Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K1.02	
	Importance Rating	3.3	

K1.02 - Knowledge of the physical connections and/or cause- effect relationships between A.C. ELECTRICAL DISTRIBUTION and the following: D.C. electrical distribution

Question: RO #1

While conducting Undervoltage Relay Testing, the technician inadvertently initiates a LOP signal on the "A" Channel 10A401 Bus.

What will be the effect on its 125 VDC bus voltage and how is the charger supply to the bus affected?

A. The battery will supply the bus with a slight drop in voltage. The charger will come back on line and supply the bus since the AC input and DC output breakers remain closed.

B. The battery will supply the bus with a slight increase in voltage. The charger will come back on line and supply the bus since the AC input and DC output breakers remain closed.

C. The charger must be manually reset. The bus will lose all power until AC is restored to the charger.

D. The battery will supply the bus with a slight drop in voltage. The operator must close the AC input and DC output breakers to the charger once the incoming AC is restored.

Proposed Answer: A

Explanation (Optional):

A: **Correct** - 125 VDC bus voltage drops without the chargers float voltage. AC input and DC output breakers remain closed

B: Incorrect – Bus voltage drops.

C: Incorrect - High Voltage Shutdown conditions need manual reset. Not expected for given conditions.

D: Incorrect - AC input and DC output breakers remain closed

Technical Reference(s): DC Electrical LP (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 68768

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 K1.06	
	Importance Rating	3.9	

K1.06 - Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) and the following: Automatic depressurization

Question: RO #2

The plant was at rated power. An event has occurred.

The "A" RHR pump is running and is the ONLY ECCS pump available.

ADS has automatically initiated.

How are the ADS valves affected if the LOGIC INIT RESET PBs are depressed and then released.

The ADS valves will close then re-open only when the _____.

A. 5 minute timer expires IF all other initiation signals are still present whether or not the RHR pump is running.

B. 5 minute timer expires IF all other initiation signals are still present and IF the RHR pump is still running.

C. 105 second timer expires IF all other initiation signals are still present IF the RHR pump is still running.

D. 105 second timer expires IF all other initiation signals are still present whether or not the RHR pump is running.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – once the 105 second timer is timed out and the RHR pump is running at a discharge pressure >125 psig. The RHR pump must be running

B: Incorrect - once the 105 second timer is timed out and the RHR pump is running at a discharge pressure >125 psig.

C: Correct – IAW HC.OP-SO.SN-0001 Step 3.3.1 - After an ADS auto initiation, the ADS valves will auto close WHEN the LOGIC INIT RESET PBs are pressed, but the valves will reopen when the 105 second timer expires, IF all other initiation signals are present.

D: Incorrect – the RHR pump must be running

Technical Reference(s): HC.OP-SO.SN-0001 Step 3.3.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

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

K2.01 - Knowledge of electrical power supplies to the following: ADS logic

Question: RO #3

Loss of ADS logic power from____________ would prevent an ADS automatic actuation. (assume all actuation setpoints were reached)

A. EITHER 125 VDC BD417 OR 125 VDC DD417

B. BOTH 125 VDC BD417 AND 125 VDC DD417

C. EITHER120 VAC BD482 OR 120 VAC DD482

D. BOTH 120 VAC BD482 AND 120 VAC DD482

Proposed Answer: B

Explanation (Optional):

A: Incorrect - Satisfaction of either ADS Channel B or ADS Channel D will result in the actuation of the ADS and the depressurization of the RPV. This ensures that a single failure will neither initiate nor inhibit the ADS function.

B. **Correct -** IAW ADS Lesson Plan NOH01ADSSYSC-08 Section V.A.3 - 125 VDC Class 1E Distribution System - The 125 VDC Class 1E Distribution System supplies electrical power to the ADS SRV pilot solenoids and the ADS logic channels.

ADS Channel B logic and the A ADS SRV pilot solenoids are powered from 1BD417.
ADS Channel D logic and the B ADS SRV pilot solenoids are powered from 1DD417.

Satisfaction of either ADS Channel B or ADS Channel D will result in the actuation of the ADS and the depressurization of the RPV. This ensures that a single failure will neither initiate nor inhibit the ADS function.

C: Incorrect – ADS Logic power is supplied from 125 VDC

D: Incorrect - ADS Logic power is supplied from 125 VDC

Technical Reference(s): PN1-B21-1060-0063 Sht.1 (Attach if not previously provided)

NOH01ADSSYSC-08 Section V.A.3

Proposed References to be provided to applicants during examination: none

Learning Objective: 6.c. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	215004	K2.01	
	Importance Rating	2.6		

K2.01 - Knowledge of electrical power supplies to the following: SRM channels/detectors

Question: RO #4

Given the following conditions:

- The plant is Refueling
- All equipment is operable
- The RPS shorting links are installed

Then, “A” SRM logic module loses power.

Which one of the following describes the SRM 'A' logic module power supply AND the plant response in regard to a scram signal due to this loss?

A. +24VDC.
A 1/2 Reactor Scram has occurred.

B. 125 VDC.
A 1/2 Reactor Scram has occurred.

C. +24VDC.
ONLY a Rod Block has occurred.

D. 125 VDC.
ONLY a Rod Block has occurred.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – removing the shorting links will cause all NMS scrams to be non-coincident. In this case the shorting links are installed (normal configuration) and a scram signal will not occur

B: Incorrect - The +24 VDC supplies the detector logic module. Removing the shorting links will cause all NMS scrams to be non-coincident. In this case the shorting links are installed (normal configuration) and a scram signal will not occur

C: **Correct** -- An SRM INOP condition will not initiate a Scram signal. IAW SRM Lesson Plan – Section VII.B.2.a. - The +24 VDC supplies the detector HVPS. Loss of this power will result in a loss of the HVPS and generate a channel INOPERATIVE trip.

IAW HC.OP-AB.ZZ-0151 Section 5.1 – With the Mode Switch not in RUN and a loss of the 24 VDC power supply a Rod Block will occur due to the loss of the High voltage power supply LOW

IAW LP Section V.A.2.- ± 24 VDC Power System - supplies detector polarizing voltage and SRM logic modules.

D: Incorrect - The +24 VDC supplies the detector logic module

Technical Reference(s): HC.OP-AB.ZZ-0151 Sect. 5.1 (Attach if not previously provided)

SRM LP Sections V.A.2, V.B.2.a.

Proposed References to be provided to applicants during examination: none

Learning Objective: 11 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 K3.09	
	Importance Rating	3.4	

K3.09 - Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Main steam system

Question: RO #5

Given the following condition:

- The plant is at 100% power with all systems normal
- “B” RPS EPA Breakers 1AN411 and 1BN411 (Normal Supply) have tripped.

2 minutes later, a NSSSS Channel "C" Manual Initiation Logic trip occurs?

Which of the following correctly describes the response of the valves below?

A. All inboard NSSSS isolation MOVs are closed and a half isolation signal is inserted to the MSIV logic.

B. All outboard NSSSS isolation MOVs are closed and all MSIVs are closed.

C. All outboard NSSSS isolation MOVs are closed and a half isolation signal is inserted to the MSIV logic.

D. All inboard NSSSS isolation MOVs are closed and all MSIVs are closed.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - A trip of the 'A' NSSSS channel is necessary to cause isolation of the inboard non-MSIV valves. The MSIVs will be closed on a complete power loss.

B: Correct - "B" RPS de-energized will cause all Non MSIV Outboard NSSSS valves to be closed (Trip of both the B and D NSSSS). The trip of "C" NSSSS channel will cause a complete loss of MSIV power and the MSIVs will close.

C: Incorrect - The MSIVs will be closed on a complete power loss.

D: Incorrect - A trip of the 'A' NSSSS channel is necessary to cause isolation of the inboard non-MSIV valves.

Technical Reference(s): PN1-B21-1090-0062 sht 16 (Attach if not previously provided)
PN1-B21-1090-0062 sht 11
NOH04NSSSS0C-04 pages 17 & 44

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X-32677

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K3.02	
	Importance Rating	3.2	

K3.02 - Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor water level: Plant-Specific

Question: RO #6

Given the following:

The plant is currently in Operational Condition 4, with RHR loop "B" aligned in the shutdown cooling mode.

A crack develops in the shutdown cooling suction piping where it connects to the reactor recirculation loop.

Assuming NO operator action, which of the following statements describes the automatic response of RPV level, the "B" RHR loop valves and the "B" RHR pump as listed below?

A. Reactor water level will lower but remain above the LOCA Level 1 setpoint
The F008, F009 (SDC suction) and F015B (SDC Return) remain open.
The "B" RHR pump trips.

B. Reactor water level will lower to below the LOCA Level 1 setpoint.
F008, F009 (SDC Suction) and F015B (SDC Return) close.
The F017B (LPCI Injection) opens.
The "B" RHR remains in service.

C. Reactor water level will lower but remain above the LOCA Level 1 setpoint
F008, F009 (SDC Suction) and F015B (SDC Return) close.
F004B (Supp Pool Suction) opens
The "B" RHR pump trips.

D. Reactor water level will lower to below the LOCA Level 1 setpoint
F008, F009 (SDC Suction) and F015B (SDC Return) close.
The F017B (LPCI Injection) opens.
F004B (Supp Pool Suction) opens
The "B" RHR pump remains in service.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – the RHR pump would trip as level lowered to below LOCA level 1 (-129") due to the leak location. F008, F009 (SDC Suction) and F015B (SDC Return) close.

B: Correct - RPV Level 3 +12.5 inches would not isolate the leak because of the leak location. Therefore, RPV level would lower until -129" is reached. B RHR would trip when F008 and F009 close.. The F017B (LPCI Injection) would open on the LOCA 1 signal. The pump remains in service because the suction valve interlock for a pump trip is overrideen by procedure (BC-0002 step 5.2.34)

C: Incorrect - F004B (Supp Pool Suction) will not open. The RHR pump trips

D: Incorrect - The RHR pump trips

Technical Reference(s): HC.OP-SO.BC-0001 – Interlocks section 3.3 & Table
HC.OP-SO.BC-0002 – step 5.2.34 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K4.01	
	Importance Rating	2.8	

K4.01 - Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Manual/automatic transfers of control

Question: RO #7

The plant is operating at rated power when the following annunciators are received:

- A2-A1, INST AIR HEADER A PRESSURE LO
- A2-A2, INST AIR HEADER B PRESSURE LO
- A2-B1, COMPRESSED AIR SYSTEM TROUBLE
- A2-B2, COMPRESSED AIR PANEL 00C188

Current air pressures are:

- Service Air pressure is 69 psig.
- Instrument air pressure at the Emergency Instrument Air Receiver is 72 psig.
- Instrument air pressure at the Instrument Air Receivers is 69 psig.

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

A. Instrument Air Dryer 1AF104 Isolation Valve, HV-11416, will be open.
The Standby Service Air Compressor will NOT be running.

B. Instrument Air Dryer 1AF104 Isolation Valve, HV-11416, will be closed.
The Standby Service Air Compressor will be running.

C. The Service Air Supply Header Isolation Valve, HV-7595, will be open.
The Standby Service Air Compressor will NOT be running.

D. The Service Air Supply Header Isolation Valve, HV-7595, will be closed.
The Standby Service Air Compressor will be running.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - Standby Service Air Compressor auto starts at < 92 psig.

B: Incorrect - HV-11416 will be open.

C: Incorrect - V-7595 will be closed, The Standby Service Air Compressor will be running

D: **Correct** – IAW HC.OP-AB.COMP-0001,

< 92 psig Service Air Pressure. - Standby Service Air Compressor Auto Start
≤ 85 psig Emergency Instrument Air Receiver Pressure
- EIAC Auto Start
- RACS Demineralizers Isolate

≤ 85 psig Instrument Air Pressure OR Loss of Power to 1A-F-104- 1-KBHV-11416 OPENS
≤ 70 psig Instrument Air Pressure 1-KAHV-7595 AUTO CLOSES

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

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

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

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

Question: RO #8

Given:

- The FRVS Vent System initiated due to a valid High Drywell pressure signal.
- "A" FRVS Vent fan is in Auto Lead
- Operators have verified FRVS response IAW HC.OP-SO.GU-0001 Filtration, Recirculation, and Ventilation System Operation.

Which one of the following describes the subsequent response, if any, of the FRVS Vent Fans to a valid High Reactor Building Exhaust Radiation signal on all channels?

A. The "A" FRVS Vent Fan will remain operating and the "B" FRVS Vent Fan will remain in standby.

B. The "A" FRVS Vent Fan will remain operating and the "B" FRVS Vent Fan will automatically start from standby.

C. The "A" FRVS Vent Fan will shutdown, then "B" FRVS Vent Fan will automatically start from standby.

D. The "A" FRVS Vent Fan will shutdown and restart from the High Reactor Building Exhaust Radiation signal and the "B" FRVS Vent Fan will remain in standby.

Proposed Answer: A

Explanation (Optional):

A: **Correct** – Per the LP - The FRVS vent fan selected for Auto Lead operation will auto start 1) on the same signals that start the FRVS recirc fans. 2) The LOCA Sequencers will start the Auto Lead fan 19 seconds after initiation. 3) The vent fan selected for Auto operation will auto start under the following conditions:a) Automatic start signal present as described in 1 and 2 above and b) A low flow condition, (6950 scfm), exists on the Auto Lead fan for more than 45 seconds.

B: Incorrect – The AUTO/AUTO Lead selection prevents both fans from starting.

C: Incorrect – 'A' is the Auto Lead Fan. "A" would be running and remain running. 'B' will start only if 'A' vent experiences a low flow condition.

D: Incorrect – 'A' is the Auto Lead Fan. "A" would be running and remain running.

Technical Reference(s): HC.OP-SO.GU-0001, NOH01RBVENTC-00 Sect.III.B.7.3.C (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 120352

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2010

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	206000	K5.08	
	Importance Rating	3.0		

K5.08 - Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM : Vacuum breaker operation: BWR-2,3,4

Question: RO #9

What is the operational implication of the HPCI vacuum breakers failing closed during system operation?

A. Overpressure protection for the HPCI Turbine steam supply piping would be lost.

B. Suppression Chamber pressure would rise to the High HPCI turbine exhaust pressure trip setpoint.

C. Torus water could be drawn into the HPCI Turbine exhaust line.

D. Condensate from the Barometric Condenser would be drawn into the HPCI Turbine exhaust line.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – overpressure protection on the supply side is provided by RPS (high reactor pressure) .

B: Incorrect – overpressure protection is provided by the rupture discs

C: **Correct** – IAW LP, Prevent the condensing action of steam in the HPCI turbine exhaust lines from creating a vacuum and drawing torus water into the exhaust line. Also, the location of the vacuum breakers is on the steam exhaust line to the Torus.

D: Incorrect – breakers on the exhaust line. Not on the piping downstream of the drain pot before the barometric condenser.

Technical Reference(s): LP NOH01HPCI00-11 Page 34 (Attach if not previously provided)
M-55-1

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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

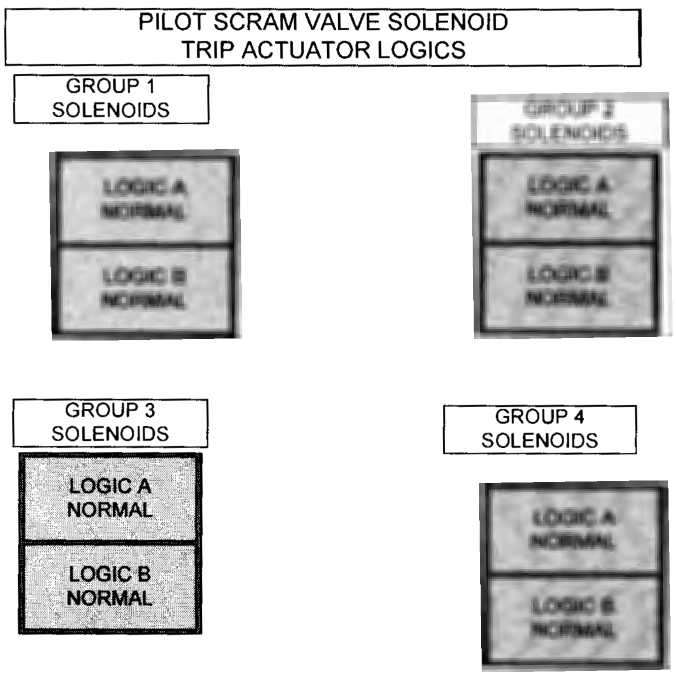
Facility:	Hope Creek 2015		
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K5.01	
	Importance Rating	3.3	

K5.02 - Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM : Specific logic arrangements

Question: RO #10

The plant is at 100% power.

The RO observes that the "GROUP 1 SOLENOIDS" "LOGIC A NORMAL" light is NOT illuminated. The cause is an actual loss of logic power to that logic. ALL other Group Solenoid Logic lights are illuminated.



If a trip of RPS logic ___(1)___ occurs, then ___(2)___ of the control rods will scram.

A. (1) B1
(2) ≈1/2

B. (1) B2
(2) ≈1/4

C. (1) A2
(2) ≈1/4

D. (1) A2
(2) ≈1/2

Proposed Answer: B

Explanation (Optional):

A: Incorrect – Only approx.. ¼ of the control rods would have a full scram signal

B: Correct - ANY 'B' side RPS trip will de-energize the 'B' scram pilot solenoid valves for the GP1 rods, resulting in their scramming in. Each group comprises approximately 1/4 of the rods.

C: Incorrect – The "A" side of RPS is already de-energized

D: Incorrect - The "A" side of RPS is already de-energized

Technical Reference(s): PN1-C71-1020-006, Shts 6,7,13,14
LP NOH01RPS00C-09 pages 12,13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: E012 (As available)

Question Source: Bank # 62627
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 6
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 K6.01	
	Importance Rating	2.7	

K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Valves

Question: RO #11

Given:

The plant is operating at 100% power.
A, B and C SACS pumps are running.

The B SACS pump trips and the D SACS pump is started.
The B2 SACS Hx inlet isolation valve HV-2494B did not open.

How will this affect B SACS Loop operation?

A. The B1 SACS Hx inlet isolation valve EG-HV-2491B is still open, allowing flow and providing cooling for the B SACS Loop.

B. The B1 SACS Hx inlet isolation valve EG-HV-2491B closes, causing the D SACS pump to eventually overheat.

C. The B1/B2 Hx Bypass valves open, allowing flow but not providing cooling for the B SACS Loop.

D. The D SACS pump will auto trip on a failure of B2 SACS Hx inlet isolation valve EG-HV-2494B to open.

Proposed Answer: A

Explanation (Optional):

A: **Correct** - The valves AUTO OPEN when their associated SACS pump is started, regardless of whether the pump control switch is in AUTO or MANUAL (HV-2491A opens when SACS pump A is started, HV-2494A opens when SACS pump C is started, etc.).

B: Incorrect - There are no AUTO CLOSE feature associated with this valve.

C: Incorrect - The Hx Bypass valves do not have any auto open features.

D: Incorrect - The pump does have a low flow trip, but the 2491B is still open, so there is flow in the B SACS Loop.

Technical Reference(s): HC.OP-SO.EG-0001 Section 3.3 & 5.2.4.B.3.b & 5.2.4.B.6
STACS Lesson Plan
NOH04STACS0C-12. Page 18,
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: 2.b. (As available)

Question Source: Bank # X - 110711

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K6.04	
	Importance Rating	2.8	

K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM : D.C. power

Question: RO #12

Given the following conditions:

125 VDC bus 1AD417 is out of service due to an electrical fault
A leak develops inside the drywell from an unidentified source
Drywell pressure is 7.3 psig and rising
RPV level is –118" and lowering
RPV pressure is 290 psig and lowering

What is the total Core Spray system injection flow?

A. ≈ 3,175 GPM

B. ≈ 6,350 GPM

C. ≈ 9,525 GPM

D. ≈12,700 GPM

Proposed Answer: B

Explanation (Optional):

A: Incorrect – this flow value would account for only one pump injecting.

B: **Correct** - the "A" CS Logic would not initiate with a loss of AD417 preventing the "A" CS pump from starting and the injection valve from opening. No injection would occur from the "A" loop. RPV pressure is below shutoff head and RPV injection will be from the B Core Spray loop @ approximately 6350 gpm.

C: Incorrect – This flow amount would require 3 pumps to be injecting. With no flow from the "A" loop only 2 pumps are injecting

D: Incorrect - No injection would occur from A loop. All Core Spray pumps would need to be running for a total injection to the RPV of approximately 12,700 gpm.

Technical Reference(s): PN1-E21-1040-0383 Sht.5 (Attach if not previously provided)

N0HO1CSSYS0C-07 page 40

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 70492

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 A1.01	
	Importance Rating	3.4	

A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: Detector position

Question: RO #13

Given:

- The plant is being shutdown for a refueling outage
- Reactor power is 9%
- "E" IRM failed downscale and is BYPASSED
- "D" IRM failed to fully drive in and its Range switch is placed in Range 7 indicating 25
- "A", "B", "C", "F", "G" and "H" IRMs are indicating between 40 and 60 on Range 8

The Mode Switch is then placed in STARTUP.

Which of the following describes a change, if any, to Reactor Manual Control and RPS status based on the position change of the Mode Switch?

A. NO status changes occur.

B. ONLY a Rod Block will be generated.

C. ONLY a ½ Scram and a Rod Block will be generated.

D. A FULL Scram and a Rod Block will be generated.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – a Rod Block will be generated

B: Correct – A Rod Block will be generated due to detector wrong position ("D" IRM not fully inserted) with the Mode Switch NOT in RUN. All IRMs are within their normal bands on the current ranges therefore no RPS Scram setpoints were reached.

C: Incorrect - ALL IRMs are within their normal band and no RPS setpoints were reached therefore no scram signal occurs

D: Incorrect - ALL IRMs are within their normal band and no RPS setpoints were reached therefore no scram signal occurs

Technical Reference(s): NOH01IRMSYS-03 – Table 2 (Attach if not previously provided)

HC.OP-SO.SE-0001 Table SE-001

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 A1.04	
	Importance Rating	3.6	

A1.04 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor water level control controller indications

Question: RO #14

Given:

- The plant is at 80% power and stable.
- RFPs "B" and "C" are in AUTO, both pump speeds indicate @ 4000 rpm.
- RFP "A" is Manual control, its speed is 1000 rpm with its discharge valve closed.

The operator then opens the "A" RFP discharge valve and depresses the AUTO push button on PDS controller for "A" RFP.

How do the RFP speed indications respond?

A. All RFP speeds remain approximately the same.

B. "A" RFP speed remains the same. Due to the additional flow to the vessel from the "A" RFP, "B" and "C" RFP speeds begin to lower.

C. "A" RFP speed rises. When "A" RFP begins to feed the vessel, "B" and "C" RFP speeds lower.

D. RFPs "B" and "C" speeds remain approximately the same. "A" RFP speed rises to match the speeds of RFPs "B" and "C".

Proposed Answer: C

Explanation (Optional):

A: Incorrect – "A" rises, "B" and "C" lower. (see explanation in "C" below)

B: Incorrect – "A" speed will increase. It cannot feed the RPV at 1000 rpm because its discharge pressure is not high enough.

C: Correct – Placing the "A" RFP in AUTO mode places its control via the Master Level Controller This will cause the "A" speed signal to match the DFCS speed signal sent to the "B" and "C" RFPs and the "A" RFP speed will rise. When RPV level begins to rise above the DFCS level setpoint, the speeds on the "B" and "C" will lower. Eventually all RFP speeds will be approximately the same (lower than 4000 rpm) in order to maintain the DFCS level setpoint.

D: Incorrect – "B" and "C" speeds will lower once "A" begins to feed the RPV

Technical Reference(s): NOH04FWCONTC-07, pages 16 & 25 (Attach if not previously provided)

HC.OP-SO.AE-0001 Step 5.7.2

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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	264000	A2.09	
	Importance Rating	3.7		

A2.09 - Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of A.C. power

Question: RO #15

Given the following conditions:

- The plant was at 100% power when a Loss of Offsite Power occurred.
- The 'A' EDG failed to automatically start.
- Operators were unable to start the 'A' EDG from the Control Room

Which of the following sets of actions describes IAW HC.OP-AB.ZZ-0135, Station Blackout/Loss of Offsite Power;

(1) how to start the 'A' EDG under these conditions

AND

(2) whether the output breaker must be manually closed or will automatically close once the EDG is at rated speed and frequency?

A. (1) AT Panel 1A-C-423 (130' El.), OBSERVE that the READY FOR AUTO START is ON. Then PLACE EMERGENCY TAKE-OVER Switch in the EMERG position. Then PLACE REMOTE ENGINE CONTROL in START.
(2) The 'A' EDG output breaker must be manually closed.

B. (1) AT Panel 10C651E (Main Control Room), Press the DIESEL ENG REMOTE pushbutton for the 'A' EDG and ensure REMOTE light is on. Then, AT Panel 1A-C-421 (102' El.), Place LOCAL ENGINE CONTROL Switch in START.
(2) The 'A' EDG output breaker must be manually closed.

C. (1) AT Panel 1A-C-421 (102' El.) Place the REM/LOC/MAINT CONTROL SELECT switch in MAINTENANCE, Then, Place LOCAL ENGINE CONTROL Switch in START,
(2) The 'A' EDG output breaker must be manually closed.

D. (1) AT Panel 1A-C-422 (130' El.) Place the EMERGENCY TAKE-OVER switch in EMERG. Then, AT Panel 1A-C-423 (130' El.), Place the REMOTE ENGINE CONTROL in START
(2) The 'A' EDG output breaker will auto-close.

Proposed Answer: A

Explanation (Optional):

A: **Correct – IAW AB.ZZ.0135 Att. 6 steps**

B: Incorrect - Pressing the DIESEL ENG REMOTE pushbutton for the 'A' EDG and ensuring the REMOTE light is on does NOT enable the Local Engine Switch on Panel 1A-C-421 (102'El). The REM/LOC/MAINT control select switch at this panel must be in LOC (Local) (see AB.ZZ-0135 Att.7)

C: Incorrect - The REM/LOC/MAINT control select switch at this panel must be in LOC (Local) (see AB.ZZ-0135 Att.7)

D: Incorrect – Placing the EMERGENCY TAKE-OVER switch in EMERG removes the auto close function of the output breaker. This is why the procedure for Manual Emergency Starting of EDG's from the Remote Panels has conditional steps for "IF DIESEL RUNNING LOADED is ON" and "IF loading of the EDG is required locally." The response of the output breaker is dependent on how Remote control was gained (from Control Room or from Emergency Takeover).

Technical Reference(s): HC.OP-AB.ZZ-0135 – Att.6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 62474

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 A2.01	
	Importance Rating	2.6	

A2.01 - 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: RO #16

The plant is operating at 100% power:

All systems are in their normal alignment and operable

Then, the breaker supplying the normal 480 VAC supply to the 1AD482 UPS trips open.

(1) What power supply is now feeding the input to the static inverter section of the 1AD482 UPS

AND

(2) What breaker on the UPS must the operator close (place in "ON" position) when restoring its' normal 480 VAC supply IAW HC.OP-SO.PN-0001 - 120 VAC Distribution?

A. (1) Backup 480 VAC
(2) the Rectifier AC Input Breaker

B. (1) Backup 480 VAC
(2) the Voltage Regulator AC Input Breaker

C. (1) Alternate 125 VDC
(2) the Rectifier AC Input Breaker

D. (1) Alternate 125 VDC
(2) the Voltage Regulator AC Input Breaker

Proposed Answer: C

Explanation (Optional):

A: Incorrect – Backup AC will only supply the inverter if the preferred sources are lost (Normal 480 VAC and Alternate 125 VDC . Normal AC is feed to the UPS through the Rectifier AC Input Breaker.

B: Incorrect - Backup AC will only supply the inverter if the preferred sources are lost (Normal 480 VAC and Alternate 125 VDC. Backup AC is supplied through the Voltage Regulator Input breaker

C: Correct - The inverter section of the UPS is fed from an auctioneered DC power supply. With the rectified AC supply to the auctioneering circuit lost, the Alternate DC will supply the inverter. These are considered the prefered sources of power. ONLY when both preferred sources are lost will the Backup 480 VAC supply the loads. Also the Backup 480 VAC does not pass through the static inverter section of the UPS. The normal AC supply is through the Rectifier AC Input Breaker.

D: Incorrect - Backup AC is supplied through the Voltage Regulator Input Breaker.

Technical Reference(s): HC.OP-SO.PN-0001, Section 5.10 and Exhibit 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: E008 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO		SRO
Tier #	2		
Group #	1		
K/A #	263000	A3.01	
Importance Rating	3.2		

A3.01 - Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights

Question: RO #17

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

A. One switchgear ground detection white light will be brighter than normal and the other will be extinguished.

B. BOTH switchgear ground detection white lights will be brighter than normal at the same intensity.

C. One switchgear ground detection white light will be brighter than normal and the other will be dimmer than normal.

D. BOTH switchgear ground detection white lights will be lit at the same intensity but dimmer than normal.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – No lights will be extinguished

B: Incorrect – One light will be dimmer than the other

C: Correct – IAW lesson Plan - Switchgear ground detection lights (2).
Normally both lights are dim.
If a ground exists, one light will be dim and the other will be bright.
Brightness is dependent upon the magnitude of the ground.

D: Incorrect - One light will be brighter

Technical Reference(s): NOH01DCELEC-05, page (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 110721- reworded
Modified Bank # (Note changes or attach parent)
New

Question History: 2012 NRC

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	211000 A3.07	
Importance Rating	3.7	

A3.07 - Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: Lights and alarms: Plant-Specific

Question: RO #18

The plant was operating at rated power when a transient occurred and required injection of Standby Liquid Control (SLC).

The RO reports the following to the CRS:

- "A" Squib Valve Continuity Light is LIT
- "B" Squib Valve Continuity Light is EXTINGUISHED
- Both SLC pumps are running

Which describes; (1) Whether SLC is injecting at rated flow AND (2) SLC Squib Valve status?

A. (1) Injecting at rated flow
(2) "A" Squib Valve has fired, "B' Squib valve has NOT fired

B. (1) Injecting at rated flow
(2) "A" Squib Valve has NOT fired, "B' Squib valve has fired

C. (1) NOT Injecting at rated flow
(2) "A" Squib Valve has fired, "B' Squib valve has NOT fired

D. (1) NOT Injecting at rated flow
(2) "A" Squib Valve has NOT fired, "B' Squib valve has fired

Proposed Answer: B

Explanation (Optional):

A: Incorrect - With "A" squib continuity light LIT , the squib valve has NOT fired, With "B" squib continuity light EXTINGUISHED , the squib valve has fired

B: Correct - With "B" squib continuity light EXTINGUISHED , the squib valve has fired. The system is designed to allow for 100% flow through a single squib valve.

C: Incorrect - With "A" squib continuity light LIT , the squib valve has NOT fired, With "B" squib continuity light EXTINGUISHED, the squib valve has fired. Rated flow was achieved

D: Incorrect - Rated flow was achieved

Technical Reference(s): NOH01SLCSYSC-07 –pages 11 & 25 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	239002	A4.01	
	Importance Rating	4.4		

A4.01 - Ability to manually operate and/or monitor in the control room: SRVs

Question: RO #19

The plant was operating at 100% power when an MSIV isolation occurred.

Assuming all systems functioned as designed, which of the following accurately describes the sequence of operation of Lo-Lo Set SRVs under these conditions?

A. PSV-F013P will open at 1017 psig. If pressure reaches 1047 psig, PSV-F013H will open. PSV-F013H will close when pressure drops below 935 psig. PSV-F013P will close at 905 psig and re-open at 1017 psig, maintaining reactor pressure 905-1017 psig.

B. PSV-F013H will open at 1017 psig. If pressure reaches 1047 psig, PSV-F013P will open. PSV-F013P will close when pressure drops below 935 psig. PSV-F013H will close at 905 psig and re-open at 1017 psig, maintaining reactor pressure 905-1017 psig.

C. PSV-F013H and PSV-F013P will open at 1047 psig. PSV-F013H will close when pressure drops below 935 psig. PSV-F013P will close at 905 psig and re-open at 1017 psig, maintaining reactor pressure 905-1017 psig.

D. PSV-F013H and PSV-F013P will open at 1047 psig. PSV-F013P will close when pressure drops below 935 psig. PSV-F013H will close at 905 psig and re-open at 1017 psig, maintaining reactor pressure 905-1017 psig

Proposed Answer: D

Explanation (Optional):

A: Incorrect - Pressure MUST reach 1047 psig to arm the Lo-Lo Set opening of PSV-F013P. When this happens, BOTH Lo-Lo Set SRVs will initially open. Additionally, setpoints for the Lo-Lo Set SRVs are reversed. Once armed, PSV-F013H cycles at 905-1017 psig, NOT PSV-F013P.

B: Incorrect - Pressure MUST reach 1047 psig to arm the Lo-Lo Set opening of PSV-F013P. When this happens, BOTH Lo-Lo Set SRVs will initially open.

C: Incorrect - Setpoints for the Lo-Lo Set SRVs are reversed. Once armed, PSV-F013H cycles at 905-1017 psig, NOT PSV-F013P.

D: Correct - Upon isolation of the MSIVs, the reactor will scram. Decay heat will cause pressure to rise. Pressure must reach 1047 psig to arm Lo-Lo Set. At this point, both Lo-Lo Set valves will open. Since SRVs can pass 5% steam flow each, and maximum decay heat is approximately 7% power, pressure will lower. At 935 psig, PSV-F013P will close. Decay heat will rapidly drop to <5% power, such that pressure will lower with only the PSV-F013P SRV open. When pressure drops to 905 psig, the PSV-F013H will close. This will cause pressure to turn and rise. Once initially armed, the PSV-F013H will re-open at 1017 psig. Pressure will cycle between 905 and 1017 psig on the PSV-F013H.

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

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 36240

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A4.07	
	Importance Rating	3.9	

A4.07 - Ability to manually operate and/or monitor in the control room: Reactor pressure

Question: RO #20

The plant was operating at rated power.

A plant event occurred and RCIC initiated due to an automatic initiation signal.

The operator observes that RCIC is not responding with the flow controller in AUTOMATIC and places the flow controller in MANUAL with flow at 600 gpm.

RCIC is now injecting at 600 gpm to the RPV.

Reactor pressure is at 700 psig and stable.

Which of the following describes RCIC operation as reactor pressure changes?

A. As reactor pressure rises, the operator must depress the "▲" (UP arrow) on RCIC flow controller FIC-600 to maintain RCIC flow to the RPV at ≈600 gpm.

B. As reactor pressure rises, RCIC injection flow will remain at ≈600 gpm and no additional operator action is required to maintain system flow to the RPV at ≈600 gpm.

C. As reactor pressure lowers, the operator must depress the "▲" (UP arrow) on RCIC flow controller FIC-600 to maintain RCIC flow to the RPV at ≈600 gpm.

D. As reactor pressure lowers, the operator must depress the "LOWER SET POINT" pushbutton to maintain RCIC flow to the RPV at ≈600 gpm.

Proposed Answer: A

Explanation (Optional):

A: Correct - Placing RCIC flow control in MAN results in an open loop control with the flow controller output becoming a fixed speed demand. Although this will provide stable, constant turbine speed, an operator will have to maintain desired vessel injection flow rate.

1) Flow Controller - The flow controller is a Bailey control station located in the control room on panel 10C650B. It allows the operator to select either the manual or automatic mode of operation.

a) MANUAL - In this mode the operator sets desired RCIC turbine speed.

b) AUTOMATIC - In the automatic mode RCIC turbine speed is automatically adjusted to maintain desired RCIC pump discharge flow established by the operator.

B: Incorrect – with FIC 600 in MANUAL, turbine speed is controlled. As reactor pressure rises it will become greater than RCIC discharge pressure and RCIC flow will begin to lower until there is no flow to the RPV.

C: Incorrect – As RPV pressure lowers, flow to the RPV will increase because RCIC speed remains constant.

D: Incorrect – The lower setpoint PB is only functional while in automatic flow control

Technical Reference(s): HC.OP-SO.BD-0001 Caution 5.3.5 NOH04RCIC00-11 Figure 3 & Page 31 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 2.1.23	
	Importance Rating	4.3	

2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation. (APRMs)
K/A Justification: The questions deals with the APRM Flow converter/comparator network which is referenced in 215005 at K1.16, K6.07, A2.06, A2.07 and A3.05

Question: RO #21

Recirculation flow unit "A" fails downscale with the plant at rated power.

Performing which of the following actions will permit resetting the resultant rod block and scram signals?

A. Bypass Recirculation flow unit "A" with the joystick.

B. Bypass Recirculation flow unit "A" and APRM Channel "A" with their respective joysticks.

C. Bypass Recirculation flow units "A" and "C" and APRM Channel "A" with their respective joysticks.

D. Bypass Recirculation flow unit "A" with the joystick and place the "A" flow unit mode switch in UNLABELED with a false high signal.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - Only removes the rod block

B: Incorrect - Only removes the rod block

C: Incorrect - downscale flow unit inputs to "A"/"C"/"E" APRMs & RPS scram is still active

D: Correct -

IAW HC.OP-SO.SE-0001 note 5.3.6 – Bypassing a Recirc Flow Unit with the joystick only removes it from the Comparator Circuit. To fully bypass unit requires I&C to bypass the unit at Panel 10C608 Pwr Range Neutron Mon Cab.

Step 5.3.6

To bypass a Recirc Flow Unit from the Comparator Circuit, Perform the following:

Place Recirc Flow Monitoring RPS Trip Channel A(B) Monitor Bypass joystick to required flow channel position.

Observe Associated Recirc Flow Monitoring RPS Trip Channel A(B) Monitor Status Flow Bypass Is On.

IAW HC.OP-AB.IC-0004 section F

Recirculation Flow Unit Failure

F.1 - Bypass the affected Flow Unit.

F.2 - Refer to DD.ZZ-0020 for a failed PPC Sensor.

F.3 - Direct Reactor Engineering to Evaluate the flow unit failure on the PPC.

F.4 - **Direct I&C to PLACE the MODE Switch, on the applicable flow unit, to the "UNLABELED" position between STANDBY and ZERO.**

Technical Reference(s): HC.OP-SO.SE-0001 5.3.6 – NOTE (Attach if not previously provided)
HC.OP-AB.IC-0004 Section F.

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 32659

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	223002	2.4.3	
	Importance Rating	3.7		

2.4.3 - Emergency Procedures / Plan: Ability to identify post-accident instrumentation. (PCIS/NS4)

Question: RO #22

Which one of the following describes;
(1) Post-Accident Monitoring level instrumentation indications on Main Control Room panel 10C650A & 10C650C

AND

(2) Why they are the preferred instruments to use during accident conditions?

A. (1) Two channels each of fuel zone level & wide range level
(2) The condensing pots are larger thus the effect of "chugging" is reduced.

B. (1) Two channels each of shutdown level & narrow range level
(2) The condensing pots are larger thus the effect of "chugging" is reduced.

C. (1) Two channels each of fuel zone level & wide range level
(2) The reference legs are shorter thus the effect of higher drywell temperatures are reduced.

D. (1) Two channels each of shutdown level & narrow range level
(2) The reference legs are shorter thus the effect of higher drywell temperatures are reduced.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - The reference legs are shorter thus the effect of higher drywell temperatures are reduced

B: Incorrect – NO narrow range is used for PAMS. The reference legs are shorter thus the effect of higher drywell temperatures are reduced

C: **Correct –** The PAM System utilizes existing reactor vessel level and pressure instrumentation for post accident monitoring. Six level and two pressure transmitters supply signals to indicators and recorders in the control room on 10C650A and C. Two channels each (A & B) of fuel zone level, wide range level, shutdown range level and wide range pressure. Also see references provided.

D: Incorrect - NO narrow range is used for PAMS.

Technical Reference(s): M-42-1 sht.2 (Attach if not previously provided)
HC.OP-ST.SH-0001 step 5.5
NOH04RXINSTC-04 page 24

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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 K4.02	
	Importance Rating	4.1	

K4.02 - Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor SCRAM signals

Question: RO #23

Given the following:

- The Reactor Mode Switch is in STARTUP.
- NO APRMs are bypassed.
- APRM "B" and "E" Meter Function switches on 10C604 are in the "COUNT" position.

Based solely on the indications of current conditions on the attached figures of 10C604 APRM "B" and "E" Count circuits, select RPS Trip status and the Rod Block for these conditions.

A. ONLY "A" Channel RPS Trips AND a Rod Block occurs.

B. NO Rod Blocks OR RPS Trips occur.

C. ONLY a Rod Block occurs.

D. "A" AND "B" Channel RPS Trip AND a Rod Block occurs.

Proposed Answer: A

Explanation (Optional):

A: Correct - In the "Count" position, the number of LPRMs feeding the APRM is indicated by reading the 0-125 scale and dividing by 5. "E" is reading 65 which equates to 13 LPRMs and "B" is reading 75 which equates to 15 LPRMs. An INOP trip is generated by <14 LPRMs. "E" APRM will generate an INOP trip which will result in an "A" RPS trip and Rod Outmotion Block.

B: Incorrect - "E" APRM has only 13 LPRMs inputting, which will result in an INOP trip.

C: Incorrect - Although "E" APRM is generating an INOP trip, this also generates an RPS trip.

D: Incorrect - "B" APRM has 15 LPRMs which is sufficient to preclude an INOP trip.

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

Proposed References to be provided to applicants during examination: Provide picture of LPRM meters (vision #84176

Learning Objective: (As available)

Question Source: Bank # 84176

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 A1.06	
	Importance Rating	3.3	

A1.06 - Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: Lights and alarms

Question: RO #24

A reactor startup is in progress.

All IRMs are on Range 8

The Mode Switch is in RUN

Prior to a Rod movement, 'E' IRM is reading 88 and stable on Range 8, 'F' IRM is reading 86 and stable on Range 8.

A Control Rod near 'E' IRM and 'F' IRM is moved

After the Rod movement, 'E' IRM reads 95 and stable on Range 8, 'F' IRM reads 110 and stable on Range 8

Which of the following describes the control room indications for these two IRMs after the control rod movement?

A. Both IRMs' 'CHANGE RANGE' lamps will illuminate, 'F' IRM will generate a rod block.

B. Both IRMs' 'CHANGE RANGE' lamps will illuminate, Both IRMs will generate a rod block, 'B' RPS 1/2 Scram.

C. ONLY 'F' 'CHANGE RANGE' lamp will illuminate, 'B' RPS 1/2 Scram, 'F' IRM will generate a rod block.

D. Both IRMs' 'CHANGE RANGE' lamps will illuminate. NO Rod Blocks, NO RPS Scram.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - With Mode Switch in RUN the upscale trip and Rod Blocks are bypassed.

B: Incorrect - With Mode Switch in RUN the upscale trip and Rod Blocks are bypassed.

C: Incorrect - With Mode Switch in RUN the upscale trip and Rod Blocks are bypassed.

D: **Correct** – IAW HC.OP-SO.SE-0001 Tables SE-001, 002 – With Mode Switch in RUN the upscale trip and Rod Blocks are bypassed. Change Range lights illuminate at 25% and 75% of scale.

Technical Reference(s): HC.OP-SO.SE-0001 Tables SE-001, 002 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K1.02	
	Importance Rating	4.0	

K1.02 - Knowledge of the physical connections and/or cause- effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: Reactor water level: BWR-2,3,4

Question: RO #25

The plant was operating at 100% power when a loss of feedwater occurred.

RPV water level dropped to below –38" and HPCI and RCIC automatically initiated.
RPV water level is currently +60" and rising.

Assuming NO operator action, which of the following HPCI valves listed below will be closed.

(1) FD-FV-4880 HPCI TURB STOP VLV
(2) BJ-HV-F006 PMP DSCH TO CS ISLN
(3) BJ-HV-8278 PMP DSCH TO FW ISLN MOV

A. (1) and (3) ONLY

B. (1) and (2) ONLY

C. (2) and (3) ONLY

D. (1), (2) and (3)

Proposed Answer: D

Explanation (Optional):

A: Incorrect - Closure of the FV-4880 causes the HV-F006 to close

B: Incorrect – Closure of the FV-4880 causes the HV-8278 to close

C: Incorrect – FD-FV-4879 closes on the +54 inch trip

D: Correct - HPCI receives a turbine trip at +54" RPV water level. This removes control oil from the FD-FV-4880 and FD-FV-4879 valves, causing them to go closed. The closure of the FD-FV-4880 causes the BJ-HV-F006 and BJ-HV-8278 PMP DSCH TO FW ISLN MOV to close. The student must recognize that the FV-4880 also closes when +54 inches is reached to understand the remaining valve closures.

Technical Reference(s): HC.OP-SO.BJ-0001, Interlocks section 3.3 (Attach if not previously provided)
AR-ZZ-0006

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 83999

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 2.1.25	
	Importance Rating	3.9	

2.1.25 - Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Question: RO #26

Given the following conditions:

- The reactor is at 100% power.
- SLCS TANK TROUBLE (C1-E1) annunciator alarms.
- The EO isolates a leaking drain line.
- 4475 gallons remain in the SLC tank.
- Chemistry takes a sample and determines boron concentration to be 13.7%.

SLC needs ____________ added to the SLC Tank to be within its Technical Specification Acceptable Operating Region.

(see attachment)

A. NEITHER Boron nor Water

B. Boron ONLY

C. Water ONLY

D. BOTH Boron & Water

Proposed Answer: D

Explanation (Optional):

A: Incorrect – Water level is outside acceptable region of curve. Boron concentration is well below acceptable level.

B: Incorrect – adding Boron would still result in water level being below the acceptable region of the curve

C: Incorrect - Adding water only will dilute the concentration below 13.7%.

D: Correct - Water must be added to bring level up to the minimum. Starting at 13.7%, this would dilute the concentration further, well less than the minimum required. Boron must also be added to bring the concentration volume within acceptable limits.

Technical Reference(s): TS 3.1.5 figure 3.1.5-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS Figure 3.1.5-1

Learning Objective: (As available)

Question Source: Bank # 66278

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201001 K1.08	
	Importance Rating	3.4	

K1.08 - Knowledge of the physical connections and/or cause- effect relationships between CONTROL ROD DRIVE HYDRAULIC SYSTEM and the following: Reactor manual control system

Question: RO #27

While performing control rod movement for rod exercise during shutdown, you notice inconsistencies in the CRD system indications.

While using continuous withdraw to move one of the control rods from notch 00 to notch 48, the control rod is initially moving slower than the previously withdrawn control rod.

Upon further investigation, drive water pressure drops several psid when the rod starts to move.

Drive water pressure returns to normal after the control rod moved several notches outward.

Which one of the following could cause the indications observed?

A. The selected pair of stabilizer valves are NOT responding during rod withdrawal.

B. One of the two pressure equalization valves is NOT closing during rod withdrawal.

C. The selected CRD Flow Control Valve has a ruptured diaphragm and will NOT move.

D. Initial drive water pressure is set too low and needs to be raised to approximately 275 psid.

Proposed Answer: A

Explanation (Optional):

A: Correct - Stabilizer valves not closing when required will cause Drive water pressure to lower while the rod is in motion.

B: Incorrect - Pressure equalizing valves are in parallel. They are normally closed and open on excessive differential pressure between drive and exhaust header.

C: Incorrect - Drive water pressure would not return to normal.

D: Incorrect - Drive water pressure would not return to normal.

Technical Reference(s): M-47-1 sht 1 (Attach if not previously provided)
NOH04CRDHYD-09 Figure 13

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X - 33444
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 6
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	226001 K2.02	
Importance Rating	2.9	

K2.02 - Knowledge of electrical power supplies to the following: Pumps (RHR)

Question: RO #28

Given:

- The reactor is at 55% power
- All vital bus normal in-feed breakers are closed.
- The vital bus alternate in-feed breakers are open. (available-NOT tagged)

Then, a high drywell pressure condition occurs (1.75 psig and rising).

After 30 seconds, the A RHR (1AP202) pump is powered from what power supply and how did it start?

A. Powered via the 1AX501 Station Service Transformer and the pump started via the LOCA Sequencer.

B. Powered from the A Emergency Diesel Generator and the pump started via the LOCA Sequencer.

C. Powered via the 1AX501 Station Service Transformer and the pump started via the RHR logic.

D. Powered from the A Emergency Diesel Generator and the pump started via the RHR logic.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - A RHR pump started from the RHR Logic, " RHR and Core Spray pump starting are not controlled by the sequencers"

B: Incorrect - With only a LOCA signal present and NOT a LOP, the normal power supply to the A RHR pump is still 1AX501, the A EDG is running unloaded and the A RHR pump started from the RHR Logic, " RHR and Core Spray pump starting are not controlled by the sequencers"

C: **Correct** - With only a LOCA signal present and NOT a LOP, the normal power supply to the A RHR pump is still 1AX501, the A EDG is running unloaded and the A RHR pump started from the RHR Logic, " RHR and Core Spray pump starting are not controlled by the sequencers"

D: Incorrect - With only a LOCA signal present and NOT a LOP, the normal power supply to the A RHR pump is still 1AX501, the A EDG is running unloaded."

Technical Reference(s): E-0001, NOH01RHRSYSC, page 19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: 1EAC00E003 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290002 K3.03	
	Importance Rating	3.3	

K3.03 - Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following: Reactor power

Question: RO #29

The plant is operating at 100 percent power.

Which one of the following describes the effect on the plant if pieces of foreign material blocked several fuel support piece flow orifices near center rods?

A. Core thermal power would lower.

B. Indicated reactor water level will fluctuate.

C. Jet pump net positive suction head would raise.

D. Steam quality exiting the reactor vessel will lower.

Proposed Answer: A

Explanation (Optional):

A: **Correct -** Low reactor coolant flow past bundles will increase voids in the channel. Reactor power will lower.

B: Incorrect - only a large change in void concentration will cause indicated RPV level to change a large amount

C: Incorrect - NPSH is related to downcomer temperature, a power reduction would lower power not temperature.

D: Incorrect - quality is effected by excessive flow through the dryers and separators, neither should be effected by this minor change

Technical Reference(s): NOH01RXVESSC-04 Page 21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: 8 (As available)

Question Source: Bank # 34115

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	245000 K4.03	
	Importance Rating	2.7	

K4.03 - Knowledge of MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS design feature(s) and/or interlocks which provide for the following: Sealing to prevent hydrogen leakage

Question: RO #30

Which of the following are designed to the ensure hydrogen does not leak along the Main Generator seals?

(1) Main Lube Oil Bearing Header
(2) Main Seal Oil Pump
(3) Emergency Seal Oil Pump
(4) Recirculating Seal Oil Pump

ONLY _________.

A. (1), (2) and (3)

B. (1), (2) and (4)

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

D. (1), (3) and (4)

Proposed Answer: A

Explanation (Optional):

A: **Correct –**
Recirculating Seal Oil Pump. *The RSOP only circulates the oil to remove any entrapped gasses.*

Emergency Seal Oil Pump. *During emergency operations the ESOP will supply oil to the seals through the normal flowpath.*

Main Turbine Lube Oil Supply Header. *With neither the MSOP or the ESOP running the MTLO bearing oil header will supply the seals with oil.*

Main Seal Oil Pump. *Normal source.*

B: Incorrect – NOT the Recirculating Seal Oil Pump

C: Incorrect – NOT Recirculating Seal Oil Pump

D: Incorrect – NOT Recirculating Seal Oil Pump

Technical Reference(s): NOH04SEALOL-05 (Attach if not previously provided)
HC.OP-SO.CD-0001

Proposed References to be provided to applicants during examination: none

Learning Objective: SEALOLE002 (As available)

Question Source: Bank # X - 30239

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	234000 K5.02	
	Importance Rating	3.1	

K5.04 - Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT : †Fuel handling equipment interlocks

Question: RO #31

Given the following conditions:

- Core reload is in progress
- The mode switch is in Refuel
- An assembly has just been latched in the fuel pool

Then, I&C testing causes a rod out signal to be generated.

What operational limitations would be encountered if the refuel bridge operator attempted to place the assembly in the core?

A. The bridge could not move in any direction.

B. The main hoist would not lower after reaching the core location.

C. The bridge would not be able to move over or near the core.

D. The main hoist could not be raised from the fuel pool location.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – Bridge movement is allowed except over or near the core.

B: Incorrect – The main hoist could not reach the core location because the refuel bridge is unable to travel to the core.

C: Correct - The refuel bridge has an interlock that prevents bridge movement towards the reactor if a control rod is withdrawn and the main hoist is loaded and the bridge is near or over the reactor vessel (as determined by refuel switches #1 and #2).

D: Incorrect – the main hoist will still function in the fuel pool area

Technical Reference(s): NOH01REFUEL-09 page 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 77466

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 4

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 K6.07	
	Importance Rating	3.8	

K6.07 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM : Reactor water level control system

Question: RO #32

The plant was operating at 80% power.

A Control Signal Failure occurred on the "C" RFPT.

"C" RFPT was in service with flow matched to the other pumps prior to the control signal failure.

Then, a Main Turbine trip without bypass valves occurs.

The following conditions exist one minute later.

- All rods failed to fully insert and Reactor power is at 6%.
- Reactor level is 10 inches and slowly lowering.
- Reactor pressure initially increased to 1100 psig and is now at 1080 psig.

Which one of the following describes the status of the Reactor Feed Pump speed?
(Assume NO operator action has been taken)
(NOTE: 650 RPM is Governor Minimum)

A. "A" is operating at 650 rpm.
"B" is operating at 650 rpm.
"C" is operating at 2500 rpm.

B. "A" is operating at 2500 rpm.
"B" is operating at 2500 rpm.
"C" is operating at 650 rpm.

C. "A" is operating at 650 rpm.
"B" is operating at 650 rpm.
"C" is operating at 650 rpm.

D. "A" is operating at 2500 rpm.
"B" is operating at 2500 rpm.
"C" is operating at 2500 rpm.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – "C" would be at governor minimum and the others at 2500 rpm due to RRCS runback signal

B: Correct - If a RFP is operating with a control signal failure, the RFP is [runback and held at governor minimum speed (650 rpm)] effectively removing the pump from service until RRCS is reset. The RRCS signal causes a runback of the other pumps to 2500 rpm.

C: Incorrect – only C would be at governor minimum

D: Incorrect – C would be at governor minimum

Technical Reference(s): HC.OP-SO.SA-0001 section 5.2.1.D (Attach if not previously provided)
NOH04RRCSS00C-03, page10

Proposed References to be provided to applicants during examination: none

Learning Objective: 2 and 3a (As available)

Question Source: Bank # X-119039

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	230000 A1.10	
	Importance Rating	3.7	

A1.10 - Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE controls including: System lineup

Question: RO #33

The plant is operating at rated power,

At T=0 minutes - a small break LOCA occurred.

Drywell pressure	2 psig slowly rising
RPV level	-26 inches slowly lowering
RPV pressure	860 psig slowly lowering
Suppression pool temperature	102 deg F slowly rising

At T=4 minutes - the "A" loop of RHR was placed in Suppression Pool Cooling/Spray.

At T= 6 minutes – the leak grows larger

At T=8 minutes -, RPV water level is -200 inches.

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

BC-HV-F048A, Hx Bypass Valve ____________

A. opens and suppression pool cooling/spray lineup isolates.

B. opens and suppression pool cooling/spray lineup is unaffected.

C. remains closed and suppression pool cooling/spray lineup isolates.

D. remains closed and suppression pool cooling/spray lineup is unaffected.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - The LPCI initiation signal has not been reset, therefore the F048A will not reset and it will remain closed. The override on the F024A/F027A will not reset, and it will remain open, SPC/Spray will not isolate.

B: Incorrect - The LPCI initiation signal has not been reset, therefore the override on the F048A will not reset, and it will remain closed.

C: Incorrect - The LPCI initiation signal has not been reset, and the F024A/F027A valve override will remain in effect, the F048A will remain closed in its in SPC position.

D: Correct – IAW HC.OP-SO.BC-0001,

The LPCI initiation signal has not been reset, and the F024A/F027A valve override will remain in effect, the F048A will remain closed in its in SPC position.

Technical Reference(s): HC.OP-SO.BC-0001, Interlocks section (Attach if not previously provided)
E11-1040 Sht 9,10,19,20

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 X

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201002 A1.02	
	Importance Rating	3.4	

A1.02 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: Control rod position

Question: RO #34

Reactor power is at 10% during a startup

I&C requests a check of the Rod Drift test circuit and requests that you depress the Rod Drift "test" pushbutton.

When you depress the "test" pushbutton, the Rod Drift overhead annunciator will alarm_______.

A. with NO rod motion.

B. with rod motion ONLY when the control rod drive mechanism passes an odd reed switch.

C. with rod motion ONLY when the control rod drive mechanism passes an even reed switch.

D. with rod motion when the control rod drive mechanism passes EITHER an odd OR even reed switch.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – there must be some rod motion past an odd reed switch with the test pb depressed

B: Correct – IAW LP NOH04MANCONC-07 Page 15 - The TEST pushbutton (yellow) - when depressed, in conjunction with the withdrawal or insertion of a control rod, a ROD DRIFT alarm will actuate as the control rod drive mechanism passes an odd reed switch. During normal control rod motion, the ROD DRIFT alarm does not actuate as odd reed switches are passed because the rod motion timers are in operation. The rod drift TEST pushbutton effectively "blinds" the ROD DRIFT alarm circuitry to timer operation which allows testing of the ROD DRIFT alarm.

C: Incorrect – an odd NOT even reed switch must be passed.

D: Incorrect – will not alarm when passing an even reed switch

Technical Reference(s): NOH04MANCONC-07 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: MANCONE005 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202002 A3.02	
	Importance Rating	3.4	

A3.02 - Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Lights and alarms

Question: RO #35

The plant is operating at 92% power.

Then, a speed control signal failure occurs on the "B" Reactor Recirc Pump and the following annunciator is received.

- C1-F5, COMPUTER PT IN ALARM

10 seconds later the "A" Reactor Feedwater Pump trips and the following annunciators are received.

- A7-D5, "RPV LEVEL 4"
- B3-E1, "RFP TURBINE TRIP"
- B3-F1, "DFCS ALARM/TRBL"
- B3-F2, "CROSSFLOW ALARM/TRBL"

Reactor water level lowers to 25 inches and then recovered to stabilize at 35 inches.

Which of the following indications would be observed on the "B" Reactor Recirc Pump control bezels concerning a Runback and Scoop Tube Lockup given these events?

A. Intermediate Runback ONLY

B. Intermediate Runback and Scoop Tube Lockup

C. Full Runback ONLY

D. Full Runback and Scoop Tube Lockup

Proposed Answer: B

Explanation (Optional):

A: Incorrect – A scoop tube lockup would also occur on the speed control signal failure. That failure also causes the Computer point in alarm annunciator.

B: Correct - The scoop tube lockup would occur on the speed control signal failure. That failure also causes the Computer point in alarm annunciator. When the RFP trips the "RFP Turbine Trip" alarm annunciates. Reactor level begins to lower to 30 inches resulting in the "RPV LEVEL 4" annunciator. When level 4 is reached, an Intermediate Runback occurs on Both Recirc Pumps. This then results in the other 2 annunciators as level and power lower.

C: Incorrect – Conditions were not met for a Full Runback. Conditions were met for a scoop tube lockup.

D: Incorrect – Conditions were not met for a Full runback

Technical Reference(s): HC.OP-AB.RPV-0004 – Auto Actions Table (Attach if not previously provided)
HC.OP-AR-ZZ-0007 – E1
NOH01RECCON-13 Page 28

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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006 A4.06	
	Importance Rating	3.2	

A4.06 - Ability to monitor automatic operations of the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) including: Selected rod position indication: P-Spec(Not-BWR6)

Question: RO #36

Two minutes following a scram the following indications appear on the Confirm Shutdown screen of the RWM.

ALL RODS IN: NO
SHUTDOWN: YES
RODS NOT FULL – IN: 003

Select the status of control rods or the RWM.

A. 3 control rods are at position 04 and the remainder are at 02 or FULL-IN. The RWM is displaying the correct SHUTDOWN Status.

B. 3 control rods are beyond position 04 and the remainder are at 00 or FULL-IN. The RWM is incorrectly displaying the SHUTDOWN status

C. 3 control rods are at position 02 and the remainder are at either 00 or FULL-IN. The indicated RWM SHUTDOWN status is correct.

D. 3 control rods are at position 02 and the remainder are at either 00 or FULL-IN. The RWM is displaying the incorrect SHUTDOWN status

Proposed Answer: C

Explanation (Optional):

A: Incorrect – The RWM display for RODS NOT FULL IN displays the rods that are at position 02 or beyond.

B: Incorrect - – The RWM display for RODS NOT FULL IN displays the rods that are at position 02 or beyond. RWM shutdown status is correct

C: **Correct** – IAW SO.SF-0003. The shutdown status will indicate "YES" if all rods are to 02 or beyond. Therefore the 3 "rods not full in" must be at 02.

D: Incorrect – RWM shutdown status is correct

Technical Reference(s): HC.OP-SO.SF-0003 step 5.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X-107634

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 2.4.31	
	Importance Rating	4.2	

2.4.31 - Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures. (Control Rod and Drive Mechanism)

Question: RO #37

Given the following conditions:

The plant is operating at 100% power

Then;

- A Reactor scram occurs due to a spurious MSIV closure
- Only about one-third of the control rods fully insert due to an undetected high water level in the Scram Discharge Volume
- Reactor Power as indicated by the APRMs is 9 %

Which one (1) of the following actions, if taken alone, as prescribed by HC.OP-EO.ZZ-101A, ATWS-RPV Control, would be most effective in inserting control rods?

A. Manually initiate ARI.

B. Vent the Scram Air Header using HC.OP-EO.ZZ-0306.

C. De-energize scram solenoids using HC.OP-EO.ZZ-302.

D. Defeat RWM interlocks, and manually insert all control rods to or beyond position 02.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - With rod insertion prevented by a full SDV, scram valves are already open and ARI will be ineffective.

B: Incorrect – Not effective with SDV full due to initial scram

C: Incorrect - Not effective with SDV full due to initial scram

D: **Correct** - With the scram failure due to a full SDV, actions associated with control rod scrams will not be effective. Manual Control Rod insertion is independent of the SDV and while slow will fully insert all rods.

Technical Reference(s): EOP-101A – RC/Q21 (Table) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X - 34558

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	233000 K6.07	
Importance Rating	2.7	

K6.07 Knowledge of the effect that a loss or malfunction of the following will have on the FUEL POOL COOLING AND CLEAN-UP : Component cooling water systems

Question: RO #38

Given the following plant conditions:

- The 'A' SACS Loop is in service supplying TACS.
- The 'D' SACS pump is running.
- The 'B' SACS Pump is Cleared and Tagged.
- The 'A' Fuel Pool Cooling Heat Exchanger (FPCC side only) is isolated for a piping leak repair.

A lightning strike results in an 'A' channel LOCA Level 1 signal and the loss of the 10A404 4KV bus.

What is the status of the Fuel Pool Cooling System for these conditions?

A. Fuel Pool Cooling heat sink is unaffected by these conditions.

B. Fuel pool Cooling heat sink is being provided by the 'B' SACS Loop.

C. Fuel Pool Cooling heat sink has been lost and can be restored when the LOCA signal is reset.

D. The Fuel Pool Cooling heat exchanger cross-tie valves auto open to provide Loop 'A' SACS flow to the 'B' FPCC heat exchanger.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - The loss of the 10A404 Bus results in the loss of the 'D' SACS Pump and the 'B' SACS loop and the loss of cooling to the only in service FPCC HX.

B: Incorrect - The loss of the 10A404 Bus results in the loss of the 'B' SACS Loop.

C: Correct - The loss of the 10A404 Bus results in the loss of the 'D' SACS Pump and the 'B' SACS loop. With the 'A' FPCC HX OOS on the FPCC side, all SACS cooling to FPCC is lost. The Cross-tie valve HV-2317A and HV-7922A receive close signals from the LOCA signal and cannot be opened without clearing the signal.

D: Incorrect - The valves close on a LOCA signal and have no auto open signals.

Technical Reference(s): HC.OP-SO.EG-0001 step 3.3.9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X - 36333

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EK1.01	
	Importance Rating	3.5	

EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE : Reactor water level measurement

Question: RO #39

Given the following conditions:

- A LOCA concurrent with a Station Blackout has occurred
- HPCI and RCIC are unavailable
- Preparations are being made to inject with Fire Water (not yet available)
- Reactor Pressure is steady at 100 psig
- Reactor Bldg Temperature is steady at 75°F
- Drywell Temperature is increasing slowly at 285°F
- Fuel Zone indicators LR-R615 and LI-R610 are reading - 209 inches and steady

Based on the above current conditions, adequate core cooling is:
[Refer to attached figure of OPA-92-039]

A. assured, since actual RPV level is above TAF.

B. NOT assured, since actual level is below -200 inches.

C. assured, since actual RPV level is between TAF and -200 inches.

D. NOT assured, since actual RPV level is between TAF and -185 inches.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – Actual water level is below TAF (-161")

B: Incorrect – actual level is above -200" and therefore with given conditions adequate core cooling is assured.

C: **Correct -** No pumps are injecting. With level below TAF and no injection, Steam Cooling is required. Adequate core cooling is assured until level reaches -200". Actual compensated level is –197" and therefore Adequate Core Cooling is assured by steam cooling at this time.

- Uncompensated level is -209".
- At 100 psig, the -200" curve corresponds to an indicated level of -210".
- RB Temp Correction: 75° - 75° = 0°
- DW Temp Correction: 285° - 135° = 150°
- The -200" Curve shifts down 1.5" for a 150°F increase in Drywell Temp
- The -200" Curve at 100 psig is -210" indicated.
- Shifting downwards 1.5" places the -200" Curve at –211.5" indicated.
- Indicated level (-209") is 2.5" ABOVE the fully compensated –200" (-211.5") curve

D: Incorrect – adequate core cooling assured with steam cooling leg and level above -200"

Technical Reference(s): OP-AID 02-39 (Attach if not previously provided)
HC.OP-EO.ZZ-0101

Proposed References to be provided to applicants during examination: OP-AID 02-39 -

Learning Objective: (As available)

Question Source: Bank # X - 48866

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AK1.03	
	Importance Rating	3.6	

AK1.03 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Thermal limits

Question: RO #40

The plant is at rated power.

Then, the "A" Reactor Recirculation Pump tripped.

The plant responded as designed.

Based on the pump trip the MCPR limit _____

A. must be lowered to ≤1.08.

B. must be raised to ≥1.10.

C. must be lowered to ≤1.10

D. must be raised to ≥1.08

Proposed Answer: B

Explanation (Optional):

A: Incorrect – the MCPR limit is raised (see safety limit 2.1.2)

B: Correct – see MCPR safety limit 2.1.2

C: Incorrect – the MCPR limit is raised

D: Incorrect – Must be raised to ≥ 1.10 in single loop operation

Technical Reference(s): COLR, TS 3.4.1.1 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295004	AK1.01	
	Importance Rating	2.9		

AK1.01 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : †Automatic load shedding: Plant-Specific

Question: RO #41

The plant is operating at full power when a LOCA and a loss of offsite power (LOP) occur. Emergency diesel generators respond as designed.

Which of the following describes the status of the 1E and Non-1E 125 VDC Battery Chargers 30 seconds after the event?

A. 1E battery chargers are in service
Non-1E battery chargers are load shed and can be manually restored by overriding the load shed and re-energizing the MCC's.

B. 1E and Non-1E battery chargers are load shed but will be automatically restored at the same time by load sequencing.

C. 1E and Non-1E battery chargers are load shed
1E battery chargers will be automatically restored by load sequencing
Non-1E battery chargers will be restored 2 minutes after the sequencer starts.

D. 1E battery chargers are in service
Non-1E battery chargers are load shed and CANNOT be returned to service.

Proposed Answer: A

Explanation (Optional):

A: **Correct** - The 1E battery chargers supply breakers are not load shed. The Non-1E battery chargers can be restored manually.

IAW DC Electrical Lesson Plan section X.C.1.b. - Upon a LOCA, the MCCs that supply the battery chargers (excluding the guardhouse battery charger 10D514) are shed from the Class 1E 480 VAC Unit Substations that normally supply their power. Shedding of the MCCs places the 125 VDC (non- 1E) power requirements on the respective batteries.

The LOCA signal for the MCC feeder breakers can be overridden in the control room at 10C650

B: **Incorrect** - The Non 1E chargers are not automatically restored after a LOCA.

C: **Incorrect** – The 1E chargers are not load shed. Non-1E battery chargers are not automatically restored.

D: **Incorrect** - The 1E battery chargers are not load shed. Non-1E battery chargers are not automatically restored.

Technical Reference(s): HC.OP-SO.SM-0001, DC Electrical Lesson Plan section X.C.1.b (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: DCELECE015 (As available)

Question Source: Bank # 54243

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AK2.03	
	Importance Rating	2.9	

AK2.03 - Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Control room HVAC

Question: RO #42

Given the following conditions at T=0:

- The plant is operating at 100 % power.
- A pipe break causes a valid high drywell pressure of 8 psig.
- Control Area Ventilation System (CAVS) responds as designed.

At T=30, smoke is detected in the outside air supply to the Control Room and begins entering the Control Room.

- Both smoke detectors are in alarm.
- FIRE PROT PANEL 10C671 alarm is received.
- The operators don respiratory equipment.
- Drywell pressure is 5 psig and lowering

Which of the following describes the Control Area Ventilation System configuration and any additional required operator actions?

A. At T=0 - CAVS must be manually placed in the ISOLATE Mode. Operator action is required to start & place CREF in Outside Air Mode.
At T=30 – NO additional actions occur or are required.

B. At T=0 - CAVS remains in the normal outside air Mode. CREF auto starts and remains in Outside Air MODE.
At T=30 – CAVs is required to be placed in the ISOLATE Mode manually. CREF remains in the Outside Air Mode

C. At T=0 – ALL CAVS fans trip. CREF auto starts and in the Outside Air.Mode
AT T=30 – CREF auto transfers to the RECIRC Mode.

D. At T=0 – CAVS auto transfers to the ISOLATE Mode. CREF auto starts in the Outside Air Mode.
At T=30 – CREF must be manually placed in the RECIRC Mode.

Proposed Answer: D

Explanation (Optional):

A: Incorrect – At T=0 CAVS transfers to ISOLATE Mode automatically. CREF also auto starts in outside air mode. CREF must be placed in RECIRC manually at T=30 per abnormal procedure.

B: Incorrect – At T=0 CAVS goes to ISOLATE & CREF starts in Outside Air mode. At T=30 CREF must be placed in RECIRC Mode.

C: Incorrect – ALL CAVS fans do no trip at T=0. At T=30 CREF does not auto transfer

D: **Correct** - At T=0 The LOCA signal at 1.68 psig Drywell Pressure results in CAVS going to ISOLATE Mode automatically. CREF also auto starts in outside air mode. CREF must be placed in RECIRC at T=30 per abnormal procedure.

Technical Reference(s): HC.OP-AB.HVAC-0002 Retainment Override (Attach if not previously provided)
NOH01CAVENTC-05 pages 14 & 24

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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 AK2.06	
	Importance Rating	2.8	

AK2.06 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Offgas system

Question: RO #43

The plant is operating at 8% power performing a power ascension IAW HC.OP-IO.ZZ-0003.

The Unit 1 Recombiner Train is in service.

Then, the instrument air supply line to the following valves completely severs:

- 1HA-HV-5646, SJAE Discharge to Gaseous Radwaste Unit 1 Recombiner
- 1HA-HV-5649-1, SJAE Discharge to Gaseous Radwaste Common Recombiner

Which of the following is an immediate action required?

A. Place a Mechanical Vacuum pump in service.

B. Trip the Main Turbine.

C. Reduce Reactor Power to maintain Main Condenser Vacuum Low Overhead Alarm Clear.

D. Place the Common Recombiner in service

Proposed Answer: C

Explanation (Optional):

A: Incorrect – The mechanical vacuum pump is prohibited from operation with power >5%.

B: Incorrect – At 8% power the Main Turbine is not online.

C: **Correct** – With the SJAEs having no discharge flowpath due to valves 1HA-HV-5646 & 1HA-HV-5649-1 failing closed on loss of instrument air, vacuum will begin to degrade and the abnormal procedure immediate action applies.

D: Incorrect – That flowpath does not exist with valve 1HA-HV-5649-1 failing closed on loss of air.

Technical Reference(s): HC.OP-IO.ZZ-0003 Step 3.5.1 (Attach if not previously provided)
HC.OP-AB.BOP-0006 – I.A.
P&ID M-69-0 Sht.2

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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 AK2.07	
	Importance Rating	4.0	

AK2.07 - Knowledge of the interrelations between SCRAM and the following: Reactor pressure control

Question: RO #44

The plant is at rated power.

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

With NO operator action the plant will scram as reactor pressure __(1)__. The Components/Systems available for reactor pressure control following the scram include ___(2)___.
(see attached DEHC Logic reference)

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

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

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

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

Proposed Answer: A

Explanation (Optional):

A: **Correct** – IAW AB.RPV-0005, 2 pressure transmitters drifting upscale will cause the Turbine Control/Bypass valves to open resulting in an Uncontrolled Lowering of RPV pressure. With the Mode switch remaining in RUN, as reactor pressure lowers to 756 psig, the MSIVs will close and the bypass valves will be unavailable for pressure control. SRVs , RCIC and HPCI could be used for pressure control in that situation.

B: Incorrect – Bypass valves are unavailable due to MSIV closure

C: Incorrect – reactor pressure lowers

D: Incorrect – reactor pressure lowers

Technical Reference(s): AV7166 (Attach if not previously provided)
HC.OP-AB.RPV-0005 – Note 2.

Proposed References to be provided to applicants during examination: DEHC control logic simplified drawing AV7166

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EK3.04	
	Importance Rating	3.7	

EK3.04 - Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: SBLC injection

Question: RO #45

The plant was at rated power when an ATWS and a Group 1 isolation occurred.

Reactor power is 3%.

Which of the following describes the reason why SLC must be injected prior to Suppression Pool temperature reaching 140°F IAW EOP-101A, ATWS RPV Control

This will ensure ______

A. SLC injection prior to reaching the suppression pool temperature at which a reactor scram is required by Technical Specifications.

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

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

D. SLC injection prior to the Suppression Pool temperature exceeding the Boron Injection Initiation Limit.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – The TS requirement for a scram is 110 degrees NOT 140 degrees. (TS 3.6.2.1

B: Correct – 140 degrees is 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.

C: Incorrect – Hot Shutdown Boron weight, NOT Cold

D: Incorrect – The concern is Hot Shutdown Boron Weight and the Heat Capacity temperature limit

Technical Reference(s): TS 3.6.2.1 (Attach if not previously provided)
EOP-101A bases Step RC/Q-11

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X
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 1
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295037	EK3.01	
	Importance Rating	4.1		

EK3.01 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Recirculation pump trip/runback: Plant-Specific

Question: RO #46

The plant was at rated power when a spurious reactor scram signal occurred.

Not all control rods fully inserted and EOP-101A, ATWS–RPV Control was entered.

The Main Turbine remained in service. Reactor power is at 28%.

Which of the following describes the reason why the recirculation pumps are required to be runback to minimum speed before being tripped under these conditions?

A. To maintain the largest margin to the APLHGR power distribution limit.

B. To prevent an RPV high level trip and ensure a HPCI thru Core Spray injection flowpath.

C. To prevent power instabilities due to operating at higher power without adequate core flow following a Main Turbine trip.

D. To prevent a Main Turbine trip and additional heat loading of the Torus from SRVs if power remains above bypass valve capacity.

Proposed Answer: D

Explanation (Optional):

A: Incorrect – APLHGR a concern at >24% power

B: Incorrect – In an ATWS condition, HPCI injection through the Core Spray piping is not used per EOP-101A Table 1 direction.

C: Incorrect – the actions taken in EOPs will remove all forced circulation and lower RPV level to lower power which should reduce any instabilities.

D: **Correct** - The most rapid flow rate reduction and, consequently, the most rapid power reduction, is achieved by tripping the recirculation pumps. However, if the recirculation pump trip is initiated from a high power level, the resulting rapid changes in steam flow, RPV pressure, and RPV water level may cause a trip of the main turbine-generator and a trip of RPV injection systems.

If the main turbine-generator trips and reactor power exceeds the turbine bypass valve capacity, RPV pressure will increase until one or more SRVs open. Heatup of the suppression pool then begins and RPV level lowering may be required. If RPV injection systems trip, the resultant RPV water level transient may require emergency depressurization of the RPV and operation of less desirable RPV injection sources.
To effect a more controlled reduction in reactor power and thereby avoid main turbine-generator and RPV injection system trips and their associated complications, a recirculation flow runback is performed prior to tripping the recirculation pumps. If an automatic runback has occurred, the operator need only confirm the action.

Technical Reference(s): EOP-101A bases stpe RC/Q-8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 56604

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AA1.03	
	Importance Rating	3.8	

AA1.03 - Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Voltage Regulator Controls

Question: RO #47

The Plant is at 28% power, when the following conditions occur:

- The AC Regulator is controlling the main generator field
- A main generator lockout occurs.
- The exciter field circuit breaker remains closed.
- All other exciter equipment functions per design.

Assuming no operator action, which of the following describes the operational effect of this condition?

A. Full excitation voltage will be applied to the main generator by the DC regulator

B. Full excitation voltage will be applied to the main generator by the AC regulator

C. Minimum (effectively zero) excitation voltage will be applied to the main generator

D. The amount of excitation voltage to the main generator is solely determined by the manual setting for the DC regulator.

Proposed Answer: C

Explanation (Optional):

A: INCORRECT, Voltage regulation will shift from the AC (automatic) Regulator to the DC (manual) Regulator, IF the field circuit breaker trips.

B: INCORRECT, When the exciter field circuit breaker trip signal is generated (either manually or automatically), both voltage regulators are placed in a de-excitation mode, causing the regulator to reduce the exciter output to minimum.

C: **CORRECT,** When the exciter field breaker trip signal is generated (either manually or automatically), both voltage regulators are placed in a de- excitation mode in which de-excitation relay contacts cause the regulator to reduce the exciter output to minimum. This acts as a backup to the exciter field breaker trip and provides minimum excitation in the event the field breaker fails to open.

D: INCORRECT, Voltage regulation will shift FROM the AC (automatic) Regulator TO the DC (manual) Regulator, IF the field circuit breaker trips.

Technical Reference(s): NOH01ALTREX – Page 20 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ALTREXE011 (As available)

Question Source: Bank # 34008

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	295038	EA1.03
	Importance Rating	3.7	

EA1.03 - Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Process liquid radiation monitoring system

Question: RO #48

Given the following:

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

Which one of these condition(s) would cause this termination?
(Assume NO operator action)

(1) Liquid Radwaste Effluent High radiation setpoint is reached
(2) Cooling Tower Blowdown dilution flow low flow setpoint is reached
(3) Liquid Radwaste Effluent sample flow rate HI setpoint is reached
(4) Cooling Tower Blowdown RMS High radiation setpoint is reached
(5) Liquid Radwaste Effluent High discharge flow setpoint is reached

A. (1) and (3) ONLY

B. (2), (4) and (5) ONLY

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

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

Proposed Answer: D

Explanation (Optional):

A: Incorrect. - (3) is incorrect. (5) is also correct

B: Incorrect. - (4) is incorrect. (1) is also correct

C: Incorrect. - (3) is incorrect. (1) is also correct

D: **Correct** IAW HC.OP-AR.SP-0001 Alarm Point 9RX508 (page 22)
AUTOMATIC ACTION - Isolation of HV-5377A&B due to any one of the following:
- High radiation (HIGH LED on 0SP-RI-4861)
- High Disch Flow (setpoint determined by Liquid Effluent Permit)
- Low Dilution Flow (setpoint determined by Liquid Effluent Permit)
- Low Sample Flow (0HBFIS-4861)
- Monitor Failure

Technical Reference(s): HC.OP-AR.SP-0001 (Attach if not previously provided)
Alarm Point 9RX508, Att 5

Proposed References to be provided to applicants during examination: none

Learning Objective: NOH04RMSYSC-07 OBJ.4 (As available)

Question Source: Bank # 120363
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Hope Creek 2015

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

AA1.04 - Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING : Alternate heat removal methods

Question: RO #49

Plant Conditions are as follows:

Alternate Shutdown Cooling is being implemented by using the C to A RHR Loop Cross-Tie IAW HC.OP-AB.RPV-0009.

If the operator opens HV-F007C, C RHR PMP MIN FL MOV during this operation, how will the plant INITIALLY respond?

A. RHR Pump C will lose NPSH.

B. The RPV will drain to the Suppression Pool.

C. Flow through the A RHR Heat Exchanger will rise.

D. SACS outlet temperature from A RHR Heat Exchanger will rise.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - C RHR Pump would eventually lose NPSH. The stem stipulates the selection of the first consequence.

B: **Correct -** Opening HV-F007 C will establish a drain path from the B Recirculation Pump Loop to the Torus via C RHR Pump Suction and HV-F007

C: Incorrect - The flow which existed initially in the A RHR Heat Exchanger will lower due to a drain path being opened to the Torus.

D: Incorrect - The loss of RHR flow to the A RHR Heat Exchanger will lower the heat burden on SACS and hence the SACS outlet temperature will not rise.

Technical Reference(s): HC.OP-AB.RPV-0009 (Attach if not previously provided)
Caution Page 29

Proposed References to be provided to applicants during examination: none

Learning Objective: ABRPV9E004 (As available)

Question Source: Bank # 61858

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	295018	AA1.02
	Importance Rating	3.3	

AA1.02 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : System loads

Question: RO #50

The plant is operating at 95% power when a loss of offsite power occurs. All systems operate as designed with the exception of 'C' EDG, which fails to start.

Which one of the following describes the status of drywell chilled water loads?

A. All RACS valves will transfer to provide cooling to drywell chilled water loads.

B. Only 'B' drywell chilled water loop loads will be transferred to RACS cooling.

C. No RACS valves will transfer to provide cooling to drywell chilled water loads.

D. Only 'A' drywell chilled water loop loads will be transferred to RACS cooling.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – The “A” Loop RACS valves will not change position due to power loss

B: **Correct** - As a result of the unavailability of ‘C’ EDG, the ‘A’ drywell chilled water loop RACS – CW transfer valves lose power, and only the ‘B’ loop valves will reposition.

C: Incorrect – The “B” loop RACS valves will still have power

D: Incorrect - The “A” Loop RACS valves will not change position due to power loss

Technical Reference(s): HC.OP-SO.ED-0001 (Attach if not previously provided)
P&ID M-87-1 Sheet 2 of 6

Proposed References to be provided to applicants during examination: none

Learning Objective: AB135E004 (As available)

Question Source: Bank # 55176

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	295030 EA2.01	
	Importance Rating	4.1	

EA2.01 - Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Suppression pool level

Question: RO #51

The plant is at rated power.

Due to a large leak, Suppression Pool is at 55 inches and lowering.

As Suppression Pool level continues to lower the first component(s) to become uncovered will be the __________.

A. HPCI turbine exhaust

B. SRV T-quenchers

C. Vent header drain pipes

D. Downcomers

Proposed Answer: C

Explanation (Optional):

A: Incorrect - 26 inches is the indicated suppression pool level which corresponds to the elevation of the top of the HPCI exhaust (see EOP bases SP/L-9)

B: Incorrect – SRV t-quenchers are below 0 inches indicated level

C: **Correct** - There is a 1¼ IN, un-valved, drain pipe attached to the low point of each of the eight vent pipes located in the torus. These drain pipes open into the Suppression Chamber at an indicated level of 50"

D: Incorrect – 38.5 inches is the elevation of the downcomer openings (see EOP bases SP/L-7)

Technical Reference(s): EOP-102 Bases step SC/L-5 (Attach if not previously provided)
NOH01PRICONC-06 Page 45

Proposed References to be provided to applicants during examination: none

Learning Objective: 3 (As available)

Question Source: Bank # 53804

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 3

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA2.04	
	Importance Rating	3.4	

AA2.04 - Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Area radiation levels

Question: RO #52

Given the following conditions:

The plant is in a refueling outage with a fuel move in progress.

The 'A' Refuel Floor Exhaust Radiation Monitor has failed upscale.

-

NO actions have been taken to address this failure.

- At time 0000 a fuel bundle is dropped and radiation levels on the refuel floor start to slowly rise.
- At time 0005 the 'B' Reactor Building Exhaust Radiation Monitor reaches its Hi Trip Setpoint.
- At time 0010 the 'C' Refuel Floor Exhaust Radiation Monitor reaches its Hi Trip Setpoint.

Under these conditions, an automatic trip of the Reactor Building Ventilation Exhaust (RBVE) fans due to Hi Refuel Floor Exhaust Radiation levels:

A. occurred at time 0005.

B. occurred at time 0010.

C. will ONLY occur when at least 1 additional Reactor Building Exhaust radiation monitor senses high radiation.

D. will NOT occur until at least 1 additional Reactor Building Exhaust radiation monitor OR 1 additional Refuel Floor Exhaust Radiation Monitor senses high radiation.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – 2/3 logic not met for RB exhaust rad monitor hi-hi

B: **Correct** - At 0010 – the 2/3 logic was satisfied for the Refuel floor exhaust rad monitor high high trip of the RBVE fans (channels "A" failed upscale and "C" hi)

C: Incorrect – it occurred at 0010. The logic would be made up if an additional RB exhaust rad monitor went high. However, it would not be the only reason for an automatic trip of the Reactor Building Ventilation Exhaust (RBVE) fans.

D: Incorrect – logic is 2 /3 on either refuel Floor OR RB Exhaust Rad monitors. This occurred on the Refuel Floor monitors at 0010

Technical Reference(s): J-102-0 sht.6 (Attach if not previously provided)
NOH04RMSYSC-10 page 32

Proposed References to be provided to applicants during examination: none

Learning Objective: 3g,4d (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 6
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA2.02	
	Importance Rating	4.2	

AA2.02 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Reactor power, pressure, and level

Question: RO #53

The plant was operating at rated power when bus lockouts occur on buses 10A110, 10A120 and 10A104 due to voltage perturbations.

Which one of the following describes the effect on the plant?

A. The plant will not scram, reactor power will lower and RFPTs will be available to control RPV level. Reactor pressure will lower and remain at a lower value.

B. The plant will scram and RPV level will be controlled by HPCI and RCIC. RPV pressure will be controlled with bypass valves augmented by SRVs as required.

C. The plant will scram and RPV level will be controlled by HPCI and RCIC. RPV pressure must be controlled by SRVs ONLY.

D. The plant will not scram, reactor power will lower and RFPTs will be available to control RPV level. Reactor pressure will initially lower and then rise to its pre-transient level.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – The plant will scram on low level due to loss of condensate/feedwater

B: Correct – Those buses power all 3 secondary condensate pumps and 2 of 3 primary condensate pumps. With the loss of condensate, the plant will scram on low level. HPCI and RCIC will be available to control RPV level. The bypass valves will be available to control RPV pressure. HPCI and RCIC function without AC power.

C: Incorrect – SRVs would be required if an MSIV closure occurred. No MSIV closure will occur due to the bus lockouts

D: Incorrect – secondary condensate pumps have lost power and will not be available. The.reactor will scram.

Technical Reference(s): NOHO1MNCONDC-09 pages 22, 29 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: MNCONDE015 (As available)

Question Source: Bank # 120364

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 2.4.1	
	Importance Rating	4.6	

2.4.1 – Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps. (High drywell pressure)

Question: RO #54

The plant was operating at rated power.

Then, a Recirc Pump dual seal failure occurs but it cannot be fully isolated.

Currently:

- Drywell pressure is 1.73 psig and rising slowly
- Drywell temperature is 147 degrees F. and rising slowly
- Suppression Pool temperature is 89 degrees F. and rising slowly
- Suppression Chamber pressure is 1.63 psig and rising slowly
- Suppression Pool level is 77 inches and stable
- Reactor level is 2 inches and rising slowly

Which of the following describes procedures that must be entered and initial actions that are required?

A. EO.ZZ-0101-RPV Control and EO.ZZ-0102-Primary Containment Control ONLY.
Lock the Mode Switch in Shutdown.

B. EO.ZZ-0101-RPV Control and EO.ZZ-0102-Primary Containment Control ONLY.
Lock the Mode Switch in Shutdown and Place Drywell Spray in service.

C. EO.ZZ-0101-RPV Control, EO.ZZ-0102-Primary Containment Control and AB-000-Reactor Scram
Lock the Mode Switch in Shutdown.

D. EO.ZZ-0101-RPV Control, EO.ZZ-0102-Primary Containment Control and AB-000-Reactor Scram.
Lock the Mode Switch in Shutdown and Place Drywell Spray in service.

Proposed Answer: C

Explanation (Optional):

A: Incorrect AB-000 is also entered.

B: Incorrect - – Entry to all steps of both EOPs is required with any entry condition. AB-000 is also entered for any scram. Drywell spray is not placed in service until Suppression Chamber Pressure exceeds 9.5 psig IAW step DW/P-7 of EOP-0102.

C: Correct – with any entry condition met for an EOP all steps are performed concurrently as stated in step PCC-2 in EO.ZZ-0101 and RC-4 in EO.ZZ-0102. In this scenario, the drywell temperature, drywell pressure and reactor level entry conditions have been reached. Locking the mode switch in shutdown is an initial action step. AB-000 is also entered on a reactor scram.

D: Incorrect – Drywell spray is not placed in service until Suppression Chamber Pressure exceeds 9.5 psig IAW step DW/P-7 of EOP-0102.

Technical Reference(s): EO.ZZ-0101 (Attach if not previously provided)
EO.ZZ-0102
AB-000

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP102E003 (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 2.4.18	
	Importance Rating	3.3	

2.4.18 - – Emergency Procedures / Plan: Knowledge of specific bases for EOPs. (reactor low water level)

Question: RO #55

Actions for Steam Cooling are being performed in accordance with HC.OP-EO.ZZ-0101, RPV Control, and compensated RPV level has dropped to -200" on Fuel Zone indication. The procedure requires emergency depressurization.

Which of the following is the reason for emergency depressurizing?

A. Maintain peak cladding temperature below 1500 degrees F.

B. Maintain peak cladding temperature below 1800 degrees F.

C. Maintain total oxidation of the cladding less than 0.17 of the total cladding thickness.

D. Maintain the maximum H2 generation less than 0.01 times the hypothetical maximum.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - - this is the MSCRWL (-185")the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F

B: **Correct** - IAW EOP-101 Bases Discussion, step ALC-21

Steam cooling is effected by allowing RPV water level to decrease through boil-off until it drops to the Minimum Zero-Injection RPV Water Level (MZIRWL). During this period the fuel temperatures in the uncovered portion of the core increase, and heat is transferred from the fuel rods to the steam. The MZIRWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1800°F.
When RPV water level drops below the MZIRWL, steam cooling may no longer be sufficient to preclude the peak clad temperature from exceeding 1800°F. **Emergency RPV depressurization is then performed in accordance with EOP-202. Unless the RPV is already depressurized, it is expected that the resulting swell will be sufficient to quench the uncovered portion of the fuel and reduce PCT almost to the value that would exist if the core were submerged.** As the swell subsides and steam flow through the open SRVs decreases, however, PCT turns and again rises.
Opening the SRVs before RPV water level reaches the MZIRWL would reduce the time over which the core remains adequately cooled with no injection. Waiting much after RPV water level reaches the MZIRWL could result in significant core damage due to excessive fuel temperatures.

C: Incorrect - This is an ECCS criteria based on < 2200 degrees F Peak cladding temperature

D: Incorrect - This is an ECCS criteria based on < 2200 degrees F Peak cladding temperature

Technical Reference(s): EOP 101 bases step ALC-21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EO101LE008 (As available)

Question Source: Bank # 56126

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 2.2.38	
	Importance Rating	3.6	

2.2.38 - Equipment Control: Knowledge of conditions and limitations in the facility license. (reactor high pressure)

Question: RO #56

Which of the following describes the Reactor Steam Dome Pressure Safety Limit and the Operational Conditions when it applies?

A. 1325 psig in OP CON 1, 2 and 3 ONLY.

B. 1325 psig in OP CON 1, 2, 3 and 4 ONLY.

C. 1300 psig in OP CON 1, 2 and 3 ONLY.

D. 1300 psig in ANY Operational Condition.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – Also applies in OP CON 4.

B: **Correct –**

Safety Limit 2.1.3 – Steam dome pressure shall not exceed 1325 psig in OP CON 1,2,3 and 4.

C: Incorrect – the Safety Limit is 1325 psig is not exceeded

D: Incorrect – 1325 psig in OP CON 1,2,3,4.

Technical Reference(s): Safety Limit 2.1.1 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AA2.13	
	Importance Rating	3.2	

AA2.13 - Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Need for emergency plant shutdown

Question: RO #57

Given:

The plant is operating at rated power.

Thick black smoke enters the Main Control Room requiring that the Control Room be abandoned.

Plant equipment has NOT been affected by the thick smoke condition.

Main Control Room actions, IAW HC.OP-AB.HVAC-002 "Control Room Environment", which have been taken include:

Locking the Mode Switch In Shutdown, tripping the Main Turbine and Recirc Pumps, closing the MSIVs.

NO other actions have been taken.

Which ONE of the following describes the plant parameter(s), if any, that will exceed Emergency Operating Procedure entry conditions?

A. None

B. Reactor Level ONLY

C. Reactor Pressure ONLY

D. Reactor Level and Reactor Pressure

Proposed Answer: D

Explanation (Optional):

A: Incorrect – Reactor pressure and level – see description of event in "C" explanation

B: Incorrect - Reactor pressure and level – see description of event in "C" explanation

C: Incorrect - Reactor pressure and level – see description of event in "C" explanation

D: Correct – IAW AB-HVAC-0002 Condition C. – MSIVs closed will result in a loss of the steam driven feed water pumps, and pressure will rise until SRVs lift since bypass valve capability is lost with the MSIVs are closed. Level will drop below 12.5 inches, and RPV Pressure will rise above 1037 psig, and these are both entry conditions to EOP-101 for RPV control.

Technical Reference(s): AB-HVAC-0002 Condition C. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AA2.08	
	Importance Rating	3.2	

AA2.08 - Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Electrical distribution status

Question: RO #58

Given:

- The plant is at 80% power
- "A" EHC pump is out of service for maintenance
- The RO observes EHC pressure at 1400 psig and lowering at 50 psig every 10 seconds
- The RO Locks the Mode Switch in Shutdown
- EHC pressure then stabilizes at 1150 psig

Based on the above:

(1) Will Locking the Mode Switch in Shutdown generate a DIRECT Turbine Trip signal?
AND
(2) Will the 500 KV BS2-6 and BS6-5 breakers automatically trip open if no operator action is taken?

A. (1) YES
(2) YES

B. (1) YES
(2) NO

C. (1) NO
(2) YES

D. (1) NO
(2) NO

Proposed Answer: C

Explanation (Optional):

A: Incorrect – EHC pressure did not lower to the turbine trip setpoint. RPS has no direct tie to the Main turbine generator or output breaker trip logic.

B: Incorrect - EHC pressure did not lower to the turbine trip setpoint. RPS has no direct tie to the Main turbine generator or output breaker trip logic. The BS2-6 and BS6-5 bkrs will trip open on reverse power (antimotoring) logic after a time delay

C: **Correct** – Taking the mode switch to shutdown is not a direct turbine trip signal. EHC pressure remained above the turbine trip setpoint of 1100 psig. The Main Turbine Generator anti motoring circuit will cause the BS2-6 and BS 5-6 to automatically trip open after a 30 second time delay (see E-3030-0 Sht.2).

D: Incorrect - The BS2-6 and BS6-5 bkrs will trip open on reverse power (antimotoring) logic after a time delay

Technical Reference(s): E-3030-2 sht 2 (Attach if not previously provided)
HC.OP-SO.CH-0001 step 3.3.3
NOH01MNGEN0C-07 Page 23

Proposed References to be provided to applicants during examination: none

Learning Objective: MNPWR E019 (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295002 AK1.03	
	Importance Rating	3.6	

AK1.03 - Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM : Loss of heat sink

K/A Match Justification – The "loss of heat sink" concept is addressed via condenser waterbox inlet valve closing and vacuum loss. The operational implication is that reactor power must be reduced.

Question: RO #59

Given:

- Reactor Power is at 95%
- The Circ Water Inlet Valve to the "A" North Waterbox has drifted closed
- HC.OP-AB.BOP-006, Main Condenser Vacuum has been entered
- Condenser Vacuum has degraded to 6.1" HgA and has stabilized and the Main Condenser Vacuum LO annunciator is LIT

For these conditions what action(s) is/are immediately required?

A. Reduce Recirc pump speed to minimum then Lock the Mode Switch in Shutdown and Close all MSIVs.

B. Reduce reactor power to maintain the Main Condenser Vacuum LO annunciator clear

C. Trip the main turbine and then Lock the Mode Switch in Shutdown.

D. Fully isolate the condenser waterbox with the suspected fouling and then reduce reactor power.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – This would be correct for a total loss of circ water IAW the Retainment Override in AB-BOP-0006

B: Correct – IAW HC.OP-AB.BOP-0006

C: Incorrect – tripping the turbine would be correct at lower power

D: Incorrect – this is subsequent and NOT an immediate action.

Technical Reference(s): HC.OP-AB.BOP-0006 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ABBOP6E003 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295007 AK2.01	
	Importance Rating	3.5	

AK2.01 - Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Reactor/turbine pressure regulating system

Question: RO #60

The plant is at 70% power.

Then, a high level trip occurs on the 6A Feedwater Heater.

Assuming no operator action, which of the following describes the initial effect on Reactor pressure and the Main Turbine pressure control system 10 minutes after the event.

Reactor pressure will ___(1)____. The DEHC system will send a signal to ___(2)____ until reactor pressure is stabilized at the pressure setpoint.

A. (1) rise.
(2) OPEN the Turbine Control Valves and Turbine Bypass Valves.

B. (1) rise.
(2) OPEN the Turbine Control Valves.

C. (1) lower.
(2) CLOSE the Turbine Control Valves and Turbine Combined Intercept Valves.

D. (1) lower.
(2) CLOSE the Turbine Control Valves.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – although reactor pressure will rise, it will not rise to the point where the Bypass valves will open.

B: Correct – When a loss of Feedwater heating occurs colder FW will be sent to the reactor resulting in a decrease of moderator temperature and an increase in reactor power due to a positive reactivity addition. As power increases, reactor pressure will increase causing the DEHC system to open the control valves further to reduce reactor pressure. The bypass valves do not open.
(this scenario was run in simulator).

C: Incorrect – See explanation "B" above. The CIVs remain open

D: Incorrect – See explanation "B" above.

Technical Reference(s): NOH01EHCLOG-09 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EHCLOGE006 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022 AK3.02	
	Importance Rating	2.9	

AK3.02 - Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: CRDM high temperature. K/A match justification: With only one justifiable distractor for why CRDM high temperature would occur, the question was written so the reader would have to determine various aspects of CRDM operation based on the stem conditions and then the reasons & subsequent effects.

Question: RO #61

The plant is operating at rated power.

The "A" CRD pump trips and the "B" cannot be started.

Which of the following describes how the CRD mechanisms respond.

A. The area under the collet piston will depressurize which allows the collett fingers to disengage the index tube and control rods may begin to drift out.

B. The CRD mechanism temperature will increase above the alarm setpoint. The control rods will remain in their current position.

C. The area under the CRDM ball check valve will depressurize to below reactor pressure. This delta-P lifts the ball check valve which in turn causes the control rods to begin to insert if reactor pressure is above 500 psig.

D. The individual HCU accumulators will provide cooling water to their associated CRD mechanisms as they depressurize. When the accumulator pressure drops below reactor pressure, individual control rods will fail to insert on a valid scram signal.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – the area under the collet piston area is at equilibrium pressure with the rest of the drive so the collet piston will not move and the rods will not drift.

B: Correct – with a loss of all CRD pumps there is a loss of cooling water flow which will cause CRDM temperatures to rise but the rods will not move.

C: Incorrect – the area above the ball check valve is at reactor pressure. It will not depressurize unless the reactor depressurizes. The rods will not insert.

D: Incorrect – the accumulators do not provide cooling water to the CRDMs. Rods will still scram.

Technical Reference(s): AR.ZZ-0011 C6-C3 (Attach if not previously provided)
NOH01CRMECHC-07 pages 19,20, 22,23,24

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

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

Question History: Brunswick

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

10 CFR Part 55 Content: 55.41 6
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	500000 EA1.03	
	Importance Rating	3.4	

EA1.03 - Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: Containment atmosphere control system

Question: RO #62

The plant was operating at rated when an event occurred.

RPV level has lowered to below top of active fuel and continues to lower.

The H2O2 analyzers must be placed in service.

Which of these describe;

(1) ALL available sample locations monitored by the Hydrogen/Oxygen Analyzers

AND

(2) whether any containment isolation signal to the associated containment isolation valves can be overridden to ensure availability?

A. (1) The upper drywell and the torus only
(2) They ALL CAN be overridden

B. (1) The upper drywell and the torus only
(2) They ALL CANNOT be overridden

C. (1) The upper drywell, lower drywell and the torus
(2) They ALL CANNOT be overridden

D. (1) The upper drywell, lower drywell and the torus
(2) They ALL CAN be overridden

Proposed Answer: D

Explanation (Optional):

A: Incorrect – the lower drywell is also sampled

B: Incorrect – the lower drywell is also sampled and all containment isolation signals to the isolation can be manually overridden

C: Incorrect - all containment isolation signals to the isolation can be manually overridden

D: **Correct** per LP NOH01H2O2AN - Each package (H2O2 analyzer) takes samples from three different locations; High - Drywell head region, Low - Drywell cylindrical region, Suppression Chamber Air Space. CIVs can be individually opened after the associated isolation override P.B. is depressed at (10C650E).

Technical Reference(s): NOH01H2O2AN-05, page 9 (Attach if not previously provided)
HC.OP-SO.GS-0002, page 9

Proposed References to be provided to applicants during examination: none

Learning Objective: H2O2ANE002, 003 (As available)

Question Source: Bank # 120390

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295020 AA2.06	
	Importance Rating	3.4	

AA2.06 - Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION : Cause of isolation
K/A match justification: The K/A is matched because the BB-SV4310 is a containment isolation valve.

Question: RO #63

The plant is at rated power.

Then, the RO initially observes:

- REAC RECIRC LOOP B SAMPLE LINE ISLN INBD VLV BB-SV4310 is closed
- MECH VAC PMP A and BP105 "STOP" lights are illuminated.

Which of the following could cause BOTH of the above?

A. A RPV -38 level trip input from PCIS Channel "A".

B. A RPV -129 level trip input from NSSSS Channel "B" and "D"

C. A Drywell Pressure Hi input from PCIS Channel "A" and "B".

D. A MSL HIGH-HIGH Radiation trip input from NSSSS Channel "A"

Proposed Answer: D

Explanation (Optional):

A: Incorrect – the BB-SV4310 and mech vacuum pumps do not receive PCIS trip inputs

B: Incorrect – the items indicated do not get inputs from "D" NSSSS

C: Incorrect – drywell pressure hi does not input to all items listed

D: Correct – Channel "A" and "B"of NSSSS causes both of the items listed to occur. See SM-0001 Table SM-002

Technical Reference(s): HC.OP-SO.SM-0001 Table-2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NSSSS0E008 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295035 2.4.11	
	Importance Rating	4.0	

2.4.11 - Emergency Procedures / Plan: Knowledge of abnormal condition procedures. (Secondary Containment High Differential Pressure)

Question: RO #64

Given:

RBVS Supply Fans BV300 and CV300 are in service

RBVS Exhaust Fans AV301 and CV301 are in service.

Then, RBVS Exhaust Fan AV301 trips on a motor fault.

Both RBVS Supply Fans BV300 and CV300 have tripped.

Reactor Building pressure is approximately 0.10" vacuum water gauge.

What actions are required IAW HC.OP-AB.CONT-0003, Reactor Building?

A. Verify Exhaust Fan BV301 auto starts and
Verify Supply Fans BV300 and CV300 auto start.
Verify Reactor Building pressure is >0.30" vacuum water gauge.

B. Manually start Exhaust Fan BV301
Manually start Supply Fan AV300.,
Verify that Reactor Building pressure is >0.25" vacuum water gauge.

C. Verify ALL FRVS Recirc fans auto start.
Verify BOTH FRVS Vent Fans auto start.
Verify Reactor Building pressure is >0.25" vacuum water gauge.

D. Manually start Exhaust Fan BV301.
Verify Supply Fan AV300 starts.
Verify Reactor Building pressure is > 0.30" vacuum water gauge.

Proposed Answer: A

Explanation (Optional):

A: Correct - BV300 and CV300 will automatically re-start after a time delay and after the start of the second exhaust fan. The abnormal procedure requires verification of RX Bldg dp at >0.30 inches water gauge.

B: Incorrect – The fans auto start. RX Bldg pressure >0.25" vacuum water gauge is associated with TS 3.6.5.1 surveillance requirement for secondary containment.

C: Incorrect – There are no auto starts of FRVS associated with this event. RX Bldg pressure >0.25" vacuum water gauge is associated with TS 3.6.5.1 surveillance requirement for secondary containment.

D: Incorrect – NO manual fan starts are required. The system logic will restart the supply fans aonce a second exhaust fan is running.

Technical Reference(s): HC.OP-AB.CONT-0003 – condition "A" (Attach if not previously provided)
H-83-0 sht 2
H-84-0 sht 2

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032 EK3.02	
	Importance Rating	3.6	

EK3.02 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Reactor SCRAM.

Question: RO #65

Given the following conditions:

An unisolable steam line leak has occurred in the RCIC room

RCIC Equipment Room Area Temperature is 207°F and rising.

Which of the following is the reason for initiating a Reactor Scram with the above conditions?

A. Emergency Depressurization is anticipated.

B. It will begin to reduce the energy that the RPV will discharge to the RCIC room.

C. It will reduce the driving head and flow through the break in the RCIC room to prevent the blowout panel from opening.

D. It will stop the radioactive release and prevent the failure of Secondary Containment due to high temperatures.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - the scram is not performed for anticipating an ED. Levels may not reach the ED requirement.

B: Correct - From EOP-103 bases:
If temperatures or floor levels in any one of the ROOMS listed in Table 1 or 2 of Reactor Building Control approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EOP actions can no longer be assured. EOP-101 must be entered to make certain the reactor is scrammed. Scramming the reactor reduces to decay heat levels the energy that the RPV may be discharging to the reactor building.

C: Incorrect - The blowout panel is not a concern per the bases and its purpose concerns building differential pressure relief.

D: Incorrect - The failure of Secondary Containment is not a concern at this point in the event with the conditions stated. Scramming in and of itself would not prevent a rad release.

Technical Reference(s): EOP 103 Bases Step RB-17 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP103E006 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.1.3	
Importance Rating	3.7	

Knowledge of shift or short-term relief turnover practices.

Question: RO #66

Which one of the following describes Reactor Operator pre-and post-shift relief actions that should be implemented by the oncoming operator IAW OP-AA-112-101-Shift Relief and Turnover?

A. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding four (4) days logs, whichever is less.
POST relief, tour the main control room back panels.
POST relief, confer with the Control Room Supervisor to determine the scope of planned shift activities and their responsibilities for that shift.

B. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding four (4) days logs, whichever is less.
PRIOR to relief, tour the main control room back panels.
POST relief, confer with the Control Room Supervisor to determine the scope of planned shift activities and their responsibilities for that shift.

C. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding seven (7) days logs, whichever is less.
PRIOR to relief, tour the main control room back panels.
POST relief, review the Daily Orders.

D. PRIOR to relief, read the Control Room logs through the last previous date on shift, or the preceding seven (7) days logs, whichever is less.
PRIOR to relief, review the Daily Orders.
POST relief, tour the main control room back panels.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - tour the main control room back panels is performed PRIOR TO relief.

B: Correct – IAW OP-AA-112-101 – the only post relief item is conferring with CRS on work activities for the shift,

C: Incorrect - review the Daily Orders is done PRIOR TO relief. The Control Room logs through the last previous date on shift, or the preceding four (4) days logs, whichever is less should be read.

D: Incorrect - tour the main control room back panels is performed PRIOR TO relief. The Control Room logs through the last previous date on shift, or the preceding four (4) days logs, whichever is less should be read.

Technical Reference(s): IAW OP-AA-112-101 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: ADMPRO102E004 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.41	
	Importance Rating	2.8	

Knowledge of the refueling process.

Question: RO #67

A Refueling outage is in progress with the reactor cavity flooded and the fuel pool gates removed.

Which one of the following is considered a CORE ALTERATION?

A. Undervessel replacement of Special Moveable Detectors

B. Movement of Source Range Monitors and Intermediate Range Monitors..

C. Control Rod movement via RMCS with fuel assemblies within the associated core cell.

D. Removing LPRM detectors for replacement and movement of Transversing Incore Probes.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – Per TS, this is not considered a Core Alteration

B: Incorrect – Per TS, this is not considered a Core Alteration

C: Correct – TS definition 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 – Per TS, this is not considered a Core Alteration

Technical Reference(s): TS definitions 1.7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: NOH04TECSPCC-09 Obj.3. (As available)

Question Source: Bank # 57156 (changed 2 distractors)
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.13	
	Importance Rating	4.1	

Knowledge of tagging and clearance procedures.

Question: RO #68

IAW OP-AA-109-115, Safety Tagging Program procedures, which of the following statements is correct regarding Worker's Blocking Tags (WBTs)?

In regard to WBTs,____________________.

A. they do not require the name of the worker to be placed on the associate tag or sticker when used on a control room bezel.

B. more than one may be installed on the same blocking point.

C. only the worker designated on the tag may perform work on the component to which the tag is affixed

D. they may be used to isolate an energy source no greater than 4.2 Kv.

Proposed Answer: A

Explanation (Optional):

A: **Correct** – IAW OP-AA-109-115 Step 4 page 74

B: Incorrect - WBTs shall not be placed on any blocking point that is tagged with any other Safety Tag except a White Caution Tag.

C: Incorrect – no work may be performed on the component to which a WBT is affixed

D: Incorrect – they may NOT be used to isolate at >600 volts.

Technical Reference(s): OP-AA-109-115 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.39	
	Importance Rating	3.9	

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

Question: RO #69

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 cooldown rate is ___(1)___ administrative limits and ___(2)___ Technical Specification limits.

A. (1) outside (2) outside

B. (1) within (2) within

C. (1) within (2) outside

D. (1) outside (2) within

Proposed Answer: A

Explanation (Optional):

A: Correct. Between 0045 and 0145, cooldown reached 104 degrees within a one hour period which exceeds the one hour TS limit of 100 degrees/hr and the admin limit of 90 degrees per hr.

B: Incorrect – both limits are exceeded.

C: Incorrect – the admin limit is exceeded.

D: Incorrect – the TS limit is exceeded.

Technical Reference(s): HC.OP-IO.ZZ-0004 , TS 3.4.6.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 80653

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	3	
K/A #	2.3.4	
Importance Rating	3.2	

Knowledge of radiation exposure limits under normal or emergency conditions.

Question: RO #70

A Hope Creek operator has received the following dose:

January 3, 2015 thru January 8, 2015 - 200 mrem - while working in a foreign nuclear plant as part of a Technical Exchange Program.

January 14, 2015 - 50 mrem - while working at Hope Creek.

January 20, 2015 - 50 mrem - while working at Hope Creek.

Which of the following describes the MAXIMUM additional Total Effective Dose Equivalent (TEDE) that this individual could receive at Hope Creek for a Planned Special Exposure (PSE) occurring today, without exceeding site or federal exposure limits?

A. 1700 mrem

B. 1900 mrem

C. 4700 mrem

D. 4900 mrem

Proposed Answer: C

Explanation (Optional):

A: Incorrect - This answer would result from using the maximum annual exposure limit for TEDE at Hope creek which is 2000 mrem – all previous annual dose (300 mrem).

B: Incorrect – This answer would result from using the maximum annual exposure limit for TEDE at Hope creek which is 2000 mrem – annual dose received only at Hope Creek. The individual may believe the dose received at the foreign site does not count toward total.

C: **Correct -** This answer would result from using the Federal maximum annual exposure limit for TEDE which is 5000 mrem – the total annual dose received this year (300 mrem). The PSE exposure limit cannot exceed the annual federal limit of 5000 mrem.

D: Incorrect - This answer would result from using the Federal maximum annual exposure limit for TEDE which is 5000 mrem – the total annual dose received this year at Hope Creek. All dose received this year no matter where received is included in the calculation for PSE.

Technical Reference(s): RP-AA-203 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 12

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.3.15	
	Importance Rating	2.9	

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question: RO #71

Concerning the Main Steam Line Radiation Monitoring System (MSLRM),

The MSLRM System ____________.

A. monitors the gross gamma radiation from the main steam lines at a location just upstream of the inboard MSIV's.

B. will activate a "Main Steam Line Radiation Downscale" annunciator to alert the operator of a potential equipment malfunction if at power.

C. will initiate a Group I isolation and a direct SCRAM when the Hi-Hi radiation trip level is reached.

D. will trip/isolate the mechanical vacuum pumps AND will initiate a Group I isolation when the Hi radiation trip level is reached.

Proposed Answer: B

Explanation (Optional):

A: **INCORRECT**, the MSLRM are physically located downstream of the outboard MSIVs as shown on M-41-1 sht 1.

B: **CORRECT,** any downscale reading that does not cause an INOP condition will cause OHA C6-B3, Main Steam Line Downscale to illuminate.

C: **INCORRECT,** 3X full power background radiation (Hi-Hi) from the MSL Rad monitors will cause an isolation of the recirc sample valves 4310 and 4311 and the direct scram input was deleted per DCP-4EC-3038.

D: **INCORRECT,** MVP pump(s) trip on a 3X full power background radiation (Hi-Hi) from the MSL Rad monitors and when they do trip the suction valves do not close, no auto isolate on hi rad condition.

Technical Reference(s): NOH04NSSSS0C, M-41-1 sht 1 (Attach if not previously provided)
HA C6-B3

Proposed References to be provided to applicants during examination: none

Learning Objective: RMSYS0E004 (As available)

Question Source: Bank # 53783

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 11

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	4	
K/A #	2.4.25	
Importance Rating	3.3	

Knowledge of fire protection procedures.

Question: RO #72

A fire has occurred on the refuel floor.

The Fire Department has requested assistance.

Which of the following describes actions required to pressurize a hose reel on the refuel floor?

A. The control room must manually start the fire pump.

B. Withdrawing the hose from the hose reel will automatically pressurize the hose.

C. Stand Pipe and Water Hose Station Isolation valves must be manually opened.

D. Valves must be opened from the control room to pressurize the reactor building fire header.

Proposed Answer: C

Explanation (Optional):

A: Incorrect, The pumps operate on pressure switches and as demand on the system increases, the header pressure will drop causing an automatic start to the electric or diesel driven fire pump.

B: Incorrect, The header is not maintained pressurized due to pipe whip concerns post earthquake

C: **Correct,** from OP-AR.QK-002, Att 2, "OPERATOR ACTION, 1. IF manual firefighting is required in the Reactor Building AND the Fire Department has requested assistance, THEN OPEN the following Stand Pipe AND Water Hose Station Isolation Valves"

D: Incorrect, The valves are manually opened locally.

Technical Reference(s): OP-AR.QK-002, Att 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: FIRPROE014 (As available)

Question Source: Bank # 77479

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.20	
	Importance Rating	3.8	

Knowledge of the operational implications of EOP warnings, cautions, and notes.

Question: RO #73

Following a LOCA plant conditions are:

- Reactor pressure is 52 psig
- Drywell Temperature is 321°F

Determine the effect on reactor water level instruments and the basis for this indication.

Indicated water levels are ______________

A. higher than actual RPV level due to boiling in the reference leg.

B. lower than actual RPV level due to boiling in the variable leg.

C. lower than actual RPV level due lower density in the reference leg.

D. higher than actual RPV level due to lower density in the variable leg.

Proposed Answer: A

Explanation (Optional):

A: Correct – EOP Caution 1 explanation in EOP Limits Conversion document –
Boiling in the reference leg (which causes boil-off) reduces the height of water in the reference leg. This lowers the pressure on the reference leg side of the ΔP cell and makes the variable leg which is the indicated level appear higher.

B: Incorrect - Boiling in the variable leg raises the pressure on the variable leg side of the ΔP cell increasing the indicated level.

C: Incorrect - Lowered density on the variable side makes the reference side appear higher resulting in a lower indicated level.

D: Incorrect - Lowered density on the variable leg makes the actual level appear lower.

Technical Reference(s): HC.OP-EO.ZZ-LIMITS-CONV. Page 46 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 120339

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.25	
	Importance Rating	3.2	

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question: RO #74

Which one of the following correctly describes the Technical Specification bases for the Suppression Pool low water level limit in Operational Condition 1, 2, and 3?

A. This limit ensures adequate ECCS NPSH and vortex protection post Emergency Depressurization.

B. This limit prevents exceeding the Suppression Pool design internal pressure limit during a DBA LOCA.

C. This limit prevents exceeding the Suppression Pool design temperature limit during a DBA LOCA.

D. This limit ensures adequate SRV T-Quencher submergence during Emergency Depressurization.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - Bases for SPL low limit in OP Con 4 and 5*. See bases for 3.5.3.

B: **Correct** - Bases for SP lower water level limit IAW HCGS TS 3.6.2 and 3/4.5.3.
3/4.6.2. Depressurization Systems-Bases
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.
suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1020 psig. ***Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum internal design pressure***

C: Incorrect - Although analyzed to determine that SPT does not exceed the design limit during a DBA LOCA, the bases specifically say this and the other limits on the SP are based on not exceeding the design pressure limit of the containment.

D: Incorrect - T-quenchers are located at -51.5 inches below 0 inches indicated. This is not the TS low level limit

Technical Reference(s): TS 3.6.2 and 3/4.5.3. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 80655

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2003

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.30	
	Importance Rating	4.4	

Ability to locate and operate components, including local controls.

Question: RO #75

Given the following conditions at T=0 minutes:

- Control Room has been abandoned
- Control has been transferred to the Remote Shutdown Panel (RSP)
- BC-HV-F008 & BC-HV-F009 SDC Suction Isolation valves are OPEN
- RPV level is +10 inches and stable
- RPV pressure is 78 psig and rising at 2 psig/minute

Which one of the following describes:

(1) At T=0 minutes, the ability to operate the BC-HV-F009 SDC Suction Isolation valve from the RSP.
AND
(2) At T=3 minutes, the position of the BC-HV-F009 SDC Suction Isolation valve.

A. The valve can be CLOSED but NOT OPENED at the RSP.
CLOSED

B. The valve can be OPENED & CLOSED at the RSP.
OPEN

C. The valve can be CLOSED but NOT OPENED at the RSP.
OPEN

D. The valve can be OPENED & CLOSED at the RSP.
CLOSED

Proposed Answer: B

Explanation (Optional):

A: Incorrect - The valve will NOT isolate if reactor pressure exceeds 82 psig.

B: Correct. When control is transferred to the RSP, both the Low RPV Water Level AND high RPV Pressure isolations for the BC-HV-F009 are defeated. There remains a pressure switch permissive in series with the opening contactor that requires reactor pressure to be below 82 psig to open the valve (this is NOT a function of NSSSS). This is identified in a Note and a Caution in HC.OP-IO.ZZ-0008 (5.10.6).

Note and a Caution in HC.OP-IO.ZZ-0008 - CAUTION
WHEN the RSP Transfer Switch is placed in EMER, RHR S/D Cooling interlocks for overpressure AND low Reactor level are inoperable. RX pressure of 80 psig should NOT be exceeded WITH Suction Valves F008 & F009 open..

C: Incorrect. The valve CAN be opened, since the RPV Low Water Level isolation is defeated and there is NO Low Water Level opening permissive.

D: Incorrect. The valve can be opened. The valve will NOT isolate if reactor pressure exceeds 82 psig.

Technical Reference(s): HC.OP-IO.ZZ-0008 step 5.10.6 caution (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: IOP008E006 (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

Level	RO		SRO
Tier #			1
Group #			1
K/A #	295037	EA2.03	
Importance Rating			4.4

EA 2.03 - Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : SBLC tank level

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #76

Given:

An ATWS occurred with the plant at rated power.

- RPV level and pressure are stable and being maintained in the required bands.
- At 14:20 hrs, "A" and "B" SLC pumps were manually started.
- "B" SLC Pump tripped immediately after start and CANNOT be restarted.

Then,

- At 14:22 hrs, the SLC TANK TROUBLE is received due to low tank level.
- Attempts to insert control rods have been unsuccessful.
- Reactor power is 10%.

Assuming the "A" SLC pump delivers the Tech Spec minimum flow rate for the next 90 minutes, which one of these actions would be required?

A. Verify the "A" SLC pump is tripped then exit EOP-101A.

B. Verify the "A" SLC pump is tripped and continue attempting rod insertion.

C. Continue in EOP-101A, monitor SLC pump operation and begin a reactor cooldown.

D. Continue in EOP-101A, monitor SLC pump operation and raise RPV level to +12.5" to +54".

Proposed Answer: C

Explanation (Optional):

A: Incorrect - 'A' pump will still be running. EOP 101A is not exited until the reactor is shutdown under all conditions without boron. (retainment Override LP-1

B: Incorrect. 'A' pump will still be running.

C: **Correct** - After 75 minutes operation with only 1 SLC pump running, level is above the 0 gallon Low Level Pump trip setpoint. 4640 Gallons at the low level alarm point with 90 minutes runtime at 41.2 gpm [4640 - (41.2 x 90)] = 932 gallons remaining. Step RC/Q-19 directs continuation at step RC/P-20 for depressurization and cooldown when CSBW is injected (1100 gallons in tank).

D: Incorrect - RPV Level cannot be raised until the reactor is shutdown under all conditions without boron and EOP -0101 is entered..

Technical Reference(s): HC.OP-EO.ZZ-0101A, 0102 (Attach if not previously provided)
TS 3.1.5

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 80580

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2010

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1, 5

Comments:

Facility: Hope Creek 2015			
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295003 2.2.12	
	Importance Rating		4.1

2.2.12 – Equipment Control: Knowledge of surveillance procedures (Partial or Complete Loss of A.C. Power)
K/A Match Justification: - the question is demonstrating the ability to determine when the surveillance procedure for loss of offsite sources is required to be performed IAW TS. This surveillance is required whenever a loss of AC power occurs

SRO Only Justification: This question is SRO only as it requires assessing facility operating limitations in the Tech Specs and their bases, cannot be answered solely by knowing <= 1hr TS actions or above the line information, cannot be answered by knowing TS safety limits, and involves application of TS required actions and surveillance requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #77

The plant is operating at 100% reactor power with all systems in a normal lineup on 8/23.

Due to a heat wave the Delaware River temperature has risen as follows:
1100 – 87.9 degrees F.
1115 – 88.1 degrees F.
1130 – 88.2 degrees F.
1200 – 88.2 degrees F.
1230 – 88.3 degrees F.

At 1235 on 08/23 the 'A' Emergency Diesel Generator is declared inoperable and tagged out for emergent oil leak repairs.

At 1300 on 8/23, the surveillance that demonstrates the OPERABILITY of the remaining AC sources was satisfactorily performed.

At 1315 on 8/23 the 500 Kv Bus 10X lockout relay actuates and the 10X bus is locked out.
At 1315 the 10X Bus is declared Inoperable.

All other plant equipment is OPERABLE.

(1) What is the NEXT time required by Technical Specifications to perform the surveillance that demonstrates the OPERABILITY of the remaining AC sources
AND
(2) When is the earliest the plant is required to be in Mode 3?

A. (1) By 1415 on 8/23,
(2) By 0035 on 8/23

B. (1) By 2100 on 8/23,
(2) By 0035 on 8/23

C. (1) By 1415 on 8/23,
(2) By 0215 on 8/27

D. (1) By 2100 on 8/23,
(2) By 0215 on 8/27

Proposed Answer: A

Explanation (Optional):

A. **Correct** – IAW TS 3.8.1.1 Action c., the surveillance must be completed within 1 hour of the offsite source becoming inoperable (Bus 10X) given that the EDG is already inoperable. Therefore it must be completed within 1 hour of 1315 on 2/23.
Mode 3 would be required within 12 hours of river temperature exceeding 88 degrees F. IAW TS 3.7.1.3.Action a. AND the EDG being declared inoperable which occurred at 1235

B. Incorrect – This would be correct if the offsite source (bus 10X) were not inoperable

C. Incorrect – This would be correct if river temperature were not high

D. Incorrect - This would be correct if river temperature were not high and the offsite source (bus 10X) were not inoperable

Technical Reference(s): T/S 3.8.1.1.1.a
3.7.1.3.a (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Tech Spec sections 3.7.1.3 and 3.8.1.1 (other non-applicable TS)

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1,2

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031	EA2.01
	Importance Rating		4.6

EA2.01 - Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Reactor water level

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #78

Plant conditions:

- A loss of offsite power (LOP) has occurred
- All control rods inserted to position "02" or beyond
- RPV pressure is 910 psig
- HPCI and RCIC failed to start
- A and C EDGs have re-energized their respective buses
- OHA C1-E5 SRV LO LO SET ARMED is illuminated
- OHA A7-C5 RPV LEVEL 4 is illuminated
- OHA A7-D5 RPV LEVEL 3 is illuminated
- OHA A7-E5 RPV LEVEL 2 is illuminated
- OHA A7-F5 RPV LEVEL 1 is illuminated
- A and C ECCS pumps have auto-started

With the above conditions which of the following is required

A. Inhibit ADS, reopen the MSIVs, and then rapidly depressurize the reactor to the main condenser.

B. Reset the ADS timers to prevent the ADS blowdown until level decreases to -200", then emergency depressurize the reactor.

C. Inhibit ADS, verify any subsystem is lined up with a pump running available for injection, before level reaches -185" emergency depressurize the reactor.

D. Do nothing with ADS since the system will not auto initiate without a HI DW pressure signal, before level reaches -185" emergency depressurize the reactor.

Proposed Answer: C

Explanation (Optional):

A: Incorrect, With the LOP the MSIVs are closed and can not be reopened until RPS power is restored, making the condenser unavailable

B: Incorrect, no direction to reset timers, if Low Pressure ECCS available blowdown is required @ -185" not -200".

C: **Correct,** Per EOP-101-steps ALC-5 through ALC-10

D: Incorrect, direction per the EOP is to inhibit ADS. ADS blowdown logic will time out if level drops below -129" without drywell pressure in 300 seconds

Technical Reference(s): EOP-101-steps ALC-5 through ALC-10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 30726

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295028	2.4.6
	Importance Rating		4.7

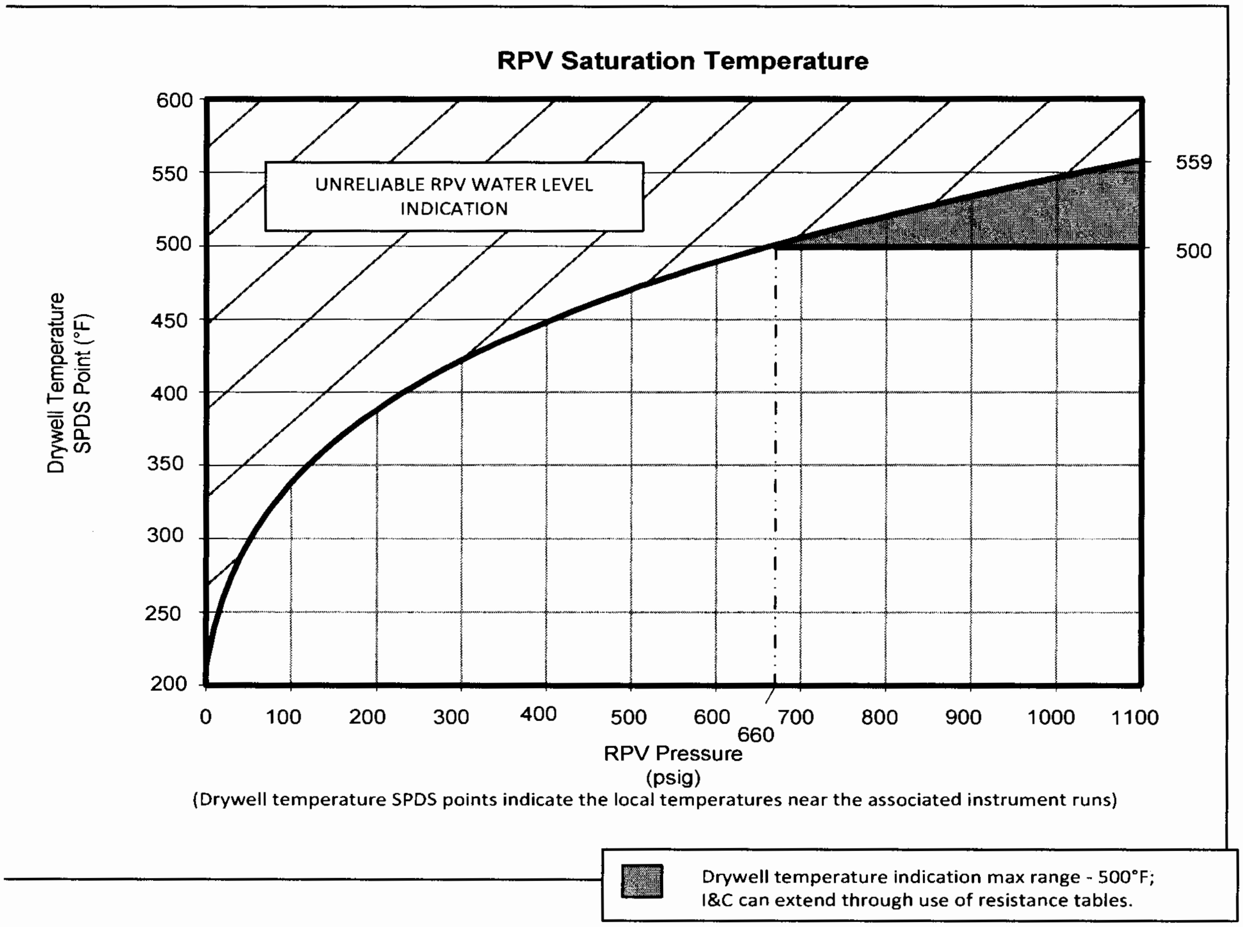
Knowledge of EOP mitigation strategies. (High Drywell Temperature)

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #79

Given:

- A Large Break LOCA has occurred concurrent with a LOP
- Only "D" EDG is running
- "D" RHR Pump is injecting to the RPV
- All control rods are fully inserted
- Drywell pressure is 32 psig and rising at 1 psig/10 minutes
- Drywell temperature is 280 F and rising at 2 degrees F/10 minutes
- Reactor pressure is 35 psig and steady
- Fuel Zone RPV level indicator LI-P615-B21 is reading (-143) inches lowering at 1"/5 minutes
- Wide Range RPV level indicator LR-623A is reading (-83) inches lowering at 2"/5 minutes
- Wide Range RPV level indicator LR-623B is reading -115 inches lowering at 4"/5 minutes
- Suppression Pool Level is 50 inches and rising slowly
- Suppression Chamber pressure is 34 psig and rising at 1 psig/10 minutes



Based on the above conditions, which one of the following actions is required at this time?

A. Maintain LPCI injection, enter EOP-206-RPV Flooding and attempt to Open 5 ADS valves.

B. Maintain LPCI injection and IAW EOP-101-RPV Control, vent the Drywell

C. Continue in all legs of EOP-101-RPV Control and all legs of EOP-102- Primary Containment Control. No other actions required until RPV level indicates -185 inches.

D. Continue LPCI injection, Emergency Depressurize IAW EOP-202, and then enter EOP-206-RPV Flooding.

Proposed Answer: A

Explanation (Optional):

A: **Correct** - with High drywell temps and low RPV pressure, per EOP caution 1 level is unreliable. Therefore it is not known and RPV flooding is required. Level is unknown due to unreliability of level instruments with high drywell temperature. RPV flooding is required. LPCI injection would continue. Se EOP -101 Step RC/L-2 retainment Override directs you to EOP-206. EOP-206 directs opening 5 ADS SRVs at Step RF-4.

B: Incorrect. Entry to EOP 206 is required due to loss of RPV level indication. DW pressure has not reached point of venting the drywell

C: Incorrect – Additional actions are required IAW EOP-101 retainment override RF-4

D: Incorrect. The Depressurization would occur based on EOP-206 actions.

Technical Reference(s): EOP-101 EOP-206 RF-4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP206E008 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	295004	AA2.03
	Importance Rating		2.9

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Battery voltage

SRO Only Justification: This question is SRO only as it requires assessing facility operating limitations in the Tech Specs and their bases, cannot be answered solely by knowing <= 1hr TS actions or above the line information, cannot be answered by knowing TS safety limits, and involves application of TS required actions and surveillance requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #80

The plant is operating at 100% power

The quarterly surveillance test (HC.MD-ST.PJ-002) for the 10D431 battery has just been performed.

Maintenance electrician reports back these results:

- Pilot Cells # 07, 14, 30 are reading 2.09 volts (float voltage)
- Connected Cells # 44, 45, 46, 91, 92 are reading 2.11 volts (float voltage)
- Specific Gravity for cell #109 is 1.173 with
- The average of all the connected cells is 1.191
- Electrolyte levels for all the cells tested were within allowable values.

All other systems are operable

Which of the following describes the Technical Specifications required actions for the above condition and the bases for these actions?

A. Enter a 14 day LCO for the HPCI system

B. Enter a 14 day LCO for RCIC system

C. Enter an 12 hr LCO for DC Distribution

D. Enter a 2 hr LCO for DC Distribution

Proposed Answer: B

Explanation (Optional):

A: Incorrect - Battery/charger listed is for RCIC, HPCI is operable

B: **Correct** - see T/S 3.8.2.1 action "b", with the RCIC 250 VDC battery inoperable the RCIC system must be declared inoperable.

C: Incorrect - the battery INOP makes RCICI INOP and is not associated with the 125 VDC The Spec

D: Incorrect - the battery INOP makes RCICI INOP and is not associated with the 125 VDC The Spec

Technical Reference(s): TS 3.8.2, 3.5.1, and 3.7.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Tech Specs 3.8.2,

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295018	AA2.03
	Importance Rating		3.5

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Cause for partial or complete loss

SRO Only Justification: This question is SRO only as it requires determining off site notifications which is a SRO ONLY function.

Question: SRO #81

The plant was operating at rated power when the Reactor Auxiliary Cooling System (RACS) head tank began to lower rapidly.

Then, a Reactor Water Cleanup (RWCU) System Non-Regenerative Heat Exchanger discharge high temperature signal was received resulting in an automatic isolation of the RWCU Inlet Outboard Isolation Valve, HV-F004,

NO other isolation valves were actuated. The plant remains stable at rated power.

Which of the following identifies:

(1) a potential cause of the RACS head tank lowering AND

(2) whether the subsequent affect on RWCU (HV-F004 isolation) was reportable under 10 CFR 50.72 requirements.

A. (1) A leak in the Reactor Recirc Pump Seal Cooler
(2) Reportable

B. (1) A leak in the Reactor Recirc Pump Seal Cooler
(2) NOT Reportable

C. (1) A leak in the RACS Heat Exchanger
(2) Reportable

D. (1) A leak in the RACS Heat Exchanger
(2) NOT Reportable

Proposed Answer: D

Explanation (Optional):

A: Incorrect - RACS system pressure is the lower pressure in recirc pump seal cooler heat exchanger. A leak would case RACS head tank level to rise. The event is not reportable.

B: Incorrect - RACS system pressure is the lower pressure in recirc pump seal cooler heat exchanger. A leak would case RACS head tank level to rise.

C: Incorrect – With only one system affected the event is not reportable.

D: **Correct** - RACS system pressure is higher than Service Water System pressure. Therefore a leak in the heat exchanger would be from RACS to SW and result in lowering head tank level. The event is NOT reportable per 10CFR50.72 as item (b)(3)(iv)(B)(2) requires containment isolation signals affecting more than 1 system. This signal only affects 1 system.

Technical Reference(s): 10 CFR 50.72 b)(3)(iv)(B)(2) (Attach if not previously provided)
NOH01RACS00C-09 Pages 13, 16,17 & 41

Proposed References to be provided to applicants during examination: 10 CFR 50.72

Learning Objective: RACS00E005 – Obj. 2 & 5 (As available)

Question Source: Bank # 120396
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 1

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295026	2.4.47
	Importance Rating		4.2

2.4.47 - Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (suppression pool water temperature high)

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #82

Given at T=0 minutes:

- An ATWS is in progress
- APRM's read 10%
- MSIV's are closed
- Manual rod insertion is in progress
- Suppression Pool level is 100" and steady
- Reactor Pressure is being controlled using SRVs
- RPV pressure is at 900 psig and rising 10 psig/minute
- Suppression Pool temperature is 188°F and rising at 1°F/5 min
- Suppression Chamber pressure is 22 psig and rising at 1 psi/15 min

AT T =10 minutes which of the following describes the required action and the reason for that action?

A. Reduce RPV pressure IAW EOP-101A-ATWS RPV Control, to prevent exceeding the SRV Tail Pipe Level limit.

B. Emergency Depressurize IAW EOP-0202, due to exceeding the Pressure Suppression Pressure limit.

C. Emergency Depressurize IAW EOP-0202, due to exceeding the Heat Capacity Temperature limit.

D. Reduce RPV pressure IAW EOP-101A-ATWS RPV Control, to prevent exceeding the Pressure Suppression Pressure limit.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - With RPV pressure at 900 psig and SP temperature at 195 °F and rising at 1°F/ 5 min, the HCTL will be exceeded at T=10 minutes, IAW Step SP/T-9 an Emergency Depressurization is required SP/T-10 and requires exiting RC/P of EOP-101A.

B: Incorrect - Pressure Suppression Pressure (SCP-L) curve is not dependent on Reactor pressure. SP pressure is at 22 psi and rising at 1 psi/15 min. and SP level is at 70 " and stable. With this rate of change, it will be about 30 minutes before the PSPL is exceeded, an ED is NOT yet appropriate. The combination of Torus level and pressure are initially within the PSP curve and NO emergency depressurization is required on these two parameters.

C: **Correct** - With RPV pressure at 900 psig and SP temperature at 188 °F and rising at 1°F/ 5 min, the HCTL will be exceeded in 10 mimutes, IAW Step SP/T-9 an Emergency Depressurization is required SP/T-10 and requires exiting RC/P of EOP-101A.

D: Incorrect - Pressure Suppression Pressure (SCP-L) curve is not dependent on Reactor pressure. SP pressure is at 22 psi and rising at 1 psi/15 min. and SP level is at 70 " and stable. With this rate of change, it will be about 30 minutes before the PSPL is exceeded, an ED is NOT yet appropriate. The combination of Torus level and pressure are initially within the PSP curve and NO emergency depressurization is required on these two parameters.

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

Proposed References to be provided to applicants during examination: HCTL, PSP, SPL-P DPSIL curves

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295033	EA2.01
	Importance Rating		3.9

EA2.01 - Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Area radiation levels

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #83

Given the following conditions:

Backwash of Clean-Up Filter Demineralizer AF-203 has just been completed.
Transfer of the RWCU Backwash Receiver Tank to Radwaste is in progress.

Then,

A catastrophic failure of Backwash Transfer pump 1AP-214 suction line causes a spill into the Reactor Building.
All attempts to isolate the leak have been unsuccessful.

Reactor Building Area Radiation conditions are as follows:

Reactor Building Area Radiation Monitor	**Beginning of Shift (Normal Rad Levels)**	**Current Conditions**
9RX706 Reactor Cleanup Demin. Sys. Equipment	2 mr/hr	2400 mr/hr - In Alarm
9RX723 Outside Reactor Bldg. Sample Station	3 mr/hr	1100 mr/hr - In Alarm
9RX708 Sample Station	3.5 mr/hr	4500 mr/hr - In Alarm
Other Reactor Building Area Radiation Monitors	2 to 5 mr/hr	3 to 7mr/hr - NOT In Alarm

Which one of the following is/are the required action(s)?

A. Commence a normal reactor shutdown to cold shutdown IAW IO.ZZ-0004, Shutdown From Rated Power to Cold Shutdown.

B. Attempt to stop the tank drain line leakage. NO power reduction is required.

C. Initiate a manual scram and Emergency Depressurize the RPV IAW EO.ZZ-0202, Emergency Depressuriztion.

D. Runback Reactor Recirc Pumps to minimum, initiate a manual scram, trip the Main Turbine and enter AB.ZZ-0000, Reactor Scram. Emergency Depressurization is not required.

Proposed Answer: A

Explanation (Optional):

A: **Correct** - RWCU Backwash Receiving tank is not a primary System (Reactor Coolant System), with 2 areas > Max Safe Operating Limit, Plant shutdown and cooldown per IO-004 is applicable. The Max Safe Op Limit for Area Radiation is defined at NOTE #1 of EOP-103/4 as 1000 times normal. In this scenario, 2 area rad monitors are greater than 1000 times normal. Therefore steps RB-21 thru RB-23 apply.

B: Incorrect. A shutdown IAW IO.ZZ-00004 is required

C: Incorrect – Applies only if a Reactor Coolant system (primary system) is discharging into the building.

D: Incorrect. If a primary system were discharging into the building then ED would be required.

Technical Reference(s): EOP 103/4 and Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 66818

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295010	2.4.41
	Importance Rating		4.6

2.4.41 - Knowledge of the emergency action level thresholds and classifications. (High DW pressure)

SRO only justification: This question is SRO as it requires determining event classification which is an SRO function only.

Question: SRO #84

The plant was operating at rated power when a main steam line break occurred in the drywell causing an MSIV isolation and a reactor scram.

Given:

- The Reactor Scram Hard Card actions have been performed
- 28 control rods are at position 48
- Reactor power is 15% and stable
- Suppression Pool temperature is 125°F and slowly rising
- SRVs are cycling
- RPV level lowered to -135" and has recovered to -75" and slowly rising
- Secondary Containment is being maintained
- Secondary Containment are temperatures and stable
- Drywell pressure is 2.3 psig and rising slowly
- ECCS has initiated as designed
- Reactor Coolant Sample Activity is 335µCi/gm dose equivalent I-131

Which of the following describes the HIGHEST classification level for this event and the reason for that classification?

A. UE due to High Drywell Pressure

B. ALERT due to High Drywell Pressure

C. SAE due to Fission Product Barrier Losses

D. GE due to Fission Product Barrier Losses

Proposed Answer: C

Explanation (Optional):

A: Incorrect – At least an Alert would be declared for High Drywell Pressure if that was the only issue. With other issues the classification is higher.

B: Incorrect – not the highest classification

C: **Correct –** Using barrier table – High DW pressure = 5 points (RB2.L), Reactor Coolant Sample Activity = 5 points (FB3.L) Total +10 points therfore SAE

D: Incorrect – Not at GE level

Technical Reference(s): EAL Barrier Table (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EALs

Learning Objective: (As available)

Question Source: Bank # 116202

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

		RO	SRO
Level		RO	SRO
Tier #			1
Group #			2
K/A #		500000	EA2.03
Importance Rating			3.8

EA2.03 - Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for Drywell

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #85

Given the following at T=0 minutes:

- A large break LOCA has occurred in the Drywell
- Multiple Equipment failures occurred
- Drywell Pressure is 16 psig and rising at 1 psig/10 minutes
- Drywell Temperature is 240 degrees F. and stable
- Steam Cooling was required until RPV level was restored above TAF with Fire Water after Emergency Depressurization.
- RPV level is -135 inches and stable
- The H2O2 Analyzers were placed in service 2 hours ago
- The Analyzers just alarmed on High Drywell H2
- H2 concentration is at 1.9% and rising at 0.2%/10 minutes
- O2 concentration is 1.3% and rising at 0.1%/10 minutes

At T=10 minutes, what actions would be required?

A. Enter SAGs and perform concurrently with EO.ZZ-0102, Primary Containment Control because the lower detonation limit for H2 has been reached and containment failure may be imminent.

B. IAW EO.ZZ-0102, Primary Containment Control, Implement EO.ZZ-0318, Containment Venting and vent containment via theSuppression Pool.

C. Exit EO.ZZ-0102, Primary Containment Control and Enter SAGs.The current H2 concentration confirms core damage beyond the requirements of 10CFR50.46 for the ECCS design.

D. IAW EO.ZZ-0102, Primary Containment Control, Implement EO.ZZ-0318, Containment Venting and vent containment via the Drywell. Enter the SAGs only when the H2 concentration has reached its combustible limit.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – No EOP direction is present to direct venting containment via drywell. Additionally, any venting would be initially attempted via the suppression pool due to the scrubbing action available but only IAW SAG procedures. EOP PC/H-2 Bases state - Hydrogen and oxygen must both be present and in sufficient concentration for combustion to occur. Excessive hydrogen concentration (≥ 6%), mixed with high oxygen concentration (≥ 5%) and ignited in the confined space of the primary containment generates peak pressures which may exceed the structural capability of the drywell, suppression chamber or drywell-suppression chamber boundary. At this time H2 concentration is less than the combustion level noted in the bases

B: Incorrect. SAG entry is required IAW EO.ZZ-0102 Retainment override PC/H-1. The EOP is exited. Venting the containment via the Suppression Pool may be viable but only IAW SAGs.

C: Correct – IAW EOP-102 Bases – Step PC/H-1 –
2% is both the alarm setpoint as well as a concentration that confirms core damage beyond the requirements of 10CFR50.46 for the ECCS design, yet is still within concentrations for which Recombiners are designed to operate. Consequently the transition to SAG where the balance of hydrogen control steps have been placed is happening
1. After core damage in excess of what EOPs are typically designed to handle has occurred <u>and</u>
2. Before more dramatic action to control hydrogen is required.

D: Incorrect – IAW EOP-0102 Retainment override, SAGs are entered at >2% Hydrogen concentration not at the combustible limit.

Technical Reference(s): EOP 102 and bases (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	217000	2.2.37
	Importance Rating		4.6

2.2.37 - Equipment Control: Ability to determine operability and/or availability of safety related equipment.

SRO Only Justification: This question is SRO only as it requires assessing facility operating limitations in the Tech Specs and their bases, cannot be answered solely by knowing <= 1hr TS actions or above the line information, cannot be answered by knowing TS safety limits, and involves application of TS required actions and surveillance requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #86

The plant is in OP CON 2 with reactor steam dome pressure at 199 psig.

The RCIC surveillance test was completed satisfactorily 4 hours ago with the following results and actions taken:

- The breaker for RCIC Pump Suction Valve from the CST, HV-F031 tripped free and would not reset. Operators have manually CLOSED the valve. The breaker has been tagged OPEN.

The HPCI surveillance test was just completed with the following results.

- A steam leak occurred at the HPCI Turbine Inlet. You directed isolating HPCI and the field operator reports that the steam leak has stopped.

What is the operational status of RCIC & HPCI and what is correct in regard to Technical Specifications? In regard to system operability and Technical Specifications ____________

A. RCIC & HPCI are INOPERABLE. Unless either HPCI or RCIC can be restored to operability, Be in COLD SHUTDOWN within the next 24 hours.

B. RCIC & HPCI are INOPERABLE. Startup may continue to normal operating pressure however OP CON 1 may not be entered

C. RCIC is Operable. HPCI is currently INOPERABLE. Startup may continue to normal operating pressure however OP CON 1 may not be entered.

D. RCIC is Operable. HPCI must be declared OPERABLE before entering OP CON 1.

Proposed Answer: D

Explanation (Optional):

A: Incorrect – RCIC is operable since a flowpath from the Suppression Pool is still available

B: Incorrect - RCIC is operable since a flowpath from the Suppression Pool is still available

C: Incorrect – HPCI is not inoperable with steam dome pressure <200 psig

D: **Correct –**
TS 3.5.1 -
3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.
ACTION: Note: LCO 3.0.4.b is not applicable to RCIC.

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; **restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours** and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

HPCI is not required to be OPERABLE until reactor steam dome pressure is greater than or equal to 200 psig.

Technical Reference(s): 3.7.4, 3.5.1 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1,2

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	230000	2.1.32
	Importance Rating		4.0

2.1.32 - Conduct of Operations: Ability to explain and apply system limits and precautions. (RHR/LPCI Torus/Suppression Pool Spray Mode)

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #87

A steam line break has occurred in the Primary Containment, with the following:

RPV level is -37 inches and rising slowly with HPCI
RPV pressure is 440 psig lowering slowly
Drywell pressure is 9.8 psig rising slowly
Drywell temperature is 180 degrees F rising slowly
Suppression Chamber pressure is 9.8 psig rising slowly
Suppression Pool temperature is 120 degrees F rising slowly
Suppression Pool level is 84 inches rising slowly

A precaution in procedure HC.OP-SO.BC-0001, RHR System Operation, states that IF RHR is in a specific alignment "the Torus suppression capability would be potentially bypassed in the event of a LOP"

Which one of the following:

Is a lineup for "A" and "B" RHR loops that meets the required EOP actions for the given conditions AND avoids the stated results in the precaution?

A. Loop "A" in drywell spray and Loop "B" in LPCI.

B. Loop "A" in drywell spray and Loop "B" in suppression pool cooling/spray.

C. Loop "A" in suppression pool cooling and Loop "B" in suppression pool spray.

D. Loop "A" in both drywell and suppression chamber spray. Loop "B" in suppression pool cooling/spray

Proposed Answer: B

Explanation (Optional):

A: Incorrect – LPCI would not be required for given conditions with RCIC recovering RPV level

B: **Correct** - IAW HC.OP-SO.BC-0001 – step 3.1.12 - IF the Drywell AND Suppression Spray are being used simultaneously on the same loop of RHR, THEN, the potential exists that, if a LOP were to occur, the associated valves would stay open AND the piping could begin to drain to the Torus until the pump restarts automatically (sequenced). The Torus suppression capability would be bypassed with those legs of piping drained completely. With SC pressure >9.5 DW spray is required. The other loop would be in SP cooling/spray due to elevated SP temperature.

C: Incorrect – Drywell spray would also be required with Suppression Chamber pressure at 9.8 psig (see EOP 102 step RW/P-10)

D: Incorrect. NO EOP direction to place one RHR loop in both drywell and suppression chamber spray

Technical Reference(s): EOP -102 (Attach if not previously provided)
HC.OP-SO.BC-0001

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	212000	A2.02
	Importance Rating		3.9

A2.02 - Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS bus power supply failure

SRO Only Justification: This question is SRO only as it requires assessing facility operating limitations in the Tech Specs and their bases, cannot be answered solely by knowing <= 1hr TS actions or above the line information, cannot be answered by knowing TS safety limits, and involves application of TS required actions and surveillance requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #88

The plant is at rated power.

Then;

- The RPS Electrical Protection Assembly (EPA) 1AN411 logic card failed, causing the breaker to trip.
- The RPS EPA 1BN411 breaker remained closed.

(1) What is the operational status of the "B" RPS MG Set EPA breakers and,
(2) Is "B" RPS bus power monitoring LCO ACTIVE or TRACKING AFTER it is transferred to the alternate supply?

A. (1) Both the 1AN411 and 1BN411 are Inoperable
(2) A Tracking LCO is entered

B. (1) ONLY the 1AN411 is Inoperable
(2) A Tracking LCO is entered

C. (1) Both the 1AN411 and 1BN411 are Inoperable
(2) An Active LCO is entered

D. (1) ONLY the 1AN411 is Inoperable
(2) An Active LCO is entered

Proposed Answer: A

Explanation (Optional):

A: **Correct** - BOTH EPAs are inoperable. The one with a failed card and also the downstream EPA breaker whichshould have tripped on undervoltage, therefore it is also inoperable. Since nothing in the stem supports any inoperability condition for the Alternate EPA breakers, and the original trip of the Normal EPA was due to a logic card failure, and not an actual Overvoltage, Undervoltage, or Underfrequency condition, the RPS bus B power monitoring is TRACKING once on the ALTERNATE supply. (TS 3.8.4.4)

B: Incorrect - Both EPAs are inoperable

C: Incorrect - ONLY a tracking LCO is required because the RPS has power.

D: Incorrect. Both EPAs are inoperable, ONLY a tracking LCO is required because the RPS has power.

Technical Reference(s): TS 3.8.4.4 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	206000	A2.12
	Importance Rating		3.5

A2.12 - Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of room cooling: BWR-2,3,4

SRO Only Justification: This question is SRO only as it requires assessing operability and technical specifications which is an SRO function.

Question: SRO #89

Given the following initial conditions:

- The Unit is in OPCON 1 at 100% power.
- HPCI is currently operable.
- 1AVH-209 HPCI Room Cooler control switch is in "AUTO LEAD".
- 1BVH-209 HPCI Room Cooler control switch is in "AUTO"

Later in the shift,

- A Maintenance Electrician reports to the Control Room and requests permission to place 1BVH-209 HPCI Room Cooler control switch in "STOP" for about 30 minutes to take voltage readings on the control switch contacts.
- The Electrician states that he will remain near the switch while performing his work and that he can restore his work and place the control switch back to "AUTO" in less than 15 minutes in the event of a HPCI start.
- The CRS gives the Electrician permission to perform the work.
- The Electrician places the 1BVH-209 HPCI Room Cooler control switch in "STOP" at 10:00 (the current time).

What is the status of HPCI?

A. HPCI is INOPERABLE since BOTH Room Coolers are NOT operable.

B. HPCI remains operable since Room Coolers are NOT required to support HPCI operability.

C. HPCI remains operable since BOTH Room Coolers are operable to support HPCI.

D. HPCI remains operable since ONE Room Cooler is operable to support HPCI.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - HPCI is operable since 1AVH-209 is unaffected by the work and only one cooler is required to be operable to support HPCI operability,
Plausible: Would be true if both coolers were required to support HPCI operability.

B: Incorrect - At least one room cooler is required to support HPCI operability.
Plausible: Some plants have performed analyses to prove that room coolers are not required for HPCI operability, so the answer is plausible.

C: Incorrect - 1BVH-209 is NOT operable. Credit can NOT be taken for manual action to restore the cooler in this case since there was no mention of an operability determination to support manual action in the stem. In addition, it is not likely that the Electrician would qualify as a "dedicated" operator for compensatory action purposes.
Plausible: Would be correct if the credit were taken for manual action. This distracter might be attractive to someone believing that both coolers are required for HPCI operability.

D: **Correct -** One cooler is required to be operable to support HPCI in accordance with OP-HC-108-115-1001 Exhibit 2. 1AVH-209 is unaffected by the work and remains operable; therefore, HPCI is operable.

Technical Reference(s): OP-HC-108-115-1001 Exhibit 2 page 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP103 - E002 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2007

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	215004	A2.01
	Importance Rating		2.9

A2.01 - Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #90

Given:

- The reactor is critical with the operable IRMs on-scale on range 7
- SRM B is bypassed with it's associated joystick due to signal cable problems
- IRMs A and B are bypassed with their associated joysticks due to failing downscale
- Reactor pressure is 60 psig.
- Heat-up is in progress
- Then, the startup is suspended because the high voltage power supply for the A SRM fails.

20 minutes later:

IRMs : 'C' is on range 3, 'D' is on range 2
'E' is on range 2, 'F' is on range 2
'G' is on range 2, 'H' is on range 1

- SRM period indicators are reading infinity.

Estimated time to repair high voltage power supply for the A SRM is 1 hour from now.

Based on the above conditions ____?

A. IAW IO.ZZ-0003, Startup From Cold Shutdown To Rated Power, insert control rods IAW the Shutdown Sequence.

B. IAW IO.ZZ-0002, Preparation For Plant Startup, lock the Mode Switch in Shutdown.

C. IAW IO.ZZ-0002, Preparation For Plant Startup, insert control rods IAW the Shutdown Sequence.

D. IAW IO-ZZ-0003, Startup From Cold Shutdown To Rated Power, lock the Mode Switch in Shutdown.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - Per IO.ZZ-003 Step 5.2.19.D, the core conditions are NOT being monitored by the minimum required nuclear instrumentation and LOCK the mode switch in shutdown is required.

B: Incorrect - actions for moving control rods are not in IO.ZZ-002. The action required is to LOCK the mode switch in shutdown, which is directed from IO-003

C: Incorrect - actions for moving control rods are not in IO.ZZ-002. The action required is to LOCK the mode switch in shutdown, which is directed from IO-003

D: **Correct -** Per IO.ZZ-003 Step 5.2.19.D, the core conditions are NOT being monitored by the minimum required nuclear instrumentation and LOCK the mode switch in shutdown is required.

Technical Reference(s): HC.OP- IO.ZZ-0003, Step 5.2.19 & Caution (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: IOP003E005 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	202001	A2.08
	Importance Rating		3.4

A2.08 - Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation flow mismatch: Plant-Specific

SRO Only Justification: This question is SRO only as it requires assessing facility operating limitations in the Tech Specs and their bases, cannot be answered solely by knowing <= 1hr TS actions or above the line information, cannot be answered by knowing TS safety limits, and involves application of TS required actions and surveillance requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #91

Given:

A Reactor Recirc pump transient has occurred.

Plant conditions have stabilized and Reactor Recirc parameters are now:
Reactor Recirc Pump A is in MANUAL speed control at 34 KGPM loop flow.
Reactor Recirc Pump B is in MANUAL speed control at 28 KGPM loop flow.
Indicated total core flow is 80 MLBM/hr.

Which one of the following describes the current limit for loop flow mismatch and the reason for the limit?

A. 10% rated core flow; Maintain MCPR operating limits.

B. 10% rated core flow; Maintain adequate core flow coast-down following a LOCA.

C. 5% rated core flow; Maintain MCPR operating limits.

D. 5% rated core flow; Maintain adequate core flow coast-down following a LOCA.

Proposed Answer: D

Explanation (Optional):

A: Incorrect - should be 5 % rated core flow based on TS. MCPR not the concern per TS bases

B: Incorrect. should be 5 % rated core flow based on TS.

C: Incorrect - MCPR not the concern per TS bases

D: **Correct -** Using footnote ** of 3.4.1.3 "Effective core flow shall be the core flow that would result if both recirculation loop flows were assumed to be at the smaller value of the 2 loop flows", the smaller loop flow is 28KGPM +28KGPM = 56KGPM. 56KGPM corresponds to 72.5 MLBM/HR on Attachment 6 of ST-BB-001. This is greater than the limit of 70 MLBM/HR (70% effective core flow) by TS 3.4.1.3. Therefore the limit is 5% of rated core flow. The basis for the limit is to ensure an adequate core flow coastdown from either recirculation loop following a LOCA. Per TS bases 3.4.1.

Technical Reference(s): OP.ST-BB-0001 Att.6 (Attach if not previously provided)
TS 3.4.1.3

Proposed References to be provided to applicants during examination: OP-ST-BB-0001 ATT.6

Learning Objective: RECIRCE015 - ((As available)
(SRO Only) Explain the bases for those Technical Specification items associated with the Recirculation System

Question Source: Bank #

Modified Bank # 120320 (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2,5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	201001	A2.10
	Importance Rating		3.6

A2.10 - the CONTROL ROD DRIVE HYDRAULIC SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: †Low HCU accumulator pressure/high level

SRO Only Justification: This question is SRO only as it requires assessing facility operating limitations in the Tech Specs and their bases, cannot be answered solely by knowing <= 1hr TS actions or above the line information, cannot be answered by knowing TS safety limits, and involves application of TS required actions and/or surveillance requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #92

A plant startup is in progress.

At 0800, RPV pressure is 950 psig.

Then, the "B" CRD pump trips and CRD Charging Water Header pressure lowers to 930 psig.

At 0815, preparations are in progress to start the "A" CRD pump.

At 0826, the following three accumulator alarms are received and the field operator calls in with their respective pressures.

- 14-11 is at position 00 with accumulator pressure at 930 psig.
- 30-23 is at position 48 with accumulator pressure at 920 psig.
- 18-27 is at position 04 with accumulator pressure at 935 psig.

At 0826, which of the following describes the status of the accumulators listed above and required actions IAW Technical Specifications?

A. ONLY 30-23 is inoperable and it must be fully inserted by 0846.

B. ONLY 30-23 is inoperable and the Mode Switch must be Locked in Shutdown by 1626.

C. ALL 3 Control Rods are inoperable and the Mode Switch must be Locked in Shutdown by 0846.

D. ALL 3 Control Rods are inoperable and 30-23 must be fully inserted no later than 0926.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - All Control rods are inoperable

B: Incorrect. All Control rods are inoperable. the mode switch must be locked in shutdown within 20 minutes. (0846)

C: Correct - 2 Control rods are inoperable IAW TS Surveillance 4.1.3.5.A. (NOTE: to be inop, rods must be withdrawn) IAW TS 3.1.3.5.a.2.a – with 2 or more control rod accumulators inoperable, the mode switch must be locked in shutdown within 20 minutes. (0846)

D: Incorrect. the mode switch must be locked in shutdown within 20 minutes. (0846)

Technical Reference(s): TS 3.1.3.5.a.2.a. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: TECSPEC010 (As available)

Question Source: Bank # X- SD

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	288000	A2.03
	Importance Rating		3.7

A2.03 - Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of coolant accident: Plant-Specific

SRO only justification: This question is SRO only as it requires determining as it requires event classification which is an SRO function only.

Question: SRO #93

The plant was at rated power.

A plant transient resulted in an un-isolable primary system leak into the Main Steam Tunnel.

The event INITIALLY leads to an Offsite Gaseous Radioactive Release Rate exceeding the Unusual Event classification criteria but below the ALERT level classification.

20 minutes later,

- The field operator informs you that Turbine Building Ventilation has tripped but is available for a restart if needed.
- The Gaseous Radioactive Release Rate has just met the SAE event classification level and is now trending lower
- The release rates are expected to continue trending lower but will remain above the ALERT level for at least the next 6 hours based on current release rates.

Which one of the following describes;
1) When EO.ZZ 0104 – Rad Release Control is entered
AND
2) Action(s) required, if any, in regard to Turbine Building ventilation and reactor depressurization.

A. (1) when the Unusual Event classification level was met
(2) Emergency Depressurization is required. DO NOT restart Turbine Building ventilation

B. (1) when you were informed that the SAE level was met
(2) Emergency Depressurization is required. DO NOT Restart Turbine Building ventilation

C. (1) when the Unusual Event classification level was met
(2) Emergency Depressurization is NOT required. Restart Turbine Building ventilation

D. (1) when you were informed that the SAE level was met
(2) Emergency Depressurization is NOT required. Restart Turbine Building ventilation

Proposed Answer: D

Explanation (Optional):

A: Incorrect – Entry to EOP-103/4 is not required until Alert level is met. An ED is not required until GE level is met and discharge is continuing. Restarting TB ventilation is required IAW step RR-4.

B: Incorrect - An ED is not required until GE level is met and discharge is continuing. Restarting TB ventilation is required IAW step RR-4.

C: Incorrect - Entry to EOP-103/4 is not required until Alert level is met.

D: **Correct** – EOP 103/4 entry is required when release level has exceeded the ALERT classification level. Until you informed that this occurred (when the SAE level was reached) entry to EOP 103/4 was not REQUIRED. Restarting TB ventilation is required IAW step RR-4

Technical Reference(s): EOP-103/4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: EOP103E006 (As available)

Question Source: Bank # X

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 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.2	
	Importance Rating		4.4

Knowledge of operator responsibilities during all modes of plant operation.

SRO only justification: This question is SRO only as it requires knowledge of responsibility for assessing operability which is an SRO function.

Question: SRO #94

IAW OP-AA-108-115 Operability Determinations, the determination of whether systems, structures, or components (SSCs) are operable is the responsibility of:

A. ONLY the Operations Director.

B. ANY active SRO available to conduct the review.

C. ANY active SRO on the operating shift who is responsible for plant operations.

D. ONLY the Shift Manager on the operating shift crew who is responsible for plant operations.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - A senior licensed operator/SRO on the operating shift crew with the responsibility for plant operations. The Operations director is NOT on the operating shift crew.

B: Incorrect. The SRO must be on the operating shift crew with the responsibility for plant operations

C: **Correct** – IAW OP-AA-108-115 Step - 3.3.1. A senior licensed operator/SRO on the operating shift crew with the responsibility for plant operations makes the declaration of operability or functionality; i.e., makes the call on whether an SSC described in TSs is operable or inoperable, or an SSC that is not described in the TSs is functional or not functional.

D: Incorrect - A senior licensed operator/SRO on the operating shift crew with the responsibility for plant operations

Technical Reference(s): OP-AA-108-115 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # 109371

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 1

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.7	
	Importance Rating		3.6

Knowledge of the process for conducting special or infrequent tests.

SRO only justification: This question is SRO only as it requires application of 10CFR55.43(b)(3), facility licensee procedures required to obtain authority for design and operating changes to the facility

Question: SRO #95

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 10CFR 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-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS

B. (1) ONLY NRC notification prior to implementation
(2) OP-AA-103-103, OPERATION OF PLANT EQUIPMENT

C. (1) ONLY NRC notification prior to implementation
(2) OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS

D. (1) NRC approval prior to implementation
(2) OP-AA-103-103, OPERATION OF PLANT EQUIPMENT

Proposed Answer: A

Explanation (Optional):

A: **Correct** - IAW LS-AA-104 – Purpose – ""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.**

Special tests or evolutions are defined in Att 1 of OP-AA-108-110. Specifically examples of special evolutions include: Appropriate portions of plant startup after an outage that involves significant change to plant systems, equipment or procedure related to the core, reactivity control, or reactor protection.

Additionally, evolutions that require the use of special tests in conjunction with existing procedures may also be classified as special evolutions, including Complex Modification Function Testing

B: 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.

C: Incorrect - NRC approval is the purpose, not notification

D: 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.

Technical Reference(s): OP-AA-108-110 (Attach if not previously provided)
LS-AA-104

Proposed References to be provided to applicants during examination: none

Learning Objective: (As available)

Question Source: Bank # X

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 1

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	2.3.6	
	Importance Rating		3.8

Ability to approve release permits.

SRO only justification: The SRO is responsible for approving any offsite release permits. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #96

Given the following:

- The plant is in Hot Shutdown
- Reactor pressure is 900 psig.
- Drywell Oxygen concentration is 4.0%.
- Primary Containment Gaseous Effluent Release permit has been obtained.

De-inerting will begin at 0800 on day shift.

As the CRS, you are given the attached valve permit from the NCO for review.
(assume "total open hours during previous year" is correct)

Based on your review, you should:

A. approve it. All information is correct.

B. NOT approve it. The "hours AVAILABLE this date" is incorrect. All other information is correct.

C. NOT approve it. The "hours AUTHORIZED this date" is incorrect. All other information is correct.

D. NOT approve it. The "hours AUTHORIZED this date & hours AVAILABLE this date" are both incorrect. All other information is correct.

Proposed Answer: C

Explanation (Optional):

A: Incorrect – The hours authorized this date should be 24 per step 4.3.9.

B: Incorrect. Those hours are correct.

C: Correct – IAW OP.HC-103-105

4.3. Initiation of Valve Permit. When it is determined that a valve permit is necessary for Containment Atmosphere Control System valves, a valve permit shall be initiated as follows:

NOTE Permits are valid for only ONE calendar day ending at 2400.

4.3.1. ENTER the valid date for the permit on Form 1 and in Section A of Form 2. The date of the permit will serve as the unique identifier for that permit.

4.3.2. CONTACT Radiation Protection to determine if a Gaseous Effluent Permit will be required for the pending valve manipulation. If a Gaseous Effluent Permit is required, OBTAIN the number from Radiation Protection and enter it in section A of Form 2. If initiating a new valve permit for commencing a new day with valves open,

USE the same Gaseous Effluent Permit number from the previous day.

NOTE An administrative limit of less than 452 hours of open time is imposed by this procedure to ensure that the Tech Spec limit is never exceeded. A running total of the time that these valves are open during the previous year will be maintained by this procedure. At the end of each day, any open time that was accumulated on or before the same date one year earlier, will be subtracted from the running total of open valve time.

4.3.3. From Form 1, OBTAIN the date and number of hours valves were open for each occasion of recorded valve operation during the previous year. ENTER this information in Section B of Form 2.

4.3.4. On Form 2, COMPUTE the total number of hours these valves have been open in the previous year

4.3.5. COMPLETE the information in Section B of Form 2.

4.3.6. The NCO performing the Section B calculations should sign in the appropriate space and enter the time and date.

4.3.7. The SM/CRS should verify the calculations, sign in the appropriate space for verification and authorization and enter the time and date.

4.3.8. On Form 1, ENTER the name of the SM/CRS authorizing the valve permit and the number of hours authorized on this permit. The NCO entering this information should initial in the appropriate space.

4.3.9 **The hours authorized this date may exceed the actual hours remaining in the day for which the permit was prepared.**

D: Incorrect. The "hours available this date" is correct

Technical Reference(s): OP-HC-103-105 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OC-HC-103-105 permit (this is required to answer question and should not be considered a reference)

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.40	
	Importance Rating		4.5

Knowledge of the SRO's responsibilities in emergency plan implementation.

SRO only justification: The SRO is responsible for implementation of the Emergency Plan and must be knowledgable of actions required as Emergency Coordinator.

Question: SRO #97

An event has occurred at the plant.

The TSC is ACTIVATED.
The EOF is MANNED and NOT ACTIVATED.

IAW NC.EP-EP.ZZ-0102 "Emergency Coordinator Response", which one of the following describes the individual responsible for escalating an emergency event level from a SAE to a GE?

A. The Shift Manager.

B. The Emergency Duty Officer.

C. The Emergency Response Manager.

D. The Site Vice President.

Proposed Answer: B

Explanation (Optional):

A: Incorrect – IF the TSC and EOF were not activated this would be correct.

B: Incorrect – With the TSC activated, the EC responsibilities shift to the EDO. The EC is responsible and cannot delegate designation of the EAL.

C: Incorrect – IF both the TSC and EOF were activated, this would be correct.

D: Incorrect – Site VP is not a designated EC

Technical Reference(s): NC.EP-EP-0102 Step 5.2 (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 X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		3
Group #		1
K/A #	2.1.9	
Importance Rating		4.5

Ability to direct personnel activities inside the control room.

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #98

Given:

- The plant was operating at rated power
- A small leak occurred the drywell
- A manual scram was initiated
- All Control Rods are Full In
- A Loss of Feedwater occurred on the scram initiation
- Drywell Pressure is 1.2 psig and is rising very slowly
- Annunciator A7-E4, Drywell Pressure HI/LO is NOT illuminated
- RPV level lowered to (-10) inches and is recovering
- RCIC & HPCI have just started to inject to the RPV

Then, these alarms annunciate:

B3-A3 CORE SPRAY PUMP B AUTO START
B3-A4 CORE SPRAY PUMP D AUTO START

Core Spray pumps "B" and "D" are verified running The "B" and "D" EDGs are also running.

What describes the actions required in response to the above annunciators and any additional actions?

A. Stop the pumps. IAW EOP-101, maintain RPV level between 12.5" and 54" using RCIC and/or HPCI.

B. Do NOT stop the pumps. IAW EOP-101, maintain RPV level between 12.5" and 54" using RCIC and/or HPCI.

C. Stop the pumps. Immediately secure the EDGs IAW HC.OP-SO.KJ-0001, Emergency Diesel Generators Operation.

D. Do NOT stop the pumps. Align Condensate Transfer to maintain RPV level >(-129") IAW HC.OP-EO.ZZ-309, Alternate Injection Using Condensate Transfer.

Proposed Answer: A

Explanation (Optional):

A: **Correct** – The ARP direction is to verify an initiation signal is present (129" or 1.68 psig dw pressure). Because no signal is present the direction is to stop the pumps. EOP 101 guidance (step RC/L-4) given the loss of feedwater would be to maintain RPV level 12.5" to 54" using the available system (RCIC/HPCI).

B: Incorrect – The pumps must be secured.

C: Incorrect - There is no direction in the given Annunciator response to immediately secure the EDGs. Additionally, given the stem conditions with RPV level rising, HPCI is not required.

D: Incorrect - The pumps must be secured. Additionally condensate transfer would not be required until it is determined that RPV level cannot be maintained between 12.5" and 54".

Technical Reference(s): EOP-101 (Attach if not previously provided)
ARPs B3-A3, B3-A4

Proposed References to be provided to applicants during examination: none

Learning Objective: CSSYS0E003 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: NRC 2009

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.38	
	Importance Rating		4.5

Knowledge of conditions and imitations in the facility license

SRO Only Justification: This question is SRO only as it requires assessing facility operating limitations in the Tech Specs and their bases, cannot be answered solely by knowing <= 1hr TS actions or above the line information, cannot be answered by knowing TS safety limits, and involves application of TS required actions and/or surveillance requirements. The question cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question: SRO #99

Given:

- Power is 89%
- At 1200 on 2/16, due to a recent procedure change, it was determined that part of a TS required surveillance was NOT performed.
- The incomplete surveillance was performed on 2/13
- The last complete satisfactory surveillance was completed at 1200 on 1/15.
- The surveillance is required to be performed at least once per 31 days

The action statement requires that the associated inoperable equipment must be restored within 72 hrs, or be in Hot Shutdown within the next 12 hrs and in Cold Shutdown within the following 24 hours.

At 1600 on 2/16, the surveillance is performed, and is determined to be UNSAT.

Which of the following is required?

A. Be in Hot Shutdown by NO later than 0400 on 2/17.

B. Be in Cold Shutdown by NO later than 1600 on 2/17.

C. Be in Hot Shutdown by NO later than 0400 on 2/20.

D. Be in Cold Shutdown by NO later than 1600 on 2/20.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - An additional 72 hours is allowed by the LCO before beginning the clock toward Hot and Cold Shutdown requirements

B: Incorrect - An additional 72 hours is allowed by the LCO before beginning the clock toward Hot and Cold Shutdown requirements

C: **Correct** – the surveillance was completed within the required surveillance time of 31 days. Therefore. A 72 hour clock starts before the 12 hour clock begins for the requirement to be in Hot Shutdown. This would be a total of 84 hours resulting in a time of 0400 on 2/20.

D: Incorrect – Cold shutdown is required within the 24 hours following when Hot shutdown is acheived

Technical Reference(s): TS 4.01 and 4.02 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: TECSPCE006 (As available)

Question Source: Bank # 53354

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Facility: Hope Creek 2015

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.30	
	Importance Rating		4.1

Emergency Procedure/Plan: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

SRO Only Justification: This question is SRO only as it requires determining off site notifications which is a SRO ONLY function.

Question: SRO #100

Given the following:

- The Unit is in OPCON 1 at 100% power.
- A loss of all drywell cooling occurs.
- Drywell pressure rises to 1.4 psig and operators manually scram the reactor.
- RPV level dropped to a low of -10 inches and was restored to +30 inches with feedwater.
- Drywell pressure continues to rise to 2.0 psig due solely to the loss of drywell cooling.
- All automatic action occur as designed.
- HPCI injected and was immediately secured since level control was established with feedwater.
- Drywell cooling was restored several minutes later, drywell pressure was reduced, all isolations were reset, and the plant was stabilized in OPCON 3.

Which one of the following is the EARLIEST event classification level that applies to the event described above?

A. 15 minute Emergency Notification to the States and Counties

B. 1 Emergency Notification to the NRC Operations Center

C. 4 hour NON-Emergency Notification to the NRC Operations Center

D. 8 hour NON-Emergency Notification to the NRC Operations Center

Proposed Answer: C

Explanation (Optional):

A: Incorrect – Although DW pressure exceeded 1.68 psig, it was due to loss of drywell cooling and not a LOCA condition. Therefore the Alert does not apply (RB2.L). no threshold exists for a 1 hour notification

B: Incorrect - no threshold exists for a 1 hour notification.

C: Correct - 4 hour non-emergency notification due to valid HPCI injection from a valid signal (1.68 psig) See RAL 11.3.1 and 11.3.2.

D: Incorrect. Primary containment isolation on 1.68 psig is an 8 hour notification and therefore is not the highest

Technical Reference(s): RALs 11.3.1 and 11.3.2 (Attach if not previously provided)
EAL RB2.L tech bases

Proposed References to be provided to applicants during examination: EALs & RALs

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1,5

Comments: