconds) after its associated TIP detector has been driven from its "in - shield" position. Power to the drive motor is disrupted under these conditions to prevent damaging the TIP detector. This light will always be illuminated during normal operation when the TIP is past the ball valve.
NOTE This indicates that the detector drive interlock has not been satisfied (Refer to section IV.C.2 for information concerning this interlock).

PURGE light

The red PURGE light illuminates when the associated PURGE switch is in the ON position. This light indicates that the purge system to the TIP Indexers has been activated and the purge solenoid valve SE5V-J009 has been energized.
NOTE: A common PURGE light is provided for TIP subsystems A and B, as well as for subsystems C and D. TIP subsystem E is provided with its own individual PURGE light.

Technical Reference(s):
HC.OP-AB.CONT-0002(Q)
NOH01TIPS00-03

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 36070 (SIG MOD'D)

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 2.2.44	
	Importance Rating	4.2	

2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question: #37

Given:

- The plant is at 90 percent power.
- #2B Feedwater Heater (FWHTR) water level is rising.

Which of the following actions occur if that FWHTR level reaches the HI-HI level setpoint?

A. #1B FWHTR, #2 Drain Cooler, and #2B FWHTR Condensate inlet and outlet valves auto close.

B. The Main Turbine trips.

C. #2B FWHTR Bleeder Trip Valves trip closed.

D. #2 Drain Cooler, and #2B FWHTR Startup and Operating Vents auto close.

Proposed Answer: A

Explanation (Optional):

A: Correct – high level auto closes the condensate inlet valves and outlet valves

B: Incorrect – high RPV level trips turbine

C: Incorrect – condenser neck heaters have internal piping within the condenser shell and do not have BTVs

D: Incorrect – vents auto open

Technical Reference(s): HC.OP-SO.AF-001

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 35515

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	233000 K5.06	
Importance Rating	2.5	

233000 Fuel Pool Cooling/Cleanup
K5.06 - Knowledge of the operational implications of the following concepts as they apply to FUEL POOL COOLING AND CLEAN-UP : Maximum normal heat load

Question: #38

Concerning the MAXIMUM anticipated heat load condition of the fuel pool, select the statement that describes the alignment of the systems to remove the heat generated in the Spent Fuel Pool.

A. One FPCCS Pump and Heat Exchanger and NO RHR Fuel Pool Cooling Assist.

B. RHR in Fuel Pool Cooling Assist and NO FPCCS Pumps or Heat Exchangers.

C. Both FPCCS Pumps and Heat Exchangers and in parallel with RHR in Fuel Pool Cooling Assist.

D. Both FPCCS Pumps and Heat Exchangers and NO RHR Fuel Pool Cooling Assist.

Answer: C

Explanation:

A. Incorrect. One FPCCS Pump and Heat Exchanger and no RHR Fuel Pool Cooling Assist. If required, one RHR pump and one RHR heat exchanger can be aligned to augment the FPCC system through the system crosstie. The maximum heat load condition would require assistance from RHR.

B. Incorrect. RHR in Fuel Pool Cooling Assist and no FPCCS Pumps or Heat Exchangers. One RHR pump and one RHR heat exchanger can be aligned to augment the FPCC system through the system crosstie. You would need FPCCS pumps and heat exchangers.

C. Correct. Both FPCCS Pumps and Heat Exchangers in parallel with RHR in Fuel Pool Cooling Assist. The FPCCS maximum heat load is 43 x 106 BTU/hr. This heat load is the discharge of one full core of fuel at the end of a fuel cycle, plus the decay heat of the reload spent fuel from all previous refuelings. If required, one RHR pump and one RHR heat exchanger can be aligned to augment the FPCC system

through the system crosstie. For this system configuration, a heat load greater than 45 million Btu/hr can be removed from the spent fuel pool with a maximum SACS inlet temperature to the RHR heat exchanger of 95F and a spent fuel pool temperature of 135F.

D. Incorrect. Both FPCCS Pumps and Heat Exchangers and no RHR Fuel Pool Cooling Assist. Need RHR

Technical Reference(s):
FSAR 91..3.1
NOH01FPCCOO-09

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 36135

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

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

AK1.02 - Knowledge of the operational implications of the following concepts as they apply to SCRAM : Shutdown Margin

Question: #39

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

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

A. Control rods are inserted to or beyond the Maximum Subcritical Banked Withdrawal limit, therefore the reactor will remain shutdown under all conditions.

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

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

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

Proposed Answer: D

Explanation:

A: Incorrect – maximum subcritical banked withdrawal limit is 02 and beyond.

B: Incorrect – shutdown margin is satisfied

C: Incorrect – maximum subcritical banked withdrawal limit is 02 and beyond. Shutdown margin is satisfied with one rod at 24.

D: Correct – design basis shutdown margin is met, therefore the reactor will remain shutdown under all conditions.

Technical Reference(s): HC.OP-EO.ZZ-0101 Bases

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 34385

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(1)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EK1.03	
	Importance Rating	3.7	

295031 Reactor Low Water Level
EK1.03 - Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL : Water level effects on reactor power

Question: #40

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

A. The level is lowered to BELOW the level of the feedwater spargers to increase core inlet sub-cooling.

B. The level is lowered to a few inches ABOVE the level of the feedwater spargers to decrease core inlet sub-cooling.

C. The level is lowered to a few inches ABOVE the level of the feedwater spargers to aid in the development of natural circulation flow.

D. The level is lowered to BELOW the level of the feedwater spargers to mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

Answer: D

Explanation:
A. Incorrect. RPV level lowering is done to <u>decrease</u> core inlet subcooling.

B. Incorrect. RPV level is lowered below the feedwater sparger to decrease core inlet subcooling.

C. Incorrect. The development of natural circulation is reduced as RPV level is lowered.

D. Correct. By lowering the RPV level below the feedwater sparger, incoming feedwater is preheated which causes subcooling to be decreased, thereby adding negative reactivity to the reactor.

Technical Reference(s):
EOP 101A bases

Proposed References to be provided to applicants during examination:

Learning Objective:

Question Source: Bank # 120300 SIG
MOD'D

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AK1.02	
	Importance Rating	2.9	

AK1.02 - Knowledge of the operation applications of the following concepts as they apply to plant fire on site: fire fighting

Question: #41

Which Fire Suppression system(s) can be activated from the Main Control room in response to a fire?

A. The Diesel Driven Fire Pump and CREF Deluge.

B. Both Motor and Diesel Driven Fire Pumps.

C. Both FRVS Deluge and CREF Deluge.

D. The Motor Driven Fire Pump and FRVS Deluge.

Proposed Answer: B

Explanation:

A: Incorrect – FVRS is indication only

B: Correct - Panel 10C671 - Fire Protection Panel, Main Control Room Pushbutton HS-7751-1 Motor Driven Fire Pump START
When depressed, Motor Driven Fire Pump 00P521 will start. Once started, the Motor Driven Fire Pump cannot be stopped from the Main Control Room.
Pushbutton HS-7759-1 Diesel Driven Fire Pump START
When depressed, Diesel Driven Fire Pump 00P521 will start. Once started, the Diesel

Driven Fire Pump cannot be stopped from the Main Control Room.

C: Incorrect – CREF is a local manual initiation for fire deluge

D: Incorrect – both are local manual imitations

Technical Reference(s):
NOH01FIREPRO-06

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 55748

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295005 AK2.03	
Importance Rating	3.2	

295005 Main Turbine Generator Trip
AK2.03 - Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Recirculation system

Question: #42

Given:

- The plant is in a normal electrical line up.
- The plant is at 45% with power ascension to 100% in progress.
- One of the Electrical Protection Assembly (EPA) breakers on the "B" Reactor Protection System (RPS) MG set has just tripped.
- Breaker investigation shows a trip on "overvoltage".

Which of the following describes the response of the Recirculation Pumps if a main turbine trip occurs before the "B" RPS Bus is reenergized for the given conditions?

A. Both Recirculation Pumps trip.

B. The “B” Recirculation Pump trips, the “A” Recirculation Pump runs back to “minimum” speed.

C. The “A” Recirculation Pump trips, the “B” Recirculation Pump runs back to “minimum” speed.

D. Both Recirculation Pumps runback to "minimum" speed.

Answer: A

Explanation:

A. Correct. See HC.OP-SO.SB-0001. A transfer to the RPS power supply with the Main Turbine Stop and/or Control Valves closed will cause an EOC-RPT operation and a Recirc Pump Trip. The EOC-RPT bypass switches should be in BYPASS prior to swapping power supplies.

B. Incorrect. See HC.OP-SO.SB-0001 explanation in distractor A. Plausible is candidate determines there is an affect upon recirculation pumps when the RPS channel is reenergized.

C. Incorrect. See HC.OP-SO.SB-0001 explanation in distractor A. Plausible is candidate determines there is an affect upon recirculation pumps when the RPS channel is reenergized.

D. Incorrect. See HC.OP-SO.SB-0001 explanation in distractor A. Plausible is candidate determines there is an affect upon recirculation pumps when the RPS channel is reenergized.

Technical Reference(s):
HC.OP-SO.SB-0001

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 34255

Question History: None

Question Cognitive Level:

Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AK2.06	
	Importance Rating	3.8	

AK2.06 - Knowledge of the interrelations between partial or complete loss of forced core flow circulation and reactor power

Question: #43

The plant was operating at 60% power when Reactor Recirculation Pump B tripped. Plant conditions following the recirculation pump trip are as follows:

- Reactor power is 50% of rated thermal power.
- Core flow is determined to be 37% of rated.
- OPRMs are Operable.
- FFWTR is **NOT** in progress.

Which of the following is the action that must be taken by the operating crew?
(see reference provided)

A. Exit Region 1 IAW Enhanced Stability Guidance.

B. Lock the Mode Switch in Shutdown.

C. Increase the speed of Recirculation Pump A to increase core flow to between 40% and 45% or rated core flow.

D. Restart Recirculation Pump B to increase core flow to greater than 46% or rated core flow.

Proposed Answer: A

Explanation (Optional):

A: Correct – by use of power to flow map

B: Incorrect – only if OPRMs are inoperable

C: Incorrect – no direction to raise flow

D: Incorrect – no direction to restart the tripped pump

Technical Reference(s):
HC.OP-RPV-0003

Proposed References to be provided to applicants during examination:
Four Power to flow maps – Attachment 4 of HC.OP-AB.RPV-0003

Learning Objective:

Question Source: Bank # 32588

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

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

AK2.14 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Plant air systems

Question: #44

Given:

The plant is at rated power. A loss of Instrument Air event has occurred. As Instrument Air pressure lowers to ≤ 70 psig and continues to lower, a manual SCRAM is required.

The basis for this action is to:

A. Prevent abnormal core flux patterns due to eventual control rods drifting in.

B. Take action before an automatic reactor scram occurs due to the effects of inboard MSIVs drifting closed.

C. Avoid a loss of condenser vacuum due to the loss of instrument air to the mechanical vacuum pump suction valves.

D. Anticipate the reactor water level control problems that will result from the effects of the loss of air to the secondary condensate pump discharge valves.

Answer: A

Explanation:

A. Correct. Rod drift is expected as instrument air pressure continues to lower. The various rod drifts could create a rod pattern that would produce abnormal core flux patterns. Therefore, the plant should be scrammed before rods drift in (expected to drift at pressures lower than 60 psig).
B. Incorrect. The inboard MSIVs will not drift close on a loss of instrument air. They are controlled with Primary Containment Instrument Gas.
C. Incorrect. The stem states the plant is at rated power; therefore, the mechanical vacuum pumps are not in service. Prior to plant startup, vacuum pumps are operated to establish main condenser vacuum. At approximately 300 psig, the SJAE is prewarmed and placed in service and the mechanical vacuum pumps are removed from service. The mechanical vacuum pump suction valves (HV-1979A, B) are air operated, fail closed valves.
D. Incorrect. Secondary Condensate Pump Discharge Valves (HV-1651A, B, C) are MOVs and are not affected by a loss of instrument air. However, SCP min flow valves will drift open on a loss of instrument air, which is expected to cause a trip of feedpumps on low suction pressure.

From Instrument Air lesson plan:
Expected effects that could be caused by a loss of instrument air at Hope Creek.
The following discussion concerns plant response from 100% power as seen in the Hope Creek Simulator on a slow loss of instrument air. It is assumed that no operator actions are taken. Actual plant response and response from lower power levels may be different.
As the loss of air event starts, there is very little effect on plant operation. As air pressure begins to decrease, some automatic actions will occur that will attempt to stop the loss of air.

- Standby Service Air Compressor starts (92 psig)
- Emergency Instrument Air Compressor starts (85 psig)
- Instrument Air Dryer 1AF104 isolation valve opens (85 psig)
- Service Air Header Isolation Valve (HV-7595) closes (70 psig Instrument Air pressure)

As instrument air pressure continues to decrease, individual air operated valves will begin to drift to their respective failed position. This begins to occur as pressure decreases to approximately 65- 70 psig.

- RHR Pump Seal Water valves drift open.
- RWCU F/D FCVs and isolation valves drift closed. This will result in a loss of the RWCU Pumps on low flow.
- Secondary Condensate Pump Minimum flow valves drift open. This will likely result in a feed pump trip on Lo-Lo suction pressure.
- Gaseous Radwaste System Isolation valves drift closed. This will cause a loss of Off gas and a loss of condenser vacuum.
- Reactor Building Isolation Dampers drift closed. This will cause a loss of the RBVS. This may result in a loss of Secondary Containment.
- **CRD FCV drifts closed. This will cause a loss of cooling to the CRD Mechanisms.**

From 100% power, the reactor will most probably scram on RPV low level (+12.5 inches). If not and pressure continues to decrease, a manual scram may need to be initiated IAW AB.RPV-0005, due to rods drifting in.
Another concern due to the loss of instrument air is the loss of the Main Condenser as a heat sink due to the outboard MSIVs drifting closed. This will occur well after the scram and will be a concern for post scram pressure control.

Technical Reference(s):
NOH01INSAIR-05
HC.OP-AB.COMP-0001

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EK3.02	
	Importance Rating	3.9	

EK3.02 Knowledge of the reasons for the flowing responses as they apply to high reactor pressure: recirculation pump trip: plant specific

Question: #45

Which of the following describes the purpose of the Recirc Pump Trip feature in RRCS?

A. The reduction in flow to the reactor vessel downcomer area reduces core inlet subcooling and causes a decrease in core power

B. Tripping of breakers results in a longer flow coastdown time, this increases heat removal from the core to allow removal of decay heat.

C. Two breakers in parallel provide for isolating the Reactor Recirc Pump motors from their electrical bus supply in response to high reactor pressure =1071 psig.

D. Tripping of breakers results in an increase in core void content which will reduce reactor power levels and limit pressure rise following reactor pressurization transient.

Proposed Answer: D

Explanation (Optional):

A: Incorrect – this is the purpose of the feedwater runback logic in RRCS

B: Incorrect – a longer flow coast down will not increase heat removal from the core.

C: Incorrect – breakers are in series

D: Correct - Tripping of the recirculation pumps in a manner which removes the power generated by recirc MG set inertia, causes a rapid cessation of reactor core flow. Subsequent voiding within the core reduces reactor power levels and limits the reactor vessel pressure rise following the transient. Upon RRCS initiation

Technical Reference(s): NOH04RRCS00C-03 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EA1.01	
	Importance Rating	4.1	

295026 Suppression Pool High Water Temp
EA1.01 Ability to operate and/or monitor the following as they apply to suppression pool high water temperature: Suppression pool cooling

Question: #46

With the plant operating at power, a small break LOCA occurs.
The plant is stabilized with the following conditions:

Drywell pressure	6 psig	increasing slowly
RPV Level	-26 inches	increasing slowly
RPV Pressure	860 psig	decreasing slowly
Suppression Pool Temp.	102 deg F	increasing slowly

RHR Pump "A" is currently in suppression pool cooling and sprays.

5 minutes later, the leak grows larger resulting in RPV water level lowering to -200 inches.

What is the response of BC-HV-F048A, Hx Bypass Valve, and the suppression pool cooling lineup? (Assume NO other operator action)

A. BC-HV-F048A remains closed and suppression pool cooling lineup is unaffected.

B. BC-HV-F048A opens and suppression pool cooling lineup isolates.

C. BC-HV-F048A opens and suppression pool cooling lineup is unaffected.

D. BC-HV-F048A remains closed and suppression pool cooling lineup isolates.

Answer: A

Explanation:

A. Correct. The LPCI initiation signal has not been reset, and the F024A and F048A valve overrides will remain in effect, valves will remain in SPC position.
B. Incorrect. The LPCI initiation signal has not been reset, therefore the override on the F048A will not reset, and it will remain closed, and the override on the F024A will not reset, and it will remain open, SPC will not isolate.
C. Incorrect. The LPCI initiation signal has not been reset, therefore the override on the F048A will not reset, and it will remain closed.
D. Incorrect. The LPCI initiation signal has not been reset, therefore the override on the F048A will not reset, and it will remain closed, SPC will not isolate.

Technical Reference(s):
E11-1040 Sht 9,10,19,20;
HC.OP-SO.BC-0001, Note
5.3.1, Step 5.3.1.b,c; Table 1

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 35796

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AK3.01	
	Importance Rating	4.1	

AK3.01 Knowledge of the reasons for the following responses as they apply to control room abandonment: reactor scram

Question: #47

Due to a fire in the Control Room console, the Control Room Supervisor orders the Control Room immediately evacuated.

The reactor was scrammed remotely from the RPS distribution panels.

Which of the following statements describes how a scram can be verified in accordance with HC.OP-IO.ZZ-0008, "Shutdown from Outside Control Room?"

A. Reactor vessel pressure stable at 920 psig.

B. HCU accumulator pressure verified to be 950 - 1000 psig at each HCU.

C. RPS Backup Scram Air Solenoids verified de-energized.

D. SPDS display terminal "Rods Full In" in the TSC.

Proposed Answer: D

Explanation (Optional):

A: Incorrect. Rx pressure at 920 indicates the reactor is at low thermal power level but not necessarily scrammed. The USFAR states "By manually cycling a safety/relief valve

from the RSP (after RSP takeover) and observing an appropriate cooldown as indicated by a reduction in steady state reactor pressure following the steam discharge. Pressure indication can be used since pressure and temperature are directly related in a saturated system. If the reactor were critical, pressure and, correspondingly, temperature, would return to approximately their initial values since the reactor would see this evolution as a power transient."

B: Incorrect. Reactor pressue will be regulating between 905 and 1047 when moving to the RSP, due to the MSIV closure and actuation of Low-Low-Set funtion of the SRV's. This will maintain the CRD HCU pressure slightly higher than that of the reactor pressure. Although the 950 - 1000 psig is still within the normal charged range of an HCU, this does not mean that all rods have made it to the requirments of maintaining the reactor shutdown.

C: Incorrect - BU scram valves are de-energized with the scram reset. If they were energized, that would indicate a reactor scram.

D: Correct - IAW IO-0008 step 5.1.3 "If the Rx scram was not verified prior to evacuating the Control Room, then verify Rods Full In (SPDS/CRIDS (TSC) or Activity Control Cards OR Other)

Technical Reference(s):
HC.OP-IO.ZZ-0008,
step 5.1.3

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 80577

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AA1.01	
	Importance Rating	3.3	

295004 Partial or Total Loss of DC Power
AA1.01 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : D.C. electrical distribution systems

Question: #48

The plant is at rated power.

125VDC Distribution Panel 1AD417 Ground Indicator switch is in REMOTE.

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

B. One white light on panel 10D410 will be dim, the other extinguished if a hard ground exists.

C. A negative ground current will be indicated on Control Room panel 10650D.

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

Answer: C

Explanation:

A. Incorrect. When a ground exists, one light will be bright and the other will be dim or extinguished depending on the magnitude of the ground.
B. Incorrect. When a ground exists, one light will be bright and the other will be dim or extinguished depending on the magnitude of the ground.
C. Correct. With the 1AD417 Ground Indicator switch in REMOTE, negative ground current will be indicated on Control Room panel 10650D.
D. Incorrect. When a ground exists, one light will be bright and the other will be dim or extinguished

depending on the magnitude of the ground.

Technical Reference(s):
NOH01DCELEC-05

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 34079

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EA1.17	
	Importance Rating	3.9	

EA1.17 Ability to operate and/or monitor the following as they apply to high drywell pressure: containment spray Plant specific

Question: #49

Given:

Following completion of a HPCI surveillance test, Reactor Power was at 85%.

"B" RHR loop was in Suppression Pool Cooling mode and was secured.

At T=0 minutes

- A high DW pressure condition occurs.
- 'B' RHR and 'B' Core Spray pumps failed to start automatically and both have been placed in service manually IAW plant procedures.

At T=10 minutes

- 'A' RHR has been placed in Suppression pool cooling and spray.
- Torus pressure is 9.5 psig and continues to slowly rise.
- The order has been given to place 'B' loop of RHR in DW Spray.
- The 'B' RHR Loop Inboard Valve F021B will open but the Outboard Drywell Spray Valve F016B remains closed (electrical power is available).

Which of the following conditions is preventing the opening of the RHR "B" Outboard Drywell Spray Valve?

A. The RHR Full Flow Test Valve (F024B) is not fully closed.

B. "B" channel Drywell pressure instrument is failed low.

C. The HX Bypass valve (HV-F048A) is not fully closed.

D. "B" channel RPV water level indicates above -129 inches.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – these valves are not interlocked together

B. Correct – requires high DW pressure
HC.OP-SO.BC-0001-Interlocks

3.3.5 ALL of the below conditions must be satisfied to allow both BC-HV-F016A(B) RHR LOOP A(B) OUTBD CONT SPRAY ISLN MOV AND BC-HV-F021A(B) RHR LOOP A(B) INBD CONT SPRAY MOV to be open simultaneously:
LPCI initiation signal present
High Drywell pressure signal present
BC-HV-F017A(B) RHR LOOP A(B) LPCI INJ MOV is CLOSED.

5.3.2 IF LPCI initiation signals are present AND LPCI auto initiation has NOT occurred,
THEN, PERFORM manual initiation as follows:
A. ARM AND PRESS LOOP A AND B AND C AND D MANUAL INIT PBs.

none

C. Incorrect – no interlock between these 2 valves

D. Incorrect – the manual initiation seals in the LPCI signal

Proposed References to be provided to applicants during examination:
None

Learning Objective:

Question Source: Bank # 73320

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA1.02	
	Importance Rating	2.9	

AA1.02 - Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS : Fuel pool cooling and cleanup system

Question: #50

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, and '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 in HC.OP-AB.COOL-0004(Q) 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.

Answer: A

Explanation:

A. Correct. This is the IMMEDIATE operator action in HC.OP-AB.COOL-0004(Q) for a significant loss of fuel pool inventory.
B. Incorrect. This is a subsequent operator action in HC.OP-AB.COOL-0004(Q) for lowering fuel pool level.
C. Incorrect. This is a subsequent operator action in HC.OP-AB.COOL-0004(Q) for loss of fuel pool heat removal capability.

D. Incorrect. This is a subsequent operator action in HC.OP-AB.COOL-0004(Q) for lowering fuel pool level.

Technical Reference(s):
HC.OP-AB.COOL-0004

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	29508 EA2.03	
	Importance Rating	3.7	

Ability to determine and/or interpret the following as they apply to high drywell temperature: Reactor Water Level

Question: #51

Given:

- RPV is at normal operating pressure.
- A steam leak exists in the Drywell.
- Drywell temperature is 170°F and rising.

Which of the following describes the effect on RPV level indication for the above conditions?

A. Indicated level currently reads below actual level.

B. Indicated level currently reads above actual level.

C. Reference leg flashing will occur when reference leg temperature reaches 212°F, causing indicated RPV level to rise.

D. Indicated level is equal to actual level as long as drywell temperature remains less than 212°F.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – with drywell temperature elevated, reference leg water density will be lower, causing the d/p across the d/p cell to be lower, which causes indicated level to be higher than actual level.

B: Correct - with drywell temperature elevated, reference leg water density will be lower, causing the d/p across the d/p cell to be lower, which causes indicated level to be higher than actual level.

C: Incorrect – reference leg flashing will not occur at 212

D: Incorrect - with drywell temperature elevated, reference leg water density will be lower, causing the d/p across the d/p cell to be lower, which causes indicated level to be higher than actual level.

Technical Reference(s):

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # LGS 562178

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038 EA2.03	
	Importance Rating	3.5	

295038 High Off-site Release Rate
EA2.03 - Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : Radiation levels

Question: #52

A discharge of the Floor Drain Sample Tank is in progress when the Liquid Radwaste Discharge Isolation Valve (HV-5377A) to the Cooling Tower Blowdown automatically closes.

Which conditions below would cause this termination when the listed setpoint was reached? (Assume NO operator action)

(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), (4) and (5) ONLY.

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

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

Answer: C

Explanation:

A. Incorrect. Isolation does not occur on "Liquid Radwaste Effluent sample flow rate HIGH"
B. Incorrect. Isolation does not occur on "Cooling Tower Blowdown RMS HIGH radiation"

C. Correct. Per Radwaste System lesson plan:

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. The liquid radwaste release system automatically isolates on the following:

a. Radwaste effluent high radiation
b. Radwaste effluent high flow
c. Cooling Tower blowdown low flow
d. Loss of operate (Liquid Radwaste Effluent Radiation Monitor Downscale)
e. Loss of sample flow

D. Incorrect. Isolation does not occur on either "Liquid Radwaste Effluent sample flow rate HIGH" or "Cooling Tower Blowdown RMS HIGH radiation".

Technical Reference(s):
Radwaste System Overview
NOH04RWOVER-02

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 120363

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments: None

Facility: Hope Creek
Vendor: GE
Exam Date: 2016
Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA2.04	
	Importance Rating	3.5	

AA2.04 Ability to determine and/or interpret the following as they apply to partial or complete loss of AC power: system lineups

Question: #53

Given:

- The plant is in Operational Condition 5 with the Electrical Distribution System aligned in the Normal lineup.
- An internal short on Transformer 1BX-501 causes a ground overcurrent on the transformer.

Which of the following is the status of power to the 1E Vital Buses?

A. Power from 1BX-501 Transformer is lost. The diesel generators for the affected 1E buses will assume the load.

B. Power from both 1AX-501 and 1BX-501 Transformers will be lost and are unavailable until the faulted transformer's incoming disconnects are manually opened.

C. Power from 1BX-501 Transformer is lost. Power from 1AX-501 is available. The diesel generators for the affected 1E buses do NOT assume the load.

D. Power from both 1AX-501 and 1BX-501 Transformers is lost. The diesel generators for the 1E buses will assume the load.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – Bus section is de-energized. EDG's do not start

B: Incorrect – bus section 7 remains energized. EDG's do not start

C: Correct – 13kv breakers trip open. Bus section 2 is de-energized.

D: Incorrect – bus section 7 remains energized

Technical Reference(s): E-0001

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 36158

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 2.4.11	
	Importance Rating	4.0	

295018 Partial or Complete Loss of Component Cooling Water
2.4.11 - Knowledge of abnormal condition procedures.

Question: #54

Given:

- All RACS flow has been lost for 10 minutes.
- Attempts to restore RACS flow have failed.

Which of the following is correct regarding the Reactor Recirculation Pumps?

Reactor Recirculation Pumps must be ________________.

A. Runback to minimum speed. No further action is required unless there is evidence of seal damage.

B. Runback to minimum speed. Then place the Mode switch in Shutdown, and trip both Recirc. pumps.

C. Runback to minimum speed, then tripped. A manual reactor trip is not required.

D. Immediately tripped prior to placing the Mode switch in Shutdown.

Answer: B

Explanation:

A. Incorrect. Subsequent Action B of HC.OP-AB.COOL-0003 is already met due to the loss of cooling for >10 min. The evidence of seal damage is an "OR" statement; therefore, Action B should be performed regardless of seal damage evidence after 10 minutes of RACS cooling loss.

B. Correct. This scenario is a total loss of RACS for >10 min. Both Subsequent Action B and the Retainment Override of HC.OP-AB.COOL-0003 instruct to runback to minimum and trip the Recirc. pumps after placing the Mode Switch in Shutdown.

B. RACS is lost to BOTH Reactor Recirculation Pumps. Date/Time: ____________	B.1 IF Cooling CANNOT be restored within 10 minutes OR Evidence of Recirculation Pump seal dama THEN **PERFORM** the following: ___ a. **REDUCE** Recirc. Pump Speed to MINIMUM. ___ b. **LOCK** the Mode Switch in Shutdown. ___ c. **TRIP** BOTH Reactor Recirculation Pum

RETAINMENT OVERRIDE	
CONDITION	**ACTION**
I. Total Loss of RACS has occurred AND CANNOT be restored. Date/Time: __________	___ **I.a** **REDUCE** Recirc. Pump Speed to MINIMUM. ___ **I.b** **LOCK** the Mode Switch in Shutdown. ___ **I.c** **TRIP** both Reactor Recirculation Pumps. ___ **I.d** **TRIP** both CRD Pumps. ___ **I.e** **TRIP** both RWCU Pumps.

C. Incorrect. A manual reactor trip is required by placing the Mode switch in Shutdown. Both Subsequent Action B and the Retainment Override of HC.OP-AB.COOL-0003 instruct to runback to minimum and trip the Recirc. pumps after placing the Mode Switch in Shutdown.

D. Incorrect. The Recirc. Pumps are runback to minimum speed before both pumps are tripped.

Technical Reference(s):
HC.OP-AB.COOL-0003

Proposed References to be provided to applicants during examination:

None

Learning Objective:

Question Source: Bank # 33310 (SIG MOD'D)

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 2.1.27	
	Importance Rating	3.9	

2.1.27 Conduct of Operations: Knowledge of system purpose and/or function

Question: #55

Question: #55

The plant was at rated power when a grid disturbance occurred causing a Generator Reg Protection Relay actuation.

Which of the following, from the list below, occur as a DIRECT result of that relay actuation?

(1) Trip of the Exciter Field Breaker.
(2) Trip of BS 2-6 and BS 6-5 breakers.
(3) Trip of the Stator Water Cooling Pump.
(4) Trip of Main Power Transformer Cooling Fans.

A. 1, 2, 3 and 4.

B. 2 and 4 ONLY.

C. 3 and 4 ONLY.

D. 1 and 2 ONLY.

Proposed Answer: D

Explanation:

A: Incorrect – 3 and 4 are incorrect, see D.

B: Incorrect – 4 is incorrect, see D

C: Incorrect – 3 and 4 are incorrect, see D.

D: Correct – 1 and 2 are among the automatic actions in HC.OP-AR.ZZ-0015, digital point D2036.

HC.OP-AR.ZZ-0015(Q)

ATTACHMENT A5

UNIT PROT
LOCKOUT
RELAY TRIP

Window Location E1-A5

OPERATOR ACTION:

REFER to HC.OP-AB.BOP-0002(Q), Main Turbine.

INPUTS

Digital Point/ Indication	Nomenclature/Condition	Automatic Action
D2036	GEN REG PROTECTION RELAY	1. Trip and block closing 500KV BUS BS6-5 BREAKER. 2. Trip and block closing 500KV BUS BS2-6 BREAKER. 3. Initiate Breakers BS2-6 and BS6-5 failure protection. 4. Trip Main Turbine. 5. Trip and block closing Exciter Field Breakers.

Technical Reference(s):
HC.OP-AR.ZZ-0015

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 2.1.7	
	Importance Rating	4.4	

295021 – Loss of Shutdown Cooling
2.1.7 - Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Question: #56

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.
- The Reactor Operator 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 increasing.

Which of the following is the cause of the given plant conditions?

A. RHR Pump B Min Flow valve BC-HV-F007B has opened.

B. Reactor Recirc Pump B Discharge valve BB-HV-F031B has opened.

C. RHR Pump B Suppression Pool Spray Header isolation valve BC-HV-F027B has opened.

D. RHR Pump B RHR loop test return MOV valve BC-HV-F024B has opened.

Answer: B

Explanation:

A. Incorrect. Opening this 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.
B. 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).
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.
D. 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.

Technical Reference(s):
HC.OP-SO.BC-0002(Q)

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 30780 (SIG MOD'D)

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EA2.04	
	Importance Rating	3.5	

EA2.04 Ability to determine and/or interpret the following as they apply to low suppression pool water level: drywell / suppression chamber differential pressure : Mark 1 & 2

Question: #57

Given:

- Suppression Pool level is lowering.
- The Reactor is shutdown in accordance with the Emergency Operating Procedures.

When Suppression Pool level drops below 50 inches, which one of the following would occur?

Drywell to Suppression Chamber __________ .

A. Vacuum Breakers close.

B. Vacuum Breakers open.

C. differential pressure increases.

D. differential pressure decreases.

Proposed Answer: D

Explanation (Optional):

A: Incorrect – requires dP to be open which would not have occurred. If anything the dP would be lower.

B: Incorrect – requires dP to open, dP would be lower

C: Incorrect – the open path allows pressure to equalize

D: Correct – the drain valve uncovers at 50"

Technical Reference(s): EOP 102, step SP/L-5 bases

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 34128

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

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

295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown
EK1.03 - 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: #58

A failure to scram has occurred and the crew is taking actions per EOP-0101A, ATWS-RPV Control.

Given:

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

Which of the following statements correctly describe the 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 limits.

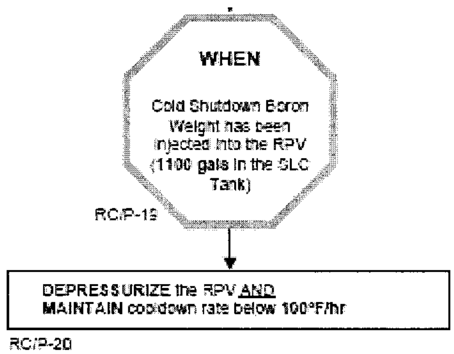
Answer: C

Explanation:

A. Incorrect. Cooldown is not permitted until Cold Shutdown Boron Weight (CSBW) is injected or only one rod not at 00.

B. Incorrect. The cooldown is permitted when SDBW is injected or if it will remain shutdown without boron.

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) 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 gal remaining in the SLC tank.



Injection 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.
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.

Technical Reference(s): (Attach if not previously provided)
EOP 101A bases
NOH01SLCSYSC-07

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 35685

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	500000 EK1.01	
	Importance Rating	3.3	

500000 High CTMT hydrogen concentration
EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: Containment integrity
Question: #59

Given:

- A LOCA has occurred.
- HC.OP-EO.ZZ-0102, Primary Containment Control, is being implemented.
- Drywell pressure is 63 psig and rising slowly.
- Containment Hydrogen concentration is 1%.
- Suppression Pool water level is >180 inches.
- HC.OP-EO.ZZ-0319, Restoring Instrument Air in an Emergency, has been completed.
- HPCI was just secured with suction aligned to the Suppression Pool.
- HPCI suction pressure is 73 psig and rising slowly.

What is the PREFERRED containment vent path for the current conditions?
(see reference provided)

A. Through the Suppression Chamber supply and ILRT Piping.

B. Through the Drywell supply and ILRT piping.

C. Via the Hard Torus Vent.

D. Vent via the Suppression Chamber 24" Exhaust.

Proposed Answer: B

Explanation:

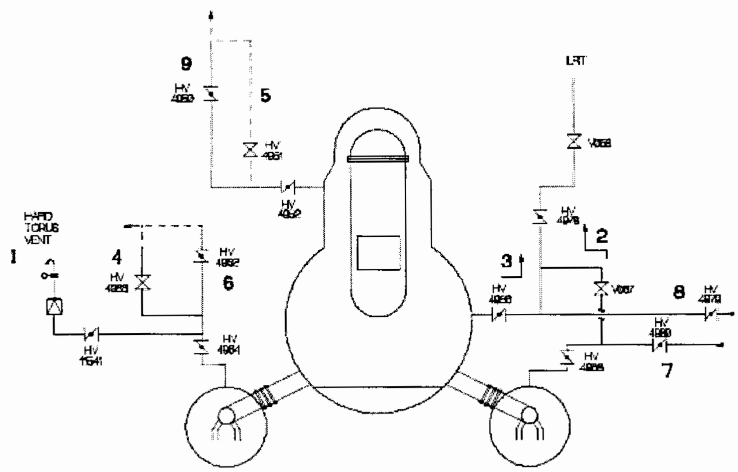
A: Incorrect. IAW HC.OP-EO.ZZ-0318 Attachment 4, the Suppression Chamber ILRT piping vent (Step 5.1.2, Path 2) can only be used when Suppression Pool level is <180".

B: Correct. Attachment 4 of HC.OP-EO.ZZ-0318 lists the preferred vent paths ranked 1 through 9 in order of having the least to most impact on Reactor Building radiological conditions. The preferred vent path for the given conditions is through the Drywell supply and ILRT piping (5.2.1, Path 3) when venting for pressure control and SPL is >180 inches with Containment level < 50 ft. The containment level calculation formula in EOP-102 (HPCI Suction Pressure - Drywell Pressure) x 2.3 ft/psi + 2.2 ft yields a containment level of 25.2 feet.

C: Incorrect. IAW HC.OP-EO.ZZ-0318 Attachment 4, the Hard Torus Vent (5.1.1, Path 1) can only be used when Suppression Pool level is <180".

D: Incorrect. IAW HC.OP-EO.ZZ-0318 Attachment 4, the Suppression Chamber 24" exhaust (5.1.6, Path 6) is used when Suppression Pool level is <180".

ATTACHMENT 4
CONTAINMENT VENT PATHS



VENT PATH NUMBER	EOP SECTION #			
	PRESS CONTROL		H2 CONTROL	
	<180"	>180"	<180"	>180"
1	5.1.1			
2	5.1.2		5.3.1	
3	5.1.3	5.2.1		5.4.1
4	5.1.4		5.3.2	
5	5.1.5	5.2.2	5.3.3	5.4.2
6	5.1.6			
7	5.1.7			
8	5.1.8	5.2.3		
9	5.1.9	5.2.4		

Technical Reference(s): HC,OP-EO.ZZ-0318

Proposed References to be provided to applicants during examination:

HC.OP-EO.ZZ-0318 - Attachment 4 embedded without table

Learning Objective:

Question Source: Bank #111420

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295010 AK2.04	
	Importance Rating	2.6	

295010 High Drywell Pressure
AK2.04 - Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Nitrogen makeup system: Plant-Specific

Question: #60

Given:

- The plant has been operating at 20% power with containment inerting in progress following a refueling outage.
- An error during a surveillance has resulted in a Group 10 (Drywell Chilled Water) isolation signal.
- The isolation goes to completion (all valves are closed).
- Drywell pressure is 1.00 psig and is slowly rising at 0.01 psig/min.

Which of the following is an appropriate action IAW HC.OP-AB.CONT-0001 for the given conditions, and why?

A. Reset the NSSSS isolation signal to permit re-opening of the Group 10 (Drywell Chilled Water) valves.

B. Swap Drywell cooling to RACS to reestablish Drywell cooling.

C. Lock the Mode Switch in Shutdown to avoid a LOCA actuation.

D. Terminate drywell inerting to prevent further pressure rise.

Answer: D

Explanation:

A. Incorrect. The valves do not go closed due to NSSSS, but rather PCIS.

B. Incorrect. The procedure directs swapping to RACS, but this will not be able to be performed until the isolation is reset or overridden (see HC.OP-AB.CONT-0002 subsequent B.4 and B.5)

C. Incorrect. Retainment Override of HC.OP-AB.CONT-0001 states to lock the mode switch in shutdown when drywell pressure is ≥1.5 psig and rising. The purpose of this action is to prevent an RPS automatic actuation and allow time to restore drywell pressure by eliminating heat addition to containment.

D. Correct. With a loss of DW cooling, inerting should be terminated to prevent further pressure rise.

Technical Reference(s):
HC.OP-AB.CONT-0001
HC.OP-AB.CONT-0002
NOH04INERTC-02

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 33300

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295014 AK3.02	
	Importance Rating	3.7	

AK3.02 Knowledge of the reasons for the following responses as they apply to inadvertent reactivity addition : control rod blocks

Question: #61

The plant is operating at 90% power.

When withdrawing control rods, _____ is a concern due to local power level increase and the __________ provides protection for this concern.

A. LHGR, Rod Worth Minimizer.

B. MCPR; Rod Block Monitor.

C. LHGR; Rod Block Monitor.

D. MCPR; Rod Worth Minimizer.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – RWM is designed to protect against a rod drop accident. LHGR is a function of exposure than local power.

B: Correct - During certain plant operating conditions, control rod withdrawal operations could lead to excessive increases in local power levels. If allowed to continue, these power levels could be increased to the point where fuel damage could occur due to localized overheating, resulting in a violation of the Minimum Critical Power Ratio

(MCPR) thermal limit.
The RBM System continuously monitors and evaluates the local power levels surrounding a control rod selected for movement and will generate signals to inhibit control rod withdrawal in areas of the core where overheating damage to the fuel could occur.

C: Incorrect – LHGR is a function of exposure than local power.

D: Incorrect - RWM is designed to protect against a rod drop accident

Technical Reference(s):

NOH04RBMSYSC-03 –
system description
FSAR – 7.7.2.

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 108100

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(1)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295015 AA1.02	
	Importance Rating	4.0	

295015 Incomplete SCRAM
AA1.02 - Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM : RPS

Question: #62

Which of the following alternate control rod insertion methods utilized during a failure-to-scram (ATWS) REQUIRES the Reactor Protection System (RPS) to be energized to successfully accomplish the task?

A. Control rod insertion by venting the control rod over-piston volume.

B. Control rod insertion by isolating and venting the scram air header.

C. Control rod insertion using the Reactor Manual Control System (RMCS).

D. Control rod insertion using the individual SRI Rod Test Switches on the HCUs.

Answer: D

Explanation:

A. Incorrect. HCU is isolated for manual vent so RPS status is not applicable.

B. Incorrect. Manual venting the scram air header is designed to insert rods with or without an RPS reset.

C. Incorrect. While rod speeds may be slower than normal, this method will work without an RPS reset.

D. Correct. HC.OP-EO.ZZ-0101A, "ATWS – RPV Control," Step RC/Q-20, directs that HC.OP-EO.ZZ-0303 be performed to allow individual rods scrams. This involves placing both rod test switches (A+B), which are located in the HCU upper transducer box, in the TEST position. Individual rod scrams using these test switches requires HCU accumulator pressure to cause a scram. This would equate to RPS needing to be energized to complete the task.

Technical Reference(s):
HC.OP-EO.ZZ-0101A
HC.OP-EO.ZZ-0303

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 36124

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

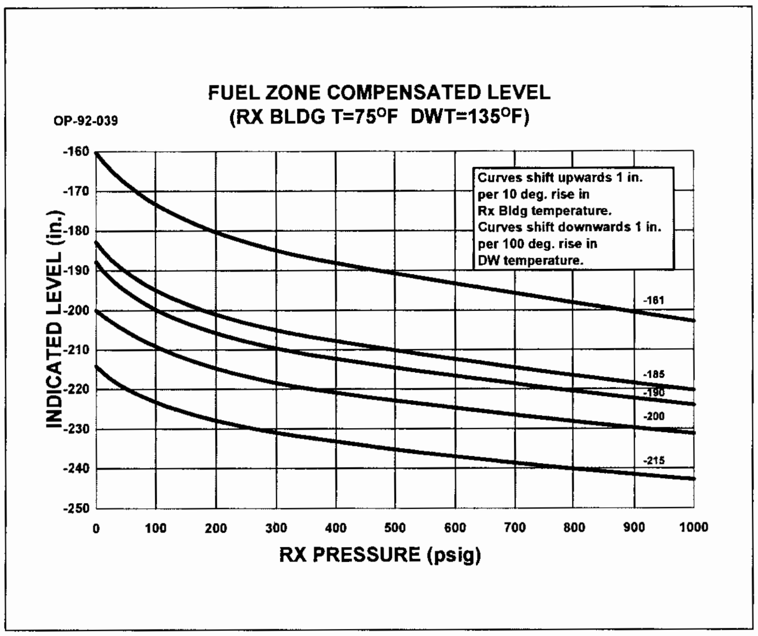
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295009 AA2.01	
	Importance Rating	4.2	

AA2.01 - Ability to determine and/or interpret the following as they apply to low reactor water level: reactor water level

Question: #63

Given:

- A LOCA concurrent with a Station Blackout has occurred.
- HPCI is unavailable.
- RCIC is being injected into the RPV.
- Reactor Pressure is steady at 500 psig.
- Reactor Building Temperature is steady at 95F.
- Drywell Temperature is 285F and increasing slowly.
- Fuel Zone indicators LR-R615 and LI-R610 are reading -187 inches and steady.



Based on the above current conditions, adequate core cooling is:
(see reference provided)

A. NOT assured, since actual RPV level is below TAF.

B. Assured, since actual RPV level is above TAF.

C. NOT assured, since actual level is below -185 inches.

D. Assured, since actual RPV level is between TAF and -185 inches.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – actual compensated level is 2.5 inches above TAF.

B: Correct – actual compensated level is 2.5 inches above of TAF (-158.5") and therefore cooling is assured by submergence.
Uncompensated level is -187"
RB temp correction: 95-75 = 20 (curve shifts up 2 inches)
DW temp correction: 285-135=150 (curve shifts down 1.5 inches)
TAF curve shifts up .5 inches at 500 psig. Normally -190 @ 500 psig, becomes -189.5, indicated level of -187 is 2.5 inches above the curve

C: Incorrect - actual compensated level is 2.5 inches above TAF. So adequate core cooling is assured

D: Incorrect - actual compensated level is 2.5 inches above TAF.

Technical Reference(s): OP-92-039, HC.OP-EO-ZZ-0101

Proposed References to be provided to applicants during examination: OP-92-039

Learning Objective:

Question Source: Modified Bank # 48863

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295008 2.4.49	
	Importance Rating	4.6	

295008 High Reactor Water Level
2.4.49 - Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Question: #64

Given:

- The reactor has been manually scrammed.
- Reactor level drops through -38" and HPCI injects
- HPCI restores vessel level to +62".

The CRS directs level to be stabilized between +12.5" and +54" inches using HPCI.

- Reactor level is now +20", lowering slowly.
- Reactor pressure is 960 psig and stable.
- Drywell pressure is 0.5 psig and stable.

Which of the following actions will result in a HPCI injection?

A. Simultaneously start the Auxiliary Oil Pump and open FD-HV-F001 turbine steam supply MOV.

B. Place E41A-S44, HPCI VESSEL WATER LEVEL 8 TRIP NORMAL/BYPASS switch to the bypass position.

C. Press HIGH REACTOR WATER LEVEL HPCI TRIP RESET PB, and then Arm and press HPCI MAN INIT PB.

D. Press RESET – INITIATION LOGIC, then Arm and press HPCI MAN INIT PB.

Answer: C

Explanation:

A. Incorrect. These are steps that are taken for manual operation of HPCI per HC.OP-SO.BJ-0001, section 5.5. However, the +54" auto trip signal is present and needs to be reset, so these steps will not result in a HPCI injection.

B. Incorrect. While a +54" high level 8 trip is present, performing only this step will not result in a HPCI injection.

C. Correct. The high level 8 trip signal must first be reset. Then, after arming and depressing the manual initiation button, HPCI will inject.

D. Incorrect. Resetting the initiation logic will not clear the +54 high level 8 trip signal, so this will not result in a HPCI injection.

Technical Reference(s):
HC.OP-SO.BJ-0001
NOH01HPCI00-12

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295012 AK2.01	
	Importance Rating	3.4	

AK2.01 - Knowledge of the interrelations between high drywell temperature and the following: Drywell ventilation.

Question: #65

Given:

- The plant was operating at rated power.
- Drywell temperature was steady at 120F.
- 8 drywell cooling fans were running with their cooling water valves open.
- 8 drywell cooling fans were in AUTO, not running.

Then, a complete loss of offsite power occurred.

- RPV level lowered initially to -35" and began to recover.
- Drywell pressure peaked at 1.58 psig and began to lower.

Currently:

- CRD and RCIC are controlling level -30 inches to +30 inches.
- RPV pressure is controlling between 800 - 1000 psig using HPCI and SRVs.
- Drywell pressure is at 1.2 psig and steady.
- Drywell temperature is 140F and steady.
- EDGs are supplying their 1E buses.

Assuming NO operator action following the initial scram actions, which of the following is the current status of the drywell cooling unit fans?

A. More drywell cooling fans are running than before the loss of offsite power.

B. The same number of drywell cooling fans are currently running now as compared to before the loss of offsite power.

C. Some of the drywell cooling fans are currently running; however, the number of drywell cooling fans running is less than before the loss of offsite power.

D. None of the drywell cooling fans are currently running.

Proposed Answer: A

Explanation (Optional):

A: Correct – IAW HC.OP-AB-ZZ-0135 Table 5.1. Upon re-energization of the 1E electrical buses, the LOP sequencers issue a start signal to ALL drywell cooling fans.

B: Incorrect – 8 additional fans will be running

C: Incorrect – all 16 fans are sequenced ON

D: Incorrect – the LOP sequencer starts all fans

Technical Reference(s): HC.OP-AB.ZZ-0135 Table 1

LOP SEQUENCER "A" (PNL AC428)		
MANUAL SWITCH	NOMENCLATURE	SEQUENCE TIMER
HS-6855-ER	AVH – 212; DRYWELL COOLER FAN BVH – 212; DRYWELL COOLER FAN CVH – 212; DRYWELL COOLER FAN DVH – 212; DRYWELL COOLER FAN EVH – 212; DRYWELL COOLER FAN FVH – 212; DRYWELL COOLER FAN GVH – 212; DRYWELL COOLER FAN HVH – 212; DRYWELL COOLER FAN	13 sec

LOP SEQUENCER "B" (PNL BC428)		
MANUAL SWITCH	NOMENCLATURE	SEQUENCE TIMER
HS-6855-FP	AVH – 212; DRYWELL COOLER FAN BVH – 212; DRYWELL COOLER FAN CVH – 212; DRYWELL COOLER FAN DVH – 212; DRYWELL COOLER FAN EVH – 212; DRYWELL COOLER FAN FVH – 212; DRYWELL COOLER FAN GVH – 212; DRYWELL COOLER FAN HVH – 212; DRYWELL COOLER FAN	13 sec

2. The Drywell Ventilation System is a 100% redundant system consisting of the following:

   a. Eight (8) cooling units with each unit consisting of:

      1) **Two 100% capacity fans each.**

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 84536

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	Conduct of Ops 2.1.29	
	Importance Rating	4.1	

2.1.29. Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Question: #66

You are performing an Independent Verification (IV) for a clearance application at the Aux Boiler House.

The IV is on an unlocked manual valve that is to be placed in the OPEN position.

IAW HU-AA-101, "Human Performance Tools and Verifications Practices," and OP-AA-108-101-1002, "Component Configuration Control," what is the proper way to perform this IV as the Verifier?

A. After the Performer positions the valve, locate and identify the component without the Performer in the area. Then by visual means, verify the valve is OPEN.

B. Accompany the Performer to initially locate and identify the component. Leave the area to allow the Performer to position the valve. Return to the component and manipulate the valve in the CLOSED direction enough to verify valve movement.

C. After the Performer positions the valve, locate and identify the component without the Performer in the area. Then manipulate the valve in the CLOSED direction enough to verify valve movement.

D. After the Performer positions the valve, locate and identify the component without the Performer in the area. Then manipulate the valve in the OPEN direction to verify no valve stem movement.

Answer: C

Explanation:

A. Incorrect. Manual valve position shall be determined by physical contact with the valve and not by observation only. See OP-AA-108-101-1002, Attachment 11, Step 2.3.

B. Incorrect. The Performer and the Verifier shall be separated by time and distance when IVs are performed and must independently locate the component per HU-AA-101, Step 4.

C. Correct. The Performer and Verifier shall be separated by time and distance. Additionally, per OP-AA-108-101-1002 Attachment 11, Step 2.6, to check an unlocked manual valve in the OPEN position, manipulate the valve in the CLOSED direction enough to verify valve movement.

D. Incorrect. Unlocked manual valves to be checked in the OPEN position are not manipulated in the OPEN position, but rather in the CLOSED direction per OP-AA-108-101-1002 Attachment 11, Step 2.6.

Technical Reference(s):
OP-AA-108-101-1002
Section 4.2

HU-AA-101 Section 4, IVs

Proposed References to be provided to applicants during examination:

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	Conduct of Ops	
	K/A #	2.1.3	
	Importance Rating	3.7	

Knowledge of shift or short-term relief turnover practices.

Question: #67

IAW OP-AA 112-101, Shift Turnover and Relief, which of the following may the oncoming RO perform **AFTER** relief?

(1) REVIEW Daily Orders.
(2) REVIEW Standing Orders for new entries.
(3) TOUR Main Control Room back panel areas.
(4) CONFER with the CRS to discuss planned shift activities.

A. (1), (2), (3) and (4).

B. (2) and (3) only.

C. (1) and (4) only.

D. (4) only.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – 1 ,2 and 3 are prior to relief
B. Incorrect – 2 and 3 are prior to relief
C. Incorrect- 1 is prior to relief
D. Correct – IAW OP-AA-112-101 step 4.8.4:

4.8.4. After relief, the on-coming RO's should **PERFORM** the following:

- **ANNOUNCE** shift turnover and relief to the Control Room Supervisor.
- **CONFER** with the Control Room Supervisor to determine the scope of planned shift activities and their responsibilities for that shift.

Technical Reference(s): OP-AA-112-101 step 4.8.4

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	Equip Control 2.2.39	
	Importance Rating	3.9	

2.2.39. Knowledge of less than one hour technical specification action statements for systems.

Question: #68

The plant is at rated power.

Which of the following required radiation monitoring equipment needs to be placed in a tripped condition within one hour of determining that it is inoperable?

A. Control Room Ventilation Radiation Monitor.

B. Safety Auxiliaries Cooling Radiation Monitor.

C. Reactor Auxiliaries Cooling Radiation Monitor.

D. Offgas Pre-treatment Radiation Monitor.

Answer: A

Explanation:

A. Correct. IAW Tech Spec Table 3.3.7.1-1. Action 71 states to place the inoperable channel in the tripped condition within one hour:

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	ACTION
1. Control Room Ventilation Radiation Monitor	2/intake	1,2,3 and *	$\leq 2\times10^{-5}$ µC/cc**	71
2. Area Monitors				
a. Criticality Monitors				
1) New Fuel Storage Vault	1	#	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr[a]	72
2) Spent Fuel Storage Pool	1	##	$\geq$ 5 mR/hr and $\leq$ 20 mR/hr[a]	72
b. Control Room Direct Radiation Monitor	1	At all times	2.5 mR/hr[a]	72
3. Reactor Auxiliaries Cooling Radiation Monitor	1	At all times	9×10^{-5} µC/cc[a]	73
4. Safety Auxiliaries Cooling Radiation Monitor	1/loop	At all times	6×10^{-5} µC/cc[a]	73
5. Offgas Pre-treatment Radiation Monitor	1	***	(b)	74

HOPE CREEK 3/4 3-63 Amendment No. 156

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 71 -

a. With one of the required monitors inoperable, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within one hour.

ACTION 72 - With the required monitor inoperable, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 73 - With the required monitor inoperable, obtain and analyze at least one sample of the monitored parameter at least once per 24 hours.

ACTION 74 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, release(s) via this pathway may continue for up to 30 days provided:

a. The offgas system is not bypassed, and

b. Grab samples are taken at least once per 8 hours and analyzed within the following 4 hours;

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

B. Incorrect. IAW Tech Spec Table 3.3.7.1-1

C. Incorrect. IAW Tech Spec Table 3.3.7.1-1.

D. Incorrect. IAW Tech Spec Table 3.3.7.1-1.

Technical Reference(s):
Tech Spec Table 3.3.7.1-1

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 34277

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	Equip. Control	
	Importance Rating	3.9	

2.2.14 Knowledge of Controlling equipment configuration

Question: #69

Per OP-AA-108-101 – Control of Equipment and System Status, in regard to the Abnormal Component Position Sheet (ACPS), when must the Work Clearance Module (WCM) be updated (1) and (2) who shall ensure this updating occurs?

	(1)	(2)
A.	Once per Day	NCO
B.	Once per Shift	NCO
C.	Once per Day	WCM Coordinator
D.	Once per Shift	WCM Coordinator

Proposed Answer: B

Explanation (Optional):

A: Incorrect - shiftly

B: Correct – shiftly, by NCO

C: Incorrect – by NCO

D: Incorrect – by NCO

Technical Reference(s): OP-AA-108-101

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	EPlan 2.4.49	
	Importance Rating	4.6	

2.4.49. Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Question: #70

Given:

- The plant is operating at 90% power.
- All Main Turbine Sealing Steam normal and backup supplies have been lost.
- There is no time estimate for repair/restoration.

Which of the following are the immediate operator actions for the given conditions?

A. Reduce power as necessary to maintain adequate self-sealing steam to the main turbine seals.

B. Reduce recirculation flow to minimum, unload and trip the main turbine.

C. Reduce recirculation flow to maintain power less than 25 percent.

D. Reduce power as necessary to maintain condenser pressure less than high pressure alarm setpoint.

Answer: D

Explanation:

A. Incorrect. Large turbines typically are "self" sealing at loads above 70%. Therefore, reducing power will reduce the self sealing affect of the main turbine.

B. Incorrect. The only immediate operator action directs a power reduction as necessary to maintain vacuum less than the alarm setpoint.

C. Incorrect. The only immediate operator action directs a power reduction as necessary to maintain vacuum less than the alarm setpoint. 25 percent is the bypass valve capacity.

D. Correct. Loss of sealing steam may eventually result in degraded condenser vacuum. Therefore, entry into HC.OP-AB.BOP-0006(Q) is appropriate. The immediate operator action is to reduce reactor power to maintain main condenser vacuum less than the alarm setpoint.

From HC.OP-AB.BOP-0006(Q):

IMMEDIATE OPERATOR ACTIONS

CONDITION	ACTION
Degraded Main Condenser Vacuum Date/Time: ____________	___ **REDUCE** Reactor Power to maintain MAIN CONDENSER A(B,C) VACUUM LO Overhead Alarm Clear.

Technical Reference(s):
HC.OP-AB.BOP-0006(Q):

Proposed References to be provided to applicants during examination:

None

Learning Objective:

Question Source: Bank # 33371

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.15	
	Importance Rating	2.9	

2.3.15 Knowledge of radiation monitoring systems, such as fixed radiation monitors, and alarms, portable survey instruments, personnel monitoring equipment. Etc.

Question: #71

Given:

- Plant is at rated power.
- A steam leak develops in the Reactor Building.
- Reactor Building Exhaust Radiation Monitor indicates 5×10^{-3} µci/cc.

Which of the following is the expected plant response and what actions must be taken by the control room?

A. Back draft dampers will close, no other plant response is expected. Control room personnel will take action to isolate the leak, no other actions are required.

B. Back draft dampers will close, RBVS fans will trip, FRVS will start. Control room personnel will take actions to isolate the leak and enter EO.ZZ-0103/4.

C. The secondary containment supply and exhaust dampers close, RBVS fans will trip, FRVS will start. Control room personnel will take actions to isolate the leak and enter EO.ZZ-0103/4.

D. The secondary containment supply and exhaust dampers close, RBVS fans will trip, FRVS will start. Control room personnel will take actions to isolate the leak, no other actions are required.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – plausible if unaware of set points and entry for 103.

B: Incorrect – plausible if unaware of set points and entry for 103.

C: Correct - 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. **<u>If 2 out of 3 sense a rad level of 2x10-3 µci/cc the following occurs:</u>**
The 1E breakers for the RBVE and RBVS fans trip
The secondary containment supply and exhaust dampers close
FRVS starts.
Air exhausted from the reactor building is monitored for radiation, prior to passing through the secondary containment isolation dampers, by three rad monitors. ***<u>If 2 out of 3 monitors sense a rad level of 1x10-3 µci/cc; then, the same actions occur as with the refuel floor monitors.</u>***

D: Incorrect – plausible if unaware of set points and entry criteria for EOP-103.

Technical Reference(s):

EO.ZZ-103,
NOH01RBVENTC -05
HC.OP-SO.GU-0001-
interlocks

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #		
K/A #	EPlan 2.4.6	
Importance Rating	3.7	

2.4.6. Knowledge of EOP mitigation strategies.

Question: #72

An Anticipated Transient Without a Scram (ATWS) is in progress and the operating crew is working through EOP 101A, "RPV Control – ATWS". Drywell pressure is 2.0 psig, rising slowly. Suppression pool temperature is 115°F, rising slowly. The CRS directs performance of the following:

LOWER RPV level, irrespective of any reactor power OR RPV level oscillations.

EXCEPT for SLC, CRD, AND RCIC, **TERMINATE** AND **PREVENT** all injection into the RPV UNTIL RPV level drops below -50 in AND any of the following occur:

- Rx power drops below 4%

  OR
- RPV level reaches -129 in

  OR
- All SRVs remain closed AND Drwl press remains below 1.68 psig

LP-13

Which of the following is the basis for the CRS directing to "Terminate and Prevent all injection into the RPV" given the conditions?

A. To prevent uncontrolled injection of large amounts of cold water as RPV pressure decreases below the shutoff head of operating system pumps.

B. To reduce natural circulation driving head and core flow, thereby reducing reactor power and the heat addition rate to the suppression pool.

C. To protect primary containment integrity and function from excessive injection from sources

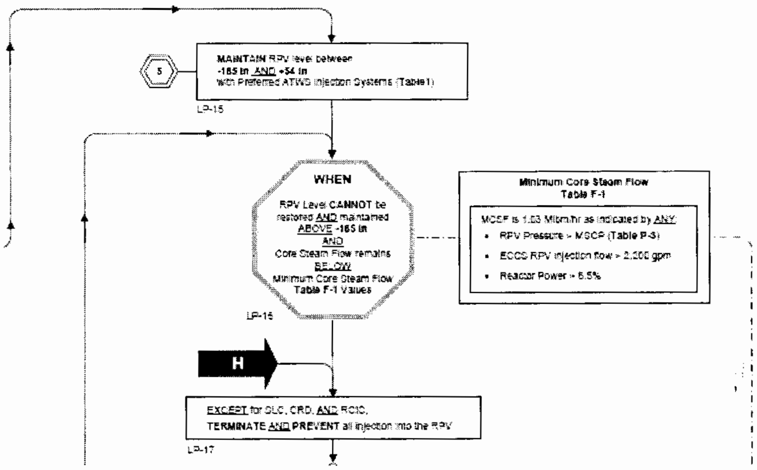
external to the primary containment.

D. To increase core inlet subcooling by lowering RPV level below the feedwater sparger nozzles.

Answer: B

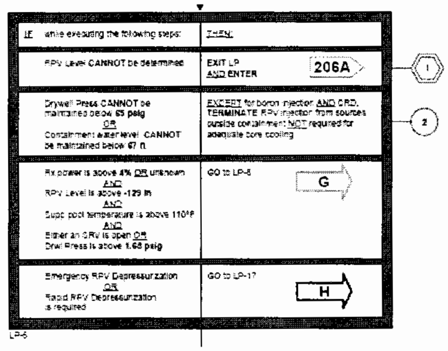
Explanation:

A. Incorrect. This is the basis for EOP 101A Step LP-17. For Step LP-17, Injection into the RPV is terminated and prevented while emergency RPV depressurization proceeds, in order to prevent uncontrolled injection of large amounts of cold water as RPV pressure decreases below the shutoff head of operating system pumps.



B. Correct. If the conditions to use Step LP-13 exist, the operator is or has been directed to reject as much heat as possible from the RPV to the main condenser (EOP-101A Step RC/P-17 first override), to place all available suppression pool cooling into operation (EOP-102 Step SP/T-3), to trip the recirculation pumps (EOP-101A Step RC/Q-9), and to concurrently inject boron and manually insert control rods (EOP-101A Step RC/Q-5). One additional action remains available to mitigate the consequences of a failure-to-scram condition: deliberately lowering RPV water level to effect a reduction in reactor power. Lowering RPV water level reduces natural circulation driving head and core flow, thereby reducing reactor power and the heat addition rate to the suppression pool.

C. Incorrect. This is the basis for EOP 101A Step LP-6 override, which is NOT met for the current drywell pressure:
"If while executing the following steps Drywell pressure cannot be maintained below 65 psig or Containment water level cannot be maintained below 67 ft., then terminate RPV injection from sources outside containment not required for adequate core cooling."



If an unisolable break exists inside the drywell, continued RPV injection from sources external to the primary containment will increase primary containment water level after RPV water level reaches the elevation of the break. The increasing primary containment water level will, in turn, increase the hydrostatic pressure over submerged components and compress the primary containment airspace, thereby increasing the atmospheric pressure. The combination of these effects tends to decrease the margin to the PCPL as the water level rises. If primary containment water level and suppression chamber pressure cannot be maintained below the PCPL, injection from external sources into the RPV is therefore terminated, provided that the injection is not needed for core cooling.

D. Incorrect. The lowering of RPV level is done to decrease core inlet subcooling (allow the temperature to increase), not increase. The desired effect is to increase temperature, not subcooling, which may be confused by the candidate.

Technical Reference(s):
EOP 101A flowchart and bases

Proposed References to be provided to applicants during examination: as shown in stem

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41 (10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	Rad Control 2.3.13	
	Importance Rating	3.4	

2.3.13. 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: #74

Which of the following is the required action if a Locked High Radiation Area key is lost by the individual who checked it out?

The individual shall immediately notify:

A. Security and establish positive control of access to the area.

B. the Shift Radiation Protection Technician and Radiation Protection Supervisor and control all access to the area.

C. the Shift Manager, re-lock the area and have Radiation Protection check for exposures in excess of those expected.

D. the Radiation Department Manager and verify the area locked after checking for unauthorized personnel.

Answer: B

Explanation:

A. Incorrect. Per RP-AA-463, step 4.1.4. “Immediately notify the SRPT and RPS if a key is lost or a lock is damaged.”
B. Correct. Per RP-AA-463, step 4.1.4. “Immediately notify the SRPT and RPS if a key is lost or a lock is damaged.”
C. Incorrect. Per RP-AA-463, step 4.1.4. “Immediately notify the SRPT and RPS if a key is lost or a lock is damaged.”
D. Incorrect. Per RP-AA-463, step 4.1.4. “Immediately notify the SRPT and RPS if a key is lost or a lock is damaged.”

Technical Reference(s):
RP-AA-463 Rev 4.

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 30961 (SIG MOD'D)

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	Conduct of Ops	
	Importance Rating	2.8	

2.1.41 Knowledge of the refueling process

Question: #75

Refueling is in progress with the reactor cavity flooded and the fuel pool gates removed.

Which of the following is a CORE ALTERATION?

A. LPRM detector removal for replacement with fuel in the vessel.

B. Removal of a double blade guide after fuel bundle insertion.

C. Inserting a control rod that had previously been withdrawn 2 notches with fuel in the vessel.

D. Vertical repositioning of Traversing Incore Probe (TIP) within its dry tube with fuel in the vessel.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – not IAW TS definition

B: Incorrect – not IAW TS definition

C: Correct –
HC Tech Spec Amendment 163 section

1.7 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors

(including undervessel replacement), and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

D: Incorrect – not IAW TS definition

Technical Reference(s): Technical Specification definition 1.7

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Modified Bank #33505

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(2)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	EA2.02	
Importance Rating		4.0

295024 High Drywell Pressure
EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell temperature

Question: #76

Given:

- Due to a large reactor coolant leak, suppression chamber pressure was rapidly increasing
- Drywell sprays have been initiated
- Drywell pressure is 10 psig and lowering
- Drywell temperature is 310°F and lowering

If drywell pressure and temperature lowering results in entering the UNSAFE region of the Drywell Spray Initiation Limit (DWT-P), what action is required?

A. Emergency Depressurize the reactor IAW EO.ZZ-0202.

B. Secure all drywell sprays when the Drywell Spray Initiation Limit curve is reached IAW EO.ZZ-0102.

C. Secure drywell sprays at 9.5 psig drywell pressure and lowering IAW EO.ZZ-0102.

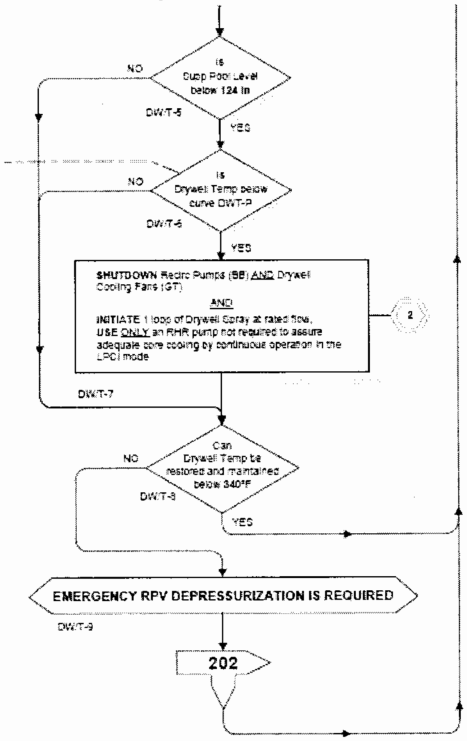
D. Continue drywell sprays until drywell pressure approaches 0 psig IAW EO.ZZ-0102.

Answer: D

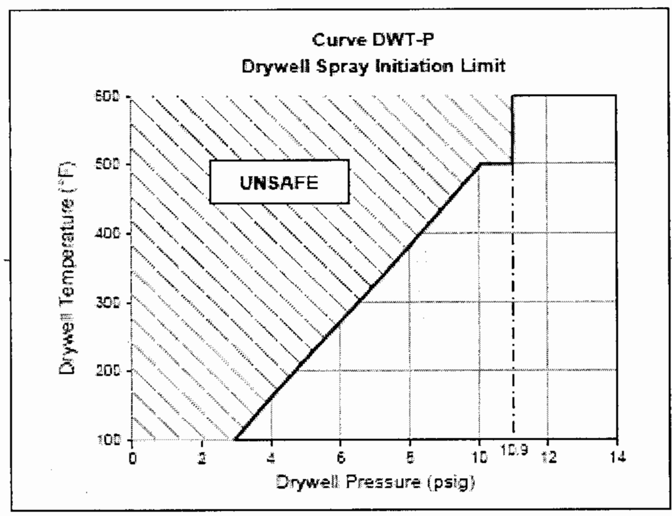
Explanation:

A. Incorrect. DW Temps are improving and are below 340 degF (Yes to DW/T-8). Therefore,

Emergency Depressurization step (DW/T-9) is not entered.



B. Incorrect. Curve is based on spray initiation, not securing spray.



C. Incorrect. DW sprays remain in service until 0 psig. There is no indication in the stem that the RHR pump is needed for adequate core cooling.

D. Correct. Once initiated, DW sprays need only be secured BEFORE DW pressure reaches 0 psig. EOP-102, PCC-1, Retainment Override:

IF while executing the following steps:	THEN:
All entry conditions have cleared	EXIT this procedure
Drywell sprays have been initiated	BEFORE Drywell press reaches 0 psig, TERMINATE Drywell sprays
Supp chamber sprays have been initiated	BEFORE suppression chamber press reaches 0 psig, TERMINATE supp chamber sprays
Only HPCI/RCIC are available for injection	EXIT this procedure AND ENTER 106
SAG entry is required.	EXIT this procedure and ENTER SAG

6

PCC-1

Technical Reference(s):
EOP 102
EOP 202

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 33995

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031 EA2.04	
	Importance Rating		4.8

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: EA2.04 Adequate Core Cooling

Question: #77

The plant was at 100% power when a loss of off-site power occurred.

- The reactor has scrammed (all control rods at 00).
- HPCI and RCIC have tripped and are not available.
- All low pressure ECCS has started on the Level 1 RPV water level signal.
- ADS is inhibited.

Current Plant conditions are:

- RPV level: -140 inches and dropping slowly
- RPV pressure: 900 psig and slowly rising
- SP level: 76 inches
- SP temperature: 113° F

IAW HC.OP-EO.ZZ-0101, what action(s) is/are required?

A. Use the main turbine bypass valves to lower reactor pressure to 400 psig to allow the ECCS systems to inject.

B. Immediately Emergency Depressurize the reactor using alternate depressurization systems.

C. Ensure that one or more low pressure ECCS systems are available, and before RPV level reaches -185 inches, Emergency Depressurize the reactor.

D. Use the Safety Relief Valves to lower reactor pressure to 600 psig and restore level with the secondary condensate pumps.

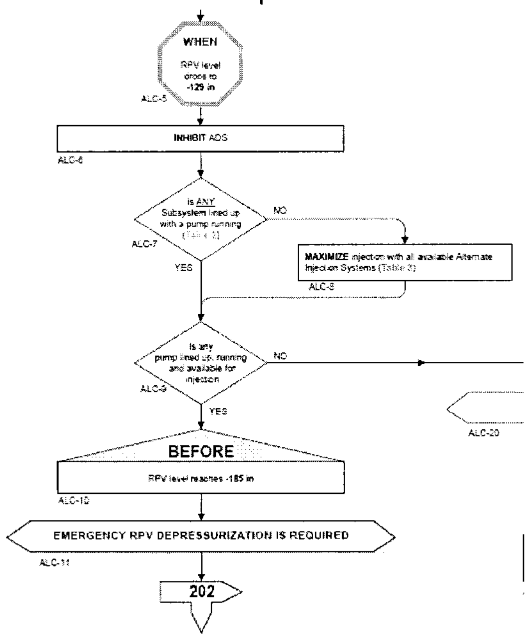
Proposed Answer: C

Explanation (Optional):

A: Incorrect –. Condenser is unavailable since the MSIVs closed at -129".

B: Incorrect – ED criteria is not currently met, ED not permitted

C: Correct – with 1 low pressure ECCS running, ED is required before -185 inches.



D: Incorrect – with the loss of offsite power, secondary condensate pumps are not available

Technical Reference(s):

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # 34404

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

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

295026 Suppression Pool High Water Temp
EA2.01 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature

Question: #78

Given:

The plant is at rated power.

HPCI is in the full flow test mode for a post maintenance retest.

Suppression Pool temperature is currently 100°F.

As the SRO, you should direct the operating crew to ___(1)___ in order to prevent ___(2)___ .

A. (1) Continue current operations. No action is required until suppression pool temperature exceeds 105°F, then terminate the HPCI test and restore water temperature to less than the TS limit within the next 24 hours.
(2) Exceeding the design pressure of 62 psig during primary containment blowdown from full operating pressure.

B. (1) Immediately terminate the HPCI full flow test and restore suppression pool temperature to less than or equal to 95°F within one hour.
(2) Exceeding the design temperature of 340°F during LOCA conditions

C. (1) Continue current operations. No action is required until suppression pool temperature exceeds 110°F, then place the Reactor Mode Switch in the SHUTDOWN position.
(2) Exceeding the peak pressure of 48.1 during LOCA conditions.

D. (1) Immediately terminate the HPCI test and then place the Reactor Mode Switch in SHUTDOWN.
(2) Exceeding the 200°F to ensure complete condensation during blowdown from full

operating pressure.

Answer: A

Explanation:

A. Part 1 is correct per TS 3.6.2.1

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. With an indicated water level between 74.5" and 78.5" and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isolation valves closed following a scram.

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of 0.80 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression chamber average water temperature greater than 95°F and THERMAL POWER greater than 1% of RATED THERMAL POWER, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95° within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Additionally, part 2 is correct per the TS bases:

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 62 psig during primary system blowdown from full operating pressure.

B. Incorrect. Part 1 action describes the requirements in TS 3.6.2.1.a, which is a temperature limit used when there is no testing in progress that adds heat to the suppression pool. The temperature limit is 105°F since there is a test which adds heat to the suppression pool.

C. Incorrect. There is an action required when the temperature exceeds 105°F (see distractor A explanation).

D. Incorrect. Given suppression pool temperature is at 100°F, and there is testing in progress which adds heat to the suppression pool, there is no requirement to immediately terminate the HPCI full flow test (see TS 3.6.2.1a)

Technical Reference(s):
TS 3.6.2.1 and bases

Proposed References to be provided to applicants during examination:

None

Learning Objective:

Question Source: Bank # 36082

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295006 2.2.39	
	Importance Rating	4.5	

2.2.39: Knowledge of less than one hour technical specification action statements for systems
Question: #79

The plant is at 20% power when the plant scrams.

HC.OP-AB.ZZ-0000, "Reactor Scram," has been entered.

Currently:

- RPV Level is +33" and stable.
- RPV Pressure is 1000 psig and stable.
- Mode Switch is Locked in Shutdown.
- All Control Rods full in.
- NI surveillances were complete as of 5 days ago.

As the Control Room Supervisor, you have reached step "S-8": "IF Conditions permit THEN RESET the Scram AND INSERT a Half Scram (if required)."

Which of the following conditions would REQUIRE ordering the RO to INSERT a Half Scram within one (1)hour?

(1) APRM channels 'A' and 'C' INOP.

(2) APRM channels 'B' and 'D' INOP.

(3) IRM channels 'E' and 'F' INOP.

(4) Reactor Vessel Steam Dome Pressure High Transmitter INOP.

A. (1) and (4) ONLY.

B. (2) and (3) ONLY.

C. (1) AND (2) ONLY.

D. (1), (2), (3) and (4).

Proposed Answer: C

Explanation (Optional):

A: Incorrect - Reactor Steam Dome Pressure High transmitter is only required in OPCON 1 or 2, since you are in OPCON 3 this would be NfA

B: Incorrect -Per TS 3.3.1-1 in OPCON 3 you are only required to have 2 IRM's OPERABLE per trip system, since you have 3 available having 1 INOP still leaves 2 that are OPERABLE and so you would NOT have to insert a Half-Scram

C: Correct -Per TS 3.3.1.a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least 1 trip system in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1. For the APRM's in OPCON 3 -Minimum OPERABLE Channels per Trip System is 2, therefore if 2 APRM were INOP you would only have 1 in that Trip System OPERABLE and would need to insert a Half-scram

D: Incorrect -see B above, also Reactor Steam Dome Pressure High transmitter is only required in OPCON 1 or 2, since you are in OPCON 3 this would be NfA

Technical Reference(s): TS 3.3.1

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source:

Modified Bank # FY11 retake question 6

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		G2.4.41
	Importance Rating		4.6

295038 High Off-site Release Rate
2.4.41 - Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications.

Question: #80

Given:

- The plant was operating at 80% power.
- A Main Steam line rupture occurred in the turbine building.
- A Group 1 Primary Containment Isolation signal was received on High Main Steam Line Flow.
- The MSIVs failed to completely isolate the leak.
- The reactor automatically scrammed.
- Reactor level dropped to -45 inches, and level was restored to +30 inches with HPCI and RCIC.
- Primary and Secondary containment parameters remained normal.
- Personnel reported a large steam plume in the turbine building upstream of the turbine stop valves.

Current conditions (75 minutes after the event began):

- Reactor pressure is 200 psig and dropping slowly.
- The steam line leak is still in progress.
- Reactor level is +30 inches with condensate injecting for level control.
- HPCI and RCIC have been secured to slow the depressurization rate and both systems were realigned for automatic operations.
- Primary and Secondary containment parameters are normal.
- Radiation Protection has just provided a dose assessment projection based on Plant effluent samples as follows:
    - o 3.0E-01 mRem TEDE 4-day dose at the MEA
    - o 1.5E-01 mRem Thyroid-CDE dose at the MEA
    - o 2.0E-04 μCi/cc Iodine-131, NPV

Which of the following describes the <u>HIGHEST</u> event classification and the EOP(s) that were or are required to be entered to mitigate the event?
(see reference provided)

A. Alert. EO.ZZ-0101 only.

B. Site Area Emergency. EO.ZZ-0101 only.

C. Alert. EO.ZZ-0101 and EO.ZZ-0103/4.

D. Site Area Emergency. EO.ZZ-0101 and EO.ZZ-0103/4.

Answer: D

Explanation:

A. Incorrect. A Site Area Emergency declaration is required (Loss of RCS Barrier and Loss of Containment Barrier) due to a Main Steam Line Break outside containment with a failure of the MSIVs to isolate leak, with a downstream pathway to the environment. EOP-0101 was required to be entered on low reactor water level. Additionally, EOP 103/4 entry is required because the I-131 level has reached the Alert level (2.0E-04 µCi/cc Iodine-131, NPV) See following tables:

Hope Creek – Fission Product Barrier Table

Reactor Coolant System Barrier		Containment Barrier (NOTE 1)	
Potential Loss (4 pts)	**Loss (5 pts)**	**Potential Loss (2 pts)**	**Loss (3 pts)**
None	RB1.L RPV level CANNOT be restored and maintained above -161 in. OR RPV level CANNOT be determined	CB1.P Primary Containment Flooding is required (SAG Entry) as indicated by ANY one of the following: • RPV Level CANNOT be restored and maintained > -185" with Core Spray unavailable (< 6150 gpm) • RPV Level CANNOT be restored and maintained > -315" with Core Spray in service (≥ 6150 gpm) • RPV Level CANNOT be determined AND core damage is occurring	None
None	RB2.L Drywell pressure > 1.68 psig due to RCS leakage	CB2.P Drywell pressure > 62 psig and rising CB3.P Indications of ≥ 6% H_2 and ≥ 5% O_2 in Drywell or Torus CB4.P RPV pressure and suppression pool temperature CANNOT be maintained below the HCTL (EOP Curve SPT-P)	CB1.L Drywell Pressure rise followed by a rapid unexplained drop in Drywell pressure CB2.L Drywell pressure response NOT consistent with LOCA conditions
RB1.P RCS leakage > 50 gpm inside the drywell RB2.P UNISOLABLE primary system leakage outside primary containment (after isolation from the Control Room has or should have been attempted) as indicated by exceeding EITHER of the following: • ANY EOP 103 Reactor Bldg room temperature Table 1, Column 1 (Max Normal) • ANY EOP 103 Reactor Bldg	RB3.L VALID isolation signal exists with an UNISOLABLE Break outside primary containment (after isolation from the Control Room has or should have been attempted) in ANY of the following systems: • Main steam line • HPCI steam line • RCIC steam line • RWCU • Feedwater	None	CB3.L UNISOLABLE leakage outside primary containment (after isolation from the Control Room has or should have been attempted) AND Direct downstream pathway to the environment exists CB4.L Intentional primary containment venting per EOPs or SAGs. CB5.L UNISOLABLE primary system leakage outside primary containment (after isolation from the Control Room has or

If sum is:	Classify as:	Emergency Action Levels (EALs)	Refer to ECG ATT#
2, 3	UNUSUAL EVENT (NOTE 1)	ANY loss or ANY potential loss of Containment	1
4, 5	ALERT	ANY loss or ANY potential loss of either Fuel Clad or RCS	2
6 - 11	SITE AREA EMERGENCY	Loss or potential loss of ANY two barriers.	3
12, 13	GENERAL EMERGENCY	Loss of ANY two barriers AND Loss or potential loss of the third barrier	4

All

RA 1.2

Liquid Release

IF

VALID Liquid Monitor reading > **the lesser** of the following thresholds on **ANY** one of the following:

For Liquid Radwaste Discharge:
- > 200X High Alarm Setpoint for 9RX508
- > 5.80E-02 uCi/cc
- > 1.50E-02 uCi/cc – CST discharge only

For Cooling Tower Blowdown:
- > 200X High Alarm Setpoint for 9RX506
- > 1.64E-03 uCi/cc

For TB Circ Water Discharge:
- > 200X High Alarm Setpoint for 9RX505
- > 4.80E-04 uCi/cc – for continuous release
- > 5.80E-02 uCi/cc – for batch release

All

RA 1.3

IF

Gaseous Sample Analysis

Confirmed sample analyses for gaseous releases indicate a concentration of > **ANY** one of the following:

- **7.10E-01 µCi/cc** – NG FRVS
- **8.20E-04 µCi/cc** – I-131 FRVS
- **1.52E-01 µCi/cc** – NG NPV
- **1.80E-04 µCi/cc** – I-131 NPV
- **1.44E-02 µCi/cc** – NG SPV
- **1.68E-05 µCi/cc** – I-131 SPV
- **200 x ODCM 3/4.11.2** – for an Unmonitored Release Point

Liquid Sample Analysis

Confirmed sample analyses for liquid releases indicate a concentration of > **200 x ODCM 3/4.11.1** for **ANY** one of the below listed Release Points:

- Liquid Radwaste Discharge
- Cooling Tower Blowdown
- TB Circ Water Discharge
- Unmonitored Release Point

AND

≥ **15 minutes** have elapsed (Note 2)

THEN

Refer to Attachment 2 **ALERT**

Note 2: The Emergency Coordinator should NOT wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown

(Turn Page for SAE EALs) R1

From EOP 104 bases document:

9.1 GASEOUS RADIOACTIVE RELEASE RATE ABOVE ALERT LEVEL EP-HC-111-103

The entry condition corresponds to an Alert action level as defined in the site Emergency Plan. This release rate is sufficiently high that it is not expected to occur during normal plant operations but is sufficiently low such that the condition does not pose an immediate threat to the health and safety of the public.

Although the Site Emergency Plan specifies Alert action levels for liquid as well as gaseous offsite radioactivity releases, it is not possible for a primary system (as the term is defined in the EPGs/SAGs) to generate a liquid offsite radioactivity release. Since this guideline is based on a primary system discharging into an area outside the primary and secondary containments, the Alert entry condition only includes gaseous offsite radioactivity releases.

B. Incorrect. An SAE is required, however, both EOP 101 and 103/4 are entered (see explanation in distractor A).

C. Incorrect. Both EOP 101 and 103/4 are entered. An Alert is declared for the I-131 level. However, A Site Area Emergency declaration is required due to a Main Steam Line Break outside containment with a failure of the MSIVs to isolate leak with a downstream pathway to the environment (see explanation in distractor A).

D. Correct. A Site Area Emergency declaration is required (Loss of RCS Barrier and Loss of Containment Barrier) due to a Main Steam Line Break outside containment with a failure of the MSIVs to isolate leak, with a downstream pathway to the environment. EOP-0101 was required to be entered on low reactor water level. Additionally, EOP 103/4 entry is required because the I-131 level has reached the Alert level (2.0E-04 μCi/cc Iodine-131, NPV) (See tables in distractor A for more EAL bases information).

Technical Reference(s):
Hope Creek Event
Classification Guide

Proposed References to be provided to applicants during examination: EALs and Barrier Table

Learning Objective:

Question Source: Modified Bank # 84350
(SIG MOD)

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295001 2.4.21	
	Importance Rating		4.6

2.4.21 Emergency Procedures /Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control etc.

Question: #81

Given:

- The plant is shutdown.
- Detensioning of the Reactor Head is in progress.
- "A" RHR is in shutdown cooling @ 10,000 gpm.
- "B" RHR in standby.
- Reactor coolant temperature is 135°F and slowly rising.
- The PO determines that the "A" RHR Heat Exchanger has failed due to leakage to SACs.
- The PO removes "A" RHR Loop from service and isolates the heat exchanger.

In accordance with Technical Specifications and station procedures, what action(s) are required for this failure?

A. Declare the ‘A’ RHR Heat Exchanger in operable. Within one hour and at least once per 12 hours thereafter, demonstrate capability of at least one method of reactor coolant circulation.

B.

B. Declare the ‘A’ loop of RHR inoperable. Immediately place the ‘B’ Loop of Shutdown Cooling in service, no tech spec actions are required.

C. Declare the ‘A’ loop of RHR inoperable. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

D. Declare the ‘A’ RHR Heat Exchanger in operable. Immediately establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once every 12 hours.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – loop is inoperable. Must demonstrate another operable method every 24 hours

B: Incorrect – would be correct if level was above reactor flange. TS 3.9.11.1

C: Correct - T.S. 3.9.11.2 Action a. - Within one hour and at least once per 24 hours thereafter, demonstrate operability of at least one alternate method capable of removing decay heat.

With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

D: Incorrect – loop is inoperable. Must demonstrate another operable method every 24 hours

Technical Reference(s):
T.S. 3.9.11.2 Action a.

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Modified Bank # 51957

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #		G2.2.40
	Importance Rating		4.7

295004 Partial or Total Loss of DC Pwr
2.2.40 - Equipment Control: Ability to apply technical specifications for a system.

Question: #82

Given:

- The plant is operating at rated power.
- 125 VDC Battery 1CD411 was DECLARED INOPERABLE 1 hour ago.
- The Quarterly Surveillance Tests for 250 VDC battery 10D431 and 125 VDC battery 1CD447 have just been completed with the following results:
    - 10D431 - Float voltage of one connected cell was 2.05 VDC.
    - 1CD447 - Float voltage of one pilot cell was 2.10 VDC.
    - All other cells tested were within allowable values.

Which of the following describes Technical Specification required actions at this time?
(see reference provided)

A. Declare the 10D431 battery and RCIC inoperable.
Restore 125 VDC batteries 1CD411 and 1CD447 to operable status within 2 (TWO) hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

B. Declare the 10D431 battery and HPCI inoperable.
Restore 125 VDC batteries 1CD411 and 1CD447 to operable status within 2 (TWO) hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

C. Declare the 10D431 battery and HPCI inoperable.
Restore 1CD411 to operable status within 1 (ONE) hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

D. Declare the 10D431 battery and RCIC inoperable.
Restore 1CD411 to operable status within 1 (ONE) hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Answer: D

Explanation:

A. Incorrect. ONLY 1 hour remains before Action A. applies because 1CD411 was declared inoperable 1 hour ago

B. Incorrect. ONLY 1 hour remains before Action A. applies because 1CD411 was declared inoperable 1 hour ago. The system affected by the inop 250 VDC battery (10D431) is RCIC not HPCI.

C. Incorrect. The system affected by the inoperable 250 VDC battery (10D431) is RCIC, not HPCI

D. Correct. Channel B 250VDC battery 10D431 is inoperable, so RCIC must be declared inoperable in accordance with TS 3.8.2.1.b. 1CD411 was declared inoperable an hour ago. Therefore, in accordance with TS 3.8.2.1.a, it must be restored within one hour to meet the two hour requirement. See TS and table below.

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Channel A, consisting of:
   1. 125 volt battery 1AD411
   2. 125 volt full capacity charger 1AD413 or 1AD414
   3. 250 volt battery 10D421;
   4. 250 volt full capacity charger 10D423

b. Channel B, consisting of:
   1. 125 volt battery 1BD411
   2. 125 volt full capacity charger 1BD413 or 1BD414
   3. 250 volt battery 10D431;
   4. 250 volt full capacity charger 10D433

c. Channel C, consisting of:
   1. 125 volt battery 1CD411
   2. 125 volt full capacity charger 1CD413 or 1CD414
   3. 125 volt battery 1CD447
   4. 125 volt full capacity charger 1CD444

d. Channel D, consisting of:
   1. 125 volt battery 1DD411
   2. 125 volt full capacity charger 1DD413 or 1DD414
   3. 125 volt battery 1DD447
   4. 125 volt full capacity charger 1DD444

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any 125v battery and/or all associated chargers of the above required D.C. electrical power sources inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With any 250v battery and/or charger of the above required DC electrical power sources inoperable, declare the associated HPCI or RCIC system inoperable and take the appropriate ACTION required by the applicable Specification.

4.8.2.1 Each of the above required batteries and chargers shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:
   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and
   2. Total battery terminal voltage for each 125-volt battery is greater than or equal to 129 volts on float charge and for each 250-volt battery the terminal voltage is greater than or equal to 258 volts on float charge.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

PARAMETER	CATEGORY A: (*) LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: (*) LIMITS FOR EACH CONNECTED CELL	CATEGORY C: (#) ALLOWABLE VALUE FOR EACH CONNECTED CELL
Electrolyte Level	≥Minimum level indication mark and <¼" above maximum level indication mark[d]	≥Minimum level indication mark and ≤¼" above maximum level indication mark[d]	Above top of plates and not overflowing
Float Voltage	≥2.13 volts	≥2.13 volts[c]	>2.07 volts
Specific Gravity[a]	≥1.200[b]	≥1.195 AND Average of all connected cells >1.205[b]	Not more than .020 below the average of all connected cells AND Average of all connected cells ≥1.195[b]

Technical Reference(s):

Proposed References to be provided to applicants during examination:
TS 3.8.2.1.A and B. and Table 4.8.2.1-1

Learning Objective:

Question Source: Bank # 119577

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	500000 EA2.03	
	Importance Rating		3.8

Ability to determine and/or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: combustible limits for drywell.

Question: #83

Given:

- A large break LOCA has occurred inside the Drywell.
- Multiple equipment failures occurred.
- Drywell pressure is 15 psig.
- Steam cooling was required until water level was restored above TAF with Fire Water.
- The Containment H2/O2 Analyzers were placed in-service 2 hours ago.
- The Containment H2/O2 Analyzers have just alarmed on High Drywell H2 concentration and the trend is upward at 0.5 percent per hour.
- H2 concentration is at 2.1% and O2 concentration is 1.3%.
- The H2 Recombiners are currently NOT in service.

Which of the following actions is required IAW HC.OP-EO.ZZ.0102?

A. Exit EOP-102 and enter SAG since core damage is in excess of what the Emergency Operating Procedures were typically designed to handle.

B. Defeat isolations and place the H2 Recombiners in service IAW EOP-102 and continue to monitor H2 & O2 concentrations.

C. Exit EOP-102 and enter SAG since the lower H2 detonation limit has been reached and containment failure may occur with known fuel damage.

D. Vent containment via the Suppression Chamber to reduce Drywell H2 concentration since the Recombiners could initiate an H2 detonation at these concentrations.

Proposed Answer: A

Explanation (Optional):

A: Correct –. EOP-102 PC/H-1 directs that the EOP be exited and SAG entered is H2 concentration exceeds 2%. In addition, the bases states that a 2% H2 concentration confirms fuel damage above 10 CFR 50.46 ECCS design requirements and beyond what the EOPs were typically designed to handle.

B: Incorrect – With concentrations above 2%, the direction is to exit the EOP and transition to SAG. Now that H2 concentration is above 2%, EOP-102 must be exited. Recombiner operations will be governed by SAG at this point NOT EOP-102.

C: Incorrect – Correct action but wrong reason. H2 and O2 concentrations are well below the LEL for H2.

D: Incorrect – venting is not an option in EOP-102 under the current conditions.

Technical Reference(s): EOP-102 PCH-1

Proposed References to be provided to applicants during examination: none

Learning Objective: EO102PE004

Question Source: Bank #

Question History: 2013 audit

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #		G2.1.32
Importance Rating		4.0

Question: #84

295015 Incomplete SCRAM

2.1.32 - Conduct of Operations: Ability to explain and apply all system limits and precautions.

The plant was in coastdown for a refueling outage when a reactor scram occurred. All control rods inserted to position 02, except control rod 46-31, which inserted to position 36.

Given:

- APRMs are downscale.
- Reactor water level is being maintained at -40" with CRD.
- HPCI and RCIC failed to initiate at Level 2 and are unavailable to restore level.
- There has been a loss of all condensate pumps.
- Both scram discharge volumes have high water levels.

Which of the following will shutdown the reactor?

A. Implement HC.OP-EO.ZZ-0303(Q), "Individual Control Rod Scrams," and place BOTH Rod Test Switches A & B in the TEST position.

B. Implement HC.OP-SO.BF-0002(Q), "Individual CRD Operation," and vent the CRD over-piston volume via the HCU withdraw line.

C. Implement HC.OP-EO.ZZ-0302(Q), "De-energization of Scram Solenoids," and ensure the SDV Vent and Drain Valve fuses are pulled first to isolate the SDV prior to pulling Scram Pilot Valve fuses.

D. Implement HC.OP-EO.ZZ-0306(Q), "Manual Vent of Scram Air Header," and depressurize the scram air header.

Answer: B

Explanation:

A. Incorrect. HC.OP-EO.ZZ-303(Q), "Individual Control Rod Scrams," is used for plant conditions where all or some of the scram solenoids failed to deenergize. This stem conditions do not indicate an electrical ATWAS. All the scram solenoids deenergized to depressurize the scram air header as evidence by the rod motion on all rods. In accordance with the procedure Note 5.3, p. 3, placing both SRI Rod Test Switches in the TEST position will insert the associated control rod. However, as evidence by the rod motion on all rods, the scram air header depressurized and the scram valve opened as designed. Therefore, this procedure will not result in any further rod motion.

From HC.OP-EO.ZZ-0303(Q):

<u>NOTE</u> 5.3
A. The following operations are performed locally at the North and South Bank HCUs, Rx Bldg, Elev. 102' (see Attachment 2).
B. The Rod Test Switches (A & B) are three-position switches (NORMAL-TEST-SRI). Placing both switches in **TEST** inserts the associated control rod. (The SRI position is not connected). The Rod Test switches must be pulled out to be repositioned.

B. Correct. HC.OP-SO.BF-0002(Q), "Individual CRD Operation," is used for plant conditions where a manual control rod insertion method is needed during an ATWAS. With reactor water level currently at -40" and unable to be recovered with anything but currently CRD, there is an active RRCS signal energizing the ARI valves to open and depressurize the scram air header. RRCS is automatically initiated on a low reactor vessel water level (Level 2 @ 38 in.) OR high reactor pressure (1071 psig). Therefore, the scram air header is depressurized (all scram solenoids deenergized, and scram valves opened). The stem indicates a hydraulic lock condition. Performance of HC.OP-SO.BF-0002(Q) offers the best option to further insert the rods.

C. Incorrect. HC.OP-EO.ZZ-302(Q), "De-energization of Scram Solenoids," is used for plant conditions where all or some of the scram solenoids failed to deenergize. This stem conditions do not indicate an electrical ATWAS. All the scram solenoids deenergized to depressurize the scram air header as evidence by the rod motion on all rods. A precaution of this procedure is to:

3.3 Ensure SDV Vent and Drain Valve fuses are pulled first to isolate the SDV prior to pulling Scram Pilot Valve fuses.

D. Incorrect. Procedure HC.OP-EO.ZZ-306(Q), "Manual Vent of Scram Air Header;" is used for plant conditions where the scram valves failed to open because the scram air header did not depressurize. The stem indicates that rod motion was achieved on all rods, indicating that the scram air header was depressurized as designed, and that the scram valves opened. Performing this procedure will not result in any further rod motion since the scram air header is already depressurized. Additionally, an active RRCS signal is energizing and opening the ARI valves, leaving no question as to whether or not the scram air header is depressurized.

From EOP101A:

INSERT control rods using 1 OR more of the following methods as applicable (Refer to RE-AB.ZZ-001 for rod insert sequence)	
Plant Conditions	**Available Methods**
Scram valves failed to open	• **ISOLATE** AND **VENT** the scram air header **USE** OP-EO.ZZ-306
Scram valves opened but SDV is full	• **RESET** the scram, **DRAIN** the SDV, AND **INITIATE** a Manual Scram, **DEFEAT** RPS interlocks **USE** OP-EO.ZZ-320, if necessary • **RESTORE** Instrument Air **USE** OP-EO.ZZ-319
All OR some of the scram solenoids failed to deenergize	• **DEENERGIZE** scram solenoids **USE** OP-EO.ZZ-302 • Individually **SCRAM** control rods **NOT** inserted to OR beyond position 02 **USE** OP-EO.ZZ-303
Manual control rod insertion methods	• **BYPASS** RWM AND manually **INSERT** control rods **USE** HC.OP-AB.ZZ-0001 • **RESTORE** Instrument Air **USE** OP-EO.ZZ-319 • **VENT** control rod over piston volumes **USE** OP-SO.BF-002 • **INCREASE** CRD cooling water header differential pressure. It is not necessary to restore differential pressure for each control rod

1

RC/Q-20

Technical Reference(s):

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295008 AA2.04	
	Importance Rating		3.3

AA2.04 Ability to determine and/or interpret the following as they apply to high reactor water level: heat up rate: plant specific

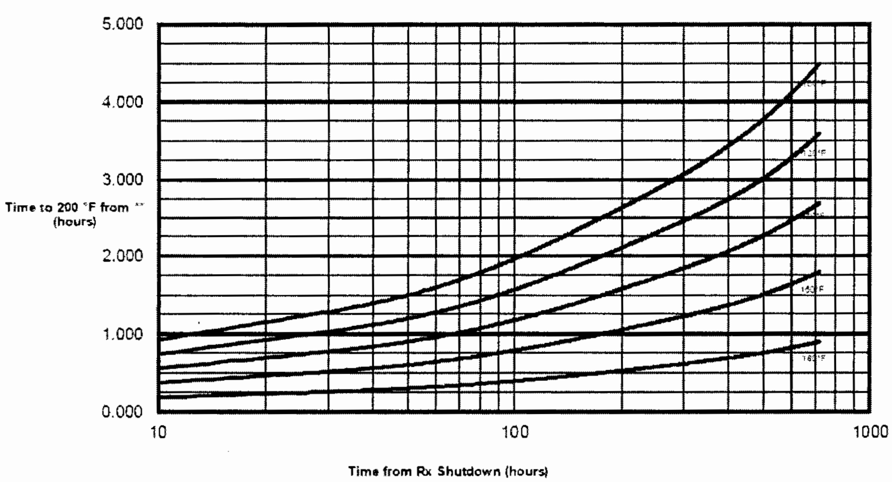
Question #85

Given:

- The plant was shutdown for refueling three (3) days ago.
- Drywell equipment hatch has been opened.
- The Reactor Head will be de-tensioned in 24 hours.
- "A" RHR is in Shutdown Cooling.
- Vessel water temperature is 120°F.

At T=0 minutes, BC-HV-F008 SHUTDOWN COOLING SUCTION OUTBOARD valve, fails closed.

Total Loss of Heat Removal from Rx Vessel
1000 EFPD of Operations



(1) How long will it take for RCS temperature to reach 200°F?
AND
(2) Assuming Shutdown Cooling is restored and RCS temperature begins to lower at T=90 minutes, what emergency declaration must be made?
(see reference provided)

	(1) Time to reach 200°F.	(2) EAL Declaration
A.	75 to 85 minutes	ALERT
B.	90 to 100 minutes	ALERT
C.	75 to 85 minutes	UNUSUAL EVENT
D.	90 to 100 minutes	UNUSUAL EVENT

Proposed Answer: C

Explanation (Optional):

A: Incorrect – by chart and EALs. Shutdown cooling must be unavailable for greater than an hour with reactor temperature greater than 200 degrees

B: Incorrect

C: Correct – only greater than 200 degrees without shutdown cooling for 15 minutes.

D: Incorrect

Technical Reference(s): HC.OP-AB.RPV-0009(Q)
EP-HC-111-227

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #		A2.01
	Importance Rating		2.8

262002 UPS (AC/DC)
A2.03 - Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Under voltage

Question: #86

The plant is at 85% power. Then, the DC supply breaker to the AD481 Inverter (72-41022) trips open due to undervoltage.

(1) Determine which procedure provides the required actions to mitigate this plant condition.
(2) What Technical Specification action(s), if any, is(are) required?
(see reference provided)

A. (1) HC.OP-SO.PK-0001(Q), "125 VDC Electrical Distribution System Operation"
(2) Enter a tracking LCO. No Technical Specification LCO entry is required because the backup power source is still available to supply power to the distribution panel.

B. (1) HC.OP-SO.PK-0001(Q), "125 VDC Electrical Distribution System Operation"
(2) Be in Cold Shutdown within 44 hours of the time breaker 72-41022 was opened, if the associated inverter cannot be made Operable.

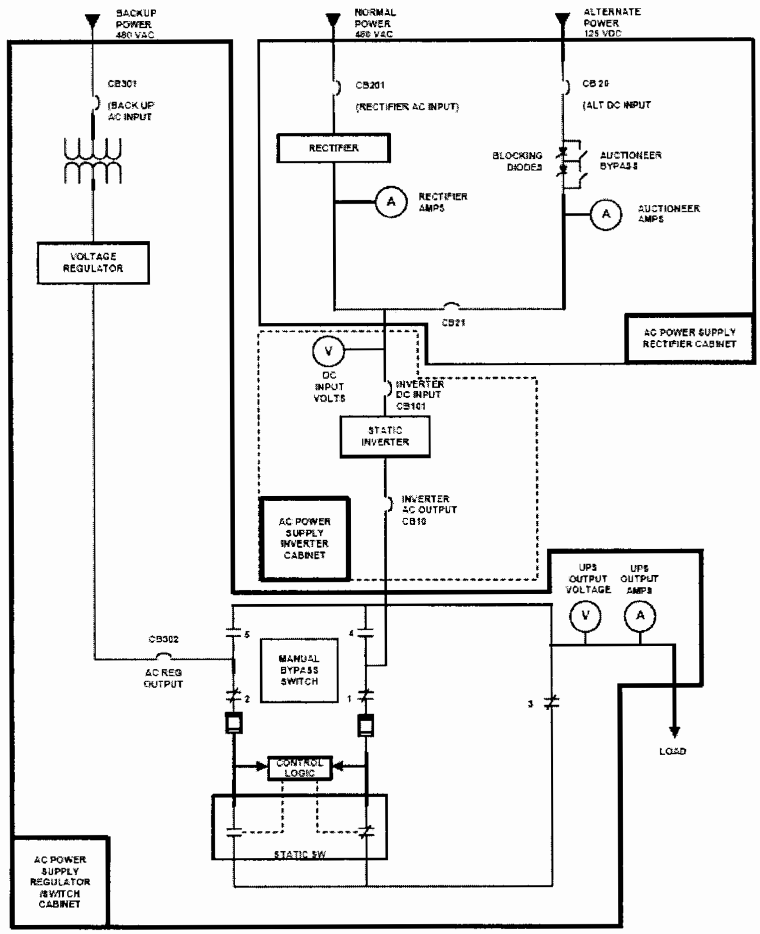
C. (1) HC.OP-AB.ZZ-0136(Q), "Loss of 120 VAC Inverter"
(2) The associated 120 VAC distribution panel must be made Operable within 8 hours to prevent additional required actions.

D. (1) HC.OP-AB.ZZ-0136(Q), "Loss of 120 VAC Inverter"
(2) The associated inverter must be made Operable within 24 hours to prevent additional required actions.

Answer: D

Explanation:

The power supply arrangement to loads having an Uninterruptable Power Supply (UPS) is as follows: 480 VAC (Normal) is rectified and auctioneered with 125 VDC to supply the static inverters. The static switch selects either the inverter output or the regulated 120 VAC backup supply to provide power to the loads.



TS 3.8.3.1 Action d:
With one or both inverters in one channel inoperable, energize the associated 120 volt A.C. distribution panel(s) within 8 hours, and restore the inverter(s) to OPERABLE status within 24 hours; or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

From TS Bases 3.8.3.1:
"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. 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. Alternatively, the power supply may be from an internal AC source via rectifier as long as the OPERABLE station battery is available as the

uninterruptible power supply.

HC.OP-SO.PN-0001- P&L 3.1.2:
For the purpose of defining the OPERABILITY requirements of the 120 VAC distribution panels specified in T/S 3.8.3.1 & 3.8.3.2, the phrase "energized" is defined in the Technical Specification Basis.

A. Incorrect. HC.OP-SO.PK-0001(Q), "125 VDC Electrical Distribution System Operation," would be used to remove or place a 125 VDC electrical distribution system in service. Troubleshooting and repair should be completed before this procedure is implemented. HC.OP-AB.ZZ-0136(Q), "Loss of 120 VAC Inverter" is more appropriate for this condition and the expected alarms. Additionally, a TS entry is required.

B. Incorrect. The 120 VAC distribution panel is energized, but the inverter is inoperable based upon not having the DC input. The requirement for COLD SHUTDOWN in this event would be in 60 hours.

C. Incorrect. This describes the correct TS (3.8.3.1.d), but the 120 VAC distribution panel is energized, so the 8 hour requirement in the first part of this TS is not applicable. The DC supply breaker is open, which requires the inverter to be declared inoperable and restored in 24 hours.

D. Correct. HC.OP-AB.ZZ-0136(Q), "Loss of 120 VAC Inverter," and appropriate Attachment should be entered to determine verify automatic plant response and. TS 3.8.3.1.d - The 120 VAC distribution panel must be energized within 8 hours (it is). The remaining action is to restore the inverter to OPERABLE status within 24 hours.

Technical Reference(s):
TS 3.8.3.1 and bases
HC.OP-SO.PN-0001- 3.1.2.

Proposed References to be provided to applicants during examination:
Supply TS 3.8.3.1

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.43 (2)

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	264000 A2.07	
	Importance Rating		3.7

Ability to (a) predict the impacts of the following of the EDG and (b) based on those predictions use procedures to correct control and mitigate the consequences of those abnormal conditions or operations: loss of offsite power during full load testing.

Question: #87

The plant was operating at 30% power. A monthly surveillance test on the 'C' EDG was in progress.

The output breaker is closed to synchronize the EDG to the grid and within a second, a Station Blackout occurred (loss of all onsite and offsite power). The reactor has scrammed.

Current plant conditions are:

- Drywell temperature 339°F rising slowly
- Drywell pressure 6.7 psig rising slowly
- Torus level 100" and steady
- RPV pressure 320 psig lowering slowly
- Reactor power all rods fully inserted
- Reactor level (-70") lowering slowly
- RCIC tagged out and disassembled for emergent maintenance work
- HPCI tripped on over-speed and will NOT restart
- 'A' EDG tripped -maintenance investigating
- 'B' EDG tripped on Bus differential over-current cause unknown
- 'C' EDG running unloaded -output breaker failed open on antipump circuitry
- 'D' EDG start failure, low starting air pressure -20 psig

What action is required?

A. IAW HC.OP-SO.KJ-0001, Emergency Diesel Generator Operation, direct the NEO to reset the Bus Differential Over-current on the "B" EDG and restart the "B" EDG. Then, start a plant cool down at <100°F/hour IAW EOP-101.

B. IAW HC,OP-AB.ZZ-0135, Station Blackout, Loss of Offsite Power, Diesel Generator malfunction, direct the NEO to reset the Bus Differential Over-current on the "B" EDG and restart the "B" EDG. Then, start a plant cool down at <100°F/hour IAW EOP-101.

C. IAW HC.OP-EO.ZZ-0202, Emergency Depressurization is required based on high Drywell temperature. Then, IAW HC.OP-AB.ZZ-0135, Station Blackout, Loss of Offsite Power, Diesel Generator malfunction, direct the RO to depress the TRIP pushbutton on the "C" EDG output breaker and verify output breaker closes.

D. IAW HC.OP-EO.ZZ-0202, Emergency Depressurization is required before Reactor Water Level decreases to -129". Then, IAW HC.OP-AB.ZZ-0135, Station Blackout, Loss of Offsite Power, Diesel Generator malfunction, direct the RO to depress the TRIP pushbutton on the "C" EDG output breaker and verify output breaker closes.

Proposed Answer: C

Explanation (Optional):

A: Incorrect -bus differential current should NOT be reset without electrical maintenance determining and correcting the cause. An ED must be performed first.

B: Incorrect -This action is not directed from the abnormal. An ED must be performed first.

C: Correct -With Drywell temperature approaching 340^0 F, an ED is performed IAW EOP - 102 step DWIT-S. IAW HC.OP.AB.ZZ-0135, Station Blackout p. 22 step 5.16 -The Anti-pump circuitry on the DIG output breaker could cause the output breaker to fail open. To load the DIG under this condition the operator must depress the TRIP push-button (even though the breaker is already tripped) to reset the logic. When the TRIP push-button is released, then the breaker will close and the DIG will load.

D: Incorrect -Emergency Depressurization procedure should NOT be entered until level is less than -129" but before level decreases to -185"

Technical Reference(s):

Proposed References to be provided to applicants during examination: none

Learning Objective:

Question Source: Bank # FY11 retake question 13

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #		G2.2.37
	Importance Rating		4.6

263000 DC Electrical Distribution
2.2.37 Ability to determine operability and/or availability of safety related equipment.

Question: #88

Given:

- 4.16Kv breaker 52-40101 breaker is tagged in DISCONNECT for Maintenance work.
- All other systems are in a normal lineup.

Then, a phase to ground fault occurs on transformer BX501.

Assuming NO operator actions, what is the resulting impact to the operability of Battery 1AD411?

Battery 1AD411 is...

A. Operable; because 10A401 4.16Kv bus is energized by normal power.

B. Inoperable; because 10A401 4.16Kv bus is de-energized.

C. Inoperable; because one of two battery chargers is de-energized.

D. Operable; because both battery chargers are energized by the "A" EDG.

Answer: A

Explanation:

A. Correct. Normally closed breaker 52-40108 would remain closed and powered from offsite power. The battery charger needs to be energized from the 1E bus to be operable.

B. Incorrect. 10A401 would still be energized. Normally closed breaker 52-40108 would remain

closed and powered from offsite power.

C. Incorrect. Both chargers would be energized.

D. Incorrect. A EDG would not be running on a loss of only one in-feed.

Technical Reference(s):

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 119578

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	212000 2.2.22	
	Importance Rating		4.7

2.2.22 Equipment Control: Knowledge of limiting conditions for operations and safety limits

Question: #89

Given:

- The plant is operating at 80% power.
- The manual scram pushbutton for RPS channel A1 is determined to be INOPERABLE (will NOT cause a half scram when armed and depressed).
- All other systems and components are operable.

IAW Technical Specifications, what action(s) is(are) required, if any?
(see reference provided)

A. Insert a half scram in the RPS "A" logic within one hour and be in HOT SHUTDOWN within 12 hours.

B. Insert a half scram in the RPS "A" logic within 12 hours.

C. Insert a half scram in the RPS "A" logic within 12 hours and be in at least HOT SHUTDOWN within the next 12 hours.

D. NO further operator action.

Proposed Answer: B

Explanation:

With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours.

A. Incorrect
B. Correct – see above
C. Incorrect
D. Incorrect

Technical Reference(s):

TS 3.3.1.a and Table 3.3.1-1

Proposed References to be provided to applicants during examination: TS 3.3.1.a and Table 3.3.1-1

Learning Objective:

Question Source: Bank # 35915

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #		G2.2.42
Importance Rating		4.6

215003 IRM
2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question: #90

Given:

- A reactor startup from COLD SHUTDOWN is in progress.
- The point of adding heat has just been reached.
- IRM B is inoperable and was bypassed with the joystick prior to startup.
- IRM H is observed by the operators to fail downscale, producing a DOWNSCALE alarm but no INOP trip.

The reactor startup:
(see reference provided)

A. may continue if IRM H is placed on Range 1. The RPS trip system with fewer operable channels than required must be placed in the tripped condition within 12 hours.

B. may continue but a mode change cannot be made until one of the inoperable IRMs is returned to service.

C. may NOT continue as it is NOT permitted by Technical Specifications. The control rods may be inserted in the reverse sequence to shutdown the reactor.

D. may NOT continue as it is NOT permitted by Technical Specifications. The control rods may be inserted after IRM H is bypassed with the joystick.

Answer: A

Explanation:

A. Correct. There are 8 IRMs in total. There are 4 IRMs in each trip system: [A, C, E, G] and [B, D, F, H]. The stem states IRM B was already bypassed, and now IRM H has failed downscale. This leaves only two operable channels in the one trip system. The plant is currently in OPCON 2 (point of adding heat). Table 3.3.1-1 states for OPCON 2 you need 3 IRMs per trip system. Since this is not met, the action of TS 3.3.1.a must be taken. Additionally, there is a Rod Withdrawal Block active due to the IRM H downscale failure. To continue startup (rod withdrawal), this needs to be bypassed by placing the IRM H on Range 1.

3/4.3 INSTRUMENTATION
3/4 4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours.

* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	ACTION
1. Intermediate Range Monitors[(b)]:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5[(c)]	3[(d)]	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3[(d)]	3

The provisions of Specification 3.0.4 are not applicable:

3.0.4 When an LCO is not met, entry into an OPERATIONAL CONDITION or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the OPERATIONAL CONDITION or other specified condition in the Applicability for an unlimited period of time; or

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the OPERATIONAL CONDITION or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in OPERATIONAL CONDITIONS or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

B. Incorrect. Startup cannot continue because there is a Rod Withdrawal Block as a result of the IRM H downscale.

C. Incorrect. Rods will not insert because there is a Rod Withdrawal Block active due to the IRM H downscale failure. To continue startup (rod withdrawal), this needs to be bypassed by placing the IRM H on Range 1.

D. Incorrect. Rods will not insert because there is a Rod Withdrawal Block active due to the IRM H downscale failure. To continue startup (rod withdrawal), this needs to be bypassed by placing the IRM H on Range 1. Additionally, since IRM B was already bypassed with the joystick (as stated in stem), you cannot physically bypass IRM H with the joystick.

Technical Reference(s):

Proposed References to be provided to applicants during examination: TS 3.3.1 and table 3.3.1-1

Learning Objective:

Question Source: Bank # 35990

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #		A2.02
	Importance Rating		3.7

290001 Secondary CTMT
A2.02 - Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Excessive outleakage

Question: #91

Given:

- The plant is operating at 70% power.
- Computer Point D9812 shows "NCLSD" on the RM-11.
- Visual observation by the Reactor Building EO confirms that Reactor Building truck bay door (4323A) is NOT closed.
- Damper alignment in the truck bay is NORMAL.
- Reactor Building ΔP is stable at -0.24" WG.

What describes the operational impact and actions, if any, are required?

A. Loss of secondary containment integrity. Take actions IAW HC.OP-AB.CONT-0003, "Reactor Building."

B. Loss of secondary containment integrity. Take actions IAW EOP-103/4, "Reactor Building and Rad Release Control."

C. Potential loss of secondary containment. Start an additional RBVS exhaust fan IAW HC.OP-SO.GR-0001, "Reactor Building Ventilation System Operation."

D. Potential loss of secondary containment. NO additional actions are required unless Reactor Building ΔP reaches -0.30" WG.

Proposed Answer: A

Explanation:

A. Correct. TS 3.6.5.1, secondary containment integrity is lost because the truck bay door is not closed and sealed. With the damper alignment in normal, direct communication to secondary containment exists. HC.OP-AB.CONT-0003, "Reactor Building," discusses mitigation actions.

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend handling of recently irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying in accordance with the Surveillance Frequency Control Program that the reactor building is at a negative pressure.

b. Verifying in accordance with the Surveillance Frequency Control Program that:

1. All secondary containment equipment hatches and blowout panels are closed and sealed.

2. a. For double door arrangements, at least one door in each access to the secondary containment is closed.

   b. For single door arrangements, the door in each access to the secondary containment is closed except for routine entry and exit.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

* When recently irradiated fuel is being handled in the secondary containment and during operations with a potential for draining the reactor vessel.

B. Incorrect. No entry condition exists for EOP 103/4.

C. Incorrect. Secondary containment integrity is lost because the truck bay door is not closed and sealed with the dampers open.

D. Incorrect. Secondary containment integrity is lost because the truck bay door is not closed and sealed with the dampers open.

Technical Reference(s):
TS 3.6.5.1, definitions
HC.OP-AB.CONT-0003

Proposed References to be provided to applicants during examination:
None

Learning Objective:

Question Source: New

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #		G2.4.45
	Importance Rating		4.3

245000 Main Turbine Gen. / Aux.
2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.

Question: #92

Given:

The plant is at rated power. 'B' EHC pump (1BP116) is tagged out for maintenance.

Then, the following alarms are received in the main control room:

- GEN H2/SEAL OIL SYS TROUBLE (E1-C3)
- TURB HYDR PUMP TROUBLE (D3-F5)
- TURBINE GENERATOR VIB HI (D3-C5)

At T=0, the crew reports the following:

- The Main Seal Oil pump tripped. Main Seal Oil discharge pressure is 115 psig and lowering at 1 psig/minute.
- The 'A' EHC pump tripped and wont restart.
- Current EHC pressure is 1315 psig and lowering at 2 psig/second.
- Main Turbine Bearing 10 is reading 7 mils and is steady.

At T= 1 minute, which of the following identifies the HIGHEST priority procedure AND action for this event?

A. Enter HC.OP-AB.BOP-0002, "Main Turbine".
Direct an operator to start the Emergency Seal Oil Pump.

B. Enter HC.OP-AB.BOP-0003, "Turbine Hydraulic Pressure"
Dispatch an operator to investigate the 'A' EHC pump.

C. Enter HC.OP-AB.BOP-0002, “Main Turbine”.
Dispatch and operator to panel 10C366 to monitor main turbine vibration.

D. Enter HC.OP-AB.BOP-0003, “Turbine Hydraulic Pressure”
Lock the Mode Switch in shutdown.

Answer: D

Explanation:

A. Incorrect. The emergency seal oil pump will auto start at 110 psig generator seal oil pressure, which will occur in 5 minutes (1605.00). HC.OP-AB.BOP-0002 action (E.1) is taken in the event the emergency seal oil pump fails to auto start, but its setpoint has not been reached yet. This would not be the priority action to take.

CONDITION	ACTION
E. Hydrogen Seal Oil System Malfunction. Date/Time: ______________	___ **NOTE 9** ___ E.1 IF the Emergency Seal Oil Pump fails to auto start, THEN **START** the Emergency Seal Oil Pump.

B. Incorrect. There is not enough time to send an operator to assess the pump's condition before this limit is exceeded and an automatic reactor scram occurs. With no EHC pumps running and EHC pressure lowering at 2 psig/second, a main turbine trip at 1100 psi is only 108 seconds from 1600 (1601.48).

C. Incorrect. Although the vibrations are elevated, they are steady so this would not be a priority.

D. Correct. HC.OP-AB.BOP-0003 directs locking the mode switch in shutdown when EHC pressure is ≤1200 psig and lowering. At 1601.00, this limit has just been exceeded (1195 psig EHC pressure). The Retainment override directs this action:

PSEG Internal Use Only

HC.OP-AB.BOP-0003(Q)
TURBINE HYDRAULIC PRESSURE

RETAINMENT OVERRIDE	
CONDITION	ACTION
I. EHC Pressure ≤1200 psig AND Lowering Date/Time: ________	___ I.a **LOCK** the Mode Switch in Shutdown.

Technical Reference(s):
HC.OP-AB.BOP-0002
HC.OP-AB.BOP-0003

Proposed References to be provided to applicants during examination: None

Learning Objective: N/A

Question Source: New

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #		A2.03
Importance Rating		3.3

215002 RBM
A2.03 - 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: #93

Given:

- The plant is operating at rated power
- APRM Channel 'D' has failed, and the associated joystick has been placed in bypass.
- Control rod 30-31 is selected, which is located in the center of the core.
- All other plant systems are operating as designed

Then, APRM Channels 'C' AND 'F' fail full "downscale.

(1) Determine the status of the Rod Block Monitor, AND
(2) Which of the following is a required action IAW Technical Specifications?
(see reference provided)

A. (1) ONLY Rod Block Monitor Channel 'A' is bypassed
(2) Verify the reactor is not operating on a Limiting Control Rod Pattern AND restore RBM 'B' to OPERABLE within 24 hours.

B. (1) ONLY Rod Block Monitor Channel 'B' is bypassed
(2) Place at least one inoperable Rod Block Monitor Channel in the tripped condition within one hour.

C. (1) BOTH Rod Block Monitor Channels 'A' and 'B' are bypassed
(2) Verify the reactor is not operating on a Limiting Control Rod Pattern AND restore the inoperable channel to OPERABLE within 24 hours.

D. (1) BOTH Rod Block Monitor Channels 'A' and 'B' are bypassed
(2) Place at least one inoperable Rod Block Monitor Channel in the tripped condition within one hour.

Answer: D

Explanation:

The APRM System supplies normal (APRM channels C and D) and alternate (APRM channels E and F) average core power signals to the RBM channel (A and B) Gain Change Circuit Comparators and reference APRM DNSC Trip Units. The reference APRM signal supplied to a RBM channel will automatically shift from the normal APRM channel (APRM channel C(D)) to the alternate APRM channel (APRM channel E(F)) when the APRM channel BYPASS joystick is placed in the CH. C(D) position ONLY (i.e., the reference APRM channel will NOT be automatically shifted to the alternate APRM channel if a malfunction exists within the normal APRM channel).

APRM 'D' has been bypassed with the joystick, which results in Rod Block Monitor channel 'B' now being supplied by APRM 'F'. The stem indicates that APRM 'F' fails downscale, which now results in RBM 'B' being bypassed (sees <30% power from APRM 'F') AND rendered inoperable because it has no APRM input.

The stem also states that APRM 'C' has failed downscale. Since APRM 'C' is the normal input to RBM 'A', RBM 'A' is currently bypassed (sees <30% power from APRM 'C') AND rendered inoperable because it currently has no APRM input (the reference APRM channel does automatically shift).

Both RBM 'A' and 'B' are bypassed and inoperable. With both RBM 'A' and 'B' inoperable, TS 3.1.4.3b applies:

REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable:

1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

Explanation (see more detail above):

A. Incorrect. Both RBM ‘A’ and ‘B’ are automatically bypassed since their reference APRM channels (C and F) have failed downscale, producing a <30% power signal. Additionally, both RBM ‘A’ and ‘B’ are inoperable, so the correct TS is 3.1.4.3b.

B. Incorrect. Both RBM ‘A’ and ‘B’ are automatically bypassed since their reference APRM channels (C and F) have failed downscale, producing a <30% power signal. Additionally, both RBM ‘A’ and ‘B’ are inoperable, so the correct TS is 3.1.4.3b.

C. Incorrect. Both RBM ‘A’ and ‘B’ are automatically bypassed since their reference APRM channels (C and F) have failed downscale, producing a <30% power signal. Additionally, both RBM ‘A’ and ‘B’ are inoperable, so the correct TS is 3.1.4.3b.

D. Correct. See previous explanation. TS 3.1.4.3b applies a both RBM ‘A’ and ‘B’ are bypassed and inoperable in this situation.

Technical Reference(s):
TS 3.1.4.3b
RBM lesson plan:
NOH04RBMSYSC-03

Proposed References to be provided to applicants during examination:
Provide segment of TS 3.1.4.3 (a and b) as shown above

Learning Objective:

Question Source: SIG MOD’D Bank
116125

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43 (2)
55.43 (5)

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		G2.1.41 Conduct of Ops
	Importance Rating		3.7

Conduct of Operations
2.1.41 – Knowledge of the refueling process.

Question: #94

Given:

- The plant is in Operational Condition 4.
- The Reactor Head detensioning machine is being lowered in place to detension the reactor head.

Which of the following personnel grants permission to begin detensioning of the first RPV Head Stud IAW HC.OP-IO.ZZ-0005 "Cold Shutdown to Refueling"?

A. Control Room Supervisor.

B. Reactor Engineer.

C. Refueling Floor SRO.

D. Refueling Outage Manager.

Answer: A

Explanation:

A. Correct. In accordance with HC.OP-IO.ZZ-0005, Attachment 2, "Entering Operational Condition 5 Final Checks," the "SM/CRS" is responsible to grant permission for entering OPERATIONAL CONDITION 5. There is a Caution in the body of the procedure that references this Attachment:

CAUTION
Do not proceed with Reactor head de-tensioning UNTIL: • **ALL systems AND equipment required to enter Mode 5 are operable.** • **RPV Metal Temps should be recorded on a 30 min. interval.** • **ALL departments have signed Attachment 2, Section A.** ___

5.1.11. IF all systems requirements for entering Operational Condition 5 are satisfied, AND all departments have signed Attachment 2 Section A, THEN **COMPLETE** Section A of Attachment 2. ___ SM/CRS

B. Incorrect. A non-licensed individual is not authorized to grant permission for MODE changes.

C. Incorrect. The licensed SM/CRS in the Control Room has the authority.

D. Incorrect. This individual can recommend a MODE change but cannot grant permission.

Technical Reference(s):
HC.OP-IO.ZZ-0005,
Attachment 2

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 35519

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(6)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #		2.2.44
	Importance Rating		4.4

2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives effect plant and system conditions.

Question: #95

The plant was operating at rated power when a loss of 10A-110 7.2KV Bus occurred during the previous shift.

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 are lower than they were last hour.

As the control room supervisor what actions do you take?

A. Enter HC.OP-AB.RPV-0003(Q), declare the jet pumps inoperable, and commence a unit shutdown per HC.OP-IO.ZZ-0004, be in hot shutdown within 12 hours.

B. Enter HC.OP-AB.RPV-0003(Q), direct the reactor operator to perform HC.OP-ST.BB-0007, Recirculation Jet Pump Operability – Single Loop Daily.

C. Enter HC.OP-AB.RPV-0003(Q), direct the reactor operator to perform HC.OP-ST.BB-0001, Recirculation Jet Pump Operability.

D. Enter HC.OP-AB.RPV-0003(Q), declare the jet pumps inoperable, and immediately insert a full scram, be in hot shutdown within one hour.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - RPV 3 directs a plant shutdown only if the jet pumps are inoperable.

B: Correct – RPV 3 directs assessment of jet pump operability per BB-0007 for single loop (the plant is in single loop operation due to loss of the A recirc pump on the loss of power)

C: Incorrect – RPV 3 is entered on indication of a failed jet pump, BB-0001 is required to assess operability with both loops of recirc running

D: Incorrect – operability needs to be assessed, tech specs require 12 hours to hot shutdown.

Justification

- Core d/p will decrease due to a drop in core flow.
- Reactor power will decrease due to the decrease in core flow
- Steam flow decrease due to the decrease in reactor power

Technical Reference(s):

HC.OP-AB.RPV-0003
HC.OP-ST.BB-0007

Proposed References to be provided to applicants during examination:
None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		
K/A #		G2.3.4 Rad Control
Importance Rating		3.7

Radiation Control
2.3.4 – Knowledge of radiation exposure limits under normal or emergency conditions.

Question: #96

Given:

You are filling out a dose extension IAW RP-AA-203 for an Operator to enter the reactor building steam tunnel.

The Operator's administrative dose control level (ADCL) must be raised to 3000 mrem.

Which of the following personnel must approve this extension?

1. Work Group Supervisor
2. RP Manager
3. Station Manager
4. Site Vice President

A. 1 and 2

B. 3 and 4

C. 1, 2, and 3

D. 1, 2, 3 and 4

Answer: A

Explanation:

A. Correct. Per RP-AA-203 step 4.2.2: To raise the ADCL to 3000 mrem, signatures 1 & 2 are required. To raise the ADCL to 4000 mrem, signatures 1,2, & 3 are required. To raise the ADCL

above 4000 mrem, signatures 1,2,3, & 4 are required.

B. Incorrect, see A
C. Incorrect, see A
D. Incorrect, see A

From RP-AA-203, rev 6:

ATTACHMENT 1
Dose Control Level Extension Form
Page 1 of 1

Section I: Reason for Extension

NAME: ____________________________ SSN: ____________________________

INDIVIDUAL SIGNATURE: _________________________

1. Are other qualified individuals with a lower current year routine TEDE available to perform this work?
   Yes___ No___ N/A ___
   Remarks____

2. State why an extension above 2000 mrem routine TEDE for the year is necessary for this individual.
   ____
   ____

3. It is requested that the individual named above be permitted to receive a TEDE for the current year of ____________mrem.

   Requestor ____________________________ Date ___/___/___

Section II: Dose Summary

CURRENT YEAR ROUTINE RECORD TEDE (mrem): ____________
CURRENT YEAR ROUTINE ESTIMATED TEDE (mrem): ____________
CURRENT YEAR ROUTINE TOTAL TEDE (mrem): ____________
LIFETIME DOSE (mrem): ____________
Verified By: ________________ Date: ___________

Section III: Approvals**

The individual named above is approved to exceed 2000 mrem but must remain below 5000 mrem routine TEDE for the current year.

Specific Approval Level in mrem TEDE: ____________

1. Work Group Supervisor ________________________ Date ___/___/___
2. RP Manager ________________________________ Date ___/___/___
3. Station Manager _____________________________ Date ___/___/___
4. Site Vice President ___________________________ Date ___/___/___

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* (single * note removed due to deleting Exelon references and terminology)

** To raise the ADCL to 3000 mrem, signatures 1 & 2 are required. To raise the ADCL to 4000 mrem, signatures 1,2, & 3 are required. To raise the ADCL above 4000 mrem, signatures 1,2,3, & 4 are required.

Technical Reference(s):
RP-AA-203

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: NRC 2012 exam

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		2.4.11
	Importance Rating		4.2

2.4.11 Knowledge of abnormal condition procedures

Question: #97

Given:

- The plant is operating at rated power and pressure.
- The A CRD pump is tagged out for maintenance.

Then:

- The B CRD pump trips on phase over current, and will **NOT** restart.
- MCR receives multiple CRD accumulator trouble alarms.
- Three (3) withdrawn Control Rods have local accumulator pressures at 935 psig.

Which of the following describes the required actions that must be taken?
(see reference provided)

A. Restart one CRD Pump within 20 minutes.

B. Be in at least HOT SHUTDOWN within one (1) hour.

C. Immediately LOCK the Mode Switch in SHUTDOWN.

D. Within 20 minutes from the point of discovery, LOCK mode switch in SHUTDOWN.

Proposed Answer: D

Explanation (Optional):

A: Incorrect – no available CRD pumps

B: Incorrect – per direction in HC.OP-AB.IC-0001

C: Incorrect – This would be correct if RPV pressure was below 900 psig (Retainment Override)

D: Correct – per direction in HC.OP-AB.IC-0001, Subsequent Action A:

HC.OP-AB.IC-0001(Q)
CONTROL ROD

SUBSEQUENT OPERATOR ACTIONS

CONDITION	ACTION
A. Reactor Pressure is ≥ 900 psig AND Charging Water Header Pressure is < 940 psig AND two or more Control Rod Scram Accumulators are INOPERABLE, at least one of which is associated with a WITHDRAWN Control Rod. **[TS 3.1.3.5.a.2]** Date/Time: ________________	___ A.1 WITHIN 20 Minutes from the point of discovery, **LOCK** Mode Switch in SHUTDOWN.

Technical Reference(s): HC.OP-AB.IC-0001
TS 3.1.3.5

Proposed References to be provided to applicants during examination:
TS 3.1.3.5

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		G2.2.40 Equipment Control
	Importance Rating		4.7

Equipment Control
2.2.40 – Ability to apply technical specifications for a system.

Question: #98

The plant is in MODE 3 at 600 psig.

HC.OP-IS.BE-0002(Q), B&D Core Spray (CS) Pumps - BP206 and DP206 Inservice Testing, is in progress.

The RO reports that CS pump B, discharge pressure transmitter 1BEPT-N055B-E21, does NOT actuate trip unit 1BEPISH-N655B-E21 on Panel 10C618 with Core Spray Pump B operating.

ALL OTHER expected alarms actuated.

- Core Spray Pump "B" Discharge Pressure is 325 psig.
- Core Spray Pump "B" Flow is ~3200 gpm.

Which of the following describes the Technical Specifications impact of this condition?
(see reference provided)

A. "B" ADS logic is INOPERABLE due to failure of the "B" Core Spray pump discharge pressure switch.

B. "B" Core Spray pump is INOPERABLE due to developing insufficient flow.

C. "B" Core Spray pump is INOPERABLE due to developing insufficient discharge pressure.

D. "B" and "D" ADS logic are INOPERABLE due to failure of the "B" Core Spray pump discharge pressure switch.

Answer: A

Explanation:

A. Correct. ADS Logic B is INOPERABLE due to missing ONE of TWO required signals. The automatic actuation of the ADS also requires the operation of one RHR pump or two Core Spray pumps (one loop). The automatic initiation of ADS will be inhibited if the pump discharge pressure permissive is NOT met at the expiration of the appropriate time delays.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. CORE SPRAY SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)(e)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)(e)	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	4/division(f)	1, 2, 3	31
		4*, 5*	32
d. Core Spray Pump Discharge Flow - Low (Bypass)	1/subsystem	1, 2, 3, 4*, 5*	37
e. Core Spray Pump Start Time Delay - Normal Power	1/subsystem	1, 2, 3, 4*, 5*	31
f. Core Spray Pump Start Time Delay - Emergency Power	1/subsystem	1, 2, 3, 4*, 5*	31
g. Manual Initiation	1/division(b)(g)	1, 2, 3, 4*, 5*	33
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2/valve	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2/valve	1, 2, 3	30
c. Reactor Vessel Pressure - Low (Permissive)	1/valve	1, 2, 3	31
		4*, 5*	32
d. LPCI Pump Discharge Flow - Low (Bypass)	1/pump(i)	1, 2, 3, 4*, 5*	37
e. LPCI Pump Start Time Delay - Normal Power	1/pump	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/subsystem	1, 2, 3, 4*, 5*	33
3. HIGH PRESSURE COOLANT INJECTION SYSTEM#			
a. Reactor Vessel Water Level - Low Low Level 2	4	1, 2, 3	34
b. Drywell Pressure - High	4	1, 2, 3	34
c. Condensate Storage Tank Level - Low	2(c)	1, 2, 3	35
d. Suppression Pool Water Level - High	2(c)	1, 2, 3	35
e. Reactor Vessel Water Level - High, Level 8	4(d)	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	1	1, 2, 3	37
g. Manual Initiation	1/system	1, 2, 3	33
4. AUTOMATIC DEPRESSURIZATION SYSTEM##			
a. Reactor Vessel Water Level - Low Low Low, Level 1	4	1, 2, 3	30
b. Drywell Pressure - High	4	1, 2, 3	30
c. ADS Timer	2	1, 2, 3	31
d. Core Spray Pump Discharge Pressure - High (Permissive)	1/pump	1, 2, 3	31

B. Incorrect. Core Spray rated flow is 3175 gpm (6350 gpm per loop at 138.6 psid)

C. Incorrect. Core Spray discharge pressure is normal. ADS auto initiation signal requires 145 psig core spray discharge pressure

D. Incorrect. Core Spray Pumps B and D input the “B” ADS Logic, Core Spray Pumps A and C input the “A” logic. With only Core Spray Pump B switch inoperable only ADS logic “B” is inoperable.

Technical Reference(s):
TS Table 3.3.3-1, pg 3/4 3-33
ADS Lesson Plan,
HC.OP-SO.SN-0001, pg 5

Proposed References to be provided to applicants during examination:
TS Table 3.3.1-1

Learning Objective:

Question Source: Bank # 139921

Question History: None

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: None

Facility:	Hope Creek
Vendor:	GE
Exam Date:	2016
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #	Emergency Plan	
	K/A #	2.4.22	
	Importance Rating		4.4

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations
Question: #99

A large break LOCA has occurred with the following conditions:

- Reactor shutdown, all rods fully inserted.
- RPV pressure has lowered to 40 psig.
- Drywell pressure is 29 psig and rising slowly.
- Drywell temperature is 315 deg. F and rising slowly.
- Suppression Pool pressure 32 psig and rising slowly.
- Suppression Pool temperature 160°F and rising slowly.
- Suppression Pool level is 80 inches and steady.
- All RPV level indicators are reading off-scale high.

Which of the following describes the required utilization of "A" and "B" RHR loops and the basis for this decision?
(see reference provided)

A. Inject into the RPV with both loops of RHR per HC.OP-EO.ZZ-0206(Q)-FC; Adequate Core Cooling is not assured.

B. Place both loops of RHR in Suppression Pool Cooling per HC.OP-EO.ZZ-0102FC; Primary Containment is threatened by exceeding heat capacity temperature limit of the Suppression Pool.

C. Spray the DW with one loop of RHR and spray the Suppression Pool with the other loop per HC.OP-EO.ZZ-0102FC; Primary Containment is threatened by direct pressurization of the Suppression Pool.

D. Inject into the RPV with one loop of RHR per HC.OP-EO.ZZ-0206(Q)-FC and spray the DW with the other loop of RHR per HC.OP-EO.ZZ-0102FC; Adequate

Core Cooling is not assured AND Primary Containment is threatened by direct pressurization of the Suppression Pool.

Proposed Answer: A

Explanation (Optional):

A: Correct – level is unknown due to elevated drywell temperatures.

B: Incorrect – RHR is needed to assure core cooling

C: Incorrect – RHR is needed to assure core cooling

D: Incorrect – both loops of RHR are needed to assure core cooling

Technical Reference(s): HC.OP-EO.ZZ-0102FC, HC.OP-EO.ZZ-0206(Q)-FC

Proposed References to be provided to applicants during examination: EOP Caution 1

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: None

Facility: Hope Creek

Vendor: GE

Exam Date: 2016

Exam Type: SRO

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		G2.2.5
	Importance Rating		3.2

2.2.5 – Knowledge of the process for making design or operating changes to the facility.

Question: #100

The plant is planning to install a new flow control system for HPCI during the next outage.

In accordance with LS-AA-104, "50.59 Review Process," a 10 CFR 50.59 Evaluation would determine if the new flow control system requires ___(1)___.

The administrative process to perform initial acceptance testing of HPCI with the new flow control system is controlled by ___(2)___.

A. (1) NRC approval prior to implementation
(2) OP-AA-103-103, OPERATION OF PLANT EQUIPMENT

B. (1) NRC approval prior to implementation
(2) OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS

C. (1) ONLY NRC notification prior to implementation
(2) OP-AA-103-103, OPERATION OF PLANT EQUIPMENT

D. (1) ONLY NRC notification prior to implementation
(2) OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS

Answer: B

Explanation:

A. Incorrect. The purpose of the OP-AA-103-103 procedure is to provide clear policies regarding

who is authorized to manipulate or operate plant equipment.

B. Correct. IAW LS-AA-104, "This procedure establishes the requirements for preparing, reviewing, approving, and documenting evaluations performed pursuant to the requirements of 10 CFR 50.59, "Changes, tests, and experiments," for determining if a facility or procedure change, test, or experiment requires NRC approval prior to implementation."

   Additionally, IAW OP-AA-108-110, evolutions that require the use of special tests in conjunction with existing procedures may also be classified as special evolutions. Specifically, Attachment 1, "Guidelines for Identifying Special Tests and Evolutions," lists the following as consideration for a special test or evolution:

   > ECCS Operability: Any activity that may prevent the actuation of, or ability of the Core Spray system, Low Pressure Coolant Injection system, High Pressure Coolant Injection System, or the Automatic Depressurization System from performing their ECCS function.

C. Incorrect. NRC approval is the purpose, not notification. The purpose of the OP-AA-103-103 procedure is to provide clear policies regarding who is authorized to manipulate or operate plant equipment.

D. Incorrect. NRC approval is the purpose, not notification.

Technical Reference(s):
LS-AA-104, OP-AA-108-110

Proposed References to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank # 125434

Question History: None

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments: None