Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	WA #	211000, K1.03	
	Importance Rating	2.5	

(K&A Statement) Knowledge of the physical connections and/or cause- effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Plant air systems: Plant-Specific

Proposed Question: Common 1

The plant is operating at 100% power with the following conditions:

- Instrument air was inadvertently isolated to the Standby Liquid Control (SLC) system

  Reactor Building temperature is 84°F and steady

- No other evolutions are being performed on the SLC system

Which of the following Control Room Annunciators initially alert the Control Room of this isolation?

- 9-5 A-1, SLC SQUIB VLV CON-PINUITY LOSS
- 9-5 A-3, SLC TANK LVL HI/LO
- 9-5 A-4, SLC TANWSUCT LINE TEMP HI/LO

Annunciator(s):

A. 9-5-A-1 only

B. 9-5-A-3 only

C. 9-5-A-1 and 9-5-A-4

D. 9-5-A-3 and 9-5-A-4

Proposed Answer: B.

Explanation (Optional):

B. Correct – The SLC Level instruments require instrument air for operation. The level instrument fails low on a loss of Instrument Air.

A. Incorrect – There is no association between the instrument air system and the Squib Valves

C. Incorrect – There is no association between the instrument air system and the Squib Valves and although the low SLC tank level will trip the heater, But with Reactor Building temperatures at 84ºF SLC tank temps will not reach the alarm setpoint of 75ºF.

D. Incorrect – At SLC tank indicated level <17% the low SLC tank level will trip the heater. But with Reactor Building temperatures at 84ºF SLC tank temps will not reach the alarm setpoint of 75ºF.

Technical Reference(s): ARS 21003 (9-5) A-3, A-4
P&ID G-191171 (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: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000, K1.03	
	Importance Rating	2.7	

(K&A Statement) Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: Radiation monitoring systems

Proposed Question: Common 2

In accordance with ON-3153, Excessive Radiation Levels, which one of the following indications would alert an operator of a possible leak into the Reactor Building Closed Cooling Water (RBCCW) system?

A. A rise in suction pressure to the RBCCW Pumps and a concurrently lowering of the Fuel Pool water level.

B. Hi radiation alarm on the RBCCW Process Radiation Monitor and rising level in the RBCCW Surge Tank.

C. A rise in the radiation levels in the vicinity of RBCCW system piping or components and high temperatures on the operating CRD Pump.

D. Hi radiation alarm on the Service Water (SW) Process Radiation Monitor with concurrent indication of a RBCCW heat exchanger tube leak.

Proposed Answer: B.

Explanation (Optional):

B. Correct - Per ON-3153, If the RBCCW radiation monitor indicates a high radiation level:

a. Isolate the RCU system and check surge tank level indication to determine if the leak has been isolated,

b. If the RBCCW surge tank level continues to increase, shift to the standby fuel pool cooling heat exchanger and continue to monitor surge tank level.

A. There is no correlation between a lowering fuel pool level and rising RBCCW suction pressure.

C. Rising rad levels around RBCCW piping is a valid indication of a leak per OP-2182, P&L #1 Be aware of normal radiation levels in the vicinity of the system. Any appreciable rise in these radiation levels can indicate a possible leak into the RBCCW system. RBCCW supplies bearing and oil coolers on the CRD Pump, therefore, high CRD pump temperatures could indicate a lower RBCCW flow and not a leak into the RBCCW system.

D. A high radiation level in the SW system would indicate a leak into the SW system however by OP 2181 App B the SW system is maintained at a higher pressure than the RBCCW system, so leakage would be into the RBCCW system.

Technical Reference(s): ON-3153
9 - 3-E-6
OP-2182, P&L #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: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	WA #	263000, K2.01	
	Importance Rating	3.1	

(K&A Statement) **Knowledge of electrical power supplies to the following: Major D.C. loads**

Proposed Question: Common 3

While shutting down using Alternate Shutdown methods, Transfer Switch MTS-13-1 is placed to the EMERGENCY position. This aligns RCIC controls to be powered from:

A. 24 VDC ECCS Bus (Panel "A").

B. 48 VDC Microwave Telemetering Bus (DC-6A).

C. 125 VDC Emergency Bus (DC-1AS).

D. 125 VDC Control Power Bus (DC-5A).

Proposed Answer: C.

Explanation (Optional):

C. Correct - Two alternate shutdown (AS) battery systems are provided. DC-1AS provides control power to Vernon Tie Breaker 3V4, to the alternate shutdown RCIC panel (CP-82-I), (with the exception of V13-16) the RCIC system valves (via transfer switch MTS-13-I), and Appendix R Converter ES-24DC-3. By placing MTS-13-1 to EMERGENCY, all controls for RCIC can be operated locally if system operation from the Control Room is lost.

A. B. & D. Incorrect – The transfer switch shifts power to DC-1AS.

Technical Reference(s): OP-2145, Discussion section page 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

Question Source:	Bank #	604	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000, K2.02	
	Importance Rating	2.5	

(K&A Statement) Knowledge of electrical power supplies to the following: Motor operated valves

Proposed Question: Common **4**

The plant is Shutdown for a refueling outage and the following conditions exist:

- Shutdown cooling is aligned to the "A" Loop of RHR using "C" RHR Pump
- Primary Containment is not in effect
- Reactor Coolant temperature is 110°F

A plant transient results in the loss of DC-2.

Which of the following describes the effect on the RHR System?

A. RHR-17 & 18 remain open and the "C" RHR pump trips.

B. RHR-18 ONLY will fail shut on a loss of power, tripping the "C" RHR pump on interlock.

C. RHR-17 ONLY will fail shut on a loss of power, tripping the "C" RHR pump on interlock.

D. System remains aligned as is with no effect from loss of DC-2.

Proposed Answer: A.

Explanation (Optional):

A. Correct – IAW ON-3160, Loss of DC 2 and 3, on a loss of DC-2, If in S/D cooling, RHR pumps trip, RHR-17 de-energized.

B. Incorrect - RHR -18 will remain open however the "C" RHR pump logic will see the RHR-17 closed due to a loss of power to the sensing relay

C. Incorrect - RHR -17 will remain open however the "C" RHR pump logic will see the valve closed due to a loss of power to the sensing relay

D. Incorrect – RHR 17 and 18 both remain open.

Technical Reference(s): ON 3160, auto actions pg 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-205; K1.07.b, K2.02.b, K5.02.b, K6.02.b, K12.01 (As available)

Question Source: Bank # 6162

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001, K3.04	
	Importance Rating	3.9	

(K&A Statement) Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: Uninterruptible power supply

Proposed Question: Common 5

The UPS 1A/B Units are in their normal at power alignment.

Which one of the following describes how the UPS 1A/B Unit will respond when a loss of ALL normal AC power supplies occurs?

First the UPS Feeder Breaker __(1)__ then the AC drive motor loses power and the DC drive motor is powered from UPS battery. When NORMAL power is restored UPS ______(2)______

A. (1) trips
(2) will automatically return to AC power

B. (1) trips
(2) must be locally returned to AC power.

C. (1) does NOT trip
(2) must be locally returned to AC power.

D. (1) does NOT trip
(2) will automatically return to AC power.

Proposed Answer: D.

Explanation (Optional):

D. Correct – IAW OT 3122, the UPS Feeder Breaker does NOT trip and UPS 1A/B shift to DC drive when power is lost to 4 Kv Buses 3 and 4 (loss of all normal power supplies). IAW OP 2143, Sect H. The AC drive will automatically return upon restoration of AC power.

A. Incorrect - UPS Feeder Breaker does NOT trip.

B. Incorrect - UPS Feeder Breaker does NOT trip and the AC drive will automatically return upon restoration of AC power.

C. Incorrect - The AC drive will automatically return upon restoration of AC power.

Technical Reference(s): OT-3122, auto actions pg 11 (Attach if not previously provided)
OP 2143, Sect H pg 16

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-03-262 (As available)

Question Source: Bank # 1729
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	WA #	264000, K3.02	
	Importance Rating	3.9	

(K&A Statement) Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEUJET) will have on following: A.C. electrical distribution

Proposed Question: Common 6

After a Loss of Normal Power, the "B" Emergency Diesel Generator (EDG) receives an automatic start signal. The diesel fails to reach the required speed during the start attempt and the Start Failure Timer energizes the Shutdown Relay.

In accordance with OP 2126, Diesel Generators, which one of the following actions is required to restore power to the "B" Emergency Diesel Generator's respective 4KV Bus?

A. Immediately depress the local RESET pushbutton, place the control switch at the instrument panel in REMOTE, then attempt another start.

B. Immediately place the engine control in AT ENGINE, wait 100 seconds, reset the lockout relay, then direct an AO to start the DG locally.

C. Wait 100 seconds, reset the lockout relay, depress the local RESET pushbutton, then place the control switch at the instrument panel in REMOTE.

D. Immediately place the engine control in AT ENGINE, depress the local RESET pushbutton, wait 100 seconds, then place the control switch at the instrument panel in REMOTE.

Proposed Answer: D.

Explanation (Optional):

A. Incorrect – Must place the engine control in AT ENGINE and wait 100 seconds for shutdown relay to time out.

B. Incorrect – Must wait 100 seconds after resetting the shutdown relay, to allow the Stopping Relay to time out. It is not necessary to reset the lockout relay, it does not trip.

C. Incorrect – Must wait 100 seconds after resetting the shutdown relay, to allow the Stopping Relay to time out. It is not necessary to reset the lockout relay, it does not trip.

D. Correct – If a diesel generator start signal (such as an accident signal or LNP signal) occurs while the Stopping Relay is energized (timing out), the engine control must be placed in AT ENGINE, the shutdown relay must be reset as described above, then, following a period of 100 seconds, to allow the Stopping Relay to time out, the engine can be started locally, or automatically started by placing the Control switch at the instrument panel in REMOTE.

Technical Reference(s):	OP-2126, P & L [5], pg 7	(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: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

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

(K&A Statement) **Knowledge of STANDBY GAS TREATMENT SYSTEM design** feature(s) and/or **interlocks which provide for the following: Automatic system initiation**

Proposed Question: Common 7

The plant has been shutdown following a small steam leak inside the drywell. The following conditions exist:

- Drywell pressure: 3.1 psig, rising slowly
- Drywell temperature: 167°F, rising slowly
- Reactor water level: 142 inches and stable
- Reactor Building ventilation exhaust radiation: 1.2 mr/hr, steady

Assuming all plant equipment operated as designed, which one of the following is the present status of Secondary Containment atmospheric control?

Secondary Containment is at a ..

A. Positive pressure, being exhausted through a filtered and monitored path.

B. Negative pressure, being exhausted through a filtered and monitored path.

C. Positive pressure, being exhausted through an unfiltered and unmonitored path.

D. Negative pressure, being exhausted through an unfiltered and unmonitored path.

Proposed Answer: B.

Explanation (Optional):

A. Incorrect - pressure is negative
B. Correct – Hi Drywell pressure initiated a secondary containment isolation and SBGT start.
C. Incorrect - pressure is negative exhausting through SBGT
D. Incorrect - exhausting through SBGT

Technical Reference(s): OP 2117, Rev. 17, page 5 (Attach if not previously

LOT-00-261, Rev. 32, pages 12-13 provided)

EOP 1

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-261, K4.01 (As available)

Question Source: Bank # 3563

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000, K4.05	
	Importance Rating	3.2	

(K&A Statement) Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(~)and/or interlocks which provide for the following: Prevention of water hammer

Proposed Question: Common 8

Following a plant trip that involved a lengthy High Pressure Coolant Injection run, the Residual Heat Removal (RHR) system was placed in Torus Cooling. The following conditions exist:

- RHR Pump " A running
- Minimum Flow Valve, RHR-16A is CLOSED
- Torus Cooling Valve, RHR-34A is THROTTLED
- Torus Spray/Cooling Valve, RHR-39A is OPEN

RHR Pump "A" motor breaker trips on a breaker fault.

IAW OP 2124, which one of the following describes a concern with starting the "C" RHR Pump at this time and how may this concern be averted?

Starting the "C" RHR Pump may result in...

A. water hammer damage because the piping will drain. Immediately close RHR-34A and RHR-39A.

B. tripping the "C" pump on high current. Verify RHR-16A is closed and close RHR-34A / RHR-39A.

C. overheating the "C" pump because RHR-16A will remain closed and RHR-39A will close on the pump trip. Open RHR-16A.

D. dead heading the "C" pump because RHR-16A is closed and both RHR 34A / RHR-39A close on the pump trip. Open RHR-16A.

Proposed Answer: A.

Explanation (Optional):

A. Correct - During Torus spray or Torus cooling operation, if RHR pump flow is lost, the discharge valves, either RHR-39A(B) or 34A(B) and 38A(B) should be immediately closed to prevent drain down from the high points in the RHR system to the Torus. This can result in void formation at the high points and upon RHR pump restart, water hammer could result.

B. C. and D. Incorrect – The "C" pump min flow will open and provide flow/cooling. RHR 34A AND RHR-39A do NOT close on the pump trip. The problem is the water hammer from starting the "C" pump.

Technical Reference(s): OP-2124, P&L #10 (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: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000, K5.01	
	Importance Rating	2.7	

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM : Fuel thermal time constant

Proposed Question: Common 9

Which one of the following RPS trip signals uses a "Thermal Time Constant" to ensure fuel cladding integrity?

A. High Neutron Flux

B. High Reactor Pressure

C. Turbine Control Valve Closure

D. Main Steam Isolation Valve Closure

Proposed Answer: A.

Explanation (Optional):
A. Correct – From T.S. 2.1 FUEL CLADDING INTEGRITY
A. Trip Settings
1. Neutron Flux Trip Settings
a. APRM Flux Scram Allowable Value (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1912 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux.

During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses are performed to demonstrate that the APRM flux scram over the range of settings from a maximum of 120% to the minimum flow biased setting provide protection from the fuel safety limit for all abnormal operational transients including those that may result in a thermal hydraulic instability.

B. C. and D. Although these scrams affect Reactor power the "Time Constant" is only used for the Neutron Flux scrams.

Technical Reference(s): T.S. 2.1 (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: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____________

10 CFR Part 55 Content: 55.41 X

55.43 ____________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	**1**	
	K/A #	**218000, K5.01**	
	Importance Rating	**3.8**	

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM : ADS logic operation

Proposed Question: Common **10**

The plant has experienced a small break LOCA and station blackout. The following conditions exist at **0900** hrs:

- Reactor water level is + **82.5"** lowering slowly
- Drywell pressure is **5** psig, increasing slowly
- Reactor pressure is **750** psig, lowering slowly
- ALL ECCS pumps are in Pull-To-Lock

Given these conditions how will Automatic Depressurization System (ADS) logic and Safety Relief Valves (SRVs) respond?

A. ADS logic initiates at **0902**
ALL SRVs open

B. ADS logic initiates at **0908**
ALL SRVs open

C. ADS logic will time out but will not initiate
NO SRVs open

D. ADS logic will not initiate because the timers will not start.
NO SRVs open

Proposed Answer: C.

Explanation (Optional):

C. Correct – With the combination of high DW Pressure and Lo-Lo RPV Water Level the ADS timer will begin timing and time-out at 0902 hrs. However, with all the ECCS pumps in PTL the ECCS logic will NOT see any ECCS pumps available and therefore will NOT initiate and the SRVs will not open.

A. Incorrect – The timer will time out but with the ECCS pumps in PTL the ECCS logic will NOT see ECCS pumps available and will NOT initiate

B. Incorrect Response - ADS logic is NOT satisfied and will NOT initiate after 2 minutes because the ECCS pumps are in PTL.

D. Incorrect – The timers will start and time out but the SRVs will not open

Technical Reference(s): OP 2122 Discussion pgs 3 & 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-218, K3.02 (As available)

Question Source: Bank # 3474 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

12-15-08, underlined the starting time.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	230000, K4.02	
Importance Rating	3.1	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001, K6.01	
	Importance Rating	3.4	

(K&A Statement) Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM : A.C. power

Proposed Question: Common 11

The plant and its systems are normally aligned for full power operations. A LOCA has occurred with a loss of normal power (LNP) and the following conditions exist:

- Emergency Diesel Generator (EDG) "B" fails to start
- Reactor pressure is 250 psig and is slowly lowering
- Drywell pressure is 14 psig and is slowly rising
- NONE of the Bus 3 pumps are in the Pull-To-Lock position
- The CRO is able to energize 4KV Bus 3 from the Vernon Tie.

Which one of the following is the ECCS pump response upon re-energizing Bus 3?

A. A and B RHR Pumps start 5 seconds after power is restored.

B. A and B RHR Pumps start immediately after power is restored.

C. B Core Spray Pump starts 10 seconds after power is restored.

D. B Core Spray Pump starts immediately after power is restored.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - A RHR Pump starts immediately.
B. Incorrect - B RHR Pump starts AFTER a 5 second time delay
C. Correct Response
D. Incorrect - B Core Spray Pump starts AFTER a 10 second time delay.

Technical Reference(s): Table 7.4.3 UFSAR
Table 14.6.4.A
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # 1206 Lot more

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002, K6.04	
	Importance Rating	3.0	

(K&A Statement) Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES : D.C. power: Plant-Specific

Proposed Question: Common 12

Which one of the following describes the effect of a loss of 125 VDC-2C on the SRVs?

A. All SRVs become inoperable in both manual and ADS mode.

B. SRVs “71A” and “71C” become inoperable in both manual and ADS mode.

C. SRVs "71B” and "71D become inoperable in both manual and ADS mode.

D. All SRVs remain operable in both manual and ADS mode.

Proposed Answer: D.

Explanation (Optional):

D. Correct - If DC-2C is lost the relief valve power will swap to DC-1C, and the 4 SRVs will open.
A. Incorrect – All SRVs remain operable in both manual and ADS mode.
C. Incorrect – All SRVs remain operable in both manual and ADS mode.
D. Incorrect - All SRVs remain operable in both manual and ADS mode.

Technical Reference(s): ON 3159, Auto Actions pg 3
LOT-00-239, pg 24
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____________ Lot more

Modified Bank # ____________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43

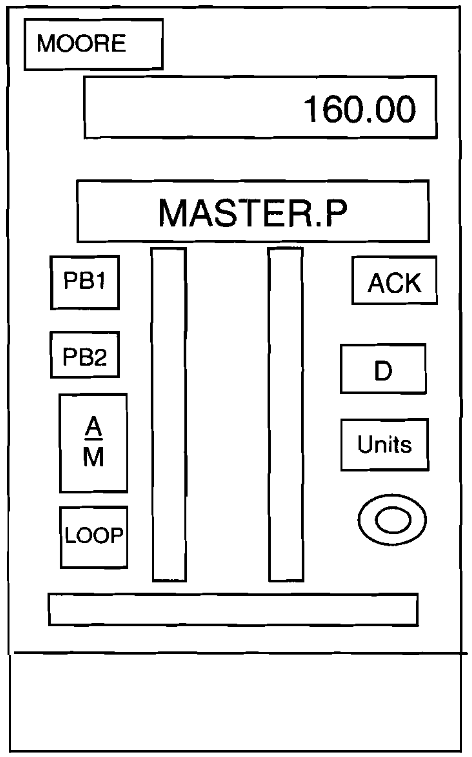
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002, A1.01	
	Importance Rating	3.8	

(K&A Statement) Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor water level

Proposed Question: Common 13

The plant is operating at 100% power, with the Reactor Vessel Level Master Controller in AUTO when PB1 is inadvertently depressed. The RO then immediately depresses the PB2 pushbutton.



Which one of the following is the RPV water level response?

Water level...

A. remains at 160 inches

B. lowers to 155 inches.

C. lowers to 133 inches.

D. rises above 177 inches

Proposed Answer: B.

Explanation (Optional):

B. Correct - Master controller push button PB1 is configured to set the master controller setpoint to 133 inches when depressed. Master controller pushbutton PB2 is configured to set the master controller setpoint to 155 inches when depressed. This feature may be used to input a setpoint reduction (set down) should the automatic setpoint set down fail to actuate during the loss of a feedwater pump or condensate pump recirculation system runback. Additionally, should PB1 be accidentally depressed, PB2 may be used to rapidly restore level setpoint to 155 inches.

A. Incorrect – Depressing the buttons changes the controller's setpoint to 133 then 155.

C. Since PB2 is immediately depressed the level will not reach 133.

D. Incorrect level will not rise

Technical Reference(s): FIGURE 1 of OP-2172 and text on page 6 of the Discussion Section. (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____________ Lot more

Modified Bank # ____________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

55.43 ____

Comments:

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

(K&A Statement) Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including: Control rod block status

Proposed Question: Common 14

The plant is subcritical withdrawing control rods for a reactor startup. The following conditions exist:

- Reactor power is 75 counts per second (CPS) in the source range
- Intermediate Range Monitors (IRMs) are downscale on Range 1.
- CH A SRM SELECT Switch on CRP 9-5 Benchboard is illuminated

The CRO depresses and holds the DRIVE OUT pushbutton.

Which of the following describes the system response?

The A SRM detector will...

A. NOT withdraw due to the current power level.

B. NOT withdraw while the IRMs are on range 1.

C. fully withdraw, however a Rod Block will occur.

D. partially withdraw until the Full In indication is lost then the CH A SRM SELECT light will extinguish.

Proposed Answer: C.

Explanation (Optional):

C. Correct - Any SRM channel will generate a rod block if a detector is not fully inserted and the level indicator for that channel drops below 100 cps.

A. Incorrect – The SRM will withdraw
B. Incorrect - The SRM will withdraw
D. Incorrect - The SRM will withdraw

Technical Reference(s): OP 2130 Page 3 (Attach if not previously

provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #	292	Lot more
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003, A2.04	
	Importance Rating	3.7	

(K&A Statement) Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Up scale or down scale trips

Proposed Question: Common 15

A reactor startup is in progress with following conditions:

- Power has risen from 20140 on IRM Range 3 to 40/125 on IRM Range 4 in 20 seconds.
- Annunciator 9-5 N-3, IRM HI alarms for IRM "D" prior to selecting Range 4.
- A half scram occurred on RPS Channel B
- NO rod motion is in progress (Assume a sustained period).

Based on the above conditions the reactor period is about __1__ . In accordance with OP 0105, the CRO should __(2)__.

A. (1) 58 seconds
(2) reset the half scram ONLY WHEN directed by the SRO.

B. (1) 29 seconds
(2) IMMEDIATELY insert control rods to make the reactor sub-critical.

C. (1) 29 seconds
(2) reset the half scram ONLY WHEN directed by the SRO.

D. (1) 58 seconds
(2) IMMEDIATELY insert control rods to make the reactor sub-critical.

Proposed Answer: B

Explanation (Optional):

B. Correct - IAW OP-0105 - Note 1 from VYOPF 0105-03: Stable Period = Time for power to double x 1.445. power doubled (125 scale is expanded 0-40 scale) over 20 sec therefore 20 x 1.445 = 28.9 sec.
If the sustained period becomes shorter than 30 seconds; use the EMERGENCY IN switch to turn the period and insert control rods until the reactor is subcritical. CRO must await direction to reset the half scram.

A. Incorrect – Period is 29 seconds, reactor must be made subcritical
C. Incorrect - Reactor must be made subcritical
D. Incorrect - Period is 29 seconds

Technical Reference(s): OP 0105 Page 21, (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Comments:

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

(K&A Statement) 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

Proposed Question: Common 16

A loss of MCC-8B resulted in a loss of the Vital MG Set when the DC Drive motor failed to power the Vital AC Generator. In addition the auto transfer to the Vital Bus Alternate source failed.

IN ACCORDANCE WITH ON 3168, Loss of Vital AC,

(1) What Control Room actions are taken to attempt to recover Vital AC? AND
(2) Which one of the following actions are required if Vital AC CANNOT be restored?

A. (1) Place the Vital Bus Manual Transfer Switch on CRP 9-8 to the ALT position, holding momentarily, and releasing.
(2) Initiate/verify a reactor SCRAM and enter OT 3100.

B. (1) Position the UPS FDR TRIP, 10A-S36A keylock switch on CRP 9-32 to BLOCK and close the UPS FDR breaker on Switchgear Bus 9.
(2) Initiate/verify a reactor SCRAM and enter OT 3100.

C. (1) Place the Vital Bus Manual Transfer Switch on CRP 9-8 to the ALT position, holding momentarily, and releasing.
(2) Verify RPV pressure control shifts to the MPR and place the Master Feedwater Controller in Manual and control RPV water level.

D. (1) Position the UPS FDR TRIP, 10A-S36A keylock switch on CRP 9-32 to BLOCK and close the UPS FDR breaker on Switchgear Bus 9.
(2) Verify RPV pressure control shifts to the MPR and place the Master Feedwater Controller in Manual and control RPV water level.

Proposed Answer: A.

Explanation (Optional):

A. Correct – If Vital AC cannot be restored the plant must be scrammed. If an automatic transfer did not occur, then attempt to re-energize the Vital bus by placing the Vital bus manual transfer switch on CRP 9-8 to the ALT position, holding momentarily, and releasing.

B. Incorrect - UPS FDR TRIP, 10A-S36A keylock switch on CRP 9-32 is for an ECCS LIPS.

C. Incorrect - If Vital AC cannot be restored the plant must be scrammed.

D. Incorrect - If Vital AC cannot be restored the plant must be scrammed and UPS FDR TRIP, 10A-S36A keylock switch on CRP 9-32 is for an ECCS UPS.

Technical Reference(s): ON 3168, Steps A and B on pg 5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____________ Lot more

Modified Bank # ____________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____________

10 CFR Part 55 Content: 55.41 X

55.43 ____________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	WA #	223002, A3.01	
	Importance Rating	3.4	

(K&A Statement) Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: System indicating lights and alarms

Proposed Question: Common 17

The six Primary Containment Isolation System (PCIS) red lights on the vertical section of CRP 9-5, lower right hand corner, are illuminated.

Which one of the following is indicated by these indicating lights?

A. A PCIS Group 1, 2 and 3 isolation signal is active and the isolations can NOT be reset.

B. Isolation signals associated with PCIS Group 1, 2 and 3 isolations are clear and the valves are still in the isolated condition.

C. The valve position indications necessary to satisfy PCIS Group 1, 2 and 3 Inadvertent Opening Protect Logic (IOPL) are satisfied.

D. The control switches required to satisfy the Inadvertent Opening Protect Logic (IOPL), for PCIS Group 1, 2 and 3, are in the closed position.

Proposed Answer: D.

Explanation (Optional):

D. Correct - The lights indicate the valves affected by the IOPL logic have had their control switches placed in the "closed" position. The correct switch position will satisfy the logic requirement even if the valves fail to close.

A. and B. Incorrect - The red lights indicate the IOPL logic has been satisfied.

C. Incorrect -The lights indicate the control switches for the valves are in the "closed" position the IOPL logic is satisfied by switch position and is independent of valve position.

Technical Reference(s): CWDs 1100-1123 (Attach if not previously provided)
LOT-01-223, page 89

OP-2115, Sect G. pg 45.

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-01-223, K16 (As available)

Question Source: Bank # 1158 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	WA #	215005, A3.05	
	Importance Rating	3.3	

(K&A Statement) Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Flow converter/comparator alarms

Proposed Question: Common 18

A plant startup is in progress with the following conditions:

- The Recirc flow input signal to the APRMs is 25%
- As Recirc flow is raised, the "B" Flow ConverterIComparator output remains at 25%
- Actual recirculation loop flows respond as designed.

What will be the effect on plant operation if recirculation flow continues to be raised to 32%?

A. Full scram on flow-biased APRM Hi-Hi flux.

B. Half scram on flow biased APRM Hi-Hi flux.

C. Control rod block on Flow Converter/Comparator "inop" signal.

D. Control rod block on Flow ConverterIComparator "unbalance" signal.

Proposed Answer: D.

Explanation (Optional):

D. Correct - An increase in actual flow will increase the output of the "A" Flow Converter/Comparator. The lowest point a flow biased rod block could happen would be 42%. When flow increases to approximately 32%, the comparators will have a 7% mismatch. The mismatch causes an out of limits trip which causes a rod block. The inop is only caused by a loss of power.

A. and B. Incorrect - The Flow Biased Scram signal for the "B" RPS Channel APRMs (APRMs B, D, F) will remain unchanged as 70.5% based on the failure of the "B" Flow Converter/Comparator. Flow will be increased to 32% on the "A" Flow Converter/Comparator before actual reactor power is increased to 70.5%. The 7% difference between the 2 Converter/Comparator units will cause a comparator unbalance ROD BLOCK before actual power reaches the artificially low trip setpoints for APRMs B, D, and F.

C. Incorrect - The inop flow converter/comparator rod block would occur if one of the comparators had lost power

Technical Reference(s): LOT-05-215, ARS 5-M-5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-05-215, CRO 2d, 5 (As available)

Question Source: Bank # 3892 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000, A4.05	
	Importance Rating	4.1	

(K&A Statement) Ability to manually operate and/or monitor in the control room: Reactor water level

Proposed Question: Common 19

The plant has scrammed following a loss of Feedwater. The following conditions exist:

- High Pressure Coolant Injection (HPCI) is inoperable
- Reactor Core Isolation Cooling (RCIC) is injecting
- RPV water level is 150 inches and increasing

If no further operator action is taken, which one of the following will occur?

The RCIC System will

A. trip. The trip must be manually reset to allow for RCIC to automatically inject.

B. isolate. The isolation will automatically reset when reactor water level lowers to 127 inches and RCIC will inject.

C. isolate. The isolation will automatically reset when reactor water level lowers to 82.5 inches and RCIC will inject.

D. trip. The trip will automatically reset when Steam Supply Valve (RCIC-131) closes. RCIC will automatically inject when reactor water level lowers to 82.5 inches.

Proposed Answer: D.

Explanation (Optional):

A. Incorrect - RCIC will inject if level falls back to 82.5 inches
B. Incorrect - RCIC does not isolate and isolations do not reset automatically
C. Incorrect - RCIC does not isolate and isolations do not reset automatically
D. Correct Response

Technical Reference(s): OP 2121, page 4 (Attach if not previously

provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-217, K5.06, K8 (As available)

Question Source: Bank # 6402 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	WA #	206000, A4.02	
	Importance Rating	4.0	

(K&A Statement) Ability to manually operate and/or monitor in the control room: Flow controller: BWR-2,3,4

Proposed Question: Common 20

A LOCA has occurred with the following plant conditions:

- RPV water level 90 inches lowering slowly
- RPV pressure 900 psig lowering slowly
- DW pressure 3.0 psig rising slowly

HPCI was in normal standby before initiation and initiated with the following parameters:

- HPCI pump flow 0 gpm
- HPCI turbine speed 2000 rpm
- HPCI pump discharge pressure 550 psig
- HPCI minimum flow valve OPEN

IAW OP 2120, which one of the following is the likely cause of this condition and what actions are required to raise RPV water level?

A. The Controller has failed low, place the Overspeed Test Selector Switch in TEST and raise HPCI turbine speed with the Test Control knob.

B. The Ramp Generator did not transfer to the HPCI controller; raise the auto flow controller setpoint tape to raise HPCI discharge pressure.

C. The controller has lost power, place the Overspeed Test Selector Switch in TEST and raise HPCI turbine speed with the Test Control knob.

D. The Ramp Generator has failed, place the flow controller in manual control and raise the turbine speed and discharge pressure with the manual control potentiometer.

Proposed Answer: D.

Explanation (Optional):

D. Correct - IAW OP Discussion Section - Upon system initiation and turbine start, the ramp generator Ramp function is initiated by the mechanical movement of the turbine stop valve as sufficient oil pressure is developed by the Auxiliary Oil Pump. Valve movement to the fully open position actuates a valve limit switch that initiates the Ramp function. Throughout the transient time period (12 seconds) of the Ramp function from idle to rated speed setpoint, the HPCI System flow controller calls for maximum pump flow and turbine speed until pump flow reaches the flow controller setpoint. The transition from the Ramp function to flow controller control is automatic and is accomplished by the Low Signal Selector in the ramp generator signal converter box. IAW OP, App. B. If HPCI PUMP FLOW CONTROLLER FIC 23-108 automatic feature fails or significant flow oscillations occur:

a) Place HPCI PUMP FLOW CONTROLLER FIC 23-108 in MANUAL.
b) Control injection flow using the MANUAL knob.

A. and C. Incorrect – There is no procedural guidance to use the overspeed test controller to control HPCI speed or discharge pressure.
B. Incorrect- Raising the setpoint tape would have no effect.

Technical Reference(s):	OP 2120, Discussion pg 3 App B, pg 1 of 3	(Attach if not previously provided)
Proposed references to be provided to applicants during examination:		None
Learning Objective:		(As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000, 2.4.4	
	Importance Rating	4.5	

(K&A Statement) Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (Instrument Air)

Proposed Question: Common 21

While operating at full power, the following annunciators are received:

- CRP 6-D-1 Inst Air Receiver HDR Press LO
- CRP 5-C-8 Scram Pilot Air Hdr Press Hi/Lo
- CRP 5-E-2 FW VLV Lockup Signal/Air Fail

Instrument air header pressure is 80 psig and lowering slowly.

IAW ON 3146, Low Instrument/Scram Air Header Pressure, and the Annunciator Response Procedures you are to confirm:

A. Both Lead and Lag Compressors running and SA-PCV-1 closed.

B. Only Lead Compressors running and SA-PCV-1 open.

C. Both Lead and Lag Compressors running and SA-PCV-1 open.

D. Only Lead Compressors running and SA-PCV-1 closed.

Proposed Answer: A.

Explanation (Optional):

A. Correct Response - Low instrument air 90 psig, lead compressors are normally running, lag compressors start at 95 psig, SA-PCV-1 starts shut at 85 psig in the instrument air header and is full shut at 80 psig. The continuing lower pressure causes SA-PCV-1 to fully shut.

B. Incorrect - The procedure assumes lead compressors running and does not require them checked, SA-PCV-1 should be closed

C. Incorrect - SA-PCV-1 should be closed

D. Incorrect - The procedure assumes lead compressors running and does not require them checked.

Technical Reference(s):	ON 3146, pg 2 ARS-6-D-1		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Learning Objective:	LOT-00-279,		(As available)
Question Source:	Bank #	5717	Lot more
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001, 2.4.30	
	Importance Rating	2.7	

(K&A Statement) Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator. (AC Distribution).

Proposed Question: Common 22

In which of the following situations may you reclose the breaker for the " A Core Spray Pump without prior permission from Electrical Maintenance?

In accordance with OP 2142, 4 KV Electrical System the breaker may be reclosed:

A. Any time the Pump tripped following a loss of DC Control Power

B. Only during an extreme emergency if the Pump tripped on Timed Overcurrent.

C. Any time the Pump tripped on Timed Overcurrent following a 30 minute cooldown.

D. Only during an extreme emergency if the Pump tripped following a loss of DC Control Power.

Proposed Answer: B.

Explanation (Optional):

A. and D. Incorrect - Under no circumstances will the breakers be operated to closed if DC control power for the breaker is unavailable.
B. Correct - operation of a 4 KV breaker will only be performed in an extreme emergency and then only with permission of the Shift Manager.
C. Incorrect - operation of a 4 KV breaker will only be performed in an extreme emergency and then only with permission of the Shift Manager.

Technical Reference(s): OP 2142, sect. J pg 25 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	LOT-01-*262*, K18		(As available)
Question Source:	Bank #	1805	Lot more
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000, A2.14	
	Importance Rating	3.8	

(K&A Statement) Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Initiating logic failure

Proposed Question: Common 23

Prior to a valid Low Pressure Coolant Injection (LPCI) system initiation a loss of the "B" side 24 VDC ECCS system occurred.

How will this failure affect the LPCI system response?

A. The A loop LPCI injection valves will not automatically open and must be manually opened from the Control Room.

B. The B loop LPCI injection valves will not automatically open and must be manually opened from the Control Room.

C. The A loop LPCI injection valves will not automatically or manually open from the Control Room; the other LPCI loop must be used for injection.

D. The B loop LPCI injection valves will not automatically or manually open from the Control Room; the other LPCI loop must be used for injection.

Proposed Answer: C.

Explanation (Optional):

A. Incorrect – The valves cannot be manually opened from the control room.
B. Incorrect – The valves cannot be manually opened from the control room.
C. Correct – Upon loss of the B side ECCS 24 VDC ECCS system, the A loop LPCI injection valves will not manually open from the Control Room because the power that provides the logic to open reactor low pressure relays will not make up.
D. Incorrect – The A loop valve won't open.

Technical Reference(s): OP 2124, Discussion pg 6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source: Bank # ______________ Lot more

Modified Bank # ______________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000, K4.03	
	Importance Rating	3.8	

(K&A Statement) Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: ADS logic control

Proposed Question: Common 24

During a LOCA, a valid ADS signal exists and the ADS system has initiated and the Safety Relief Valves have opened. With the initiation signal still present both Initiation Signal Timer Reset Pushbuttons are depressed.

Which ONE of the following describes the automatic response of the ADS system?

All ADS valves will:

A. remain open

B. close and remain closed indefinitely

C. close and remain closed for 8 minutes then reopen.

D. close and remain closed for 2 minutes then reopen.

Proposed Answer: D.

Explanation (Optional):

D. Correct – IAW OP-2122, ADS Auto Logic A(B) pushbuttons on CRP 9-3 have the following functions. The pushbuttons are normally in the AUTO position and will allow automatic initiation of the ADS system. The buttons can be depressed to reset the 120 sec. initiation timers. The timers will restart when the buttons are released if the initiating conditions are still present. If the initiating signal is clear the timers will not restart.

A. Incorrect - the valves will close
B. Incorrect – the valves will re-open after 120 seconds.
C. Incorrect - the valves will re-open after 120 seconds.

Technical Reference(s): OP-2122, Discussion pg 5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:		None	
Learning Objective:	LOT-00-218, K 9.02	(As available)	
Question Source:	Bank #	3253	Lot more
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004, A3.02	
	Importance Rating	3.4	

(K&A Statement) Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: Annunciator and alarm signals

Proposed Question: Common 25

The Mode Switch is in START UP/HOT STANDBY.

The following Annunciators and indications are indicated on the Control Room 9-5 Panel:

- 5-D-3, ROD WITHDRW BLOCK
- 5-P-3, SRM HI/INOP

  All IRMs are mid-scale on Range 2.
- SRMs are being withdrawn from the core in accordance with OP 0105.
- Reactor period has dropped to 600 seconds.

SRMs (cps)

A	B	C	D
2.0×10^3	2.0×10^5	3.0×10^3	5.0×10^5

The CRS has directed the next control rod be withdrawn to continue the startup.

Given the above, which one of the following is necessary to withdraw control rods?

A. Place the IRMs on Range 3

B. Withdraw All the SRM detectors.

C. Insert SRM Detectors A and C

D. Withdraw SRM Detectors B and D

Proposed Answer: D.

Explanation (Optional):

Clean up explanation...........

D. Correct – IAW OP-2130 and ARS for 9-5 any SRM channel will generate a rod block if one of the following conditions exist:

1. Any SRM downscale (3 cps) or any SRM detector not fully inserted and the level indicator for that channel drops below 100 cps. (With all IRM range switches at position 3 or higher, these rod blocks are bypassed automatically.)
2. SRM count greater than the high setpoint unless the IRM range switches are on Range 8 or above.
3. SRM Inoperable

Annunciator 5-P-3 Hi Setpoint is 2.2 x 10^5 CPS. In addition guidance in OP 0105 (Phase 1A, Step 33) directs SRMs be withdrawn from the core once IRM overlap has been confirmed. Detectors are to withdrawn two at a time to maintain SRM count rates between 10^3 to 10^5 cps. All SRM detectors are not fully withdrawn from the core until all IRMs are approximately mid-scale on Range 4.

A. Incorrect - Going to range 3 will not bypass an SRM hi condition.

B. Incorrect – Withdrawing all the SRMs is not necessary and if all the SRMs were withdrawn SRM " B would fall below 100 cps and cause a rod block.

C. Incorrect – Bypassing SRM " B will have no effect

Technical Reference(s): OP-2130, Discussion pg 3; ARS for 9-P-3; OP 0105, Phase 1A, Step 33 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____________ Lot more

Modified Bank # ____________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	**1**	
	WA #	**215005, K2.02**	
	Importance Rating	2.6	

(K&A Statement) Knowledge of electrical power supplies to the following: APRM channels

Proposed Question: Common 26

The reactor is operating at 100% power.

RPS Bus " A was placed on it's alternate power source. All required operator actions were completed for realigning APRM power supplies.

Subsequently, a voltage transient causes RPS MG Set " B Output Breaker to trip.

Which one of the following describes the power supplies for the APRMs?

A. APRMs A, C, and E are powered from MCC-8B.
APRMs B, D, and F are powered from Vital AC.

B. APRMs A, C, and E are powered from Vital AC.
APRMs B, D, and F are powered from MCC-8B.

C. APRMs A, C, and E are powered from Vital AC.
APRMs B, D, and F are powered from Instrument AC.

D. APRMs A, C, and E are powered from MCC-8B.
APRMs B, D, and F are powered from Instrument AC.

Proposed Answer: D.

Explanation (Optional):

A. Incorrect - A,C,E APRMs are supplied from MCC-8A; RPS Bus B is de-energized

B. Incorrect - A,C,E APRMs were shifted back to RPS Bus "A" power; A,C,E APRMs are supplied from MCC-8A, B,D,F APRMs are powered from Instrument AC.

C. Incorrect - A,C,E APRMs were shifted back to RPS Bus "A" power; A,C,E APRMs are supplied from MCC-8A

D. Correct Response – IAW OP-2134, Trip system A receives power from MG-3-1A and trip system B receives power from MG-3-1B. Alternate power is available to either RPS bus from 480V MCC 8B. When de-energizing either RPS bus, the APRMs will automatically transfer to a loss of normal power (LNP) alternate power source. APRMs associated with RPS Bus A will transfer to the Vital AC source and RPS Bus B will transfer to an instrument AC source.

Technical Reference(s): OP 2134, Discussion pg 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-05-215 (As available)

Question Source: Bank # ______ Lot more

Modified Bank # 581 (Note changes or attach parent)

New ______

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006, K1.04	
	Importance Rating	3.1	

(K&A Statement) Knowledge of the physical connections and/or **cause- effect relationships between ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) and the following: Steam** flow/reactor **power:** P-Spec(Not-BWR6)

Proposed Question: Common 27

During a reactor shutdown at what point will the Rod Worth Minimizer (RWM) first begin to enforce Control Rod Blocks IAW OP 2450?

A. Steam Flow OR Feed Flow < 21%

B. Steam Flow AND Feed Flow < 21%

C. Steam Flow OR Feed Flow <25%

D. Steam Flow AND Feed Flow <25%

Proposed Answer: A.

Explanation (Optional):

A. Correct - When either Steam Flow or Feed Flow drops below the Low Power Set Point the RWM will start imposing select, insert and withdraw blocks. The LPSP is 21% in accordance with OP 2450 (Discussion Section and Definition 6).

B. Incorrect - When <u>either</u> Steam Flow or Feed Flow drops below the Low Power Set Point the RWM will start imposing insert and withdraw blocks.

C. Incorrect - APRM power readings do not affect the RWM. (They affect the RBM.)

D. Incorrect - Steam flow at <25% corresponds to the Low Power Alarm Set Point in accordance with OP 2450 (Discussion Section and Definition 5). The RWM will generate alarms and indicate rod insert & withdraw errors but will not enforce Rod Blocks. APRM power readings do not affect the RWM. (They affect the RBM.)

Technical Reference(s): OP 2450, Discussion, pages 3 & 4 (Attach if not previously provided)
OP 2450, Definitions, #5 & 6, pg 7

Proposed references to be provided to applicants during examination: None

Learning Objective: ________________ (As available)

Question Source: Bank # 3255 Lot more

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 X

55.43 ________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	WA #	223001, K2.08	
	Importance Rating	2.7	

(K&A Statement) Knowledge of electrical power supplies to the following: Containment cooling air handling units: Plant. Specific

Proposed Question: Common 28

A Loss of Normal Power (LNP) has occurred and the following conditions exist:

- The "A" Emergency Diesel Generator failed to start.
- All Drywell Reactor Recirculation Units (RRUs) are selected to RUN.
- All other equipment functioned normally
- No operator actions have been taken

Which one of the following conditions will the CRO observe when verifying the drywell RRU status on CRP 9-25?

A. RRUs 1A/B running

B. RRUs 2A/B running

C. RRUs 1A/B and 3A/B running

D. RRUs 2A/B and 4A/B running

Proposed Answer: A.

Explanation (Optional):

A. Correct - If an LNP condition still exists, RRU 1 and 3 will restart immediately; RRU 2 and 4 may be restarted by momentarily pushing the DRYWELL CLG AND CTRL A/C BLOCKING RESET pushbutton. However in this case EDG " A has tripped so no power is available to RRU-3. RRU-2 will not restart until reset. RRU-3&4 do not have power
B. Incorrect - RRU-2 will not restart until reset
C. Incorrect - RRU-3 does not have power
D. Incorrect - RRU-2 will not restart until reset. RRU-4 does not have power

Technical Reference(s): OP 2115, Discussion pg 5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-288, K10 (As available)

Question Source: Bank # 1169 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	256000, K3.04	
	Importance Rating	3.6	

(K&A Statement) Knowledge of the effect that a loss or malfunction of the REACTOR CONDENSATE SYSTEM will have on following: Reactor Feedwater System

Proposed Question: Common 29

Plant conditions are as follows:

- Reactor Power = 87%
- Feed Water Level Control is in Automatic, 3-element control with level controlling at 160"
- The U S lights are NOT LIT on the individual recirc controllers

"C" Condensate Pump trips on thermal overload, resulting in a trip of " B Reactor Feedwater Pump.

IAW OT 3113, Reactor Low Level, the correct response is to:

A. Lower power by reducing recirculation flow 10% then restart the " B Reactor Feed Pump.

B. Lower recirculation flow at 10% per minute to clear the Feed Pump low suction annunciator.

C. Manually runback both Recirculation Pumps and verify the Master Feedwater Level controller sets down to 155".

D. Verify both Recirculation Pumps runback to minimum speed and manually set down the Master Feedwater Controller to 155".

Proposed Answer: C.

Explanation (Optional):

C. Correct – IAW OT 3113, If reactor power is greater than 1593 megawatts thermal and any condensate or feed pump trips: Manually runback Reactor Recirc Pumps to a 40% demand signal and verify the master vessel level controller setdown to 155. Per OP 2110, whenever PB-1 is depressed on the individual recirc pump controllers, a level setdown to 155 is auto generated.
87% power is >1593 MWth. There is no low flow, level condition or Feed Pump Breaker tripping to automatically initiate a Recirc Runback, so the runback must be done manually.

A. and B. incorrect – Recirc pumps must be runback to 40% demand
D. Incorrect – Because it was a Condensate Pump that tripped, an automatic runback will not occur because of the lower power level and actions must be manually taken.

Technical Reference(s): OT 3113, Immediate Actions pg 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-602, K3.05, K4.10, CRO 2 (As available)

Question Source: Bank # 6209 Lot more
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	230000, K4.02	
	Importance Rating	3.1	

(K&A Statement) Knowledge of RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE design feature(s) and/or interlocks which provide for the following: Redundancy

Proposed Question: Common 30

The plant was at power with the breaker for RHR Pump "D" tagged out for maintenance when the following occur:

- A reactor coolant leak resulted in a Drywell pressure of 6 psig
- RHR Pump " A was started
- RHR-39A, TORUS SPRAY/CLG was opened, for Torus Spray

No other actions are taken.

Following these actions, a loss of power of Bus 4 occurs (DG fails to supply the bus and NO buses are cross-tied).

What is the current status of the RHR loops?

One RHR Pump is available for LPCI injection on ___(1)____ RHR Loop(s)

Torus sprays are ____(2)____ .

A. (1) " A
(2) NOT available on either RHR loop

B. (1) "B"
(2) NOT available on either RHR loop

C. (1) " A and " B
(2) available ONLY on 'A' RHR loop

D. (1) " A and "B"
(2) available ONLY on 'B' RHR loop

Proposed Answer: A.

Explanation (Optional):

A. Correct - RHR Loop A consists of pumps A & C, RHR Loop B consists of pumps B & D. Bus 3 supplies RHR Pumps C & D, Bus 4 supplies RHR Pumps A & B. With the present plant conditions only the "C" RHR Pump has power which is being supplied from Bus 3 since the " D RHR Pump was previously tagged out of service. With RHR-39A, TORUS SPRAY/CLG previously opened for Torus Spray it is necessary to open RHR-38A to initiate Torus Spray on the " A RHR Loop. However, with Bus 4 being de-energized and no buses cross-tied there is no power to the "A" Loop Valves so sprays are not available on " A RHR Loop. Although there is power to the "B" RHR Loop valves from Bus 8 there is no RHR Pump available in the loop to supply torus spray.

B. C. & D. Incorrect – Both pumps in loop B are inoperable (B pump de-energized D pump tagged-out). With the loss of power to Bus 4 and Bus 9, RHR containment cooling valves are deenergized on RHR Loop A. Cannot remotely open RHR-38A, TORUS SPRAY

Technical Reference(s): P&IDs 191172, 191299
OP 2143, App A Page 42 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290002, K5.07	
	Importance Rating	3.9	

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS . Safety limits

Proposed Question: Common 31

Which OWE of the following result in a violation of a TS Safety Limit?

A. MCPR = 1.08 when Reactor Pressure = 810 psia AND core flow = 12% of rated while in Two Loop operation.

B. MCPR = 1.10 when Reactor Pressure = 810 psia AND core flow = 11% of rated in Single Loop operation.

C. The reactor shutdown with irradiated fuel in the vessel and reactor vessel water level 1 0 above the top of the enriched fuel when it is seated in the core.

D. The reactor shutdown with enriched fuel in the vessel and reactor vessel water level 1 2 above the top of the irradiated fuel when it is seated in the core.

Proposed Answer: C.

Explanation (Optional):

A. Incorrect: This meets the TS Safety Limit 1.1.A.
B. Incorrect: This meets the TS Safety Limit 1.1.A.

C. Correct response - this is a violation of TS Safety Limit 1.1.D. which states that whenever the reactor shutdown with irradiated fuel in the vessel the water level shall not be less than 12 inches above the top of the enriched fuel when it is seated in the core.

D. Incorrect – Not <12", also, the limit concerns level above enriched and not irradiated fuel. 1 2 above the top of the irradiated fuel corresponds to an indicated level of 1 8 (i.e. TAF + 12" where at VY TAF = +6". Therefore TAF + 1 2 = 18.)

Technical Reference(s): T.S. 1.1.D (Attach if not previously provided)

Proposed references to be provided to applicants during examination:			None
Learning Objective:			(As available)
Question Source:	Bank #		Lot more
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

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

(K&A Statement) Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR MANUAL CONTROL SYSTEM : Select matrix power

Proposed Question: Common **32**

During operation at **100%** power, which one of the following is the effect of placing the Rod Select Power Switch in OFF?

A. The Rod Worth Minimizer indicates a Select Error.

B. The Rod Position Indication System becomes inoperative.

C. A control rod CANNOT be selected for display on the four rod display.

D. Control rod positions CANNOT be determined on both the full core and four rod displays.

Proposed Answer: C.

Explanation (Optional):

A. Incorrect – the RWM has no effect at this power level.

B. Incorrect – The RPIS does NOT become inoperable, but no control rods can be selected.

C. Correct – Rod select power must be turned on to select a control rod. If a rod cannot be selected, it cannot be displayed on the four rod display.

D. Incorrect – Control rod positions are still displayed

Technical Reference(s): OP **2147,** App B Page 18 (Attach if not previously provided)
ON **3174,** Page **2**

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Lot more

	Modified Bank # ________	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam No	
Question Cognitive Level:	Memory or Fundamental Knowledge ________	
	Comprehension or Analysis X	
10 CFR Part 55 Content:	55.41 X	
	55.43 ________	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	233000, A1.07	
	Importance Rating	2.7	

(K&A Statement) Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: System temperature

Proposed Question: Common 33

The plant has just resumed power operations following a refueling outage. The following conditions exist:

- One train of Normal Fuel Pool Cooling (NFPC) is in service aligned to the Spent Fuel Pool with maximum cooling water flow through the in-service heat exchanger.
- River temperatures are unusually high resulting in high Service Water and Reactor Building Closed Cooling Water (RBCCW) temperatures.
- All trains of NFPC and Standby Fuel Pool Cooling (SFPC) are available.
- Spent Fuel Pool temperature is 139°F and rising.

In accordance with OP 2179, Standby Fuel Pool Cooling which one of the following actions is required?

A. Place the OTHER train of NFPC in service.

B. Maintain ONE train of NFPC in service and place ONE train of SFPC in service.

C. Place ONE train of SFPC in service and secure NFPC.

D. Maintain ONE train of NFPC in service and place BOTH trains of SFPC in service.

Proposed Answer: C.

Explanation (Optional):

C. Correct – From OP-2179, P&L 8, Prior to the Spent Fuel Pool water temperature exceeding 140°F, secure the Normal FPC and place the SFPC System in service. SFPC is cooled by SW, FPC is cooled by RBCCW.

A. Incorrect – SFPC needs to be placed in service and so NFPC must be secured. It is physically impossible for NFPC and SFPC to be in service on the Spent Fuel Pool at the same time.

B. Incorrect –SFPC needs to be placed in service and Normal FPC has to be secured. It is physically impossible for NFPC and SFPC to be in service on the Spent Fuel Pool at the same time.

D. Incorrect - SFPC need to be placed in service. It is physically impossible for NFPC and SFPC to be in service on the Spent Fuel Pool at the same time.

Technical Reference(s): OP-2179, P&L 8, pg 6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____ (As available)

Question Source: Bank # ____ Lot more

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 X

55.43 ____

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001, A2.08	
	Importance Rating	3.1	

(K&A Statement) 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

Proposed Question: Common 34

The reactor was operating at 90% power when the "B" Recirculation Pump tripped.

In response reactor power has been lowered to 43% by inserting control rods.

In accordance with OT 3118, Recirculation Pump Trip, which one of the following subsequent actions is required for speed control on the " A Recirculation Pump?

A. Lower the pump speed to between 65% and 70%.

B. Lower the pump speed to between 98% and 100%.

C. Raise pump speed as necessary to insure Core flow is >34%.

D. Raise pump speed as necessary to insure Core flow is >45%.

Proposed Answer: A.

Explanation (Optional):

A. Correct – IAW OT 3118 (similar statements in OT-3117 and OP-2110) IF running pump is operating >70% rated speed, THEN REDUCE speed to 65 to 70% rated speed.
B. Incorrect – The initial portion of the procedure directs the operator to insure the recirc pump is not exceeding 98 - 100% speed; this is prior to inserting the control rods.
C. and D. Pump speed should not be raised because the trip of the B pump occurred at 90% power the " A pump is already over 70% speed.

Technical Reference(s): OT 3118/3117 and OP-2110 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	______________		(As available)
Question Source:	Bank #	______	Lot more
	Modified Bank #	1651	(Note changes or attach parent)
	New	______	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge		______
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	X	
	55.43	______	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001, A3.10	
	Importance Rating	3.4	

(K&A Statement) Ability to monitor automatic operations of the REACTOR FEEDWATER SYSTEM including: Pump trips

Proposed Question: Common 35

During power ascension at ~40% feedwater flow, a second feedwater pump is started.

Shortly thereafter the discharge valve for the first running reactor feedwater pump is inadvertently shut and the associated minimum flow valve fails to open.

What are the effects on the first running feedwater pump?

A. The pump will continue to run until cavitation causes the pump to trip on high vibration after a six-second time delay.

B. Cavitation in the pump will require operator action in 180 seconds to prevent pump damage.

C. The pump will auto trip when suction flow drops below 300,000 lbm/hr after a thirteen second time delay.

D. The discharge valve will automatically re-open when discharge flow drops below 300,000 lbm/hr.

Proposed Answer: C.

Explanation (Optional):

A. Incorrect - There is no high vibration trip. Six second TD is to allow the pump to develop sufficient flow (>300,000 lbm/hr).

B. Incorrect -Operation near or at min flow conditions are only allowed for <90 seconds. This pump is running with no flow path. Per Precaution 3 of OP 2172: "At operating speed, the energy expended by a pump running at shutoff head conditions will result in excess vibration of the pump casing and internals. This will result in pump seal damage in approximately 20 sec. when running at speed or slightly longer during a pump start. Trip the pump if flow isolates or minimum flow valve does not open when pump is started. " Therefore the pump should be manually tripped as soon as pump isolation is diagnosed. Allowing the pump to run for up to 180 seconds would result in pump seal damage.

C. Correct Response

D. Incorrect -The minimum flow valve opens on a low flow, not the discharge valve

Technical Reference(s): OP 2172 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # 1648 (requal) Lot more

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	234000, A4.02	
	Importance Rating	3.4	

(K&A Statement) Ability to manually operate and/or **monitor in the control room: Control rod drive system (Fuel Handling Equipment)**

Proposed Question: Common 36

The reactor mode switch is in the REFUEL position, and the refueling platform (bridge) is over the reactor pressure vessel. For which one of following reasons would a control rod block be generated?

A. The Grapple is NOT in the FULL UP position.

B. All rods are full-In, except for a selected rod at position 48.

C. The Frame Mounted Hoist is unloaded and NOT FULL UP.

D. The Grapple is loaded with a fuel assembly and in the FULL UP position.

Proposed Answer: D.

Explanation (Optional):

A. Incorrect - The Grapple not full up will not cause a block unless it is loaded.

B. Incorrect –The fuel grapple will not cause a block unless it is loaded.

C. Incorrect – For a rod block the control room must select a second rod for movement with any other rod withdrawn from the fully inserted position.

D. Correct – IAW OP-1100, Control rod withdrawal is prevented with the mode switch in REFUEL and Refuel platform over the core with fuel loaded on any refueling platform hoist.

Technical Reference(s): OP 1100, pg 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source:	Bank #		Lot more
	Modified Bank #	3408	(Note changes or attach parent)
	New		

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290001, 2.2.37	
	Importance Rating	3.6	

(K&A Statement) Equipment Control: Ability to determine operability and / or availability of safety related equipment (Secondary Containment)

Proposed Question: Common 37

With the plant operating at 100% power, a leak has developed on the "A" Reactor Water Cleanup Pump (RWCU) discharge line. The leak CANNOT be isolated and temperatures on the Reactor Building 280 ft level are approaching the Maximum Safe Value for that area.

Which one of the following actions is required and why is this action required?

A. Immediately scram the reactor because EOP related equipment may fail.

B. Commence a plant shutdown because Secondary Containment may be lost.

C. Commence a plant shutdown because of limited access to the Reactor Building.

D. Immediately scram the reactor because overheating of RWCU resin will produce increasing area radiation levels.

Proposed Answer: A.

Explanation (Optional):

A. Correct – IAW EOP-4, With a primary system discharging into Reactor Building scram the plant before any parameter reaches its Max Safe Value.
If parameters in EOP- 4, Secondary Containment 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

B. Incorrect – A scram is required
C. Incorrect - A scram is required
D. Incorrect – The location of the RWCU leak will not result in the overheating of demineralizer resin.

Technical Reference(s): EOP-4, Step SC-3 (Attach if not previously provided)
EOP Study Guide

Proposed references to be provided to applicants during examination:	None
Learning Objective:	(As available)

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

Question History:	Last NRC Exam	No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	272000, 2.4.47	
	Importance Rating	4.2	

(K&A Statement) Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (Radiation Monitoring System)

Proposed Question: Common 38

The plant is operating at 100% and SJAE radiation levels are increasing.

No alarms are currently in but SJAE OFF GAS RAD recorder RR-17-152 indicates higher than normal levels. Activity levels are fluctuating between 5000 µCi/sec and 5500 µCi/sec.

Which one of the following provides operational guidance for the current conditions?

A. ON 3152, MSL and Off Gas High Radiation

B. ON 3153, Excessive Radiation Levels

C. EOP-4, Radioactivity Release Control

D. Core Operating Limits Report

Proposed Answer: A.

Explanation (Optional):

A. Correct – ON 3152, MSL and Off Gas High Radiation

B. Incorrect – ON 3153 provides guidance for high area radiation levels NOT high activity levels in the Off Gas System

C. Incorrect – Air Ejector Hi-Hi (3-G-1) or Hi (3-G-2) Annunciators are NOT in alarm so there currently is no entry condition for EOP-4, Radiation Release.

D. Incorrect - The Core Operating Limits Report contains guidance regarding operation constraints placed on the reactor core such as thermal limits. It does not provide any constraints or guidance pertaining to high Off Gas activity.

Technical Reference(s): ON-3152, Sect. 4, pg 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005, AK1.01	
	Importance Rating	4.0	

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP : Pressure effects on reactor power

Proposed Question: Common 39

Which one of the following is TRUE concerning the pressure transient that results from a design basis Turbine Stop Valve Closure Scram?

A. Minimum Critical Power Ratio is NOT exceeded ONLY if the Turbine Bypass Valves open.

B. Minimum Critical Power Ratio is NOT exceeded REGARDLESS of the Turbine Bypass Valves position.

C. Technical Specifications Pressure Safety Limit is NOT exceeded ONLY if the Turbine Bypass Valves open.

D. Technical Specifications Pressure Safety Limit is NOT exceeded REGARDLESS of the control rod positions following the scram.

Proposed Answer: B.

Explanation (Optional):

B. Correct – IAW TS, The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed.

A. Incorrect - MCPR remains above the fuel cladding integrity safety limit even when the Turbine Bypass Valves fail to open.

C. Incorrect – Power (MCPR) is the bases, safety limit is not exceeded regardless of bypass valve positions.

D. Incorrect - Power (MCPR) is the bases. The transient analysis assumes control rods insert to reduce power and limit rise in reactor pressure and reduction in MCPR.

Technical Reference(s): TS Bases 2.1.E, pg 17 (Attach if not previously

provided)

Proposed references to be provided to applicants during examination:		None
Learning Objective:		(As available)
Question Source:	Bank #	Lot more
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam No	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 X	
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028, EK1.O1	
	Importance Rating	3.5	

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

Proposed Question: Common Proposed Substitute Question 40

A LOCA has occurred and the following conditions exist:

- All control rods have inserted
- Recirc Pumps have tripped
- RPV Emergency Depressurization has just been commenced
- Drywell pressure is 5.3 psig
- RPV pressure is 700 psig and lowering
- Reference Leg Temperatures are reading 300°F

Which of the following is the most accurate RPV level indication for the CURRENT conditions?

A. + 7 8 on ECCS Level Indicator LI-2-3-72A on CRP 9-5

B. + 7 9 on RPS Level Indicator LI-2-3-57B on CRP 9-5

C. - *5* on ERFIS point WIDEM071, Compensated Rx Level Wide 70

D. - 45" on ERFIS point SHDAB046, Compensated Rx Level Shroud 73A

Proposed Answer: C.

Explanation (Optional):
C. Correct - DP 0166 (page 7) states: "During rapid reactor de-pressurization transients, the narrow range (NR) RPV level instruments may indicate high or off-scale. Because they are compensated, both Shroud and Wide Range level indications should be monitored. During rapid depressurization Wide Range is preferred above 350 psig, while Shroud Level is preferred below 350 psig." DP 0166 then specifies that ERFIS point WIDEM071, COMPENSATED RX LEVEL WIDE 70 be used for this purpose

A. Incorrect – Based on the Minimum Indicated Level for RPS/ECCS/Transients curve. The minimum indicated level for the ECCS instruments with the Reference Leg at 300°F is greater than 81. Since the indicated ECCS level is 78" this level indicator cannot be used.

B. Incorrect - Based on the Minimum Indicated Level for RPS/ECCS/Transients curve. The minimum indicated level for the RPS instruments with the Reference Leg at 300°F is greater than 81. Since the indicated RPS level is 7 9 , this level indicator cannot be used.

D. Incorrect – The Shroud Level Indicators are Cold Calibrated and therefore should not be used until reactor pressure is < 350 psig. DP 0166 specifies that during rapid depressurization the Compensated Wide Range ERFIS point is preferred above 350 psig,

Technical Reference(s):	EOP-1 and EOP-3 Minimum Indicated Level - RPS/ECCS/Transients EOP Study Guide Sect 13, pg 47 DP 0166 (Rev. 19) Pg. 7	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: Graph of Minimum Indicated Level - RPS/ECCS/Transients from EOP-1

Learning Objective: LOT-00-622, 8 (As available)

Question Source:	Bank #	3222	Lot more
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

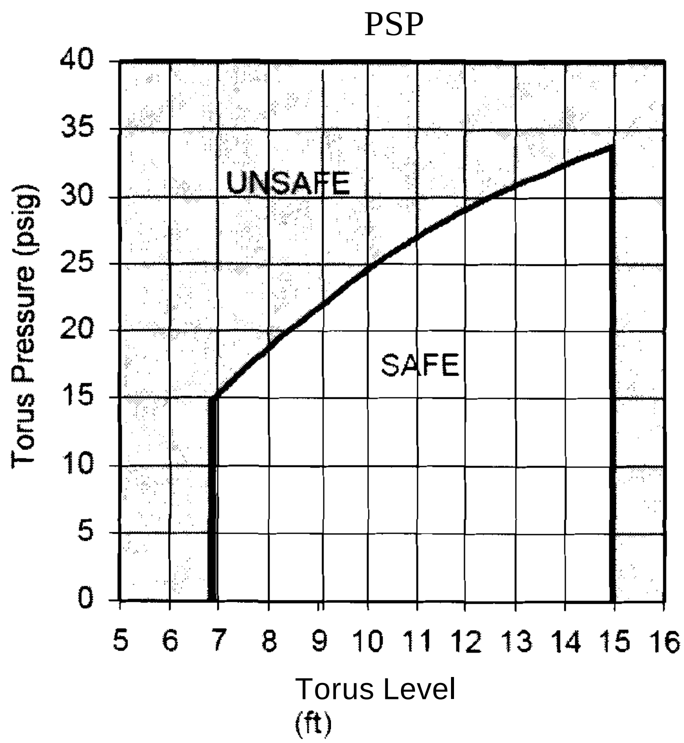
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030, EK1.O1	
	Importance Rating	3.8	

(K&A Statement) Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Steam condensation

Proposed Question: Common 41



Refer to the PSP figure above and determine:

1) Which listed event challenges the LOW TORUS LEVEL portion of the graph? AND
2) What are the consequences of violating the limit?

Event 1: Suppression Pool level uncovering of Torus Downcomers.
Event 2: Suppression Pool level uncovering Safety Relief Valve T-Quenchers

A. 1) Event 1
2) Violating the limit allows steam to flow directly into the torus free air space.

B. 1) Event 2
2) Violating the limit allows steam to flow directly into the torus free air space.

C. 1) Event 1
2) Violating the limit allows gases from the torus free air space to flow into the drywell.

D. 1) Event 2
2) Violating the limit allows gases from the torus free air space to flow into the drywell.

Proposed Answer: A.

Explanation (Optional):

A. Correct - 7 ft (the elevation of the downcomer openings). When the downcomer vent openings are not adequately submerged, any steam discharged From the RPV into the drywell may not condense in the torus before torus pressure reaches unacceptable levels.

B. Incorrect – The SRVs may be used down to a torus level of 5.5 feet.

C. Incorrect – The bases for the limit is to insure condensation of steam.

D. Incorrect - The SRVs may be used down to a torus level of 5.5 feet and The bases for the limit is to insure condensation of steam.

Technical Reference(s): EOP-Study Guide, Sect 13, pg 39 of 54
Sect 5, pg 8 of 19
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

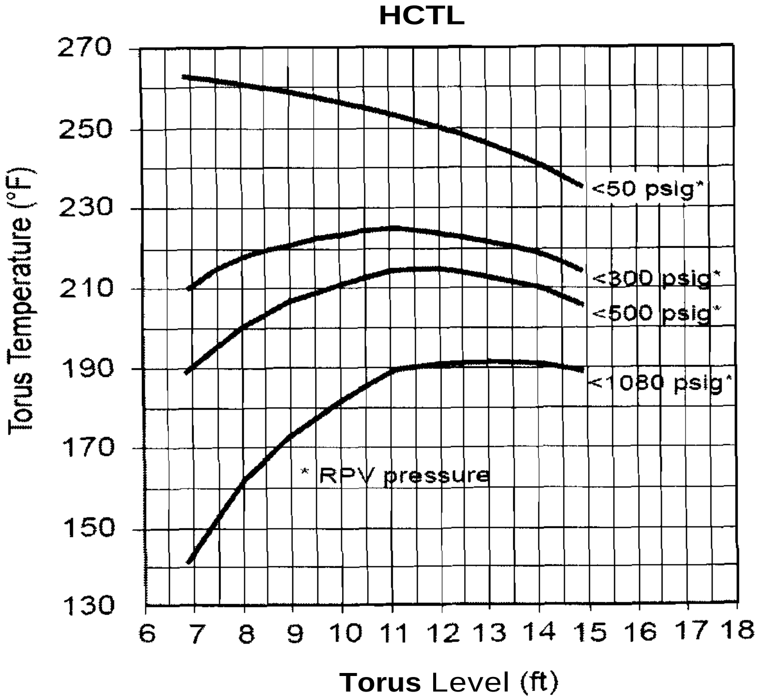
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026, EK2.06	
	Importance Rating	3.5	

(K&A Statement) Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Suppression pool level

Proposed Question: Common 42

Using the Heat Capacity Temperature Limit diagram below determine which one of the following sets of parameters VIOLATES the HCTL.



	Torus Level	Torus Temperature	RPV Pressure
A.	9	180	1050
B.	10	205	480
C.	11	220	250
D.	12	245	47

Proposed Answer: A.

Explanation (Optional):

A. Correct – 180 degrees exceeds the 1080 psig curve for 9' torus level.
B. Incorrect – 205 degrees is below the 500 psig curve for 10' torus level.
C. incorrect – 220 degrees is below the 300 psig curve for 11' torus level.
D. Incorrect - 245 degrees is below the 50 psig curve for 12' torus level.

Technical Reference(s): EOP Study Guide, Sect. 6, pg 41 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019, AK2.04	
	Importance Rating	2.8	

(K&A Statement) Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Reactor water cleanup

Proposed Question: Common 43

Given the following plant conditions:

- A reactor startup and heatup is in progress
- Reactor water level is being controlled via RWCU letdown to the main condenser
- The RWCU Demin Bypass Valve (CU-74) is open
- Main condenser vacuum has been established using the Mechanical Vacuum Pump
- The Control Rod Drive System is in service.

A loss of instrument air ONLY to the RWCU System will have which of the following effects under these conditions?

A. The running RWCU Pump will trip on low flow.

B. A loss of RWCU letdown flow will result in rising reactor water level.

C. The running RWCU Pump will trip on overcurrent due to pump runout.

D. RWCU will isolate on high Non-Regenerative Heat Exchanger outlet temperature.

Proposed Answer: B.

Explanation (Optional):

A. Incorrect - The running RWCU Pump will NOT trip on low flow because flow will still flow through the filterldemineralizers and back to the reactor.
B. Correct - RCU Dump Flow Regulator will fail closed on loss of air. Use of cleanup system to lower or control Rx water level will not be available. With CRD in service water level will increase.
C. Incorrect - The running RWCU Pump will NOT trip on high flow because flow will be controlled by flow through the filterldernineralizers and back to the reactor.
D. Incorrect – This may occur if the RCU Dump Flow Regulator failed open.

Technical Reference(s): ON 3146, Low Instrument/Scram Air Header Pressure, Rev 20 (Appendix A, page 5) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # 3876 Lot more

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000, AK2.04	
	Importance Rating	2.5	

(K&A Statement) Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Breakers, relays, and disconnects

Proposed Question: Common 44

The plant is operating at 100% power when the following alarm occurs:

7-D-2, MAIN XFMR T1 HI PRESS TRIP

7-D-3, MAIN XFMR T1 TEMP HI

Zone 40 Fire Alarm on Control Room fire panel (CP-115-3).

Which one of the following is the status of the Main Transformer?

The Main Transformer deluge sprays are ___(1)____ and the transformer ____(2)____

A. (1) on
(2) energized

B. (1) on
(2) de-energized

C. (1) off
(2) energized

D. (1) off
(2) de-energized

Proposed Answer: B.

Explanation (Optional):

B. Correct – IAW OP 2186, At 280°F a continuous linear Protectowire loop activates Zone 40 ALARM, CP-115-3, Control Room fire panel and deluge valve, FP-DV-202, actuates water flows to the Main Transformer spray system. IAW 7-D-2, Check open 81-IT, 1T, T1 MOD, exciter field breaker and turbine tripped. Check 86GP, 86GB lockout relays de-energized.
A. Incorrect –Transformer is de-energized.
B Incorrect – Sprays are on and transformer is de-energized.
C. Incorrect – Sprays are on.

Technical Reference(s): ARS 21005, 7-D-2 (Attach if not previously provided)
OP 2186, Sect. L, pg 106

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023, AK3.03	
	Importance Rating	3.3	

(K&A Statement) AK3.03 - Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: Ventilation isolation.

Proposed Question: Common 45

Which one of the following is the reason for the Reactor Building Ventilation (RBHVAC) isolation during a Design Bases Refueling Accident IAW the UFSAR?

A. Insures containment integrity and an elevated release.

B. Insures a filtered release, regardless of Secondary Containment.

C. Blocks the intake and exhaust of unfiltered air from the Reactor Building.

D. Blocks the RBHVAC spent fuel pool ductwork to minimize the pickup of radioactive particulates.

Proposed Answer: A.

Explanation (Optional):

A. Correct - A design bases refueling accident involves dropping a fuel assembly the UFSAR analysis states (regarding the radiological release) "Credit is taken for containment, collection, and elevated release for 100% of the activity escaping the fuel pool. No credit is needed (or taken) for SGTS filters".

B. Incorrect – No credit is needed (or taken) for SGTS filters, however secondary containment is maintained.
C. Incorrect – No credit is needed (or taken) for SGTS filters
D. Incorrect – This ductwork is not isolated and this is not the basis for the isolation.

Technical Reference(s): UFSAR Section 14.6.4.4, pg 1558. (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	____________	(As available)	
Question Source:	Bank #	____________	Lot more
	Modified Bank #	____________	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	____________	
10 CFR Part 55 Content:	55.41	X	
	55.43	____________	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018, AK3.04	
	Importance Rating	3.3	

(K&A Statement) Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Starting standby pump

Proposed Question: Common 46

The running RBCCW pump has tripped and the standby pump did NOT start. If the standby pump cannot be started which of the following componentslsystems must be MANUALLY shutdown/tripped in accordance with ON 3147, Loss of RBCCW?

A. The Reactor Water Cleanup System ONLY

B. The Recirculation Pumps, the operating CRD Pump ONLY

C. The operating CRD Pump and the Reactor Water Cleanup System ONLY

D. The Recirculation Pumps, the operating CRD Pump AND the Reactor Water Cleanup System.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - All three componentslsystems must be shutdown.
B. Incorrect - All three components/systems must be shutdown.
C. Incorrect – All three componentslsystems must be shutdown.
D. RWCU must be shutdownlverified isolated and shutdown. The CRD pumps supplying seal water and hi temperature effects on resin (demineralizers are protected by an automatic isolation at 140°F). Recirc pump seals will be damaged and the recirc pumps are required to be shutdown within 2 minutes after RBCCW is lost. CRD pump bearings and reduction gear are cooled by RBCCW and must be manually shutdown to prevent damage.

Technical Reference(s): ON 3147, Rev 11, pgs 2 & 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	____________	(As available)	
Question Source:	Bank #	5677	Lot more
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016, AK3.03	
	Importance Rating	3.5	

(K&A Statement) Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Disabling control room controls

Proposed Question: Common 47

In accordance with OP-3126, Shutdown Using Alternate Shutdown Methods, which one of the following occurs when the 4KV/480V Switchgear transfer switches are placed in EMER and what is the reason for this transfer?

A. MOST automatic functions and system interlocks are defeated because a fire may cause inadvertent automatic actions.

B. ONLY breaker control functions from the Control Room are defeated because Appendix R requires that components must be controlled only from one area.

C. ALL automatic functions are available but system interlocks are defeated because operators have less indications and controls available to them during alternate shutdown.

D. ALL breaker control functions are defeated but most automatic functions and system interlocks are available because Appendix R requires the majority of interlocks and automatic functions to remain functional.

Proposed Answer: A.

Explanation (Optional):

A. Correct - OP-3126, P & L #1 Placing the RCIC, RHR, 4KV/480V Switchgear, or the DG transfer switches in EMER removes control function from the Control Room and defeats most automatic functions and system interlocks. OP-3126, PG 51, Local Operation of 4KV Bus 4 Breaker, states Place the alternate shutdown transfer switch to the emergency position and then place the emergency breaker control switch to the desired position. If not shifted to alternate various automatic actions may or may not occur as intended or they may occur inadvertently.
B. Incorrect –Most automatic functions and system interlocks are defeated and If not shifted to alternate various automatic actions may or may not occur as intended or they may occur inadvertently.
C. Incorrect – Most automatic functions and system interlocks are defeated and If not shifted to alternate various automatic actions may or may not occur as intended or they may occur inadvertently.
D. Incorrect – Most automatic functions and system interlocks are defeated and If not shifted to alternate various automatic actions may or may not occur as intended or they may occur inadvertently.

Technical Reference(s): OP-3126, Rev 18 pg 6, 8 and 51 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000, AA1.05	
	Importance Rating	3.9	

(K&A Statement) Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Engineered safety features.

Proposed Question: Common 48

Following loss of the Auto Transformer, voltage on Emergency Buses 3 and 4 CANNOT be restored to above 3700 VOLTS.

In accordance with ON-3155, Loss off the Auto Transformer, which one of the following methods is used to restore voltage on Buses 3 and 4?

Note: Assume all actions are completed for one bus before completing the same actions for the second bus.

A. START and PARALLEL each diesel generator to its 4 KV bus.

B. OPEN Breaker 12 (22) to de-energize Bus 1 (2) initiating a full LNP sequence for each bus.

C. START and PARALLEL the associated diesel generator to its 4 KV bus. Then separate the bus from the electrical system by opening Breaker 3T1 (4T2).

D. START the associated diesel generator, then transfer Bus 3 (4) to the diesel generator by de-energizing the bus by opening Breaker 3T1 (4T2).

Proposed Answer: D.

Explanation (Optional):

A. Incorrect -The procedure warns not to attempt manually connecting DG-1-1B to Bus 3 because of the low voltage condition.
B. Incorrect -The sequence/steps are not in accordance with procedural direction and would cause a temporary loss of power and challenge the electrical system.
C. Incorrect - The procedure warns not to attempt manually connecting DG-1-1B to Bus 3 because of the low voltage condition.
D. Correct Response - Correct per the procedure

Technical Reference(s): ON 3155, Rev 10, pg 6 Step (Attach if not previously

12.0 provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-601, CRO 3 (As available)

Question Source: Bank # 205 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

9-9-08 – EB – Revised question for readability and plausibility.

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

(K&A Statement) Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : D.C. electrical distribution systems

Proposed Question: Common 49

A 4KV breaker that is normally operated from the Control Room loses DC control power.

What breaker responses, if any, are functional?

A. Auto trips on an LNP
Can be closed from the Control Room

B. Auto trips on an LNP
CANNOT be closed from the Control Room

C. Does NOT auto trip on an LNP
Can be opened from the Control Room

D. Does NOT auto trip on an LNP
CANNOT be opened from the Control Room

Proposed Answer: D.

Explanation (Optional):

D. Correct – The breaker will lose DC Control power and therefore not be operable from the control room. Additionally remote automatic functions like the load shed signal will not open the breaker.

A. B. & C. Incorrect the loss DC Control power prevents remote operation and remote automatic trips.

Technical Reference(s): CWD Drawings 4KV breakers
General Physics Fundamentals
LOT-00-121
ON-3159, Mentions specific breakers but not Load Shed breakers. See Note pg 5

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-263 (As available)

Question Source: Bank # 5650 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006, AA1.06	
	Importance Rating	3.5	

(K&A Statement) Ability to operate and/or monitor the following as they apply to SCRAM : CRD hydraulic system

Proposed Question: Common 50

A hydraulic ATWS has occurred due to blockage of both scram discharge volumes. OE 3107 Appendix G, Manual Insertion of Individual Control Rods has been directed. The following conditions exist:

- Scram air header pressure is 0 psig.
- The scram valves for all 89 control rods are open.
- Only one CRD Pump is available
- CRD-56, CRD Charging Water Header Supply, is stuck in the OPEN position.

Under these conditions, control rods will:

A. NOT move inward because the CRD flow control valve is fully open.

B. MOVE inward even though both scram discharge volumes are blocked.

C. MOVE inward since the CRD flow control valve will not be affected by the CRD 56 failure.

D. NOT move inward because drive water differential pressure will not be sufficient to move the control rods.

Proposed Answer: D.

Explanation (Optional):

A. Incorrect - the rods will NOT move inward because the CRD flow control valve is fully shut.
B. Incorrect - the CRD flow control valve is fully shut.
C. Incorrect – The blockage of the scram discharge volume only affects the discharge of scram water and not CRD drive flow which returns to the reactor.
D. Correct Response

Technical Reference(s): P&ID 191170 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-610, CRO-2, 3 (As available)

Question Source: Bank # 408 requal Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021, AA2.05	
	Importance Rating	3.4	

(K&A Statement) Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : Reactor vessel metal temperature

Proposed Question: Common 51

A loss of shutdown cooling has occurred and forced circulation through the core cannot be established. Which one of the following is required to monitor reactor vessel metal temperature?

A. Restore reactor water level to > 160" and monitor reactor vessel skin temperatures at least every 15 minutes on PLC 2-166, RPV/SV/RV screen and TR-2-3-90 (CRP 9-3).

B. Restore reactor water level to > 185" and monitor reactor vessel skin temperatures at least every 30 minutes on PLC 2-166, RPV/SV/RV screen and TR-2-3-90 (CRP 9-3).

C. Restore reactor water level to > 160" and monitor reactor vessel bottom drain temperatures least every 15 minutes on Recirculation loop temperature recorder, TR-2-165.

D. Restore reactor water level to > 185" and monitor reactor vessel bottom drain temperatures least every 30 minutes on Recirculation loop temperature recorder, TR-2-165.

Proposed Answer: B.

Explanation (Optional):

B. Correct – IAW ON-3156, Sect. B.9,
If forced circulation through the core cannot be established proceed as follows:

- Periodically monitor reactor vessel skin temperatures on PLC 2-166, RPV/SV/RV screen, (CRP 9-21) AND TR-2-3-90 (CRP 9-3) at least once per 30 minutes or more frequently dependent on the time to boil estimate.

A. Incorrect - 1 6 0 is approximately the height reactor water level is raised to for normal SDC.

C. Incorrect – 1 6 0 is approximately the height reactor water level is raised to for normal SDC and with no flow in the RPV the recirc loop temperatures would be meaningless.

D. Incorrect – There is no guidance to use bottom head temperatures and with no flow in the RPV the recirc loop temperatures would be meaningless.

Technical Reference(s): ON-3156, Sect. B.9, Rev 6, pg 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037, EA2.04	
	Importance Rating	4.0	

(K&A Statement) Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Suppression pool temperature

Proposed Question: Common 52

In accordance with EOP 2 during an ATWS boron must be injected before Torus water temperature reaches ____(1)____ to ensure injection of ________(2)________

A. (1) 110°F
(2) Hot Shutdown Boron Weight to prevent additional containment heat input which would cause a loss of primary containment integrity.

B. (1) 120°F
(2) Hot Shutdown Boron Weight to prevent additional containment heat input which would cause a loss of primary containment integrity.

C. (1) 110°F
(2) Cold Shutdown Boron Weight before torus temperature exceeds the Heat Capacity Temperature Limit.

D. (1) 120°F
(2) Cold Shutdown Boron Weight before torus temperature exceeds the Heat Capacity Temperature Limit.

Proposed Answer: A.

Explanation (Optional):

A. Correct - The combination of high reactor power (above the APRM downscale trip), high torus temperature (above 110°F, the Boron Injection Initiation Temperature), and an open SRV or high drywell pressure (2. 5 psig), are symptomatic of heat being rejected to the torus at a rate in excess of that which can be removed by the torus cooling system. Unless mitigated, these conditions ultimately result in loss of NPSH for ECCS pumps taking suction on the torus, containment overpressurization, and (ultimately) loss of primary containment integrity.

The Boron Injection Initiation Temperature (BIIT) is he highest torus temperature at which initiation of boron injection will permit the injection of the Hot Shutdown Boron Weight of boron before torus temperature exceeds the Heat Capacity Temperature Limit
B. Incorrect – Boron injection should be started prior to exceeding 110°F.
C. Incorrect –By starting injection at 110°F Hot, NOT COLD, Shutdown Boron Weight can be injected prior to exceeding the HCTL.
D. Incorrect – Boron injection should be started prior to exceeding 110°F. By starting injection at 110°F Hot, NOT COLD, Shutdown Boron Weight can be injected prior to exceeding the HCTL.

Technical Reference(s): EOP Study Guide, Rev 13, Sect 13.1, pg 3 of 54 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001, AA2.06	
	Importance Rating	3.2	

(K&A Statement) Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Nuclear boiler instrumentation

Proposed Question: Common 53

The reactor is in single loop following a trip of the "B" Recirc Pump at 100% power. The following conditions exist:

- Rapid Shutdown Sequence is latched
- APRMs are fluctuating between 58% and 70% power
- Total Core Flow: 20.5 mlb/hr
- "A" Recirc Pump Speed: 69.5%
- "B" Recirc Pump Discharge Valve (RV-53B): Closed

Determine the correct action based on this information and the enclosed power to flow map.

A. Immediately insert a manual scram

B. Increase the speed of the "A" Recirc Pump.

C. Insert rods using the rapid shutdown sequence.

D. Open "B" Recirc Pump Discharge Valve (RV-53B).

Proposed Answer: A.

Explanation (Optional):

A. is correct
B. C. & D. Incorrect -The reactor is operating in the Exclusion Area of the Power to Flow Map. Signs of reactor instability are present because APRMs are oscillating 10% peak-to-peak. The operator should immediately insert a manual reactor scram IAW OT 3117.

Technical Reference(s): OT 3117, Rev 16, Immediate (Attach if not previously

Operator Action #2.c provided)

Proposed references to be provided to applicants during examination: P/F Map

Learning Objective: LOT-00-602, CRO 2 (As available)

Question Source: Bank # 5924 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024, 2.1.30	
	Importance Rating	4.4	

(K&A Statement) Conduct of Operations: Ability to locate and operate components, including local controls. (High Drywell Pressure)

Proposed Question: Common 54

The plant is in OP-3126, Shutdown Using Alternate Shutdown Methods. Drywell temperature and pressure are rising.

In accordance with OP-3126 which method is available for cooling the Drywell?

A. Place the "B" RHR system in the Drywell Spray Mode from the RHR Alternate Shutdown Panel.

B. Place the "A" RHR system in the Torus Cooling Mode then manually open RHR - 26A, DWL SPRAY OUTBD and RHR-31A, DWL SPRAY INBD.

C. At panel HVSGP A, start additional RRUs and vent the Torus using the Containment Ventilation System.

D. At panel HVSGP A, start additional RRUs and vent the Drywell using the Containment Ventilation System.

Proposed Answer: B.

Explanation (Optional):

B. Correct – IAW OP-3126, App A, pg 7 , If Drywell temperature exceeds 260°F direct Operator #2 to place the "A" RHR system in the Torus Cooling Mode and manually open RHR-26A and RHR-31A.

A. Incorrect – There are no controls for the " B RHR System or either set of DW Spray Valves on the RHR Alternate Shutdown Panel.

B. Incorrect – There are no additional RRUs to start and all Containment Ventilation valves are failed closed during alternate shutdown.

C. Incorrect – There are no additional RRUs to start and all Containment Ventilation valves are failed closed during alternate shutdown.

Technical Reference(s): OP-3126, Rev 17 App A, pg (Attach if not previously

7 provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Lot more

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038, 2.1.23	
	Importance Rating	4.3	

(K&A Statement) Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation. (High Off-Site Release Rate)

Proposed Question: Common 55

Following entry into the Severe Accident Guidelines (SAG), the SAGs provided direction to Vent the Containment using OE 3107 Appendix "HH", Primary Containment Venting Methods.

The TSC recommends using a filtered flow path.

What section in Appendix "HH" provides a filtered flow path?

A. Section 5, Torus Hardened Vent to Stack

B. Section 6, 4" Sprays to Waste Collector to RW to Stack

C. Section 10, 3" Vent to RB HVAC via RTF-5 to Stack

D. Section 12, CAD via 1" Vent to SBGT to Stack

Proposed Answer: D.

Explanation (Optional):

Ans D. This is the only flowpath through filtration (SBGT).

A. Incorrect - Using Hardened Vent path to stack provides scrubbing but no filtration.
B. Incorrect - Provides a 4 inch vent path from drywell or torus sprays through the RHR system letdown to radwaste (RHR-57 and RHR-66) to the waste collector tank to Radwaste Ventilation HEPA to the stack. Although this path includes the HEPA filter OE-3107 does NOT consider this a vented flow path.
C. Incorrect -Using 3" Vent to RB HVAC via RTF-5 to Stack provides no filtration.

Technical Reference(s): EOP-3, OE 3107, Rev 17, App HH, Table II, Pg 218 of 227 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:			None
Learning Objective:	LOT-01-626, CRO **4**	(As available)	
Question Source:	Bank **#**		Lot more
	Modified Bank #	3107	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Similar to 2007 common 19	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031, 2.4.46	
	Importance Rating	4.2	

(K&A Statement) Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions. (Reactor Low Water Level)

Proposed Question: Common 56

The plant is raising power when the following events occur:

- 5-E-2, FW VLV LOCKUP SIGNAL/AIR FAIL Annunciator alarms
- Attempts to reset the Feedwater Regulating Valves have failed
- At the time of the lock-up power is still slowly rising from the effect of the last recirculation flow adjustment

What is the affect of the valve lock-up on reactor water level and what action IAW OT 3113 is required in response to this condition?

Indicated reactor water level will be (1). Level is controlled by (2).

A. (1) lowering.
(2) opening the AUX FEED REG VLV FDW-13.

B. (1) rising.
(2) raising core flow as necessary to stabilize level at a rate ≤10% RTP/min.

C. (1) lowering.
(2) placing a FEEDWATER REG VLV FDW-12A(B) CONTROLLER in manual and raise feedwater flow.

D. (1) rising.
(2) placing a FEEDWATER REG VLV FDW-12A(B) CONTROLLER in manual and lower feedwater flow.

Proposed Answer: A.

Explanation (Optional):

A. Correct – Because power was being raised steam flow will be still rising when the FWRVs lockup. Because of the lockup the feedwater Reg. valves will lockup and not respond resulting in a lowering water level.
IF a lockup of the Feedwater Reg Valves has occurred (loss of air or control signal) THEN: IF level is slowly trending downward, THEN attempt to control level by opening the AUX FEED REG VLV FDW-13.

B. and D. Incorrect - RPV water level will lower.
C. Incorrect – placing the FEEDWATER REG VLV FDW-12A(B) CONTROLLER in manual will have no effect with the valves locked-up.

Technical Reference(s): OT 3113, Rev. 22, pg 3 of 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

Question Source: Bank # ________ Lot more

Modified Bank # ________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ________

Comments:

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

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

Proposed Question: Common 57

A Loss of Offsite power has occurred and the following conditions exist:

- The " A Diesel Generator started and is supplying its buses
- The "B" Diesel Generator started but did NOT close in on its bus due to a fault on the bus.
- No buses have been cross-tied at this time.

IAW OT 3122, Loss of Normal Power which one of the following actions is required for supplying Vital AC Bus and why is this action taken?

A. Transfer the Vital AC Bus to AC Drive motor to minimize station DC loads on the Station Battery.

B. Transfer the Vital AC Bus to its alternate source to minimize station DC loads on the Station Battery.

C. Adjust the EDG frequency as required to obtain a steady state 60.0-60.2 Hz to permit the Vital AC Bus to automatic transfer to the AC Drive.

D. Adjust the EDG frequency as required to obtain a steady state 60.0-60.2 Hz to prevent the Vital AC Bus from automatically transferring to the DC Drive.

Proposed Answer: B.

Explanation (Optional):

B. Correct – With the loss of 480 VAC Bus 8 (8B) the Vital AC MG set swaps from the AC supply to the DC Supply to minimize the load on the station batteries OT 3122 directs shifting to the alternate supply.

A. Incorrect there is no power supply to 480 VAC Bus 8 (8B)

C. and D. Incorrect the A EDG supplies 480 VAC Bus 9, adjusting voltage on that EDG will have no effect.

Technical Reference(s): OT 3122, Rev 21, Step 5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Lot more

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

12-15-08, Removed the two DG statuses from the stem and put them in bullets.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025, 2.4.34	
	Importance Rating	4.2	

(K&A Statement) Emergency Procedures / Plan: Knowledge ot RO tasks performed outside the main control room during an emergency and the resultant operational effects. (High Reactor Pressure)

Proposed Question: Common 58

The control room has been abandoned and OP 3126, Shutdown Using Alternate Shutdown Methods, is being performed. Reactor pressure is currently 1000 psig. You are directed to commence a cooldown, lowering reactor pressure the maximum amount, while remaining within the reactor cooldown rate limit IAW OP 3126.

1) Which one of the following methods is used for the cooldown? AND
2) Calculate the new reactor pressure and control band. (Figure 1 of Appendix C is attached for your use)

Calculate the new reactor pressure then:

A. 1) Place RCIC in a pressure control lineup then slowly lower reactor pressure to 430 psig
2) Operate the SRV as necessary to maintain pressure 430 - 530 psig.

B. 1) Place RCIC in a pressure control lineup, then slowly lower reactor pressure to 530 psig.
2) Operate the SRV as necessary to maintain pressure 350 - 530 psig.

C. 1) Open Safety Relief Valve RV2-71A or RV2-71B to reduce reactor pressure to 430 psig.
2) Operate the SRV as necessary to maintain pressure 430 - 530 psig.

D. 1) Open Safety Relief Valve RV2-71A or RV2-71B to reduce reactor pressure to 530 psig.
2) Operate the SRV as necessary to maintain pressure 430 - 530 psig.

Proposed Answer: C.

Explanation (Optional):

C. Correct – IAW OP-3126, RV2-71A or RV2-71B are opened to lower pressure to the value determined using Figure 1 of Appendix C. The correct pressure corresponding to 456 deg F (90 deg F drop from original temperature of 546 deg F) is 430 psig.
A. and B. Incorrect – The SRVs are used to lower pressure and the correct pressure corresponding to 456 deg F (90 deg .F drop from original temperature of 546 deg F) is 430 psig
D. Incorrect - The correct pressure corresponding to 456 deg F (90 deg F drop from original temperature of 546 deg F) is 430 psig.

Technical Reference(s): OP 3126, Rev. 18 (Appendix C, Step 16,), pg 33 of 53 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: OP-3126, figure 1 of Appendix C

Learning Objective: LOT-00-612 EO-10.2, 9.1, 8.1 (As available)

Question Source: Bank # Lot more
Modified Bank # Requal 1149 (Note changes or attach parent)
New

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43

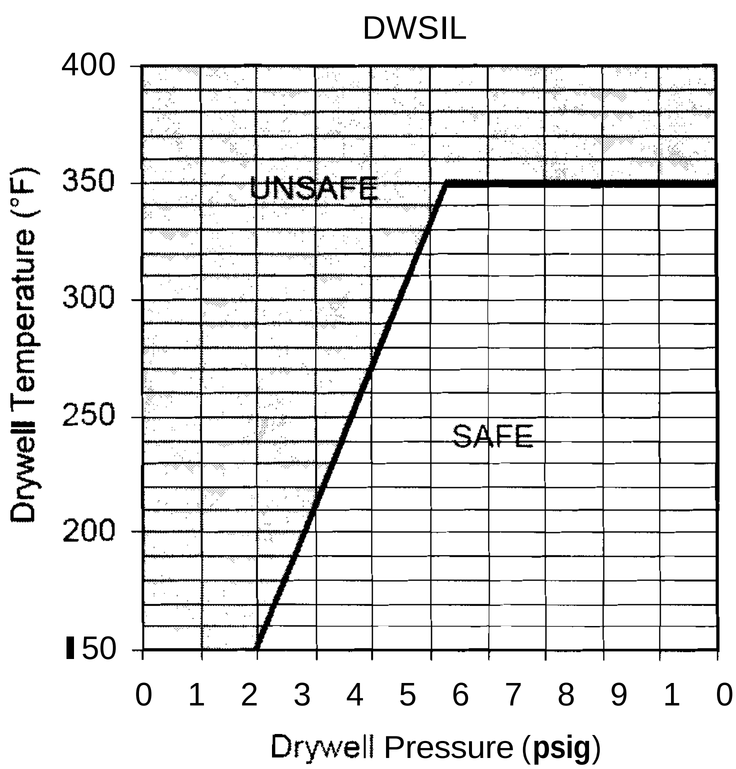
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295010, AK1.03	
	Importance Rating	3.2	

(K&A Statement) **Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : Temperature increases**

Proposed Question: Common 59

An accident occurred and Drywell pressure is 6 psig with Drywell temperature 360°F and rising.



Based upon these plant conditions and the Drywell Spray Initiation Limit Curve, what adverse conditions could result from drywell spray initiation?

A. The evaporative cooling pressure drop may result in de-inerting the drywell following initiation of drywell sprays.

B. The delta pressure between the drywell and torus will prevent the vacuum breaker from functioning as designed.

C. The evaporative cooling pressure drop following initiation of drywell sprays may result in exceeding Torus design internal pressure.

D. Lowering drywell pressure will reopen the SRVs, causing RPV pressure to drop below the saturation pressure for this temperature.

Proposed Answer: A.

Explanation (Optional):

A. Correct – IAW EOP Study Guide, The DWSIL is a function of drywell pressure. It is utilized to preclude de-inertion following initiation of drywell sprays.

B. Incorrect – This is the bases for the level limit

C. Incorrect - the vacuum breakers are designed to prevent exceeding the DW external pressure limit (2 psig)

D. Incorrect – The SRVs may open in lowering PC pressure, however PC temperature will also be lowering and the SRVs will close at a sufficient D/P to prevent falling below the saturation pressure for the new DW temperature.

Technical Reference(s): EOP Study Guide, Rev 13 Section 8, pg 186 of 346 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029, EK2.02	
	Importance Rating	3.4	

(K&A Statement) Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: HPCI: Plant-Specific

Proposed Question: Common 60

The HPCI System is in a normal standby lineup when a spurious initiation signal is received. The system auto initiates. The Balance-Of-Plant (BOP) operator stops HPCI without referring to OP 2120 for guidance.

Fifteen minutes later, the crew detects Suppression Pool level is rising.

Which one of the following is the cause of the high Suppression Pool level?

A. The BOP operator secured HPCI injection, by inadvertently closing the Pump Discharge Valve, HPCI-19.

B. The spurious initiation signal caused the HPCI suction to shift to the Suppression Pool.

C. The BOP operator inadvertently opened the HPCI suction from the Suppression Pool, HPCI-58.

D. The discharge from the HPCI gland seal condenser was pumped into the Suppression Pool.

Proposed Answer: A.

Explanation (Optional):

A. Correct – When HPCI initiates, it takes a suction from the CST. When the discharge valve is closed, the minimum flow valve will open providing flow (CST water) to the Torus.
B. Incorrect – A HCPI pump suction from the Suppression Pool would not transfer water from the CST to the Suppression Pool.
C. Incorrect – If the Suppression Pool suction was opened, the CST suction would close. If both valves were open, there are check valves in the suction lines to prevent cross-connecting the CST and Suppression Pool.
D. Incorrect - The discharge from the HPCI gland seal condenser is pumped into the HPCI suction.

Technical Reference(s):	OP-2120, Rev 55 App B, pg 31 of 37 DWG 191169, shts 1 & 2	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	WA #	295033, EK3.01	
	Importance Rating	3.3	

(K&A Statement) Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Emergency depressurization

Proposed Question: Common 61

Which one of the following is the reason for performing an emergency depressurization when required by High Secondary Containment Radiation levels?

EOP action(s) are taken to...

A. mitigate the loss of ECCS pumps taking suction on the torus which in turn could lead to a loss of adequate core cooling.

B. permit the use of secondary containment ventilation to filter both primary and secondary containment atmospheres and allow access to the reactor building.

C. mitigate the consequences of unisolable leakage from a primary system into the secondary containment that may pose a direct and immediate threat to primary containment integrity

D. mitigate the consequences of unisolable leakage from a primary system into the secondary containment that may pose a direct and immediate threat to secondary containment integrity

Proposed Answer: D

Explanation (Optional):

D. Correct – IAW the EOP Study Guide, When the rise in secondary containment temperature, radiation or water level spreads to more than one area, a direct threat exists relative to secondary containment integrity, to equipment located in the reactor building, and to continued safe operation of the plant.
A. Incorrect – Adequate core cooling is not the basis.
B. Incorrect – This is the basis for ED for an Off-Site release.
C. Incorrect – The basis is not for Primary Containment integrity.

Technical Reference(s): Ref EOP Study Guide (Attach if not previously

Section 9 page 14 of 16 provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-611, RO-3 (As available)

Question Source: Bank # Lot more

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

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

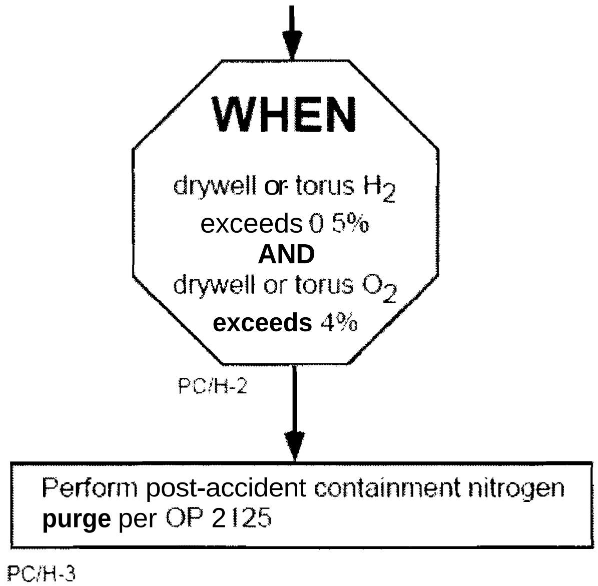
(K&A Statement) Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: Containment atmosphere control system

Proposed Question: Common 62

Given the following conditions:

- A large break LOCA has occurred inside the Drywell.
- The Containment H^2/O^2 Analyzers were placed in-service 2 hours ago.
- The Containment H^2/O^2 Analyzers readings are:

Drywell H^2 – 0.6% O^2 – 3.0%
Torus H^2 – 0.2% O^2 – 7.0%



Which one of the following actions is required?

A. Perform a containment purge by injecting nitrogen into the drywell and venting from the torus.

B. Perform a containment purge by injecting nitrogen into the torus and venting from the drywell.

C. Continue to monitor containment H^2/O^2 Analyzers because post accident nitrogen purging is not required at this time.

D. Perform a containment purge by injecting nitrogen into the torus and drywell while simultaneously venting from the torus and drywell.

Proposed Answer: A.

Explanation (Optional):

A. Correct – Drywell H2 levels and torus 0 2 levels require purging IAW OP 2125. Offsite dose will be minimized by injecting nitrogen into the drywell and venting from the torus.
B. Incorrect – Offsite dose will be minimized by injecting nitrogen into the drywell and venting from the torus.
C. Incorrect – Drywell H2 levels and torus 0 2 levels require purging IAW OP 2125.
D. Incorrect – The drywell and torus purge and vents should never be simultaneously opened because that would cross connect the two containments.

Technical Reference(s): EOP-3 (Attach if not previously provided)
OP-2125, Pg 15

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ____

Comments:

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032, EA2.01	
	Importance Rating	3.8	

(K&A Statement) Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Area temperature

Proposed Question: Common 63

The plant is at 100% power when a fire is reported in material placed next to the Drywell Personnel Air Lock Area.

The following events occur:

- Annunciator 4-H-1, STEAM LEAK DET PANEL TEMP HI alarms
- STEAM LEAK DETECTION TOUCHSCREEN MONITOR on CRP 9-21 indicates Rx/Turb Ch. 19 is above its max normal operating limit
- The crew has entered OP 3020, Fire Emergency Response Procedure

In accordance with the Vermont Yankee Procedures, the operators shall:

A. Enter EOP-4, Secondary Containment Control and initiate a shutdown per OP 0105.

B. Enter EOP-4, Secondary Containment Control and isolate and systems that might be affected.

C. Enter ON 3158, Reactor Building High Area Temperature/Water Level and initiate a shutdown per OP 0105.

D. Enter ON 3158, Reactor Building High Area Temperature/Water Level and determine the equipment that may be affected.

Proposed Answer: D.

Explanation (Optional):

D. Correct – This location is not an EOP-4 entry. Entry into ON-3158 is required and the procedure directs the operators to determine the affected systems.
A. Incorrect - This location is not an EOP-4 entry and a shutdown is not required.
B. Incorrect – This location is not an EOP-4 entry.
C. Incorrect - A shutdown is not required by ON 3158.

Technical Reference(s): ARS 21002, 4-H-1 (Attach if not previously provided)
ON 3158,

Proposed references to be provided to applicants during examination: Table of Secondary Containment Limits

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	WA #	295015, 2.1.28	
	Importance Rating	4.1	

(K&A Statement) Conduct of Operations: Knowledge of the purpose and function of major system components and controls. (Incomplete Scram)

Proposed Question: Common 64

Which ONE of the following is the purpose for bypassing the Low-Low RPV Water Level logic in EOP-2, RPV Control (implementing Appendix P)?

A. To maintain the Main Condenser as a heat sink if reactor water level is to be lowered later.

B. To preclude inadvertent positive reactivity addition by closure of the Main Steam Isolation Valves.

C. To bypass the Emergency Core Cooling Systems automatic initiations to prevent uncontrolled injection.

D. To ensure Main Steam Isolation Valves can be reopened concurrent with high main steam line radiation.

Proposed Answer: A.

Explanation (Optional):

A. Correct – Subsequent actions in EOP-2 may require that RPV water level be lowered to or below the low RPV water level MSIV isolation setpoint. To prepare for this possibility and prevent unintended loss of the main condenser, direction is given to bypass selected interlocks in advance of subsequent actions.

B. Incorrect - Incorrect - Bypassing the LO-LO Level Water Level does not prevent a large positive reactivity insertion. This is accomplished by inhibiting ADS.
C. Incorrect – This does NOT bypass ECCS initiations.
D. Incorrect - EOP-2, ARC/OR-5 states that MSIVs should only be reopened if MSL Hi Rad signal is not present.

Technical Reference(s): EOP Study Guide, Sect. 7 pg 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:		None
Learning Objective:	______________	(As available)
Question Source:	Bank # 2226	Lot more
	Modified Bank # ______	(Note changes or attach parent)
	New ______	
Question History:	Last NRC Exam No	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 X	
	55.43 ______	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295036, EA2.03	
	Importance Rating	3.4	

(K&A Statement) Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : Cause of the high water level

Proposed Question: Common 65

The plant is operating at full power when the following annunciators alarm.

- RX BLDG EQMT DRN SUMP SOUTH LVL HI (4-L-4)
- RBCCW SURGE TANK LVL HI/LO (6-M-8)

1) What is the probable cause of these annunciators? AND
2) What action is required?

Gross failure of the.

A. 1) sump cooling coils.
2) Enter EOP-4 Secondary Containment Control.

B. 1) Reactor Recirc Unit Coolers.
2) Enter EOP-4 Secondary Containment Control.

C. 1) sump cooling coils.
2) Enter ON-3158, Reactor Building High Area Temperature/Water Level.

D. 1) Reactor Recirc Unit Coolers.
2) Enter ON-3158, Reactor Building High Area Temperature/Water Level.

Proposed Answer: C.

Explanation (Optional):

C. Correct – 4-L-4 alarms one foot below the top of the sump. The Rx Bldg. Equipment Drain Sumps have sump coolers supplied by RBCCW, a gross failure of the tubes would allow large amounts of RBCCW into the sump causing a high level alarm and lowering the inventory in the RBCCW system causing the low level in the surge tank.

A. Incorrect – this is an entry into ON-3158. To enter EOP-4 and operator must locally verify that the level in a corner room is 1" above the floor. 4-L-4 alarms one foot below the top of the sump.

B. and D. Incorrect - Because the Reactor Recirc Unit Coolers drain to the Drywell Sump.

Technical Reference(s): P&ID 191177
EOP-4
ON-3158, Rev 10, pg 1.
9-4 ARS for 4-L-4

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: Table of Secondary Containment Limits

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.8	
	Importance Rating	3.4	

(K&A Statement) Ability to coordinate personnel activities outside the control room.

Proposed Question: Common 66

The reactor has just scrammed from full power.

IAW DP-0166, which one of the following are the minimum actions directed by the Control Room for the Reactor Building Auxiliary Operator?

A. Monitor the Recirc MG lube oil temperatures and the Control Rod Drive system including the Flow Control Station.

B. Monitor the Recirc MG lube oil temperatures and the Reactor Cleanup System, including the Cleanup Demineralizers.

C. Monitor the Control Rod Drive system including the Flow Control Station and manually adjust cooling water valves for shutdown operation.

D. Monitor the Control Rod Drive system including the Scram Discharge Volume and the Reactor Cleanup System, including the Cleanup Demineralizers.

Proposed Answer: B.

Explanation (Optional):

B. Correct – IAW DP-166, Sect A.2.m, Auxiliary Operator actions on a reactor scram or plant transient include, but are not limited to, the following:

- The Reactor Building AO shall, **when directed by the Control Room,** monitor the Recirc MG lube oil temperatures and the Reactor Cleanup System, including the Cleanup Demineralizers.

A. C. and D. Incorrect – There is no direction to monitor the CRD system.

Technical Reference(s): DP-166, Rev 19, pg 10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:	None	
Learning Objective:		(As available)
Question Source:	Bank #	Lot more
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam No	
Question Cognitive Level:	Memory or Fundamental Knowledge X	
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 X	
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.4	
	Importance Rating	3.3	

(K&A Statement) Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: Common 67

The plant is conducting a startup. Three (3) of the five *(5)* on-shift Auxiliary Operators (AOs) are stricken with food poisoning and are transported off-site to the hospital.

What is the REQUIRED number of additional AOs needed to meet the minimum shift staffing requirements of AP 0894?

A. None

B. One

C. Two

D. Three

Proposed Answer: B.

Explanation (Optional):

B. Correct – three AOs are needed during startup and power operations

A. Incorrect – only one is needed to make the required 3 AOs
C. Incorrect – only one is needed to make the required 3 AOs
D. Incorrect – only one is needed to make the required 3 AOs

Technical Reference(s): AP 0894, Rev 8, Table 1. (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # ______________ Lot more

Modified Bank # ______________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.13	
	Importance Rating	3.6	

(K&A Statement) Knowledge of tagging and clearance procedures

Proposed Question: Common 68

A tagout is in place for the A RFP Oil pump at the MCC 6B. The individual, who accepted the tagout, is off-site and cannot be contacted. Plant startup is in progress and this tagout must be released.

IAW EN-OP-102, which of the following describes the minimum authorization required for an alternate release authorization?

A. The Tagout Holders Supervisor

B. The Shift Manager or his/her designee.

C. The Supervisory Control Room Operator AND the Shift Manager..

D. The Shift Manager or hislher designee AND another Senior Licensed operator.

Proposed Answer: B.

Explanation (Optional):
B. Correct - EN-OP-102, Sect 5.17, In the event that release is required and a Tagout/Work Order Holder cannot be contacted and is not on site, the release can be authorized by the Shift Manager or hislher designee.

A. Incorrect – Not authorized by EN-OP-102.
C. Incorrect – Not authorized by EN-OP-102.
D. Incorrect – Not authorized by EN-OP-102.

Technical Reference(s): EN-OP-102, Sect 5.17 (Attach if not previously provided)

Proposed references to be provided to applicants during None examination: ______________________

Learning Objective: ____________________________ (As available)

Question Source: Bank # ____________________ Lot more

Modified Bank # ____________________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.2.40	
	Importance Rating	3.4	

(K&A Statement) **Ability to apply technical specifications for a system.**

Proposed Question: Common 69

During an outage an Auxiliary Operator contacts the Control Room and notifies you that both Inner and Outer Reactor Building Railroad Doors are OPEN.

Which of the following evolutions, if initiated in conjunction with the above, would violate Technical Specifications?
(Assume none of the evolutions has yet been done, or currently exists)

A. Moving irradiated fuel in the Spent Fuel Pool.

B. Opening the Primary Containment Airlock Doors.

C. Removing the reactor steam separator.

D. Performing Control Rod Timing.

Proposed Answer: A.

Explanation (Optional):
A. Correct - IAW T.S. Section 3.7.C 1.b, Secondary Containment is required whenever moving irradiated fuel assemblies.

B. Incorrect – Because only irradiated fuel movement requires Secondary Containment.
C. Incorrect – Because Primary Containment does not exist in this condition.
D. Incorrect – Because it is not a core alteration and does not have the potential for draining the reactor vessel

Technical Reference(s): T.S. 3.7.C.l .b (Attach if not previously provided)

Proposed references to be provided to applicants during None

examination:			________
Learning Objective:	________	(As available)	
Question Source:	Bank #	________	Lot more
	Modified Bank #	________	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge		________
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	X	
	55.43	________	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.12	
	Importance Rating	3.2	

(K&A Statement) Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: Common 70

IAW RP-0507, Primary Containment Entry, which one of the following are the MINIMUM requirements that must be met before personnel may enter the Drywell at power to investigate a problem?

They may enter after reactor power is less than __(1)__ and Oxygen (O^2) concentration is greater than __(2)__.

A. (1) 30 %
(2) 17.5%

B. (1) 70%.
(2) 17.5%

C. (1) 30%
(2) 19.5%.

D. (1) 70%
(2) 19.5%.

Proposed Answer: D.

Explanation (Optional):

D. Correct – IAW RP-0507, Max power may be 70%.The drywell must be de-inerted and the drywell is not considered de-inerted until two consecutive air samples, taken at least 15 minutes apart on both the drywell and torus, indicate an oxygen concentration of greater than 19.5%

In all cases drywell entry will not be made above 70% power.

B. Incorrect - Drywell AND Torus must be de-inerted to >19.5% Oxygen.

A. Incorrect - Max power may be 70%, Drywell AND Torus must be de-inerted to >19.5% Oxygen.

C. Incorrect - Max power may be 70%.

Technical Reference(s):	RP-0507, Rev 25, P&L statements pg 4 of 5.	(Attach if not previously provided)
Proposed references to be provided to applicants during examination:		None
Learning Objective:	______	(As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.3.4	
	Importance Rating	3.2	

(K&A Statement) Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: Common 71

During an emergency, tasks are being conducted in areas that show no significant increase in general area dose rates from normal plant radiological conditions.

IAW OP 3507, Emergency Radiation Exposure Control, ...

(1) What is the maximum exposure an individual can receive in these areas without Emergency Exposure authorization? AND

(2) Who must approve an Emergency Exposure above this limit?

A. (1) 4.5 Rem
(2) The General Manager Plant Operations and Senior Radiation Protection Representative.

B. (1) 9.5 Rem
(2) The General Manager Plant Operations and Senior Radiation Protection Representative.

C. (1) 4.5 Rem
(2) Shift Manager/Plant Emergency Coordinator (or TSC Coordinator) and Senior Radiation Protection Representative.

D. (1) 9.5 Rem
(2) Shift Manager/Plant Emergency Coordinator (or TSC Coordinator) and Senior Radiation Protection Representative.

Proposed Answer: C.

Explanation (Optional):

B. Correct –<u>IF</u> tasks are being conducted within areas that show no significant increase in general area dose rates from normal plant radiological conditions, <u>AND</u> dose commitment to any individual of less than or equal to 4.5 Rem is required, <u>THEN</u> the normal work process will be used to control radiation exposure of personnel.

Authorization to the 10 Rem limit (Protecting Valuable Property) 25 or 75 Rem limit (Lifesaving or Protection of a Large Population) may only be made with the joint concurrence of the Shift Supervisor/Plant Emergency Coordinator (or TSC Coordinator) and Senior Radiation Protection Representative.

A. Incorrect – Shift Supervisor/Plant Emergency Coordinator (or TSC Coordinator) and Senior Radiation Protection Representative.

B. and D. Incorrect – 4.5 Rem is the limit.

Technical Reference(s): OP 3507, pgs 4 & 6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-404 (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.46	
	Importance Rating	4.2	

(K&A Statement) Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: Common 72

The plant is operating at 100% power when Annunciator 5-A-1, SLC SQUIB VLV CONTINUITY LOSS, alarms. Which one of the following indications is consistent with this alarm condition?

A. Squib valve ready lights on CRP 9-5 are on.

B. Squib valve continuity meters behind CRP 9-5 indicate 0.4 and 0.5 ma.

C. The Red indicating light on the " A SLC pump is on because the pump has been started locally.

D. The Red and the Green " B SLC pump indicating lights are off indicating the breaker has tripped.

Proposed Answer: D.

Explanation (Optional):

D. Correct - The SLC Pump breaker supplies power to the squib valve, tripping the SLC pump breaker will cause the alarm.

A. Incorrect – These lights are energized when the valve is energized.

B. Incorrect - These are acceptable currents (>.1ma)

C. Incorrect – Starting the pump locally has no affect on the squib valve.

Technical Reference(s): ARS-21003, Rev 7, pg 2
OP-2114, Rev 33, pgs 11 & (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:		______	(As available)
Question Source:	Bank #	______	Lot more
	Modified Bank #	______	(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	No	
Question Cognitive Level:	Memory or Fundamental Knowledge		______
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	X	
	55.43	______	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.42	
	Importance Rating	2.6	

(K&A Statement) Knowledge of emergency response facilities.

Proposed Question: Common 73

Following a plant scram conditions deteriorated and an ALERT has been declared.

Based on this emergency classification _______(1)_______ will be activated. If On-Site and Off-Site Boundary Teams are needed they will be dispatched by the ______(2)______

A. (1) ONLY the Technical Support Center (TSC)
(2) TSC

B. (1) ONLY the TSC and Operations Support Center (OSC)
(2) OSC

C. (1) TSC, OSC and Emergency Operations Facility/Recovery Center (EOFIRC)
(2) EOFIRC

D. (1) TSC, OSC, and EOFIRC
(2) OSC

Proposed Answer: D.

Explanation (Optional):

D. Correct – During an Alert all the facilities are activated the EOFIRC, TSC and OSC are activated, Boundary Teams are assembled and dispatched out of the OSC.

A. Incorrect - Both the TSC and OSC are activated, Boundary Teams are assembled and dispatched out of the OSC.

B. Incorrect - During an Alert all the facilities are activated the EOFIRC, TSC and OSC are activated (although the EOFIRC may not be manned until the recovery actions.

C. Incorrect – The Boundary Teams are assembled and dispatched out of the OSC.

Technical Reference(s):	AP 3554, Sect. 5.2.2, pg 6 OP-3542, pg 11 of 89, pg 16 of 89. OP-3544, OSC Checklist # 8 pg 3 of 12 LOT-00-900, pg 40	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.4.6	
	Importance Rating	3.7	

(K&A Statement) **Knowledge of** EOP **mitigation strategies.**

Proposed Question: Common 74

An ATWS has occurred with the following conditions:

Reactor power is 20% and slowly lowering

- Torus temperature is 115°F and slowly rising
- Reactor pressure control is on SRVs 800-1000 psig
- Reactor water level is +25" and slowly lowering
- Injection has been terminated & prevented IAW OE 3107 Appendix GG

Which of the following conditions would establish the upper end of the RPV level control band?

A. APRM downscales come in.

B. Reactor water level reaches -19 inches,

C. Only one SRV is open for reactor pressure control.

D. Reactor power reaches the heating range with a negative period.

Proposed Answer: A.

Explanation (Optional):

A. Correct – IAW EOP-2

B. Incorrect - This is the bottom of the ATWS level band of +6 to -19", injection should recommence at +6" - TAF.

C. Incorrect – Regardless of SRV conditions, injection is recommenced at TAF to insure adequate core cooling.

D. Incorrect – This is indication of the reactor being shutdown but is applicable when a cooldown is commenced and no boron has been injected.

Technical Reference(s): EOP-2 (Attach if not previously

provided)

Proposed references to be provided to applicants during examination: EOP-2, Level leg

Learning Objective: ______ (As available)

Question Source: Bank # 5767 Lot more

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43 ______

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	WA #	2.2.37	
	Importance Rating	3.6	

(K&A Statement) Ability to determine operability and/or availability of safety related Jequipment.

Proposed Question: Common 75

During a plant startup, with RPV pressure steady at 850 psig, the in-service CRD pump, pump " A trips. The "B" CRD pump is out of service for maintenance.

After one minute, which one of the following statements is correct?

Rod motion with RMCS is __(1)__ AND __(2)__.

A. (1) NOT available.
(2) scram times will be within technical specification limits.

B. (1) available.
(2) scram times will exceed technical specification limits.

C. (1) available.
(2) scram times will be within technical specification limits.

D. (1) NOT available.
(2) scram times will exceed technical specification limits.

Proposed Answer: A.

Explanation (Optional):

A. Correct – Normal rod motion is lost. Scram times are OK as long as accumulators are charged. IAW TS Bases Sect. 3.3, Accumulators, At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure.

B. Incorrect – Normal rod motion is lost. Scram times are OK as long as reactor pressure is >800 psig

C. Incorrect – Normal rod motion is lost.

D. Incorrect - Scram times are OK as long as reactor pressure is >800 psig

Technical Reference(s): TS Bases 3.3 (Attach if not previously

provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Lot more

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295019, AA2.02	
	Importance Rating		3.7

(K&A Statement) Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

Proposed Question: SRO 76

The plant is operating at 100% power when the following occurs:

- A TOTAL LOSS of Instrument Air has occurred

  Instrument Air Header Pressure is 0 psig

  RPV water level is lowering

Which one of the following is the overall reactor water level control and pressure control strategy?

In accordance with...

A. OT-3100, use HPCI/RCIC to control reactor water level and control reactor pressure below 1055 psig by manually cycling SRVs per OT-3100.

B. OT-3100, use HPCI/RCIC to control reactor water level while pressure is controlled by placing the SRVs control switches to AUTO and allowing them to cycle on their relief setpoints.

C. ON-3146, Low Instrument/Scram Air Header Pressure, start one feed pump and control reactor water level with FDW-5, HP HTR BYPASS, control reactor pressure below 1055 psig by manually cycling SRVs per EOP-1.

D. ON-3146, Low Instrument/Scram Air Header Pressure, start one feed pump and control reactor water level with FDW-5, HP HTR BYPASS, by placing the SRVs control switches to AUTO and allowing them to cycle on their relief setpoints.

Proposed Answer: C.

Explanation (Optional):

C. Correct – The outboard MSIVs will go closed on a loss of air the feedwater control valves will fail as is (100% flow). The feedwater pumps will trip on high level (or be manually tripped) one feed pump is started (ON-3146) and the manual feedwater isolation valves are closed and feed flow controlled using FDW-5. Air pressure will be lost to the SRVs, the Containment Air System will provide operating pressure to the SRVs. .

A. Incorrect - Although HPCI/RCIC could be used the preferred method is using Feedwater as directed in ON-3146.

B. Incorrect – Although HPCI/RCIC could be used the preferred method is using Feedwater as directed in ON-3146. Although all IA pressure is lost N2 bottles provide sufficient pressure to manually operate the SRVs until another method of pressure control is available.

D. Incorrect - Although all IA pressure is lost N2 bottles provide sufficient pressure to manually operate the SRVs until another method of pressure control is available.

Technical Reference(s): ON-3146 (Attach if not previously provided)
OT-3100

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295005, AA2.02	
	Importance Rating		2.7

(K&A Statement) Ability to determine and/or **interpret the following as they apply to** MAIN **TURBINE GENERATOR** TRIP
Turbine vibration

Proposed Question: SRO 77

During a plant startup the turbine is being brought up to speed using OP-0105, Phase 3, Turbine Startup and Synchronization. The following conditions exist:

- This is a Cold Startup
- Turbine speed is being raised at 90 rpm/minute.
- Turbine speed is at 1000 rpm and slowly rising.
- 7-E-9, TURBINE SUPERVISORY CABINETS TROUBLE is in alarm
- 7-F-2, TURB EXCESSIVE VIBRATION has just alarmed
- ERFIS indicates the low pressure turbine and generator vibrations are below the turbine trip setpoints.

Which one of the following actions is required for these conditions?

A. Continue using OP-0105 to raise turbine speed through the critical speeds and then verify turbine vibration lowers.

B. In accordance with the guidance in the ARS for 7-F-2, lower speed below 800 rpm then hold speed steady until the turbine vibration alarm clears.

C. Continue using OP-0105 to raise the rate of speed until above 1300 rpm then hold speed to allow turbine shell warming and the vibration alarm to clear.

D. In accordance with the ARS for 7-E-9, trip the turbine and if vibrations do not lower below turbine supervisory limits break vacuum on the main condenser.

Proposed Answer: A.

Explanation (Optional):

A. Correct – Turbine vibration trips are bypassed during a startup by placing TURBINE VIB BYP SW-110-10 switch to the BYPASS position. When this is done Ann 7-E-9, TURBINE SUPERVISORY CABINETS TROUBLE alarms. IAW OP-0105, Phase 3, As the turbine is accelerated, it is normal for the generator bearing vibration to increase as the first critical point is approached, approximately 1000 rpm. Less than 6 mils vibration is considered desirable and 10 mils is acceptable. The turbine vibration alarm alarms at 7 and 8 mils. The turbine startup should continue through the critical speeds (1100 – 1400 rprn).

B. Incorrect - Do not operate the turbine at speeds below 800 rprn for greater than 5 minutes. Vibration detectors are inaccurate below approximately 800 rpm, with accuracy dropping off very rapidly below approximately 600 rpm.

C. Incorrect - At critical speeds of the unit (1000 to 1400 rprn), special care should be used in passing at a reasonably high rate of speed change. Prolonged operation at or close to a critical speed is positively not permitted. Additionally turbine speed is limited to 90 rpm/minute because this is a cold startup.

D. Incorrect – There is no procedural guidance to trip the turbine at this time. If after turbine speed is raised above the critical speeds and turbine vibration does nor lower than the turbine may be tripped.

Technical Reference(s): OP-0105 (Attach if not previously provided)
ARS-21005, rev 15

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____________ Lot more

Modified Bank # ____________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	WA #	295004, AA2.03	
	Importance Rating		2.9

(K&A Statement) Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Battery voltage

Proposed Question: SRO 78

The plant is operating at 100% power with no equipment out of service when the Annunciator 8-P-1, BATT VOLTAGE LO alarms. None of the actions of ON 3159, Loss of DC-1, have been completed.

For this situation which one of the following is required?

A. Battery charger BC-1-1A or BC-1-1C will supply power to DC Bus 1 and its loads.

The "A" EDG must be declared inoperable and Tech. Spec. 3.10.B.1 entered, requiring the " B EDG demonstrated to be operable within 24 hours and the " A EDG returned to service within the succeeding seven days.

B. Battery charger BC-1-1A or BC-1-1C will supply power to DC Bus 1 and its loads.

The "B" EDG must be declared inoperable and Tech. Spec. 3.10.B.1 entered, requiring the " A EDG demonstrated to be operable within 24 hours and the " B EDG returned to service within the succeeding seven days.

C. Both the A-1 Battery and the " A EDG must be declared inoperable and Tech. Specs. 3.10.B.2 and 3.10.B.1 entered.

This will allow operation for the succeeding three days provided all required systems, subsystems, trains, components and devices supported by the operable 125 volt Station Battery System are operable and the " B EDG demonstrated to be operable within 24 hours.

D. Both the A-1 Battery and the " B EDG must be declared inoperable and Tech. Specs. 3.10.B.2 and 3.1O.B.I entered.

This will allow operation for the succeeding three days provided all required systems, subsystems, trains, components and devices supported by the operable 125 volt Station Battery System are operable and the " A EDG demonstrated to be operable within 24 hours.

Proposed Answer: D.

Explanation (Optional):

D. Correct – IAW OP-2145, DC-1 for ACB control power and DG control power must be operable for DG-1B to be considered operable. Both the B EDG and the DC Bus must be declared inoperable requiring verifying operability of alternate equipment.
A. and B. Incorrect - The charger is not allowed to power the bus alone due to charger may not handle any load changes. Additionally all the components of the 125 VDC system are required to be operable therefore entry into 3.10.2.b is required.
C. Incorrect – the A EDG receives power from the other battery (DC -2-AS).

Technical Reference(s): OP-2145, Rev 45, pg 7.
Preq. 5
T.S. Sect 3.10.2.b
ARS-21006, 8-P-1
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: T.S. Sect 3.10

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295006, 2.1.19	
	Importance Rating		3.8

(K&A Statement) Conduct of Operations: Ability to use plant computers to evaluate system or component status. (Scram)

Proposed Question: SRO 79

The plant was operating at 100% power when loss of feedwater causes a Low RPV Water Level scram coincident with a loss of the Rod Position Indicating System (RPIS) for several control rods. The following events occur:

- Control Room Operator reports that reactor power is observed on the SRMs; and counts are lowering.
- The ERFIS Post Scram Rod Position screen indication (PSRP) indicates all rods inserted.
- A Control Rod scan on the Plant Process Computer is NOT available.

Which one of the following actions is required?

A. Perform EOP-1 actions and make a four hour notification to the NRC.

B. Perform EOP-1 actions and make a one hour notification to the NRC.

C. Perform EOP-2 actions and make a four hour notification to the NRC.

D. Perform EOP-2 actions and make a one hour notification to the NRC.

Proposed Answer: A.

Explanation (Optional):
A. Correct – EOP-2, ATWS is entered from EOP-1, RPV Control. When the crew has transitioned to OP-0109 there are no entry conditions for EOP-1, therefore no entry conditions for EOP-2 exist. The crew should continue in OP-0109.
B. Incorrect there is no requirement to delay the Cooldown.
C. and D. There is no need to enter EOP-2 and therefore no need to enter the E-Plan.

Technical Reference(s): OP 0109 (Attach if not previously

EOP-Study Guide. provided)

Proposed references to be provided to applicants during examination: AP 0156 - Appendix G & H

Learning Objective: (As available)

Question Source: Bank # Lot more

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295001, 2.2.40	
	Importance Rating		4.7

(K&A Statement) Equipment Control: Ability to apply technical specifications for a system. (Partial or Complete Loss of Forced Core Flow Circulation)

Proposed Question: SRO 80

During the preparation for single loop operation following the trip of the "A" Recirculation Pump you are informed that drive flow for " B Recirculation Loop is MORE than it would be if both loops were still in service producing the same amount of core flow.

Which one of the following actions must be taken for these conditions?

A. The reactor must be immediately brought to a Hot Standby condition.

B. Adjustments must be completed on the MCPR and MAPLHGR limits ONLY within the next 24 hours.

C. Adjustments must be completed on the APRM flux scram, APRM rod block settings and the rod block monitor trip settings ONLY within the next 12 hours.

D. Adjustments to the APRM flux scram, APRM rod block settings, rod block monitor trip settings, MCPR and MAPLHGR limits must be initiated within 8 hours following the trip of the " A Recirculation Pump.

Proposed Answer: D.

Explanation (Optional):

D. Correct - T.S. 3.6.G, The reactor may be started and operated or operation may continue with a single recirculation loop provided that the designated adjustments for APRM flux scram setting (Specification 2.1.A.1.a and Table 3.1.1), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit (Specification 1.1.A), and MCPR operating limits and MAPLHGR limits, provided in the Core Operating Limits Report, are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

A. Incorrect – The specified actions of T.S. 3.6.G.1 must be completed within 20 hours following the trip of a Recirculation Loop. If the specified actions are not completed within 20 hours then the plant must be placed in Hot Standby condition.
B. Incorrect – Adjustments are required on the APRM flux scram, APRM rod block settings, rod block monitor trip settings.
C. Incorrect - Adjustments are required on the APRM flux scram, APRM rod block settings, rod block monitor trip settings.

Technical Reference(s): T.S. 2.1.A, 3.2.E, 4.2.E, 3.6.F,3.6.G, 3.11.C, TRM 2,1.B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more
Modified Bank # 293 requal (Note changes or attach parent)
New ______

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295025, 2.4.45	
	Importance Rating		4.3

(K&A Statement) Ability to prioritize and interpret the significance of each annunciator or alarm. (High Reactor Pressure)

Proposed Question: SRO 81

The plant has just shutdown with the following conditions:

- "A" Loop of RHR is operating in Shutdown Cooling.
- The Main Turbine is cooling down on the Turning Gear
- Primary Containment is still being maintained
- Operations to de-inert Primary Containment are in progress

Which of the following annunciators would be the PRIORITY concern and why?

A. 3-L-9, RX PRESS S/D CLG PERMISSIVE ON, clears, because this indicates an entry condition for ON 3156, Loss of Shutdown Cooling.

B. 7-H-6, LIFT PUMP PRESS LO, alarms because this indicates a loss of the Turbine Bearing Lift Pumps.

C. 5-G-1, DRYWELL PRESS HI/LO, low alarm clears because this indicates an entry condition for OT 3111, High Drywell Pressure.

D. 7-H-3, COND VAC LO, alarms because this indicates an entry condition for OT 3120, Condenser High Back Pressure

Proposed Answer: A.

Explanation (Optional):

A. Correct - The loss of the pressure permissive results in the isolation and loss of shutdown cooling requiring entry into ON-3156, Loss of Shutdown Cooling.

B. Incorrect – This annunciator indicates the loss of any lift pump, since the lift pumps lower the power requirements of the turning gear there is no abnormal response other than to investigate and if necessary restart the lift pump or direct maintenance to investigate. Low LO pressure is an entry condition for OT 3115, Reactor Pressure Transients, NOT LIFT PUMP PRESS LO.

C. Incorrect – Drywell Low Pressure clearing does not cause any automatic actuations or isolations to occur. There is ample time to respond to the slowly rising DW pressure. This condition can be easily corrected by restarting DW RRUs. It is NOT an entry condition to OT 3111, High Drywell Pressure.

D. Incorrect – This annunciator indicates a loss of vacuum however in this plant condition there will be no negative effect.

Technical Reference(s): DP 0166, A.2.b.3,(c) pg 8
ARS21001, rev 14 pg 104 of 180.
OP 2124, pg 18
(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: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	WA #	295028, EA2.02	
	Importance Rating		3.9

(K&A Statement) Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Reactor pressure

Proposed Question: SRO 82

A plant transient occurs resulting in a successful reactor scram and appropriate PCIS isolations. The crew is currently implementing EOP-3. The following conditions exist:

- Reactor pressure is 750 psig
- Reactor water level is being maintained above TAF

  Drywell Pressure 25.0 psig and rising.
- Drywell temperature 250°F and rising slowly.
- Torus pressure 24.0 psig and rising slowly.
- Torus level 11.0 ft.
- Torus water temperature is 183°F
- Drywell and Torus H^2 at 0.3% respectively and steady.

For the above, the CRS is required to:

A. Continue in EOP-3 and direct the containment sprays be initiated.

B. Direct the crew to enter EOP-5 RPV-ED and blowdown.

C. Enter SAGs Appendix E, Section 4 to re-assess containment conditions.

D. Enter SAGs Appendix E, Section 7 and direct venting the containment via CAD.

Proposed Answer: A.

Explanation (Optional):

A. Correct – With torus pressure greater than 10 psig and within the DWSIL containment sprays should be initiated. This point is within the DWSIL (drywell temp 350°F, within the max Drywell temp 280°F and within the HCTL curve 1080 psig).
B. Incorrect – RPV blowdown is not required until torus pressure reaches 27 psig. Reactor pressure is above the 500 psig HCTL line but below the 1080 psig line. With Torus water temperature at 183 ° F it is within the safe area of the 1080 psig line for the HCTL curve.
C. and D. Incorrect - There are no entry requirements for the SAGs (H^2entry is not required until 0.5% $H^{2)}$

Technical Reference(s): EOP-3 and associated curves. (Attach if not previously provided)
EOP Study Guide, rev 13, Section 13 pages 13 – 17 of 54

Proposed references to be provided to applicants during examination: EOP 3 DWSIL curve, HCTL curve, PCPL-A, and PSP curves

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295002, AA2.04	
	Importance Rating		2.9

(K&A Statement) Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : Offgas system flow

Proposed Question: SRO 83

The plant is operating at 100% power when the following annunciators alarm:

- 7-H-3, COND VAC LO
- 50-A-2, RECOMBINER INLET TEMP LOW
- 50-B-2, RECOMBINER HIGH PRESSURE DROP
- 50-D-1, PREFILTER HIGH PRESSURE DROP
- 50-H-3, DELAY PIPE FLOW HIGH

An operator at the 9-7 panel reports a slow downtrend on Main Condenser Vacuum.

Given these indications, which one of the following is required?

Enter

A. OT 3120, Condenser Low Vacuum, transfer electrical loads and manually scram the reactor and enter OT 3100.

B. ON 3151, Off Gas Explosion, reset the OG-516A(B) isolation and slowly re-open OG-516A(B) using the controller until desired back pressure is obtained.

C. ON 3151, Off Gas Explosion, commence power reduction per OP 0105, Reactor Operations, at a rate ≤10% RTP/min until vacuum stabilizes or the plant is shut down.

D. OT 3120, Condenser Low Vacuum, lower power at 10% RTP/min using recirc flow until condenser vacuum is less than 5.0 inches HgA, or core flow is 28.5-29.5 Mlbm/hr.

Proposed Answer: D.

Explanation (Optional):

A. Incorrect – Entry requirements for OT-3120 are met however the plant would only be scrammed if condenser vacuum was dropping quickly.

B. Incorrect – None of the annunciators that are entry requirements for ON-3151 are in alarm and OG-516A(B) will NOT have isolated.

C. Incorrect – None of the annunciators that are entry requirements for ON-3151 are in alarm and vacuum must be established at less than 5.0 inches HgA.

D. Correct – These indications, recombiner inlet temp low and high pressure drop as well as the high d/p on the filter and delay pipe flow indicate an excess air flow which would indicate increased main condenser air in-leakage. The low vacuum alarm is an entry condition into OT-3120 and for these conditions reactor power should be lowered while the air in-leakage is investigated.

Technical Reference(s): OT 3120
ARS-21018 (AOG)
ARS-21005, 7-H-3
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	WA #	295017, 2.1.23	
	Importance Rating		4.4

(K&A Statement) Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation. (High Off-site Release Rate)

Proposed Question: SRO 84

The plant is operating at 100% power and moving irradiated fuel in the fuel pool when the following events occur:

- A bundle is dropped and damaged.
- The RM-17-453A/B Rad Monitors trip causing a Reactor Building HVAC isolation and SBGT initiation.

The following annunciators alarm:

- 3-E-3, RX BLDG RAD HI
- 5-H-1 and 5-J-1, RX BLDG/REFUEL FLR CH A(B) RAD HI
- 3-E-4, REFUEL FLOOR RAD HI

Which of the following is required?

A. Enter EOP-4 Secondary Containment and Radioactivity Release Control ONLY then determine Refuel Floor and Reactor Building radiation levels and direct Chemistry to obtain stack release samples.

B. Enter EOP-4 Secondary Containment and Radioactivity Release Control <u>and</u> ON-3152, MSL and Off Gas High Radiation then direct an evacuation of personnel and verify OG-FCV-11, Off Gas to Stack Isolation.

C. Enter ON-3152, MSL and Off Gas High Radiation <u>and</u> ON-3153, Excessive Radiation Levels then re-start Reactor Building HVAC using OE-3107, Appendix AA and direct Chemistry to obtain stack release samples

D. Enter ON-3153, Excessive Radiation Levels <u>and</u> EOP-4 Secondary Containment and Radioactivity Release Control then direct an evacuation of personnel and request Radiation Protection to obtain area dose rates and air samples

Proposed Answer: D.

Explanation (Optional):

A. Incorrect – There are entry requirements for ON-3153 and an evacuation is required.

B. Incorrect - There are no entry conditions for ON 3152, and no requirement to verify OG-FCV-11 isolation for these conditions.

C. Incorrect - There are no entry conditions for ON 3152 and no requirements to re-start RB HVAC at this time or obtain stack samples.

D. Correct - There are entry conditions for these procedures and they direct an evacuation of personnel and request Radiation Protection to obtain area dose rates and air samples.

Technical Reference(s): ARS-21001, for the 9-3 Ann
ON-3153
EOP-4
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ________
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295013, 2.1.25	
	Importance Rating		4.2

(K&A Statement) Ability to interpret reference materials, such as graphs, curves, tables, etc. (High Suppression Pool Temperature)

Proposed Question: SRO 85

Given the following conditions:

- The " B and "D" RHR Pumps are aligned for Torus Pool Cooling at 10,000 gpm.
- Torus Temperature is 180°F rising slowly.
- RPV water Level is -5 inches and lowering slowly.
- Torus pressure is 4 psig.
- NO other ECCS is running.

Which one of the following actions is required?

In accordance with the requirements of...

A. EOP-3, Re-align B RHR Loop to DW Spray at 14,000 gpm flow.

B. EOP-3, Lower B RHR Loop Torus Cooling flow to 9,500 gpm.

C. EOP-1, Re-align B RHR Loop for LPCI mode at 9,500 gpm flow.

D. EOP-1, Re-align B RHR Loop for LPCI mode at 14,000 gpm flow.

Proposed Answer: C.

Explanation (Optional):

C. Correct – Per step EOP-1, RC/L-4 restore and maintain RPV level above 6 in. using Preferred Injection Systems. RHR flow must be lowered because they are above their NPSH limits. The overpressure lines indicate the minimum pressure required for that limit to be valid. , if tours pressure is reading 4 psig, the 0 psig limit must be used. Do not interpolate between the limits.

A. Incorrect – RHR Pumps must be shifted to LPCI mode.

B. Incorrect – RHR Pumps must be shifted to LPCI mode.

D. Incorrect – There is no direction to raise flow and exceed NPSH requirements.

Technical Reference(s):	EOP-1 EOP Study Guide, Rev 13, Section 13, pg 36 of 54	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP-1 RC/L leg and EOP-3 PC/TT and PC/DT legs and NPSH Limits curve.

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	203000, A2.14	
	Importance Rating		3.9

(K&A Statement) Ability to (a) predict the impacts of the following on the RHWLPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Initiating logic failure

Proposed Question: SRO 86

The plant is in Cold Shutdown with all equipment operable. The " D RHR Pump is aligned for Shutdown Cooling. Reactor water level is being controlled at 185" to 187" using CRD make-up and RWCU letdown.

The "A" RHR Logic spuriously initiates and seals in. Which one of the following procedures contains the guidance necessary to prevent RHR injection AND what minimum actions are required?

In accordance with...

A. EN-OP-115, Conduct of Operations, verify adequate reactor water level using two independent indications then place ALL four RHR pump control switches on CRP 9-3 in Pull-to-Lock

B. EN-OP-115, Conduct of Operations, verify adequate reactor water level using two independent indications then place the ONLY RHR Pumps "A" and "C" control switches on CRP 9-3 in Pull-to-Lock

C. OP-2124, Section B, System Shutdown from LPCI Operation, depress the RHR A/C LOGIC LPCI/RECIRC VALVE RESET pushbutton then close OUTBD INJECTION, RHR-27A ONLY.

D. OP-2124, Section B, System Shutdown from LPCI Operation, depress the RHR A/C LOGIC LPCI/RECIRC VALVE RESET pushbutton then close OUTBD INJECTION, RHR-27A and 27B.

Proposed Answer: B.

Explanation (Optional):

A. Incorrect – The pump start logic is one pump in each piping loop per logic. Therefore since only the " A logic initiated only the A and C Pumps would start. The " B RHR Loop is operating in shutdown cooling mode. Taking " D RHR pump to PTL would result in a loss of SDC.

B. Correct – The A and C RHR pumps will start and injection can be terminated by placing their control switches in PTL. The injection valves cannot be closed because they are on timers.

C. and D. Incorrect – the A/C LOGIC LPCI/RECIRC VALVE RESET pushbuttons are used to stop the pumps after the initiation signal has cleared. Both injection valves will receive and open signal from a single logic initiation.

Technical Reference(s): EN-OP-115, Pg. 27 (Attach if not previously provided)
OP-2124, Sect B

Proposed references to be provided to applicants during examination: None

Learning Objective: ________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ________
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	259002, A2.01	
	Importance Rating		3.4

(K&A Statement) Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of any number of main steam flow inputs

Proposed Question: SRO 87

A plant startup is in progress and reactor power is at 50%. Vessel level is being maintained at 160 inches when the "C" Main Steam Line Flow transmitter fails to a minimum (zero) output.

Which one of the following correctly describes Feed Flow / RPV level response AND the initial action required?

A. Feed flow will not be affected by this failure, because the Feedwater Control system is in SINGLE ELEMENT control at this power level. In accordance with 5-E-6, FW CONTROL SYSTEM TROUBLE, initiate a WR.

B. Feed flow will lower, and then maintain level less than 160", but greater than the scram setpoint. In accordance with 5-E-6, FW CONTROL SYSTEM TROUBLE, place the Feedwater Control System in SINGLE ELEMENT control.

C. Feed flow will lower, and then maintain level less than 160", but greater than the scram setpoint. Enter OT-3113, Reactor Low Water Level, Shift RX VESSEL MASTER CONTROLLER (FC-6-83) to MANUAL and restore and control RPV water level.

D. Feed flow will initially lower but then recover to 1 6 0 because on low reactor vessel water level, the reactor vessel water level signal overrides the Steam Flow/Feedwater Flow error signal to increase Feedwater flow. Enter OT-3113, Reactor Low Water Level, and monitor the system response.

Proposed Answer: C.

Explanation (Optional):

C. Correct – Feed flow will lower as less steam demand is sensed by the feedwater control system. The system will compensate by lowering feedwater flow however at this power level and because only one fourth of the demand signal is lost, and the level demand signal being dominant the level will not lower to the scram setpoint. OT-3113 will be entered which directs placing the Feedwater Controller in manual.

A. Incorrect – Feedwater flow will be affected because FWLC is taken from single-element to three-element control at approximately 30% power. Therefore operator actions are required to restore and properly control RPV water level.

B. Incorrect – There is no guidance in the ARS to shift to single element.

D. Incorrect - Feedwater flow will be affected and it will not automatically recover. Actions are required to restore and properly control RPV water level.

Technical Reference(s): OP 0105, Rev 85, (Phase 4B, Step 27, pg. 89)
OT-3113, Rev 22, pg 2
ARS 21003, for 5-E-6
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam Similar to 2007 common 17

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	400000, 2.4.49	
	Importance Rating		4.4

(K&A Statement) Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (Component Cooling Water System)

Proposed Question: SRO 88

With the plant operating at power, a short develops in the RBCCW Drywell Return Valve (V70-117), causing the valve to go fully closed. The valve cannot be re-opened from either the Control Room or locally.

Which one of the following actions is required?

A. Immediately trip both Recirc Pumps and within two minutes scram the reactor and enter OT 3100, Scram Procedure.

B. Immediately lower reactor power as necessary to reduce primary containment heat load and enter OT 3111, High Drywell Pressure.

C. Enter OT 3111, High Drywell Pressure, close N_2 MAKE-UP AC-20 and open the normal Torus vent path isolation valves.

D. Within 2 minutes scram the reactor, enter OT 3100, Scram Procedure, and trip the Recirc Pumps.

Proposed Answer: D.

Explanation (Optional):

A. Incorrect – Two minutes are allowed to recover RBCCW, THEN the plant must be scrammed, then the recirc pumps are tripped.
B. Incorrect – Not required by ON 3147, the initial problem is loss of cooling water to the recirc pump seals.
C. Incorrect – Not required by ON 3147, the initial problem is loss of cooling water to the recirc pump seals.
D. Correct – IAW ON 3147, If RBCCW flow is not restored in two minutes:

1) Scram the reactor,
   a) Enter OT 3100, Scram Procedure
2) Trip both recirc pumps.

Technical Reference(s): ON 3147, Sect. 1, pg 2. (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	262002, 2.1.20	
	Importance Rating		4.6

(K&A Statement) **Ability to interpret and execute procedure steps. (UPS** (AC/DC))

Proposed Question: SRO 89

With the plant operating at 75% power, a leak results in rising drywell pressure.

When drywell pressure exceeds 2.5 psig, the "B" UPS Feeder Breaker opens. Shortly thereafter an AC generator output undervoltage condition occurs. Assume normal power remains available and all other automatic actions occur as designed.

Which one of the following describes how this failure affects the "B" UPS <u>and</u> what action must be directed to comply with the procedure?

The AC generator output undervoltage trip

A. does not effect the UPS output. Position the UPS FDR TRIP, 10A-S36B keylock switch on CRP 9-33 to BLOCK. UPS remains operable.

B. causes a trip of the MCC-89B feed from UPS 1B Breaker. Re-energize MCC-89B from the Control Room by closing the Maintenance Tie Breaker, Declare UPS inoperable.

C. does not effect the UPS output. At CRP 9-3 place UPS-1B control selector keylock switch to OFF and transfer MCC-89B Power from UPS to the Maintenance Tie. Declare UPS inoperable.

D. causes a trip of the MCC-89B feed from UPS 1B Breaker. Position the UPS FDR TRIP, 10A-S36B keylock switch on CRP 9-33 to BLOCK and re-close MCC-89B feed from UPS 1B Breaker. UPS remains operable.

Proposed Answer: B.

Explanation (Optional):

A. Incorrect – UPS output is lost and UPS is inoperable.

B. Correct - there is a trip of the UPS Tie Breaker. The alternate power supplies (maintenance ties) are used to power affected LPCI subsystems in the event of the loss of UPS 1A(1B).

C. Incorrect – UPS output is lost.

D. Incorrect - There is no need to bypass the ECCS trip to lineup the maintenance tie and with the AC generator output undervoltage trip present the MCC-89B feed from UPS 1B Breaker will not close and UPS is inoperable.

Technical Reference(s): OP 2143, Rev. 117, Sect. I, pg 17 of 58 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # Requal 1475 (Note changes or attach parent)

New ______

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 5, 2

Cornments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	WA #	211000, A2.07	
	Importance Rating		3.2

(K&A Statement) Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve closures

Proposed Question: Common 90

During an ATWS the RO places the SLC control switch on CRP 9-5 to "System 1" and "System 2".

- Both pumps Red Lights energized when the respective system was started.
- Both squib valves continuity lights extinguished.
- SLC pump discharge pressure is cycling between 1275 and 1400 psig
- SLC Flow Indicator red light is NOT illuminated.

Based on these indications...
1) Which of the following is the cause? AND
2) What is the appropriate alternate boron injection method from OE-3107, EOP/SAG Appendices?

A. 1) Boron has solidified in the SLC suction piping.
2) Use App. I, Alternate SLC Injection, for local operation of SLC pumps.

B. 1) The SLC Tank Suction Valve, SLC-11, is closed.
2) Use App. O, Alternate Injection Using SLC Test Tank.

C. 1) The SLC Discharge to the Reactor, SLC-18, is closed.
2) Use App. J, Boron Injection using RWCU.

D. 1) Both squib valves have failed to fire.
2) Use App. I, Alternate SLC Injection, to locally fire at least one squib valve.

Proposed Answer: C.

Explanation (Optional):

C Correct – The discharge pressure of the SLC pumps (1275 – 1400 psig) is well above RPV pressure the relief valves providing dead headed relief would cause these indications. Using RWCU would be successful for these conditions because then the discharge is through RWCW.

A. Incorrect – The suction path blocked would not allow pump discharge pressure. This method would still not inject into the reactor.

B. Incorrect - The suction path blocked would not allow pump discharge pressure. This method if used for RPV water level makeup and would not work with SLC-18 closed.

D. Incorrect – Based on squib valves continuity lights being extinguished both squib valves have fired.

Technical Reference(s): P&ID-191171
OE-3107, Rev 17 App I, J, K, and O, EOP-2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	286000, A2.02	
	Importance Rating		3.3

(K&A Statement) Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to actuate when required

Proposed Question: SRO 91

A fire has occurred in the Cable Vault. The CO_2 discharge timer TL-30U-1 starts but fails 20 seconds into the 30 second timeout and does not timeout.

Which one of the following actions is required?

A. Enter OP 2186, Fire Suppression Systems, and place the INITIATE/ABORT switch in 1st SHOT to immediately initiate a CO_2 discharge into the Cable Vault.

B. Enter OP 3020, Fire Emergency Response Procedure, and place the INITIATE/ABORT switch in 1st SHOT to immediately initiate a CO_2 discharge into the Cable Vault.

C. Enter OP 2186, Fire Suppression Systems, and open a manual cardox valve on one of the CARDOX bottles to start the pneumatic timer allowing 70 seconds before CO_2 discharge into the Cable Vault.

D. Enter OP 3020, Fire Emergency Response Procedure, and open a manual cardox valve on one of the CARDOX bottles to start the pneumatic timer allowing 70 seconds before CO_2 discharge into the Cable Vault.

Proposed Answer: C.

"D" also correct
TF 3/6/09

Explanation (Optional):

C. Correct – OP 2186, gives detailed instructions for the operation of fire suppression systems. Since the electrical timer failed the C02 system will remain waiting for the timer to complete its 30 sec time-out. This timer can be bypassed by manually opening one of the C02 bottles and starting the pneumatic timer. Placing the INITIATEIABORT switch in 1st SHOT starts the electric timer which has failed.

A. Incorrect – Placing the INITIATEIABORT switch in 1st SHOT starts the electric timer which has failed so no initiation will occur. Additionally this timer then opens valves on two C02 bottles which start another 70 sec timer, so any initiation would not occur immediately.

B. Incorrect – OP 3020, Fire Emergency Response Procedure, contains checklists for control room actions during the fire and assume normal system operation. Placing the INITIATEIABORT switch in 1st SHOT starts the electric timer which has failed so no initiation will occur. Additionally this timer then opens valves on two C02 bottles which start another 70 sec timer, so any initiation would not occur immediately.

D. Incorrect – OP 3020, Fire Emergency Response Procedure, contains checklists for control room actions during the fire and assume normal system operation.

Technical Reference(s): OP 2186, Sect A, Pgs 15- 22 (Attach if not previously provided)
OP 3020, App W.

Proposed references to be provided to applicants during examination: None

Learning Objective: ________________ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	268000, 2.2.36	
	Importance Rating		4.2

(K&A Statement) Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (Radwaste)

Proposed Question: SRO 92

The plant is starting up and is at 30% power when Maintenance requests a clearance on Drywell Floor Drain Isolation Valve, LRW-82. The clearance will require the following:

- Closing the Instrument Air (IA) isolation valve to LRW-82
- Isolating the power supply to the IA supply solenoid

Which one of the following LCOs must be entered?

A. one (1) hour or the plant must immediately commence a shutdown and be in cold shutdown within 24 hours.

B. twenty four (24) hours or the plant must immediately commence a shutdown and be in cold shutdown within 24 hours.

C. six (6) days or the plant must immediately commence a shutdown and be in cold shutdown within 24 hours.

D. seven (7) days or the plant must immediately commence a shutdown and be in cold shutdown within 24 hours.

Proposed Answer: D.

Explanation (Optional):

D. correct - Isolating the air to 20-82 will cause the valve to close. This meets the requirement of T.S. 3.7.D.2 that reactor power operation may continue provided one containment isolation valve in each line having an inoperable valve is in the mode corresponding to the isolated condition. In this case since 20-82 is failed closed it satisfies the T.S. requirement for one valve closed in the line and therefore T.S. 3.6.C.2 applies which requires the sumps to operable. Both the sump and air sampling systems shall be operable during power operation. From and after the date that one of these systems is made or found inoperable for any reason, reactor operation is permissible only during succeeding seven days. If this specification is not met then an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

A. B. and C. Incorrect – The plant may operate for 7 days before starting the shutdown.

Technical Reference(s): P&ID 191177, Sht 1
ON-3146, Rev 20, App A, pg 8
T.S. Table 4.7.2.A., 3.7.D.2.
3.6.C.2. & bases pgs 164 & 166

(Attach if not previously provided)

Proposed references to be provided to applicants during examination: T.S. Sections 3.6.C and 3.7.D

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	204000, 2.4.6	
	Importance Rating		4.0

(K&A Statement) Emergency Procedures / Plan: Knowledge symptom based EOP mitigation strategies. (RWCU).

Proposed Question: SRO 93

The plant was operating at 100% power when a leak in the Reactor Building from the RWCU system produced the following conditions:

Reactor is shutdown with all rods fully inserted.

- PCIS Groups 2, and 3 have isolated; Group 5 Isolation valves cannot be closed.

Station electrical loads are being supplied by the Start-Up Transformers.

Reactor Building Temperatures are as follows:

RB East 252' at 147°F and rising
RB East 280' at 145°F and rising
RB NW 280' at 153°F and rising
RB East 303' at 155°F and rising

	Parameters and Areas		Maximum Normal Operating Limit	Maximum Safe Operating Limit	Value/Trend
	Area / Location	**Indicator**	**°F**	**°F**	
TEMPERATURE	**NE Corner Room**				
	NE Corner Room 213'	Channel 1	104	194	
	NE Corner Room 232'	Channel 2	104	194	
	SE Corner Room				
	SE Corner Room 213'	Channel 3	104	194	
	SE Corner Room 232'	Channel 4	104	194	
	Torus Area				
	Torus NW 213'	Channel 5	120	250	
	Torus SW 213'	Channel 6	120	250	
	Torus NE 213'	Channel 7	120	250	
	Torus SE 213'	Channel 8	120	250	
	RB 252'				
	RB East 252'	Channel 9	106	160	
	RB NE 252'	Channel 10	106	160	
	RB NW 252'	Channel 11	106	160	
	RB SW 252'	Channel 12	106	160	
	RB 280'				
	RB East 280'	Channel 13	106	160	
	RB NW 280'	Channel 14	106	160	
	RB SW 280'	Channel 15	106	160	
	RB 303'				
	RB East 303'	Channel 16	106	150	

For these conditions, which one of the following is required?

A. Enter EOP-5, RPV-ED and open all SRVs.

B. Rapidly depressurize the RPV using Bypass Valves irrespective of cooldown rate.

C. Begin a normal plant cooldown per OP-0105 using the Bypass Valves.

D. Depressurize the RPV using SRVs without exceeding the Technical Specification cooldown rate limit.

Proposed Answer: B.

Explanation (Optional):

B. Correct – only one area (303') is above its Max Safe limit, but the others are approaching their limits. If blowdown is anticipated then the CRS should direct depressurizing with the BPVs per EOP-1.

A. Incorrect - RPV-ED is not yet required due to the same parameter not being above the Max Safe level in more than one area.

C. Incorrect - A rapid depressurization is warranted per EOP-1 RC/OR-4

D. Incorrect – A rapid depressurization is warranted at this time per EOP-1 RCIOR-4. SRVs are not warranted at this time.

Technical Reference(s): EOP-1 EOP-4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Requal #1185 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Similar to 2007 SRO 77

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.4.1

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.39	
	Importance Rating		4.3

(K&A Statement) Knowledge of conservative decision making practices.

Proposed Question: SRO 94

The plant is in Cold Shutdown, HPCI was tagged out of service per EN-OP-102, Protective and Caution Tagging. Because of this tagout, HPCI status is controlled by EN-OP-102.

- AO #1 is removing the HPCI tagout in accordance with the 'Tags To Be Removed" sheet.
- This tagout removal will align HPCI valves for normal operations and HPCI status control will be transferred back to AP 0155, Current System Valve and Breaker Lineup and Identification.
- Due to manpower constraints AO #2 has been assigned to concurrently verify the tagout removal.

IAW EN-OP-012, what action is permitted for completing the Tag Out Removal and transferring HPCI status control back to AP 0155?

A. AO #2 CAN sign the "Restoration 2nd Verification" box in eSOMS. The valve restoration activities completed by the two operators MAY be considered adequate completion of the system line-up.

B. AO #2 CANNOT sign the "Restoration 2nd Verification" box in eSOMS. Before the system valve line-up can be considered complete an Independent Verification of the HPCI lineup MUST be performed by another operator.

C. AO #2 CAN sign the "Restoration 2nd Verification" box in eSOMS. Before the system valve line-up can be considered complete an Independent Verification of HPCI lineup MUST be performed by another operator.

D. AO #2 CANNOT sign the "Restoration 2nd Verification" box in eSOMS. Another operator MUST perform an independent verification of the tagout removal and MUST perform an Independent Verification of the HPCI lineup.

Proposed Answer: A.

Explanation (Optional):

A. Correct - The tag out removal can be performed with concurrent verification per Attachment 9.1 of EN-OP-102. IAW AP 0155, System status must be verified/established prior to declaring a system or component operable. The system/component status is established by performing either; a complete system valve lineup, or by completing Restoration Conditions if the systemlcomponent status is under the control of EN-OP-102. (LER9820R1_04)

B. Incorrect - AO that observes the tagout removal can sign the "Restoration 2nd Verification" box in eSOMS. The independent verification of the system line-up is not required.

C. Incorrect - The tag out removal can be performed with concurrent verification and this concurrent verification can be used to transfer status control to AP 0155.

D. Incorrect - The tag out removal can be performed with concurrent verification. The independent verification is not required.

Technical Reference(s): AP-0155, Discussion pg 4
EN-OP-102, Sect. 5.20, pg 31
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.2.35	
	Importance Rating		4.5

(K&A Statement) Ability to determine Technical Specification Mode of Operation.

Proposed Question: SRO 95

Vermont Yankee is shutdown with the following conditions:

- Reactor coolant temperature is 185°F and steady
- Maintenance is preparing to install the steam dryer assembly.
- All the requirements for Secondary Containment are met.
- The Reactor Mode Switch is in SHUTDOWN
- I&C has permission to perform Mode Switch Interlock Functional Testing.

Which of the following is correct when the reactor mode switch is placed in START/HOT STANDBY position?

The actual Reactor Mode is..

A. Startup/Hot Standby and all normal requirements for transferring to Hot Standby Mode must be met.

B. Cold Shutdown but all normal requirements for transferring to Hot Standby Mode must be met.

C. Cold Shutdown and current plant conditions allow this operation provided control rods are verified to be fully inserted.

D. Startup/Hot Standby and current plant conditions allow this operation provided control rods are verified to be fully inserted.

Proposed Answer: C.

Explanation (Optional):

C. Correct - The reactor mode switch may be changed to either the Run or Startup/Hot Standby position, and operation not considered to be in the Run Mode or Startup Mode, to allow testing of instrumentation associated with the reactor mode switch interlock functions, provided:
1. Reactor coolant temperature is < 212°F 2. All control rods remain fully inserted in core cells containing one or more fuel assemblies; and
3. No core alterations are in progress.
A. and D. Incorrect – The plant is still in Cold Shutdown
B. Incorrect - Requirement is all rods verified fully inserted

Technical Reference(s): T.S., 3.7.C, Note at bottom of pg 155A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: TS Section 3.7.C

Learning Objective: (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.3.11	
	Importance Rating		4.3

(K&A Statement) Ability to control radiation releases.

Proposed Question: SRO 96

Per OP 2610, Liquid Waste Disposal, which one of the following is MINIMUM required before the Shift Manager can authorize the BATCH Discharge of a Waste Sample Tank to the Condensate Storage Tank?

A. Sample the tank to verify it is within the limits of OP-2610 and then complete a discharge permit, VYOPF 2610.03, which must be approved by Chemistry and an SRO on shift ONLY.

B. Sample the tank to verify it is within the limits of OP-2610 and then complete a discharge permit, VYOPF 2610.03, which must be approved by an SRO on shift ONLY.

C. Chernistry must sample the tank prior to and during the discharge and then complete a discharge permit, VYOPF 2610.03, which must be approved by the Chemistry Manager and the Shift Manager ONLY.

D. Chemistry must sample the tank prior to and during the discharge and then complete a discharge permit, VYOPF 2610.03, which must be approved by the Operations Manager ONLY.

Proposed Answer: A.

Explanation (Optional):

A. Correct – IAW OP-2610 the tank must be sampled before it is pumped to verify the limits on VYOPF 2610.03 are met then a discharge permit must be signed by Chemistry and the Shift Manager or an SRO.

B. Incorrect – Any SRO on shift can sign the discharge permit.

C. and D. A continuous discharge should be initiated by the Operations Manager and approved by the Chemistry Manager; they are not required for a "Batch" discharge. Additionally continuous sampling (or once per shift) is only required for a "Continuous" discharge.

Technical Reference(s): OP-2610, Section A.3 and (Attach if not previously

VYOPF 2610.03, rev 27 provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______ Lot more

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.49	
	Importance Rating		4.4

(K&A Statement) Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question: SRO 97

A plant startup is in progress. Reactor power is 390 MWt and plant loads are on the Auxiliary Transformer. Reactor pressure is 950 psig. The CRO reports EPR Stroke is failing to zero.

As the CRS you must direct entry into:

A. OT 3115, Reactor Pressure Transients, Verify pressure lowers until MPR takes control, take the EPR to cutout, and stop power increase.

B. OT 3110, Positive Reactivity Insertion, Verify pressure lowers until MPR takes control, take the EPR to cutout, and continue power increase.

C. OT 3115, Reactor Pressure Transients, Verify pressure rises until MPR takes control, take EPR to cutout, lower MPR setpoint, and stop power ascension.

D. OT 3110, Positive Reactivity Insertion, Verify pressure rises until MPR takes control, take EPR to cutout, lower MPR setpoint, and continue power ascension.

Proposed Answer: C.

Explanation (Optional):

A. and B. Incorrect -Because pressure will increase to the MPR setpoint when the EPR stroke fails to zero and for B. Because with reactor power < 25% RTP with the EPR out of service thermal limits should be considered suspect and power should not be increased above 25% RTP.

C. Correct Response. – Enter OP-3115, Verify pressure lowers until MPR takes control, take the EPR to cutout, and stop power increase until core limits are verified.

D. Incorrect – Because with reactor power < 25% RTP with the EPR out of service thermal limits should be considered suspect and power should not be increased above 25% RTP.

Technical Reference(s): OT 3115, rev 9, Follow-Up (Attach if not previously

Action 2.a provided)

OP 0105, Rev. 11, Phase 4, Precaution 13

Proposed references to be provided to applicants during examination: None

Learning Objective: LOT-00-602, SRO 6 (As available)

Question Source: Bank # 5694 Lot more

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.4.20	
	Importance Rating		4.3

(K&A Statement) Knowledge of operational implications of EOP warnings, cautions, and notes.

Proposed Question: SRO 98

An ATWS has occurred and reactor water level has been deliberately lowered for power control. The following conditions exist:

- Initial SLC Tank level was 86% and is now 71% and lowering
- Reactor pressure is 880 psig and rising slowly
- Reactor water level has been lowered to TAF
- Drywell pressure is 1.8 psig and steady
- CST level is 10% and lowering slowly
- Torus water level is 11.0 ft. and steady
- Torus temperature is 140°F and rising slowly

In accordance with EOP-2, which one of the following actions is required?

A. Restore HPCI injection using OE 3107, Sect GG with suction from the Torus and control RPV water level between 6" and 90".

B. Restore HPCI injection using OE 3107, Sect GG with suction from the Torus and control RPV water level between -19 and 177".

C. Place Feedwater in service by adjusting controllers to throttle FDW-12A/B and/or FDW-13 as necessary to control RPV water level between 127" and 177".

D. Place Feedwater in service by adjusting controllers to throttle FDW-12A/B and/or FDW-13 as necessary to control RPV water level between -19" and the level at which power went below 2%.

Proposed Answer: C.

Explanation (Optional):

C. Correct IAW the note in EOP-2 Hot Shutdown Boron Weight is 15% of the SLC tank volume so level can be restored to 127 – 177 using preferred ATWS injection system. Although HPCI is a preferred system a Caution states Operation of HPCI or RCIC turbines with suction temperatures above 140°F may result in equipment damage. Therefore HPCI should not be started and Feedwater used.

A. and B. Incorrect - Although HPCI is a preferred system a Caution states Operation of HPCI or RCIC turbines with suction temperatures above 140°F may result in equipment damage. Therefore HPCI should not be started and Feedwater used.

D. Incorrect - IAW the note in EOP-2 Hot Shutdown Boron Weight is **15%** of the SLC tank volume so level can be restored to 127 – 177

Technical Reference(s): EOP-2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP-2, Level leg

Learning Objective: ________ (As available)

Question Source: Bank # ________ Lot more

Modified Bank # ________ (Note changes or attach parent)

New X

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge ________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ________

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	WA #	**2.2.21**	
	Importance Rating		**4.1**

(K&A Statement) Knowledge of pre and post-maintenance operability requirements.

Proposed Question: SRO 99

RPS "A" System has been tripped for replacement of backup scram relay contacts. The work party has informed the Shift Manager that work is complete and the equipment can be returned to service.

IAW AP **0125,** the RPS A trip can be reset under __(1)__ to allow the post maintenance testing required to demonstrate its operability provided that __(2)__ is completed first.

A. (1) administrative controls,
(2) all other required testing

B. (1) administrative controls,
(2) all other preventive maintenance

C. **(1)** Test & Maintenance Tags
(2) all other required testing

D. **(1)** Test & Maintenance Tags
(2) all other preventive maintenance

Proposed Answer: A.

Explanation (Optional):

A. Correct – Per Sect. 5.2 of AP-0125, Technical Specification related equipment or components removed from service or declared inoperable may be returned to service under administrative control solely to perform post maintenance testing (PMT) required to demonstrate its operability. All other required checks that support verifying operability shall be completed prior to returning the equipment or component to service under administrative control. Following successful completion of the PMT under administrative controls, the equipment/components can be considered fully operable.

B. Incorrect – All other testing must be completed. Preventive maintenance can be deferred provided appropriate approvals are obtained.

C. and D. Incorrect – RPS may be returned to service under administrative control and for D. All other testing must be completed. Preventive maintenance can be deferred provided appropriate approvals are obtained.

Technical Reference(s): AP 0125, rev 12, Admin Limit 5.2 (Attach if not previously provided)

Proposed references to be provided to applicants durirlg examination: None

Learning Objective: ________ (As available)

Question Source: Bank # 5709 Lot more

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam No

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 ________

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.25	
	Importance Rating		4.2

(K&A Statement) **Ability to** interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: SRO 100

The plant experienced a large LOCA. Adequate core cooling has been established in accordance with applicable EOPs.

deleted from exam
TF 3/6/09

The following plant conditions currently exist:

- Reactor pressure is 170 psig and steady
- Drywell pressure is 23.3 psig and rising slowly
- Torus pressure is 22.1 psig and rising slowly
- Torus level – 11.8 ft and steady
- DW Temp 190°F and steady

Using the EOP charts determine the LOWEST pressure at which containment integrity could no longer be assured?

A. 27 psig Torus pressure

B. 62 psig Torus pressure

C. 77 psig Torus pressure

D. +1.5 psid DW/Torus Differential Pressure

Proposed Answer: A.

Explanation (Optional):

A. Correct - The PSP pressure limit is based on the maximum pressure that can exist in the Torus that will prevent exceeding PCPL during a blowdown. Since RPV pressure is above 50 psid with the Torus a blowdown could exceed the PCPL based on PSP.
B. Incorrect - For Vermont Yankee s PCPL- , the maximum pressure (62 psig) at which the primary containment vent valves can be opened and closed however the lowest pressure is 27 psig.
C. Incorrect – This pressure is above the design limit for the Drywell based on the upper limit of the PCPL-A for water levels above 57 feet.
D. Incorrect – The Torus Level Limit is based on the TS value for minimum and maximum torus level and is concerned with the ability of the torus to withstand a LOCA prior to the LOCA and is not concerned with Containment capabilities in these conditions.

Technical Reference(s): EOP 3 Basis document
Must integrate the following:
PCPL-A graph maintains containment integrity.
T.S. Bases 3.7. A
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP Graphs for PCPL-A, PSP, Torus Level Limit

Learning Objective: LOT-00-607, SRO 4 (As available)

Question Source: Bank # 5652 Lot more
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41
55.43 5, 2

Comments: