RO Written Exam Reference Index

1. 0EOP-01-UG, User's Guide, Attachment 5, Figure 3, Heat Capacity Temperature Limit
2. 0EOP-01-UG, User's Guide, Attachment 5, Figure 5, Core Spray NPSH Limit
3. 0EOP-01-UG, User's Guide, Attachment 5, Figure 6, RHR NPSH Limit
4. 0EOP-01-UG, User's Guide, Attachment 5, Figure 7, Pressure Suppression Pressure

1. 201001 1

With Unit Two operating at rated power which one of the following choices completes the statements below concerning the *ROD DRIFT* annunciator for a selected control rod?

The *ROD DRIFT* annunciator will be received when an ____(1)____ reed switch position is detected with no motion signal demanded.

The appropriate action to take IAW the APP for one drifting control rod is to ____(2)____.

A. 1) odd-numbered
   2) insert a manual reactor scram

B. 1) odd-numbered
   2) attempt to arrest the drift and latch the control rod

C. 1) even-numbered
   2) insert a manual reactor scram

D. 1) even-numbered
   2) attempt to arrest the drift and latch the control rod

Answer: B

K/A: 201001 Control Rod Drive Hydraulic System
G2.04.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Objective: LOI-CLS-LP-302-B, Obj. 4 - Given plant conditions, determine the required supplementary actions IAW 0AOP-02, Control Rod Malfunction/Misposition.

Reference: None

Cog Level: High

Explanation: The rod drift alarm is actuated when no motion is demanded and an odd numbered reed switch is detected. If a rod is drifting the APP attempts to latch the rod by giving it an insert or withdraw signal as appropriate. If more than one rod is drifting then a reactor manual scram is inserted.

Distractor Analysis:

Choice A: Plausible because the first part of the answer is correct and if more than one rod is drifting then a reactor manual scram is required.

Choice B: Correct Answer, see explanation

Choice C: Plausible because control rods are latched on even numbered positions and a reactor scram is inserted if more than one rod is drifting.

Choice D: Plausible because control rods are latched on even numbered positions and the second part of the answer is correct.

SRO Basis: N/A

ROD DRIFT

CAUSE

1. Rod in uneven position due to:
   - Leaking scram valve
   - High cooling water pressure
   - Failure of directional control valves
   - Slow to settle due to fuel bundle channel bow
2. Malfunction in alarm circuit

ACTIONS

1. Determine if affected control rod(s) is drifting or if rod(s) has scrammed using full core display, RPIS, and RWM.
2. If more than one control rod is drifting, manually scram the reactor and refer to 1EOP-01-RSP.
3. If more than one control rod has scrammed, then perform the following:
   a. If reactor power is less than or equal to 25%, then perform the following:
      (1) Manually scram reactor.
      (2) Refer to 1EOP-01-RSP.
   b. If reactor power is greater than 25%, and nine or more rods have scrammed, then perform the following:
      (1) Manually scram reactor.
      (2) Refer to 1EOP-01-RSP.
   c. If the sum of scrammed and inoperable control rods is greater than eight, then refer to Technical Specification 3.1.3 for shutdown requirements.
   d. If reactor power is greater than 25% and the sum of scrammed and inoperable control rods is less than nine, then refer to 0AOP-02.0, Supplementary Actions.
4. Select the drifting rod and determine direction of drift.
5. Attempt to arrest the drift and latch rod by giving appropriate insert or withdrawal signals to the rod using RMCS controls and bypassing RWM if necessary.

2. 203000 1

Which one of the following identifies the power supply to 2C RHR Pump?

A. E1

B. E2

C. E3

D. E4

Answer: A

K/A: 203000 Residual Heat Removal /Low Pressure Coolant Injection: Injection Mode
K2.01 - Knowledge of electrical power supplies to the following: Pumps (CFR: 41.7)

RO/SRO Rating: 3.5 / 3.5

Objective: LOI-CLS-LP-017-A Obj. 17a - List the normal and emergency power sources for the following: RHR Pumps

Reference: None

Modified question 230000_4 which was used on the 10-2 exam. Changed pumps to 2C which changed the answer and removed the second question of the power supply to the e bus.

Cog Level: Low

Explanation: 2C RHR pump is a Div I pump with a power supply from E1

Distractor Analysis:

Choice A: Correct Answer, see explanation

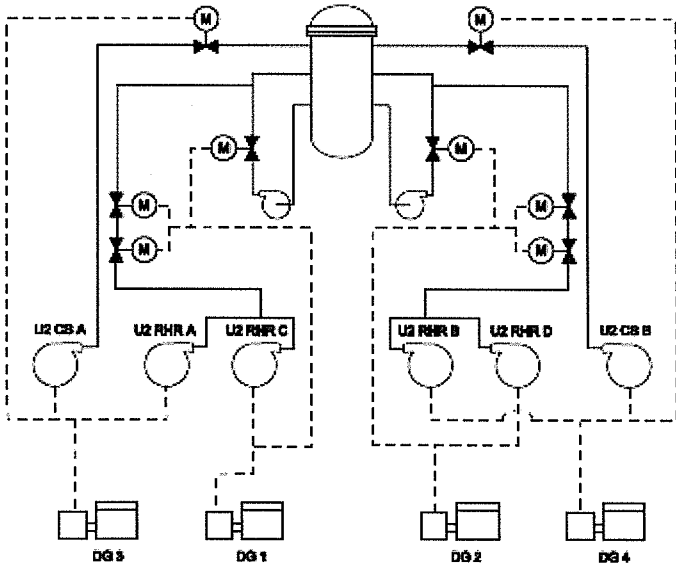
Choice B: Plausible because E2 is a Unit One bus that supplies power to Unit One and Unit Two loads. RHR Pumps 1D and 2D are supplied from this bus.

Choice C: Plausible because E3 is a Unit Two bus that supplies power to Unit One and Unit Two loads. RHR Pumps 1A and 2A are supplied from this bus.

Choice D: Plausible because E4 is a Unit Two bus that supplies power to Unit One and Unit Two loads. RHR Pumps 1B and 2B are supplied from this bus.

SRO Basis: N/A

UNIT 2 LOW PRESSURE ECCS



NOTE: INJECTION FLOW PATH AND POWER SUPPLIES SHOWN.
LOGIC & OTHER FLOW PATHS NOT SHOWN.

3. 205000 1

RHR Loop A is operating in the Shutdown Cooling mode of operation. It becomes desired to reduce the reactor cooldown rate.

Which one of the following identifies the action necessary to reduce the cooldown rate IAW 2OP-17, Residual Heat Removal System Operating Procedure?

A. Throttle open E11-F003A, HX 2A Outlet Valve.

B. Throttle closed E11-F047A, HX 2A Inlet Valve.

C. Throttle open E11-F048A, HX 2A Bypass Valve.

D. Throttle closed E11-F017A, Outboard Injection Valve.

Answer: C

K/A: 205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)
A1.10 - Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: Throttle valve position (CFR: 41.5 / 45.5)

RO/SRO Rating: 3.0 / 2.9

Objective: LOI-CLS-LP-017, Obj. 15 - Describe how the reactor cool down rate is controlled when the RHR system is in the Shutdown Cooling mode.

Reference: None

Cog Level: Low

Explanation: The procedure allows throttling closed the F003 or F068 and throttling open the F048. Throttling open the F048 will reduce the flow through the heat exchanger thereby bypassing the water preventing it from being cooled down.

Distractor Analysis:

Choice A: Plausible because the F003 would be throttled closed to reduce the cooldown.

Choice B: Plausible because throttling this valve closed would reduce the cooldown, but it is not allowed IAW the procedure.

Choice C: Correct Answer, see explanation

Choice D: Plausible because throttling this valve closed would reduce the cooldown, but it is not allowed IAW the procedure.

SRO Basis: N/A

From OP-17 Section 8.12

4. **IF** a lower cooldown rate is desired, **THEN PERFORM** the following, as necessary, for each operating RHR loop while maintaining desired flow rate, **NOT** to exceed 10,000 gpm per loop:

> **CAUTION**
>
> **IF** *E11-F003A(B)* is closed, **THEN** *RHR HEAT EXCHANGER 2A(2B)* inlet temperature, located on *E41-TR-R605*, Point *1(2)*, is **NOT** a valid indication of reactor coolant temperature.

a. **SLOWLY THROTTLE CLOSE** *HX 2A(2B) OUTLET VLV, E11-F003A(B)*, as necessary. ☐

b. **THROTTLE CLOSED** *HX 2A(2B) SW DISCH VLV, E11-PDV-F068A(B)*, as necessary, to reduce RHRSW flow rate. ☐

c. **SLOWLY THROTTLE OPEN** *HX 2A(2B) BYPASS VLV, E11-F048A(B)*, as necessary, maintaining RHR flowrate greater than 4500 gpm. ☐

4. 205000 2

Unit Two is in Mode 3 when a loss of shutdown cooling occurs. The operating crew is performing alternate shutdown cooling with the SRVs IAW 0AOP-15.0, Loss of Shutdown Cooling.

Reactor pressure is 205 psig above torus pressure.

Which one of the following identifies the status of cooldown rate and the operator actions required IAW 0AOP-15.0?

Cooldown rate is ____(1)____ and the operator is required to ____(2)____.

A. (1) inadequate
   (2) start additional RHR pumps

B. (1) excessive
   (2) secure the running RHR pump(s)

C. (1) inadequate
   (2) open an additional SRV

D. (1) excessive
   (2) close SRVs

Answer: C

K/A: 205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)
A2.12 - Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow (CFR: 41.5 / 45.6)

RO/SRO Rating: 2.9/3.0

Objective: LOI-CLS-LP-302-L, Obj. 07c - State the reason(s) for the following actions taken during the use of Alternate Shutdown Cooling: Opening an additional SRV if reactor pressure rises to >164 psig above Suppression Chamber pressure.

Reference: None

Cog Level: High

Explanation: Loss of shutdown cooling is addressed in 0AOP-15 which provides for multiple methods of providing decay heat removal. Using alternate shutdown cooling with the SRVs requires raising reactor water level to the steam lines and opening an SRV. A low pressure pump (preferably RHR) is then used to inject into the reactor with the coolant flowing to the torus via the open SRV. RHR pumps will deadhead at approximately 165 psig. IAW 0AOP-15, if pressure reaches shutoff head, an additional SRV must be opened. This relieves vessel pressure, allowing additional flow and preventing deadheading of the RHR pumps thereby maintaining decay heat removal.

Distractor Analysis:

Choice A: Plausible because high pressure will deadhead the RHR pump and result in inadequate cooling. If pumps were not deadheaded, starting an additional pump would increase flow and cooldown rate.

Choice B: Plausible because high pressure will increase the flow through the SRV. If cause of pressure increase was due to to excessive RHR pump flow, and shutoff head were not reached, then excessive cooldown could be achieved and stopping an RHR pump might be the appropriate action.

Choice C: Correct answer, see explanation.

Choice D: Plausible because high flow could increase cooldown rate. Closing the SRV would stop cooling flow.

SRO Basis: N/A

AOP-15.0

This procedure addresses a loss of normal decay heat removal capability during shutdown conditions. The procedure provides contingencies for the following methods of decay heat removal:

- RHRSW loop failure
- RHR loop failure
- Condenser cooling failure
- Feed and bleed combinations
- Alternate shutdown cooling with SRVs

3.2.14 **IF ALL** of the above methods can **NOT** maintain reactor vessel coolant temperature less than 212°F, **THEN INITIATE** alternate shutdown cooling with the SRVs as follows:

	RHR A/C	RHR B/D	CS A	CS B
HIGHEST **<u>COOLDOWN</u>**	B21-F013F B21-F013H	B21-F013A B21-F013B	B21-F013K	B21-F013E B21-F013L
	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D
	B21-F013A B21-F013B B21-F013K	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013A B21-F013B B21-F013K
	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J
LOWEST **<u>COOLDOWN</u>**	B21-F013E B21-F013L	B21-F013K	B21-F013A B21-F013B	B21-F013F B21-F013H

9. **PLACE** the control switch for the desired SRV to *OPEN*. ☐

10. **RAISE AND MAINTAIN** reactor water level greater than 254 inches. ☐

13. **IF** reactor pressure can **NOT** be maintained less than 164 psig above Suppression Chamber pressure, **THEN** **PLACE** another SRV control switch to *OPEN*. ☐

5. 206000 1

Unit Two is operating at rated power when the circuit breaker on 120V Distribution Panel 32A to the Div I Steam Leak Detection Numacs trips.

Which one of the following identifies the effect of this condition on the HPCI system?

A. E41-F002, HPCI Inboard Steam Line Isolation Valve, will isolate.

B. E41-F003, HPCI Outboard Steam Line Isolation Valve, will isolate.

C. E41-F002, HPCI Inboard Steam Line Isolation Valve, area temperature isolation signal is disabled.

D. E41-F003, HPCI Outboard Steam Line Isolation Valve, area temperature isolation signal is disabled.

Answer: D

K/A: 206000 High Pressure Coolant Injection System
K6.10 - Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM: PCIS: BWR-2,3,4 (CFR: 41.7 / 45.7)

RO/SRO Rating: 3.8/4.0

Objective: LOI-CLS-LP-019, Obj. 03q - Given plant conditions, predict how the HPCI System will respond to the following events: PCIS failure

Reference: None

Cog Level: High

Explanation: The Group 4 isolation logic contains two isolation trip systems. Trip System A provides isolation signals to the Outboard steam line isolation and torus suction valve; trip system B operates the Inboard isolation valves. The NUMACs provide indication and trip channels for RWCU, HPCI and RCIC leak detection. The NUMACs are powered from 120 Vac E bus power; Div. I from 31A (32A) and Div. II from 31B (32B). On a loss of power, the NUMAC output relays for RWCU are de-energize to trip and will initiate a RWCU isolation. The HPCI/RCIC output relays are energize to trip and will not cause a HPCI or RCIC isolation on a loss of AC power.

Certain ASSD procedures and the SBO procedure requires HPCI be maintained available for safe shutdown under conditions of elevated reactor building temperature due to fire and/or loss of ventilation. The steam leak detection inputs to the isolation logic are defeated by turning off the 120 Vac power supply breaker, 31A (32A) and 31B (32B) to the NUMAC steam leak detection modules.

Distractor Analysis:

Choice A: Plausible because if the HPCI isolation was de-energize to actuate (as is the RWCU system) then this would be correct. 32A provides to outboard not inboard.

Choice B: Plausible because if the HPCI isolation was de-energize to actuate (as is the RWCU system) then this would be correct.

Choice C: Plausible because if the loss was 32B then this would be correct.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

The isolation system is divided into two logic systems, Logic Bus A and Logic Bus B. Logic Bus A isolates the outboard valves, E41-F003 and E41-F041, while Logic Bus B isolates the inboard valves, E41-F002 and E41-F042. A Manual Isolation pushbutton is provided on Panel P601 to permit the operator to insert an isolation should plant conditions require it. The Manual Isolation only initiates a Logic A isolation and is only in effect when a Reactor Low Level Two or a High Drywell Pressure signal is present.

Power supplies are as follows:

Inboard Isolation Logic	125 VDC	Pnl. 3B(U1),4B(U2)
Outboard Isolation Logic	125 VDC	Pnl. 3A(U1),4A(U2)
Leak Detection Logic Div. I	120 VAC	Pnl. 31A(U1),32A(U2)
Leak Detection Logic Div. II	120 VAC	Pnl. 31B(U1),32B(U2)

The Steam Leak Detection System for RWCU, HPCI and RCIC consists of four NUMAC microprocessor units located in the Control Room back panel area (additional information on the operation of the NUMAC microprocessors is in Section 3). The NUMACs provide indication and trip channels for RWCU, HPCI and RCIC leak detection. In addition the NUMAC output signals interface with the ERFIS and Process Computer for various calculations and monitoring of plant parameters (i.e., Heat Balance). The NUMACs are powered from 120 Vac E bus power; Div. I from 31A (32A) and Div. II from 31B (32B). On a loss of power, the NUMAC output relays for RWCU are de-energize to trip and will initiate a RWCU isolation. The HPCI/RCIC output relays are energize to trip and will not cause a HPCI or RCIC isolation on a loss of AC power.

Steam leak detection for the Unit 1 Main Steam system is accomplished by temperature switches located along the main steam lines. A total of 16 temperature switches monitor the main steam lines for steam leaks in the Reactor Building (MSIV pit) and Turbine Building steam tunnels. Four

6. 209001 1

A transient has occurred on Unit One causing the following plant conditions:

Drywell pressure	12 psig
Reactor water level	65 inches
Reactor pressure	360 psig

Which one of the following choices completes the statement below?

Core Spray A Inboard Injection Valve (E21-F005A) is ___(1)___, and Min Flow Bypass Valve (E21-F031A) is ___(2)___.

A. (1) open
   (2) open

B. (1) open
   (2) closed

C. (1) closed
   (2) open

D. (1) closed
   (2) closed

Answer: A

K/A: 209001 Low Pressure Core Spray System
K4.08 - Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Automatic system initiation (CFR: 41.7)

RO/SRO Rating: 3.8/4.0

Objective: LOI-CLS-LP-018, Obj. 07 - Given plant conditions, determine if the Core Spray System should automatically initiate.

Reference: None

Modified question from the 07 NRC exam. (209001_1) Q#6

Cog Level: High

Explanation: Initiation signal present due to low RPV pressure and high DW pressure. Injection valves are open <410 psig, but RPV press is above 300 psig, the shutoff head of the pump, therefore the pump will be running on min flow.

Distractor Analysis:

Choice A: Correct Answer, see explanation

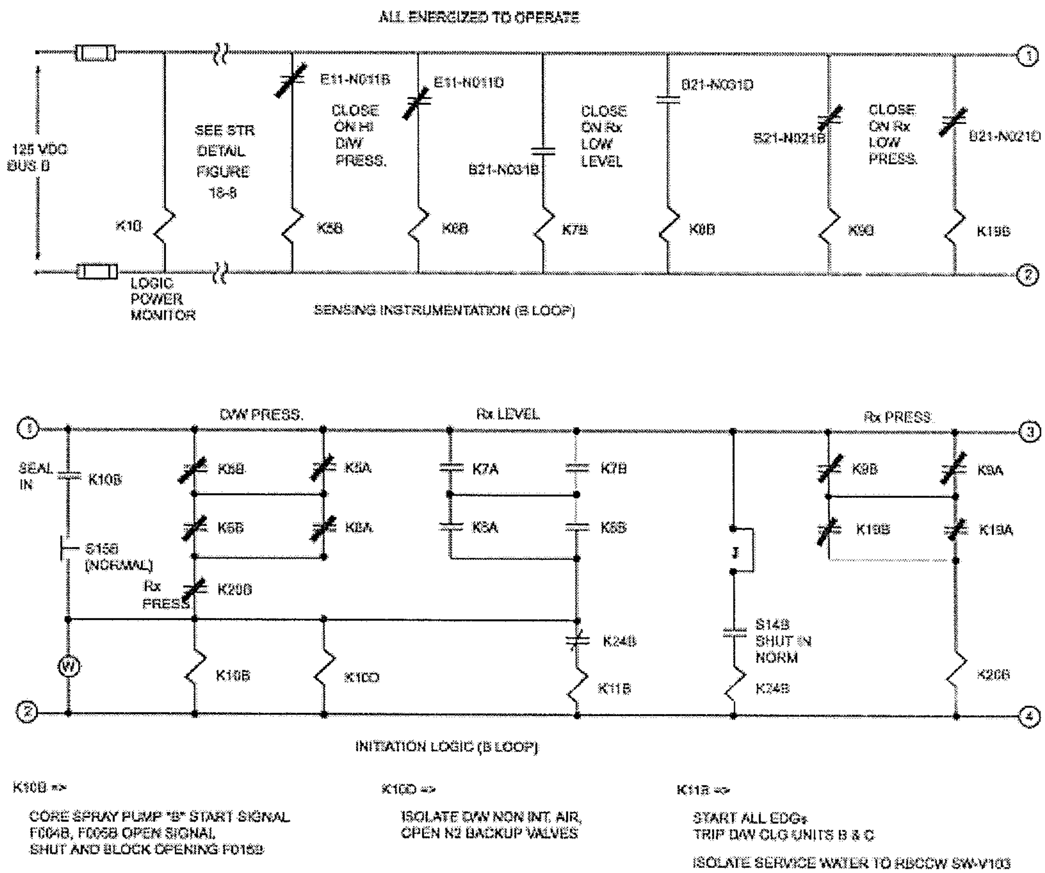
Choice B: Plausible because if reactor pressure was below 300 psig Core Spray would be injecting (discharge pressure greater than shutoff head) and the minimum flow valve would be closed.

Choice C: Plausible because if reactor pressure was above 410 psig the discharge valve F005A would be closed and the minimum flow valve F031A would be open.

Choice D: Plausible because if an initiation signal was not present the injection valve would be closed.

SRO Basis: N/A

FIGURE 18-7
Core Spray Initiation Logic



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 K1.09	
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause- effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Nuclear boiler instrumentation

Proposed Question: Common 6

A feedwater line rupture has occurred on Unit Two (2) and the following conditions exist:

Drywell pressure	8.8 psig
Reactor water level	- 35 inches
RPV pressure	500 psig

Which **ONE** of the following describes the configuration of both loops of Core Spray?

A. No Core Spray pumps are running.
Both min flow valves 2E21-F031A(B) are open.
Both inboard injection valves 2E21-F005A(B) are closed.
Both outboard injection valves 2E21-F004A(B) are open.

B. Both Core Spray pumps are running with flow to the vessel.
Both min flow valves 2E21-F031A(B) are closed.
Both inboard injection valves 2E21-F005A(B) are open.
Both outboard injection valves 2E21-F004A(B) are closed.

C. Both Core Spray pumps are running.
Both min flow valves 2E21-F031A(B) are open.
Both inboard injection valves 2E21-F005A(B) are closed.
Both outboard injection valves 2E21-F004A(B) are open.

D. Both Core Spray pumps are running with flow to the vessel.
Both min flow valves 2E21-F031A(B) are open.
Both inboard injection valves 2E21-F005A(B) are open.
Both outboard injection valves 2E21-F004A(B) are open.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Core Spray will be running due to Low Level signal.

B. Incorrect – No injection flow because F005 valves (not the F004) are closed and pressure is > pump shutoff head.

C. Correct Response - the F031A(B) will be open for min flow protection of the pump, F004A(B) are normally open and the F005A(B) are closed because the low pressure permissive of 410 psig has not been met.

D. Incorrect – F005 valves are closed. Rx pressure is > pump shutoff head. Min flow valves F031A(B) are open because the pumps are running with no flow to the vessel.

Per SD-18, Step 3.1.3 Core Spray Inboard and Outboard Injection Valves
The Core Spray Inboard and Outboard Injection Valves, E21-F005B(A) and E21-F004B(A), respectively, have automatic and manual control functions. Normally, the valves are in the automatic mode as dictated by the Control Switches, E21-S1B/A and S2B/A (CLOSE-AUTO-OPEN, spring return to AUTO), for each valve. Both valves will automatically open provided the following conditions are met:

- An initiation signal present (Low reactor water level #3 or low reactor vessel pressure coincident with a high drywell pressure). (K10)
- Low reactor pressure permissive satisfied (K20)
- 10 second start timer relay timed out (timer starts once the E-bus is energized) (K3/K4)

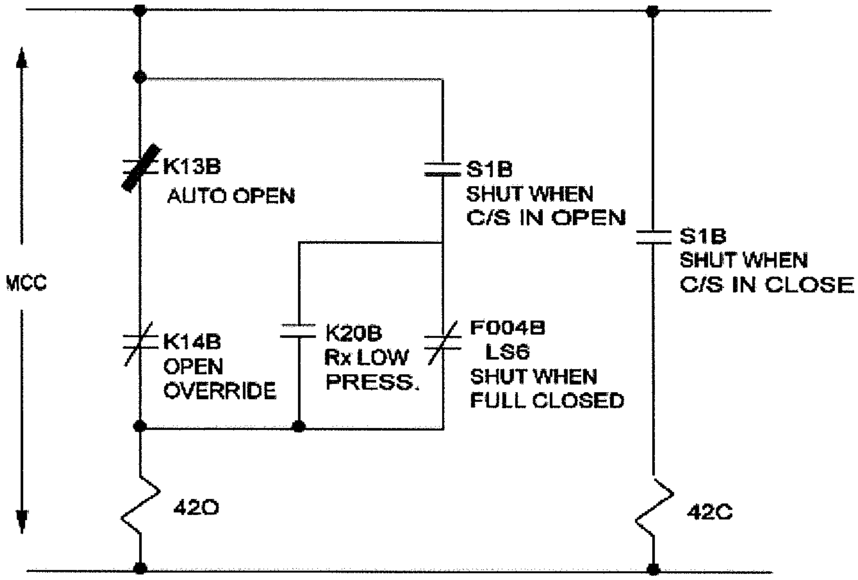
If the inboard (E21-F005B/A) valve control switch is turned to CLOSE while an initiation signal is present, the valve automatic opening function will be disabled. The manually initiated CLOSE signal will override the automatic OPEN signal thus allowing the valve to be throttled as the Operator desires. The valve's automatic opening function will remain disabled until the system initiation signal is cleared (initiation logic is reset, reactor pressure increases above 410 psig, or power is lost to the associated Emergency Bus). A white light will illuminate "CLOSE SIG SEALED VLV E21-F005" on P601 which indicates that the automatic opening is disabled.

Technical Reference(s): OP-18, 1.5.5.2 /R19 (Attach if not previously provided)

____

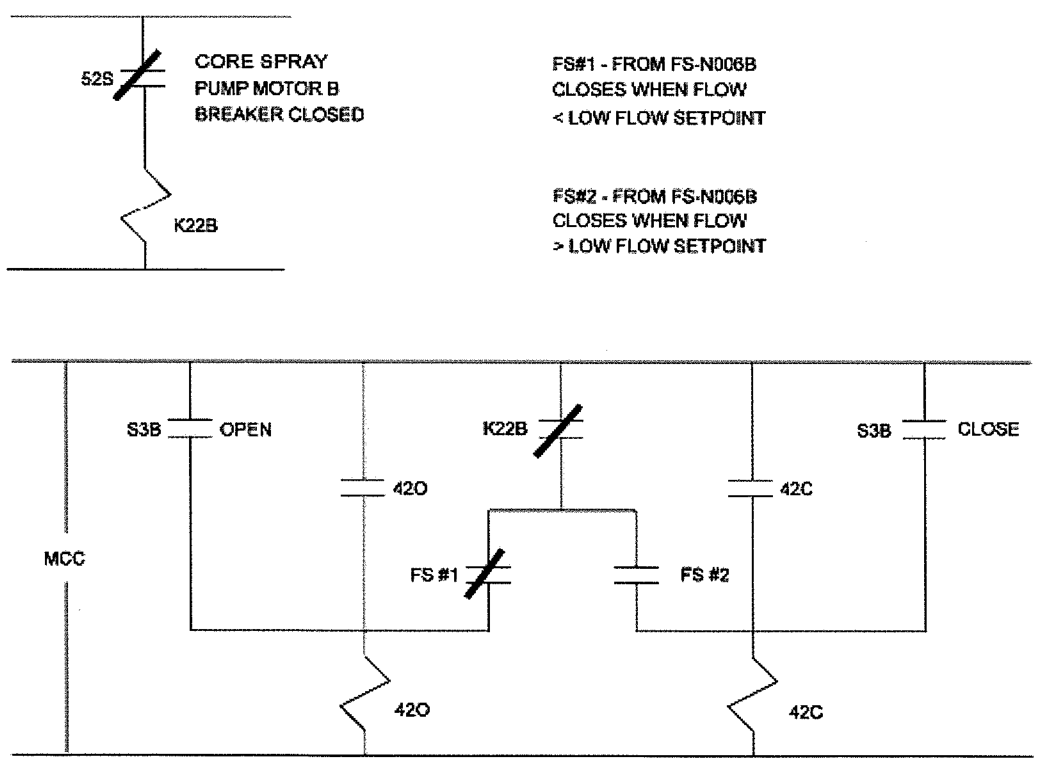
____

FIGURE 18-5
Injection Valve Interlock Logic



F005B (THROTTLEABLE)
INBOARD

FIGURE 18-4
Minimum Flow Valve Interlock Logic



F031B

Core Spray Minimum Flow Bypass Valve, E21-F031B(A), is operated by a Flow Switch (E21-FS-N006B/A) installed on the pump discharge line. This valve is normally open when the associated Core Spray pump is not running. If the flow switch senses a flow greater than 1500 gpm increasing (operational value, nominal <u>and</u> the corresponding Core Spray pump breaker is closed, indicating that the pump is running, the associated minimum flow valve will close and remain closed as long as flow remains greater than 500 gpm decreasing. If pump flow drops below 500 gpm decreasing <u>and</u> the pump breaker is closed, the flow switch opens the corresponding minimum flow valve.

The minimum flow valves may be operated manually with Control Switches E21-S3B/A (CLOSE-AUTO-OPEN, spring return to AUTO); however, if the pump breaker is closed, the valve will return to the position desired by the flow instrumentation. If the Core Spray pump breaker is open, indicating that the pump is not running, the Operator may close the valve using the control switch to allow for the isolation of primary containment.

7. 211000 1

An ATWS has occurred on Unit One and reactor water level deliberately lowered IAW LPC. The following conditions exist:

Reactor Water Level	maintained between LL4 and TAF
Reactor Power	9%
Reactor Pressure	960 psig
SLC Tank Level	2800 gallons
SLC Pumps	Both operating

Which one of the following choices completes the statements below?

Adequate mixing of the boron with reactor water ___(1)___ assured at this level.

Under the current conditions the time for the SLC tank to reach 0% would be approximately ___(2)___ minutes.

A. (1) is
   (2) 32 to 34

B. (1) is
   (2) 65 to 68

C. (1) is NOT
   (2) 32 to 34

D. (1) is NOT
   (2) 65 to 68

Answer: C

K/A: 211000 Standby Liquid Control System
K5.02 - Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Chugging (as it pertains to boron mixing)
(CFR: 41.5 / 45.3)
****Question to be written to address boron mixing vice chugging per discussion with Chief Examiner (Bruno Caballero) 2/27/12***

RO/SRO Rating: 2.8/3.0

Objective: LOI-CLS-LP-300-E, Obj.14C - Given plant conditions and the Level/Power Control Procedure, determine the following: When Boron Mixing is required after being injected into the reactor.

Reference: None

Cog Level: HIGH

Explanation: From SD-05, Section 2.11:
Adequate mixing of the solution with the reactor water should occur if the solution is injected when natural circulation exists with normal reactor water level. In this case the sodium pentaborate remains in the lower plenum of the reactor until the reactor water level is raised. The injection rate of the SLC pumps is per design 43 gpm while the PT acceptance criteria is 41.2 gpm. In the case both pumps are operating so the numbers are calculated with both numbers.

Distractor Analysis:

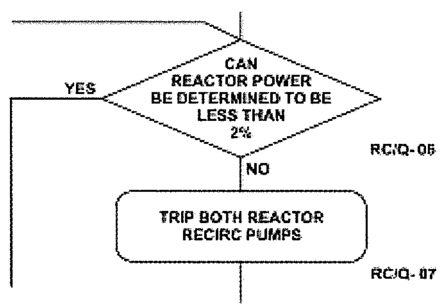
Choice A: Plausible because adequate mixing would occur with forced circulation, however during an ATWS level is deliberately lowered and recirculation pumps are not operating. 32 to 34 minutes would be the injection time of both pumps.

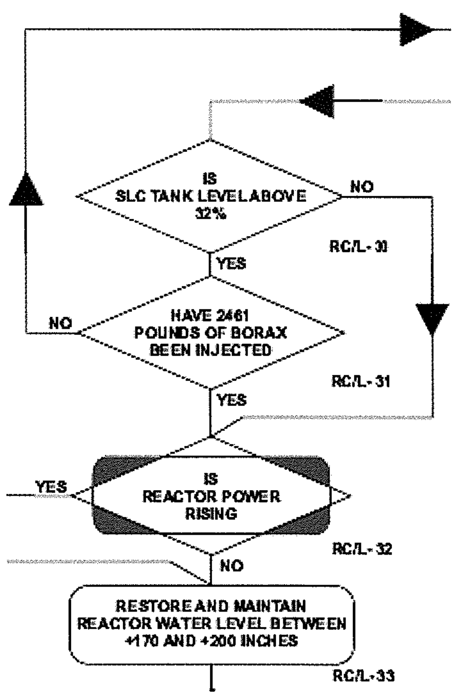
Choice B: Plausible because adequate mixing would occur with forced circulation, however during an ATWS level is deliberately lowered and recirculation pumps are not operating. 65 to 68 minutes would be the injection time of one pump.

Choice C: Correct answer, see explanation.

Choice D: Plausible because adequate mixing has not occurred and 65 to 68 minutes would be the injection time of one pump.

SRO Basis: N/A





From 0OI-37.5, Level/Power Control Procedure Basis Document:

Tripping the recirculation pumps from high reactor power effects a prompt reduction in power. If boron injection is later required, three-dimensional model tests have demonstrated that forced recirculation need not be maintained because natural circulation flow provides adequate boron mixing. Tripping the recirculation pumps is allowed in this step because the operator will have already run the speeds back, if necessary, and the resulting changes will be significantly reduced.

Maintaining reactor water level below the normal operating range suppresses reactor power by reducing natural circulation core flow. If reactor water level is lowered to and maintained near LL-4, little if any natural circulation flow exists within the reactor vessel.

Three-dimensional scale model tests have been conducted which confirm that little boron mixing occurs under these conditions if the boron is injected into the lower plenum. The injected boron concentrates in the lower plenum region of the reactor vessel and does not contribute to reactor shut down until in-core distribution (mixing) is achieved. When an amount of boron sufficient to shut down the reactor has been injected into the reactor vessel, mixing is accomplished by raising reactor water level thereby increasing natural circulation flow through the vessel.

From SD-05, Standby Liquid Control System:

2.11 SLC Injection Piping (Figure 05-6)

The solution enters the reactor vessel through the SLC/core differential pressure line penetration. The penetration consists of a pipe within a pipe, with the outer pipe welded to the reactor pressure vessel nozzle, providing an annular space between the nozzle and the inner line, used for SLC injection, to minimize thermal shock to the vessel if SLC is used. The outer pipe connects to the core support plate and senses the above core plate pressure. The inner pipe penetrates (sealed) the outer pipe inside the reactor vessel and ends below the core plate. The SLC solution is injected through the inner pipe and is dispersed beneath the lower core plate. Adequate mixing of the solution with the reactor water should occur if the solution is injected when natural circulation exists with normal reactor water level.

However, if natural circulation does not exist, during cases of extreme low water level, no mixing will occur. In this case the sodium pentaborate remains in the lower plenum of the reactor until the reactor water level is raised allowing normal natural circulation be established. In addition to providing the SLC solution injection point, this penetration provides the following functions:

2.5 SLC Pumps

There are two full capacity triplex piston positive displacement pumps that can inject the solution into the reactor at a rate of 43 gpm at 1190 psig. When both pumps are operating simultaneously, the flow rate is increased to approximately 86 gpm.

One pump will inject the tank contents in 58 to 81 minutes and two pumps will inject the solution into the reactor in 29 to 40 minutes.

6.0 ACCEPTANCE CRITERIA

NOTE:	A condition report shall be initiated for any off normal condition observed during testing.

This test may be considered satisfactory when the following criteria are met:

6.1 SLC solution is recirculated through the SLC pumps to the storage tank.

6.2 Pump Tests

6.2.1 Each Standby Liquid Control Pump develops a flowrate of greater than or equal to 41.2 gpm with a pump discharge pressure greater than or equal to 1190 psig.

8. 212000 1

Unit Two is in an ATWS with the following plant conditions:

Reactor power	31%
Mode Switch	RUN

Which one of the following choices will prevent the operator from resetting RPS prior to the LEP-02, Section 3 jumper installation with the given conditions?

A. Scram discharge volume Hi Hi level RPS trip sealed in.

B. Reactor water level is controlling at the setdown setpoint.

C. IRMs upscale Hi Hi due to being inserted but not ranged up.

D. Inboard MSIV B21-F022A and Outboard MSIV B21-F028D closed.

Answer: A

K/A: 212000 Reactor Protection System
A4.04 - Ability to manually operate and/or monitor in the control room: Bypass SCRAM instrument volume high level SCRAM signal (CFR: 41.7 / 45.5 to 45.8)

RO/SRO Rating: 3.9/3.9

Objective: LOI-CLS-LP-003, Obj. 15, Describe the method to reset a scram, including the conditions that must be met.

Reference: None

Cog Level: High

Explanation: SDV hi hi trip can be bypassed only if the Mode switch is in Shutdown or Refuel. Setdown is 170", 4" above scram setpoint, IRM trips bypassed in RUN, MSIV input to RPS for scram is not made up.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Plausible because reactor water level sets down to 170" following a scram to prevent overfeeding of the vessel following the scram. This setdown, however, is above the low level scram setpoint of 166"

Choice C: Plausible because misranging of IRMs with the reactor critical could cause a reactor scram but the IRM scram input is bypassed with the mode switch in RUN.

Choice D: Plausible because the right combination of MSIVs closed with the mode switch in RUN would result in a reactor scram.

SRO Basis: N/A

3.1.4 Scram Discharge Volume High Level (Figures 03-11, 03-12 and 03-13)

The Scram Discharge Volume High Level initiates a Scram while adequate volume is still available to receive Scram discharge water to assure that all operable Control Rods will fully insert.

Since this Scram signal will be generated following any Scram, it must be bypassed to allow for resetting a Scram condition. A two position NORMAL/BYPASS keylock switch (S4), on Panel 603, must be in BYPASS coincident with the Reactor Mode Switch (S1) in Refuel or Shutdown in order to initiate bypass of this Scram parameter.

3.1.5 Main Steam Isolation Valves Closure, Setpoint ≤ 10 % Closed (Figures 03-11, 03-12 and 03-14)

A Scram signal is initiated when specific combinations of Main Steam Line Isolation Valves (MSIVs) close. This scram acts to mitigate the positive reactivity addition transient resulting from sudden MSIV closure at high power and precludes high power reactor operation under low pressure conditions such as may occur following a main

steam line rupture while at 100% power. This scram is bypassed when the Reactor Mode Switch is <u>not</u> in RUN.

As with the Turbine Stop Valves, two Steam Lines may be isolated (≤ 10% closed) without causing a reactor scram.

- Isolating one steam line does not cause a half-Scram.
- Isolating Main Steam Lines A and D or B and C does not cause a half-Scram.
- Isolating any other combination of two Main Steam Lines will cause a half-Scram.
- Isolating any combination of three Main Steam Lines will cause a Reactor Scram.

3.1.7 Reactor Low Water Level, Setpoint 166 Inches (Figure 03-16)

The Reactor water level trip point was chosen far enough below the normal operating level to avoid spurious Scrams but high enough above the fuel to assure that there is adequate water to account for evaporation losses and displacement of coolant following the most severe transients. Signals are from four (4) level indicating switches.

Since the backup Scram valve logic requires a full Scram signal, the logic is used to isolate the Scram Discharge Volume. The Backup Scram Solenoid logic (K21 relays) is used to by Digital Feedwater Control System (DFCS) to set the DFWLC setpoint down to 170", transfer to Single Element Control, and bypass the Single Element Control Noise Filters following a scram. These DFCS post-SCRAM functions are intended to improve Reactor level control post-SCRAM. These relays are also used in the Scram Reset logic.

Only one IRM per RPS Trip System may be manually bypassed due to physical arrangement of the bypass switch. The IRMs are automatically bypassed when the Reactor Mode Switch is in Run.

9. 215001 1

A drywell entry is being made with the reactor at 10% power and rated pressure. TIPs are currently at the indexer position for decay following a TIP core scan.

Which one of the following TIP system manipulations is required by 0OI-01.03, Non Routine Activities, Attachment 11, Drywell Entry Requirements, to reduce radiation levels in the drywell?

TIPs shall be relocated from the indexer position to the:

A. core bottom limit with the TIP machine mode switch in Off.

B. core bottom limit with the TIP machine mode switch in Manual.

C. in-shield position with the TIP machine mode switch in Off.

D. in-shield position with the TIP machine mode switch in Manual.

Answer: C

K/A: 215001 Traversing In-Core Probe
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the TRAVERSING IN-CORE PROBE controls including: Radiation levels: (Not-BWR1)(CFR: 41.5/45.5)

RO/SRO Rating: 2.8/2.9

Objective: None

Reference: None

Bank question last used on the 08 NRC exam.

Cog Level: Low

Explanation: 0OI-01.03 attachment 11 requires that TIPs be in the stored position (in shield) in the TIP Room and a clearance on each TIP ball valve (closed) and the TIP machine Auto Manual Switch (Off)

Distractor Analysis:

Choice A: Plausible because the TIP must be stored in the TIP Room, but storing the TIP in the vessel would provide shielding from the reactor vessel and internals and the biological shield, and the off position would prevent withdraw (Note this would not be acceptable as an equivalent boundary since the PCIS function would be defeated)

Choice B: Plausible because the TIP must be stored in the TIP Room, but storing the TIP in the vessel would provide shielding from the reactor vessel and internals and the biological shield, and the manual position would maintain primary containment isolation capability (Note this would not be acceptable as an equivalent boundary since the TIP could automatically retract through the drywell raising the radiation levels)

Choice C: Correct answer, see explanation

Choice D: Plausible because since manual position is available on the switch and would prevent any automatic operation of the machine except for the primary containment isolation function which is already met.

SRO Basis: N/A

0OI-01.03

4.0 The TIPs shall be in the stored position (in shield) in the TIP Room, and a clearance placed on each TIP ball valve (Closed) and each TIP machine AUTO-MANUAL mode switch (OFF).

10. 215003 1

Which one of the following indicates the purpose for performing the IRM range 6/7 overlap determination?

IRM range 6/7 overlap determination is required to be performed in order to ensure that the IRM:

A. voltage preamplifier has properly transitioned from more noise immune to more sensitive operation.

B. voltage preamplifier has properly transitioned from more sensitive to more noise immune operation.

C. pulse height discriminator circuitry has properly transitioned from more sensitive to more noise immune operation.

D. pulse height discriminator circuitry has properly transitioned from more noise immune to more sensitive operation.

Answer: B

K/A: 215003 Intermediate Range Monitor System
K4.04 - Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Varying system sensitivity levels using range switches (CFR: 41.7)

RO/SRO Rating: 2.9/2.9

Objective: LOI-CLS-LP-009-A, Obj. 15e - Explain the basis or precautions associated with the following: Performing IRM range 6 to 7 overlap test.

Reference: None

Cog Level: LOW

Explanation: The IRMs utilize a preamplifier to boost the output signal and routes this signal through range switches to provide indication over varying power levels. This preamplifier operates over two discreet frequency ranges. As power increases, the output signal of the detectors increases and noise increases. When range six is exceeded, the frequency of the preamplifiers is shifted higher. In order to ensure accurate indication for the control operator, the indications resulting from this frequency shift are verified to be correct by the performance of 0MST-IRM25R, IRM Channels Range Correlation Adjustment.

Distractor Analysis:

Choice A: Plausible because overlap determination is performed when IRMs change from low frequency (more sensitive) to high frequency (more noise immune)(range six to seven). If ranging was from range seven to six, the opposite relationship would be true.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the pulse height discriminator is part of the SRM circuit that provides for the removal of the noise in the circuit.

Choice D: Plausible because the pulse height discriminator is part of the SRM circuit that provides for the removal of the noise in the circuit.

SRO Basis: N/A

SD-09.1

2.4.1 Voltage Preamplifier (IRM)

Each IRM instrument channel utilizes a voltage preamplifier which is used in conjunction with a mean square voltage wide range monitor to measure the mean square value of a current or voltage signal over a range of three decades in each of two bandwidths.

The voltage preamplifier raises the output of the detector so that it can overcome the noise picked up in the long cable to the control room. The voltage preamplifier:

1. Provides amplification of the low level signal from the detector

2. Passes only those portions of the detector signal necessary for proper operation of the IRM signal by isolating the HVPS and detector signal from one another.

The preamplifier is located in Cabinet P030 a chassis just outside the drywell. Relays are in cabinet P008. The preamplifier passes only one of two sets of frequencies depending on the detector range selected by the IRM range selector switch as follows:

Range	Frequencies Passed
1-6	0.8 - 16 Khz
7-10	300 - 600 Khz

The reason for selection of frequency bands is because these frequency bands exhibit the best proportionality to power. Correct overlap of IRM range 6 to 7 is verified during a reactor startup.

The lower band (0.8 - 16 Khz) is more sensitive. This frequency band has signals from both electron and ion collection pulses, while the upper band sees signals from only electron collection, which has a shorter (higher frequency) pulse. This lower band frequency is utilized for ranges 1-6, where neutron flux levels are relatively low.

The upper band has greater immunity to noise and develops a signal based solely on the change in frequency as a result of electron pulses from neutron interaction within the detector.

0GP-02

5.2.26 **IF** this is the initial startup following a refuel outage, **THEN PERFORM** the following:

1. 0PT-14.3.1 (if 0PT-14.3 was **NOT** completed prior to startup).

2. 0MST-IRM25R, IRM Range 6 and 7 Correlation adjustments for any IRM Channel(s) with open PMT requirements prior to reaching range 7 (SR 3.3.1.1.13).

0MST-IRM25R

1.0 PURPOSE

1.1 Range correlation adjustment of IRM channels ensures high and low frequency ranges will provide similar readings when the operator switches between Range 6 and Range 7 during reactor startup or shutdown. This procedure partially fulfills the channel calibration requirements of Technical Specification Tables 3.3.1.1-1 Item 1.a, SR 3.3.1.1.13.

TEST DESCRIPTION

Test ensures high and low frequency ranges of IRM channels agree to provide similar readings for operator when switching between Range 6 and Range 7 during reactor startup or shutdown.

11. 215004 1

Unit Two is in the process of a reactor startup IAW 0GP-02, Approach to Criticality and Pressurization of the Reactor, following a refueling outage.

The following SRM readings are indicated:

SRM Channel A	8.0E4
SRM Channel B	7.0E4
SRM Channel C	2.0E5
SRM Channel D	6.0E5

All IRMs are on range 4.

Which one of the following identifies the expected plant response?

A. Alarm ONLY.

B. Alarm and rod block ONLY.

C. Alarm, rod block and 1/2 scram.

D. Alarm, rod block and full scram.

Answer: B

K/A: 215004 Source Range Monitor System
A4.03 - Ability to manually operate and/or monitor in the control room: CRT displays: Plant-Specific (CFR: 41.7 / 45.5 to 45.8)

RO/SRO Rating: 2.9/2.7

Objective: LOI-CLS-LP-009-A, Obj. 05 - Describe the interrelationships between the SRM and IRM systems.

Reference: None

Cog Level: High

Explanation: A non-coincidence reactor scram would occur with any SRM greater than 5.0 E5 IF shorting links are removed. SRM rod block occurs if any SRM greater than 2.0 E5. SRM rod block is bypassed if mode switch is in RUN or if all IRMs are greater than range 7. Candidate must also recognize that the mode switch would be not be in RUN while executing 0GP-02.

Distractor Analysis:

Choice A: Plausible because rod block and upscale alarms would both occur.

Choice B: Correct Answer, see explanation

Choice C: Plausible because an alarm and rod block would occur. If shorting links were removed a scram would also occur. Candidate must have knowledge of the non-coincidence scram function of SRMs without shorting links installed. Most scrams from nuclear instrumentation are divisionalized with the exception of SRMs

Choice D: Plausible because an alarm and rod block would occur. If shorting links were removed a scram would also occur. Candidate must have knowledge of the non-coincidence scram function of SRMs without shorting links installed. Most scrams from nuclear instrumentation are divisionalized with the exception of SRMs

SRO Basis: N/A

SD-09.1

TABLE 09.1- 1
INSTRUMENT AND CONTROL SETPOINTS
STARTUP RANGE NEUTRON MONITORING SYSTEM

INSTRUMENT DESIGNATION AND TRIP FUNCTION	TRIP SETPOINT AND FUNCTION	FUNCTION, ADDITIONAL CONDITIONS AND COMMENTS
SRM Inop Trip C51-SRM-K600 (A-D) TRM Annunciator "SRM UPSCALE/INOP" (A-05 2-3)	HVPSS - 10% ± 1%* Switch not in OPERATE or SRM Module unplugged	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is <u>not</u> in RUN •ANY divisional IRM < Range 8 and <u>NOT</u> bypassed. <u>Note:</u> Bypassed if all divisional IRMs are above Range 7
SRM Downscale Trip C51-SRM-K600 (A-D) TRM Annunciator "SRM DOWNSCALE" (A-05 1-3)	5± 1.5 cps	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is <u>not</u> in RUN •ANY divisional IRM < Range 3 and <u>NOT</u> bypassed. Note: Bypassed if all divisional IRMs are above Range 2
SRM Retract Permissive C51-SRM-K600 (A-D) TRM Annunciator "SRM RETRACT NOT PERMITTED" (A-05 4-3)	125 cps (101 to 150)	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is <u>not</u> in RUN •SRM detector not FULL IN •ANY divisional IRM < Range 3 and <u>NOT</u> bypassed. Note: Bypassed if all divisional IRMs are above Range 2 Note: SRM Retract Permissive is BYPASSED when the Mode Switch is in RUN. Reference drawing 0FP-05852-3 REV F
SRM Upscale Alarm C51-SRM-K600 (A-D) TRM Annunciator "SRM UPSCALE/INOP" (A-05 2-3)	2 X 10^5 cps (1.3 X 10^5 – 3.0 X 10^5)	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is <u>not</u> in RUN •ANY divisional IRM < Range 8 and <u>NOT</u> bypassed. <u>Note:</u> Bypassed if all divisional IRMs are above Range 7
SRM Upscale Trip C51-SRM-K600 (A-D) TRM	5 X 10^5 cps (3.3 X 10^5 - 7.5 X 10^5)	**Full Scram if refueling shorting links removed** and Reactor Mode Switch is <u>not</u> in RUN
SRM Period C51-SRM-K600 (A-D) TRM	50 seconds -10, +15 sec	Annunciator "SRM PERIOD" (A-05 3-3)

TRM Technical Requirement Manual (TRM) related (SRM Instrumentation is Technical Specification related however trips listed are in the TRM)

* HVPSS is the high voltage power supply setting (350-600 Vdc range) and the percentages are of this value. <u>Note:</u> A complete loss of power will produce an apparent trip of all trip units (i.e. Full scram if shorting links are removed due to SRM Upscale Trip)

12. 215005 1

The quantity of operating LPRM detectors in the flux average is 15 for APRM 1.

Which one of the following identifies the impact of this condition?

A. *Rod Out Block* Annunciator ONLY

B. *APRM Upscale Trip/ Inop* Annunciator ONLY

C. *APRM Trouble* and *Rod Out Block* Annunciators.

D. *APRM Trouble* and *APRM Upscale Trip/ Inop* Annunciators.

Answer: C

K/A: 215005 Average Power Range Monitor/Local Power Range Monitor System
A3.04 - Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Annunciator and alarm signals (CFR: 41.7 / 45.7)

RO/SRO Rating: 3.2/3.2

Objective: LOI-CLS-LP-09.6, Obj. 14 - Given PRNMS settings for abnormal conditions or operation, use the Annunciator Panel Procedures (APP) to determine the probable cause(s) for the following alarms: e. A-6 3-7, APRM TROUBLE, m. A-5 2-2, ROD OUT BLOCK

Reference: None

Cog Level: Low

Explanation: An APRM with too few LPRM inputs will cause a trouble alarm and rod block. In this case, there are too few detectors since there are only 15 inputs in average and a minimum of 17 inputs in average are required.

Distractor Analysis:

Choice A: Plausible because a Rod Out Block annunciator would be received but would not be the only alarm.

Choice B: Plausible APRM Upscale Trip/Inop is one alarm that could be received in association with APRM Trouble.

Choice C: Correct answer, see explanation

Choice D: Plausible APRM Upscale Trip/Inop is one alarm that could be received in association with APRM Trouble.

SRO Basis: N/A

APRM TROUBLE

AUTO ACTIONS

1. Rod Withdrawal Block if alarm initiated by too few LPRM detectors per level or too few LPRM detectors in flux average.
2. If alarm is due to a 2/4 Logic Module (RPS Voter) power supply failure, the associated RPS Trip Channel trips

CAUSE

1. The quantity of operating LPRM detectors at any given reactor level is less than three.
2. The quantity of operating LPRM detectors in the flux average is less than 17.
3. A 2/4 Logic Module (RPS Voter) power supply failure
4. Any self-test fault.

OBSERVATIONS

1. ROD OUT BLOCK (A-05 2-2) alarm.
2. The Rod Withdrawal Permissive indicating light will be off.
3. On APRM BARGRAPH display at P608 and PPC Displays 882-885, LPRMs in average is less than 17, if this condition caused the alarm.
4. If a 2/4 Logic Module power supply failure caused the alarm, the following indications can be observed:
   - Home APRM for the 2/4 Logic Module (RPS Voter), which experienced the power supply failure, displays POWER SUPPLY ERROR REMOTE in its alarm summary.
   - REACTOR AUTO SCRAM SYS A(B) A-05 1-7 (2-7) alarm
   - NEUTRON MON SYS TRIP (A-05 4-7)alarm

ROD OUT BLOCK

AUTO ACTIONS

1. Rod withdrawal prohibited.

CAUSE

1. South SDV not drained.
2. North SDV not drained.
3. SRM downscale and any IRM is below Range 3.
4. IRM downscale and affected IRM channel is not on Range 1.
5. SRM upscale/inoperative and any IRM channel is below Range 8.
6. IRM upscale and the reactor system mode switch is not in the RUN position.
7. IRM A upscale/inoperative and the reactor system mode switch is not in the RUN position.
8. SRM detector not fully inserted and log count rate is less than or equal to 100 cps (bypassed when all IRM channels are above Range 2 or the reactor system mode switch is in the RUN position).
9. IRM B upscale/inoperative and the reactor system mode switch is not in the RUN position.
10. APRM downscale and the reactor system mode switch is in the RUN position.
11. APRM UPSCALE alarm.
12. APRM UPSCALE TRIP/INOP alarm.
13. Less than 17 LPRM inputs to any APRM or less than 3 LPRMs per axial level for any APRM.
14. RBM downscale and reactor system mode switch is in the RUN position
15. RBM upscale/inoperative.
16. Recirc flow signal to any APRM greater than or equal to 110%.
17. Discharge Volume Hi Water Level Trip Bypass switch in Bypass with the Reactor System Mode Switch in Shutdown or Refuel.
18. Reactor System Mode Switch in Refuel with a second rod selected and another rod not full in.

13. 217000 1

Unit One is operating at rated power when Division I DC Switchboard is lost.

Which one of the following identifies the impact of this power loss on Reactor vessel level control using RCIC?

RCIC ___(1)___ automatically initiate on valid low level signal.
RCIC ___(2)___ shutdown on a valid high level signal.

A. (1) will
   (2) will

B. (1) will
   (2) will NOT

C. (1) will NOT
   (2) will

D. (1) will NOT
   (2) will NOT

Answer: B

K/A: 217000 Reactor Core Isolation Cooling System
K3.01 - Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Reactor water level (CFR: 41.7 / 45.4)

RO/SRO Rating: 3.7/3.7

Objective: LOI-CLS-LP-016-A, Obj. 15e - Given plant conditions, predict the RCIC System response to the following conditions: DC power failure

Reference: None

Bank question that was modified from the 07 NRC Exam. (217000_1)

Cog Level: High

Explanation: The majority of the RCIC System components are powered from Division II 125/250 Vdc Electrical Distribution via MCC 1-XDB. A loss of Division I DC power will make the inboard isolation logic (Isolation Logic A) inoperative, and result in the failure of the Turbine Steam Supply Valve to automatically close on a high vessel level condition. RCIC operation would be otherwise unaffected. A loss of Division II DC power will render the RCIC System totally inoperative for normal use.

Distractor Analysis:

Choice A: Plausible because loss of Div II power would make this correct.

Choice B: Correct answer, see explanation

Choice C: Plausible because this is a common misapplication of knowledge of power supplies to RCIC logic (impact of the power loss is the opposite of this).

Choice D: Plausible because RCIC will not shutdown on high water level with a loss of Div I DC power.

SRO Basis: N/A

SD-16

The RCIC Relay Logic A, which includes Isolation Logic A and one of the required high level inputs to the high vessel level closure of the RCIC Turbine Steam Supply Valve, E51-F045, is powered from 125 Vdc Distribution Panel 3A (4A). Control power to the Condensate Pump Discharge Outboard Drain Valve (E51-F005) and the Supply Drain Pot Inboard Drain Valve (E51-F025) is from 125 Vdc Distribution Panel 3A (4A). The Remote Shutdown Panel RCIC Turbine EGM Control Box is powered from 125 Vdc Distribution Panel 1B (2B).

A loss of Division I DC power will make the inboard isolation logic (Isolation Logic A) inoperative, and result in the failure of the Turbine Steam Supply Valve to automatically close on a high vessel level condition. RCIC operation would be otherwise unaffected.

A loss of Division II DC power will render the RCIC System totally inoperative for normal use.

0AOP-39.0

ATTACHMENT 2
Page 1 of 1
Plant Effects from Loss of DC Panel 3A(4A)

RCIC: Will not shutdown on reactor high water level, inboard isolation logic inoperable (E51-F007, -F031, and -F062 will not auto close). Valves E51-F005 and -F025 fail closed.

ATTACHMENT 2A
Page 6 of 28

PANEL 4A Reference Drawing: LL-03024-6	LOCATION: Control Building 49' East	NORMAL SUPPLY: Switchboard 2A	ALTERNATE SUPPLY: N/A

CKT #	LOAD	EFFECT
3 (cont'd)	RCIC Div I Isolation Logic	1. RCIC Groups 5 & 9 Div I isolation valves will not auto close. 2. RCIC High Water level trip INOP. 3. Receive annunciator A-03 1-3.
	ADS Logic B Low Level Permissive	1. ADS Logic B INOP. A Logic available. 2. Receive annunciator A-03 2-2.
	RCIC Div I Initiation Logic	1. RCIC Low Level 2 initiation logic INOP, RCIC will still initiate from Div II RHR logic.

ATTACHMENT 2B
Page 7 of 27

PANEL 4B Reference Drawing: LL-03024-7	LOCATION: Control Building 49' South	NORMAL SUPPLY: Switchboard 2B	ALTERNATE SUPPLY: N/A

CKT #	LOAD	EFFECT
8	RCIC Flow controller E51-FIC-K600 (24 VDC)	1. Controller fails downscale. 2. Loss of flow indication. 3. Receive annunciator A-03 5-5.
	RCIC Supervisory Lights	1. Loss of E51-V8 and E51-V9 indication.
	RCIC Vertical Board meters (62.5 VDC)	1. Loss of pressure transmitters/meters R601, R602, R603, R604 on the RTGB.
	E51-F026, E51-F004, E51-F054	1. Fail closed. 2. Loss of indication.
	HPCI E41-F028	1. Fail closed. 2. Loss of indication.
9	RCIC EGM	1. Loss of speed control. 2. Loss of speed indication on RTGB.
	RCIC Initiation and Control Logic	1. RCIC will not auto initiate. Cannot be manually operated.

14. 218000 1

A LOCA has occurred on Unit One resulting in the start of **all** low pressure ECCS pumps and rapidly lowering reactor water level.

Auto Depress Relays Energized annunciator is received and immediately clears.
Auto Depress Control Pwr Failure annunciator alarms.

Which one of the following choices completes the statement below?

The RO will be directed to open seven ADS valves ___(1)___ after the receipt of the ___(2)___ annunciator.

A. (1) immediately
   (2) *Reactor ADS Lo Water Level* (A-03 4-2)

B. (1) 83 seconds
   (2) *Reactor ADS Lo Water Level* (A-03 4-2)

C. (1) immediately
   (2) *Reactor Low Wtr Level Initiation* (A-03 6-9)

D. (1) 83 seconds
   (2) *Reactor Low Wtr Level Initiation* (A-03 6-9)

Answer: D

K/A: 218000 Automatic Depressurization System
G2.04.50 - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

RO/SRO Rating: 4.2/4.0

Objective: LOI-CLS-LP-020, Obj. 11 - Given plant conditions, determine if an automatic initiation of ADS should occur.

Reference: None

Cog Level: High

Explanation: Low level 1 (LL1) occurs when reactor water level reaches 166 inches and Annunciator *Reactor ADS Lo Water Level* alarms. At 45 inches, LL3 reactor water level, Annunciator *Reactor Low Wtr Level Initiation* alarms. Once the above low level conditions are met, a time delay begins (83 seconds). If one Core Spray pump or one loop of RHR pumps is running ADS will initiate after the 83 second timer times out. If control power is lost, *Auto Depress Control Pwr Failure* will annunciate and power to actuate the ADS valves will be lost. Manual action will be required to open the ADS valves.

Distractor Analysis:

Choice A: Plausible because this alarm is needed for initiation and the student may get the level alarms backwards. The pumps are running but the logic is not made up without LL3 alarm and the 83 second timer.

Choice B: Plausible because this alarm is needed for initiation and the student may get the level alarms backwards.

Choice C: Plausible because the logic is made up with the exception of the timer.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

Unit 1
1APP-A-03 3-2
Page 1 of 3

AUTO DEPRESS RELAYS ENERGIZED

AUTO ACTIONS

1. Energizes one half of the ADS valve logic to allow opening of the seven ADS valves when timer elapses at 83 seconds

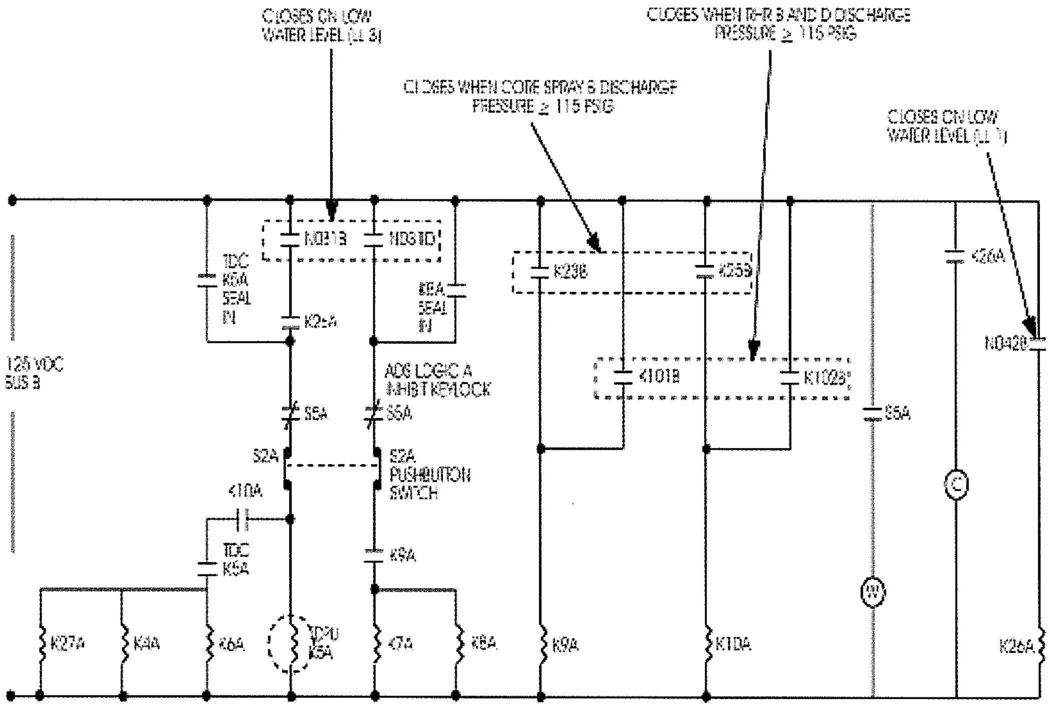
Unit 2
APP A-03 4-2
Page 1 of 1

REACTOR ADS LO WATER LEVEL

AUTO ACTIONS

1. Provides confirmatory low water level permissive for ADS initiation

SD-20



CLOSES ON LOW WATER LEVEL (LL 3)
CLOSES WHEN CORE SPRAY B DISCHARGE PRESSURE ≥ 115 PSIG
CLOSES WHEN RHR B AND D DISCHARGE PRESSURE ≥ 115 PSIG
CLOSES ON LOW WATER LEVEL (LL 1)
125 VDC BUS B
TDC K5A SEAL IN
K8A SEAL IN
ADS LOGIC A INHIBIT KEYLOCK
S5A
S2A
S2A PUSHBUTTON SWITCH
K10A
TDC K5A
K9A
K23B
K25B
K101B
K102B
K26A
C
W
K27A
K4A
K6A
TDPU K5A
K7A
K8A
K10A
K26A

15. 219000 1

Unit Two is operating at rated power with B Loop Suppression Pool Cooling in service with cooling maximized when a spurious LOCA signal occurs.

Which one of the following choices completes the statement below?

In order to re-establish Suppression Pool Cooling IAW the SPC hard card ___(1)___ must be restarted and the use of an over-ride ___(2)___ be required.

A. (1) 2B/D RHR and 2B/D RHR SW pumps
   (2) will

B. (1) 2B/D RHR and 2B/D RHR SW pumps
   (2) will NOT

C. (1) ONLY the 2B/D RHR SW Pumps
   (2) will

D. (1) ONLY the 2B/D RHR SW Pumps
   (2) will NOT

Answer: C

T2G2

K/A: 219000 RHR/LPCI: Torus/Suppression Pool Cooling Mode
A4.12 - Ability to manually operate and/or monitor in the control room: Suppression pool temperature (CFR: 41.7 / 45.5 to 45.8)

RO/SRO Rating: 4.1/4.1

Objective: LOI-CLS-LP-017, Obj. 18o - Given plant conditions, determine how the following will affect the RHR system: Loss of Coolant Accident (LOCA)

Reference: None

Modified question from 10-1 exam (295026_10). This question does not specifically ask which override is needed and asks if pumps were tripped. The previous question asked what override was needed to establish SPC.

Cog Level: High

Explanation: With the initial conditions both RHR pumps and both RHR SW pumps will be running with the HX bypass valve (F048) full closed. Based on the initial conditions no over-rides will be required at this time for Suppression Pool Cooling lineup. Upon receipt of the LOCA initiation signal, the HX bypass valve will receive an auto open signal and the torus cooling valves will auto close. Suppression pool cooling suction valves will remain in the current alignment. Running RHR pumps will continue to run (no loss of suction trip). RHR SW pumps will receive a trip signal. To re-establish cooling the RHR SW pumps will have to be restarted using a LOCA override switch and the torus cooling valves will require the Think switch to be manipulated in order to re-open them.

Distractor Analysis:

Choice A - Plausible because the RHR SW pumps will trip.

Choice B - Plausible because the RHR SW pumps will trip. Initial conditions did not require use of overrides.

Choice C - Correct Answer, see explanation

Choice D - Plausible because the RHR SW Pumps will trip. Initial conditions did not require use of overrides

SRO Basis: N/A

3.6.2 Containment Cooling/Spray Discharge Valves (Figures 17-22, 17-23, 17-24, 17-25, and 17-26)

The Suppression Pool spray and cooling valves (F028A/B and F024A/B and F027A/B) and the Containment spray valves (F016A/B and F021A/B), which automatically close on a LPCI initiation, can be opened manually when satisfying the Containment Spray Permissive Logic. The containment spray valve control logic requires Drywell pressure to be greater than 2.7 psig **AND** the reactor vessel water level inside the core shroud be above the level equivalent to 2/3 of the core's height as proven by B21-LTM-N036-1 or N037-1 with a LPCI initiation signal present, **OR** the 2/3 Core Height LPCI Initiation Manual Override keylock switch placed in OVERRIDE. Positioning the Containment Spray Valve Control (THINK) switch, in MANUAL will complete the spray permissive. The manual positioning (OPEN) command to the valves originating from the associated control switch can then be executed.

With no accident signal present, F027A(B) and F028A(B) are interlocked with each other such that one of the valves must be closed in order to open the other. Torus isolation valve F028A(B) also requires both Shutdown Cooling suction valves F006A(B) and F006C(D) to be full closed prior to opening. Drywell Spray valves F016A(B) and F021A(B) have the same interlock arrangement with no accident signal as F027A(B) and F028A(B).

Containment-side discs for E11-F028A(B) have been drilled with a vent hole to prevent thermally induced pressure-locking when required to open.

The suppression pool cooling valves which automatically close during a LPCI initiation can be opened manually with the following permissives satisfied: The water level inside the core shroud is above the level equivalent to 2/3 of the core height, as proven by B21-LTM-N036-1 or N037-1 with a LPCI initiation signal present. If it is absolutely necessary to provide cooling (e.g., to lower the suppression pool temperature), the two-thirds of the core height water level inhibit and LPCI signal can be overridden by placing 2/3 core height LPCI initiation manual override keylock override switch E11-S18A(B) to MANUAL OVERRIDE position. "Think" switch E11-S17A(B) must also be activated to MANUAL before the manual positioning (OPEN) command to the valves associated control switch can be effected and executed.

4.2.2 LPCI Initiation During Suppression Pool Cooling

If an RHR LPCI initiation signal is received while in Suppression Pool cooling, the following actions will occur in addition to the normal system response:

- HX Bypass valve F048A(B) opens automatically and cannot be closed for 3 minutes.
- Torus isolation valves F024A(B), and F028A(B) would all close automatically if open, and cannot be reopened until the Containment Cooling Spray permissives are satisfied.
- RHR pump in operation would continue to run as long as no bus undervoltage condition exists.
- RHRSW pumps in operation trip and cannot be restarted unless the RHRSW pump LOCA Override switch is placed in Manual OVERRIDE.

16. 223002 1

A LOCA has occurred on Unit Two. Subsequently a steam line leak occurs on the RCIC system. The following plant conditions are present 10 minutes after the steam line leak occurred:

Reactor water level	95 inches
Drywell pressure	3.5 psig
Reactor pressure	900 psig
RCIC Steam Line Tunnel Ambient Temp	170°F

The CRS orders the RO to manually isolate RCIC.

Which one of the following describes the effect on RCIC when the manual isolation pushbutton is depressed?

A. Inboard and Outboard Steam Supply Isolation valves F007 AND F008 close ~~and the RCIC turbine trips~~.

B. ONLY the Outboard Steam Supply Isolation valve F008 closes ~~and the RCIC turbine trips.~~

C. ONLY the Inboard Steam Supply Isolation valve F007 closes ~~and the RCIC turbine trips.~~

D. No effect on RCIC Isolation Valves.

Answer: B

K/A: 223002 Primary Containment Isolation System /Nuclear Steam Supply Shut-Off
K1.07 - Knowledge of the physical connections and/or cause effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: Reactor core isolation cooling; Plant-Specific (CFR: 41.2 to 41.9 / 45.7 to 45.8)

RO/SRO Rating: 3.4/3.6

Objective: LOI-CLS-LP-016-A, Obj. 12c - Given plant conditions with RCIC controlled from the RTGB, determine if the following automatic actions should occur: RCIC System isolation.

LOI-CLS-LP-016-A, Obj. 4b - Describe the function of the following: Manual isolation pushbutton.

Reference: None

Bank question that was last used on the 03 NRC exam.

Cog Level: High

Explanation: The Manual Isolation pushbutton is effective only if a low reactor water level signal (LL2) is present, it will close the outboard isolation valves, E51-F008 and F029. If the low reactor water level condition does not exist, the outboard isolation valves will not close. There is an isolation signal present on RCIC Steam Line Tunnel Ambient Temperature but it has a 27 minute time delay when temp is $\geq$ 165°F and this is 10 minutes into the event.

Distractor Analysis:

Choice A: Plausible because only the F008 valve is affected by the "Manual Isolation" pushbutton while there is an initiation signal present, but the F007 is not because this manual isolation pushbutton only affects the B logic valves.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the F007 valve is an isolation valve but it is not not affected by the "Manual Isolation" pushbutton because this pushbutton only affects the B logic valves.

Choice D: Plausible because there is an isolation signal present on RCIC Steam Line Tunnel Ambient Temperature but it has a 27 minute time delay when temp is $\geq$ 165°F and this is 10 minutes into the event. The Isolation pushbutton has no effect on RCIC if an initiation signal is not present.

SRO Basis: N/A

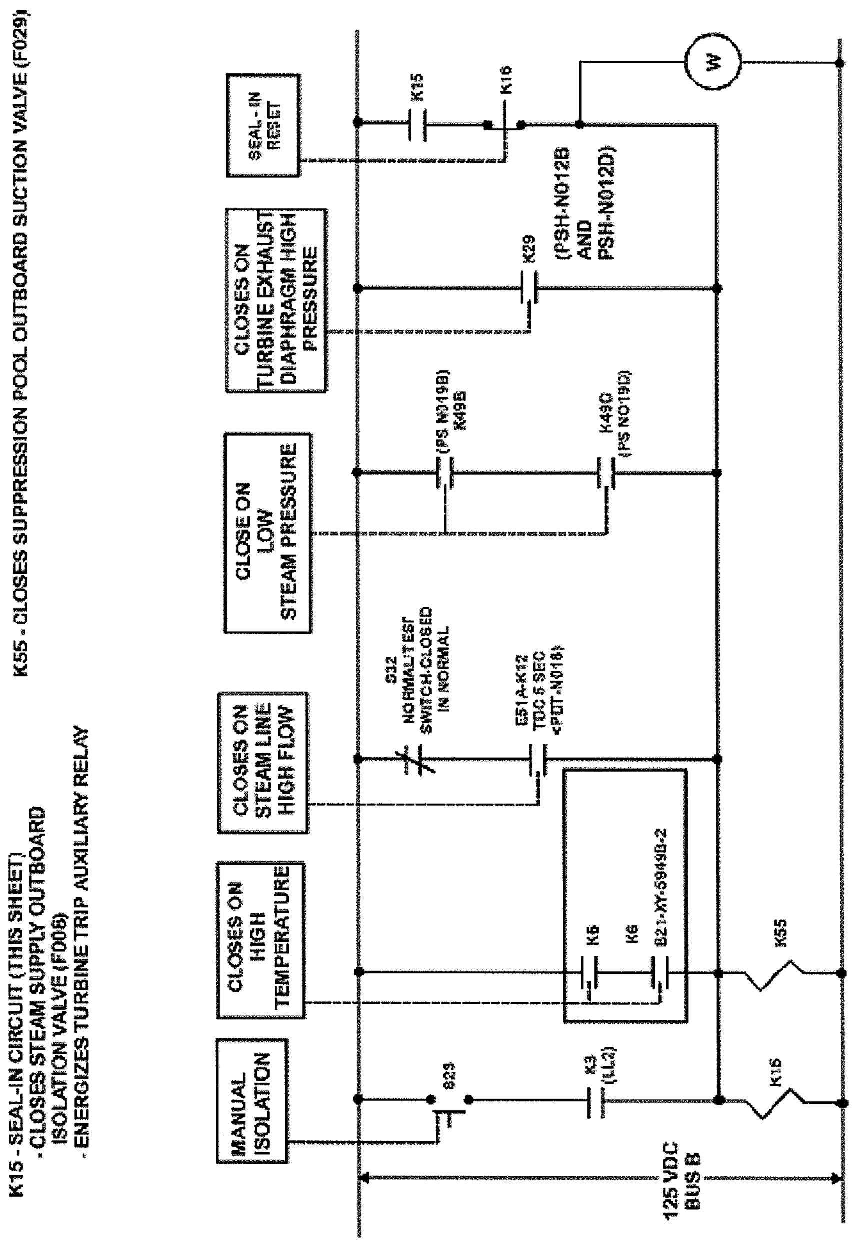
From SD-16

In addition to the automatic isolation, the RCIC System may be manually isolated. This manual isolation may be accomplished in two separate and independent ways; one is only applicable during RCIC's automatic initiation, and the other is applicable during RCIC non-automatic operation.

The first manual isolation function is accomplished by depressing the STEAM ISOLATION pushbutton. This isolation, effective only if a low reactor water level signal (LL2) is present, closes the outboard isolation valves, E51-F008 and F029. If the low reactor water level condition does not exist, the outboard isolation valves will not close.

The second manual isolation function is the regular close and open feature of the isolation valves. This function is initiated through the use of the isolation valves' individual control switches.

FIGURE 16-16
RCIC Isolation Logic B



17. 226001 1

A steam line break has occurred in the Unit Two Drywell. The CRS has directed drywell sprays be placed in service on B loop RHR IAW SEP-02, Drywell Spray Procedure. The following plant conditions exist:

Recirc Pumps	Secured
Drywell Coolers	Secured
Reactor level	100 inches
Reactor pressure	400 psig
Drywell pressure	15 psig

The RO has momentarily placed the Containment Spray Valve Control (Think) Switch to Manual.

Which one of the following choices completes the statement below?

The Ctmt Spr Ovrd light is ___(1)___ and the minimum action(s) the operator must take to initiate Drywell Spray IAW SEP-02 is to ___(2)___.

A. (1) on
   (2) place the 2/3 Core Height LPCI Initiation Override switch to Manual Overrd, then open E11-F016B and E11-F021B

B. (1) on
   (2) open E11-F016B and E11-F021B A

C. (1) off
   (2) place the 2/3 Core Height LPCI Initiation Override switch to Manual Overrd, then open E11-F016B and E11-F021B

D. (1) off
   (2) open E11-F016B and E11-F021B

Answer: B

T2 G2

K/A: 226001 RHR/LPCI: Containment Spray System Mode
K4.01 - Knowledge of RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE design feature(s) and/or interlocks which provide for the following: Testability of all operable components (CFR: 41.7)

RO/SRO Rating: 2.6/2.8

Objective: LOI-CLS-LP-017, Obj. 24 - Given plant conditions, determine if any of the white lights associated with the RHR system should be illuminated.

LOI-CLS-LP-017, Obj. 25 - Given plant conditions, ensure that all permissives are met to spray the containment.

Reference: None

Cog Level: High

Explanation: Since a LOCA signal exists (>1.7 psig DW pressure concurrent with < 410 psig reactor pressure) and reactor water level is greater than 2/3 core height, use of the 2/3 Core Height LPCI Initiation Override switch is unnecessary. In this condition, operation of the THINK switch in conjunction with DW pressure greater than 2.7 psig is sufficient to make up the spray logic and illuminate the Ctmt Spr Ovrd light. With the logic made up, the operator only has to open the spray valves E11-F016B and E11-F021B.

Distractor Analysis:

Choice A: Plausible because there are two lights associated with the cooling/spray logic. If drywell pressure was below 2.7 psig, the spray light would not be illuminated but the cooling light would be illuminated. If a LOCA signal were not present or level was below 2/3 core height, the use of the 2/3 Core Height LPCI Initiation Override switch would be required.

Choice B: Correct Answer, see explanation

Choice C: Plausible because there are two lights associated with the cooling/spray logic. If drywell pressure was below 2.7 psig, the spray light would not be illuminated but the cooling light would be illuminated. If a LOCA signal were not present or level was below 2/3 core height, the use of the 2/3 Core Height LPCI Initiation Override switch would be required.

Choice D: Plausible because there are two lights associated with the cooling/spray logic. If drywell pressure was below 2.7 psig, the spray light would not be illuminated but the cooling light would be illuminated.

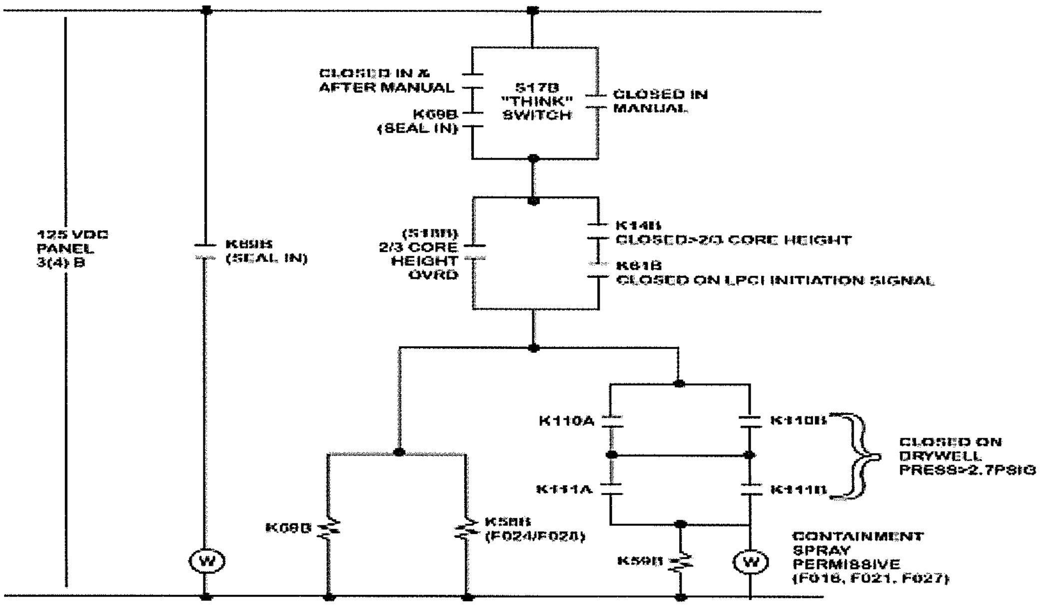
SRO Basis: N/A

SD-17

- Containment Cooling/Spray Valve Control (THINK): Two lights, one that is illuminated for Suppression Pool Cooling permissive (from K69B), the other for containment spray.

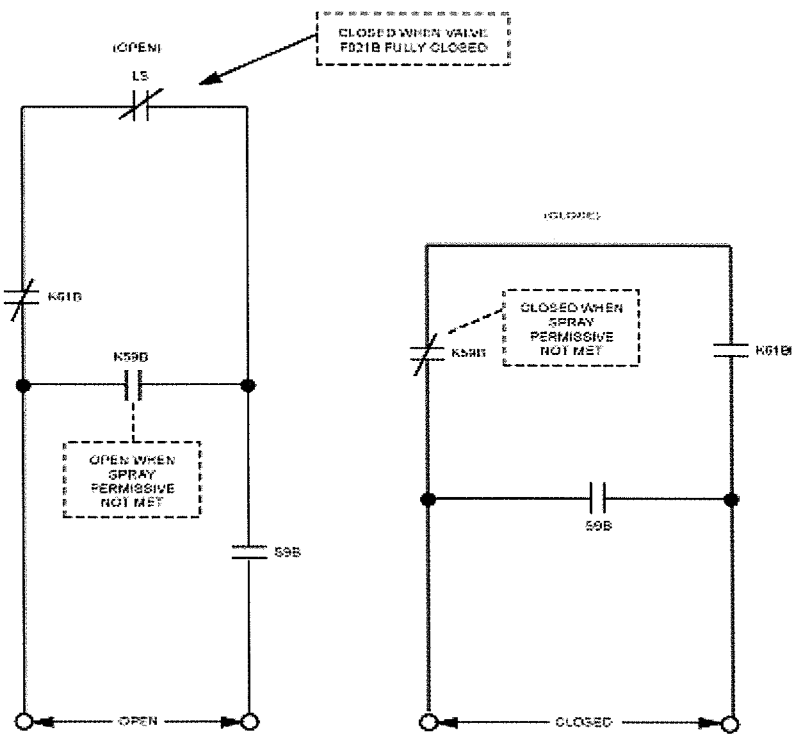
- If the Containment Spray or Suppression Pool Cooling permissive white lights are illuminated with the "Think" switch in MANUAL, then the 2/3 Core Height permissive is satisfied and placing the 2/3 Core Height LPCI Initiation Override Switch to OVERRIDE is NOT necessary.

FIGURE 17-12
Cooling/Spray Permissive Logic



F021B & F016B logic is similar for spray logic.

FIGURE 17-25
F016B Control Circuit



S9B - RTCS CONTROL SWITCH
K61B - LPCI INITIATION
K59B CONTAINMENT SPRAY PERMISSIVE

3.7 2/3 Core Height LPCI Initiation Override Switch

If it is absolutely necessary to provide spray or cooling to the primary containment (to lower the drywell pressure), the 2/3 core height water level and LPCI initiation inhibit can be overridden by placing the keylocked override switch, CS-S18A(B), to MANUAL OVERRIDE position.

3.8 Containment Spray Valve Control (THINK) Switch

A Manual switch, CS-S17A(B), that requires operator action to allow overriding the close signal sent to the containment spray and suppression pool cooling valves (F016A/B, F021A/B, F027A/B, F028A/B and F024A/B) during a LPCI initiation.

SEP-02

RO:	2.4.6	**IF** necessary, **THEN PLACE** Loop A(B) *2/3 CORE HEIGHT LPCI INITIATION OVERRIDE* switch, *E11-CS-S18A(S18B)*, to *MANUAL OVERRD*.	☐
RO:	2.4.7	**IF** the *CTMT SPR OVRD* light for Loop A(B) *CONTAINMENT SPRAY VALVE CONTROL* switch, *E11-CS-S17A(S17B)*, is **NOT** on, **THEN MOMENTARILY PLACE** Loop A(B) *CONTAINMENT SPRAY VALVE CONTROL* switch, *E11-CS-S17A(S17B)*, to *MANUAL*.	☐

18. 230000 1

Unit Two is operating at rated power when a pipe break occurs inside primary containment. A small break has also occurred in the downcomer.

Which one of the following choices completes the statements below?

IAW PCCP, Suppression Pool Spray must be initiated before ___(1)___ pressure reaches 11.5 psig.

Failure of Suppression Pool Spray may result in exceeding ___(2)___.

A. (1) drywell
   (2) RHR/Core Spray vortex limits

B. (1) drywell
   (2) Pressure Suppression Pressure limit

C. (1) suppression chamber
   (2) RHR/Core Spray vortex limits

D. (1) suppression chamber
   (2) Pressure Suppression Pressure limit

Answer: D

K/A: 230000 RHR/LPCI: Torus/Suppression Pool Spray Mode
K3.04 - Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE will have on following: Suppression chamber air temperature (CFR: 41.7 / 45.4)

RO/SRO Rating: 3.7/3.8

Objective: LOI-CLS-LP-300-L, Obj. 4e - State the effect on Primary Containment if the following limits are exceeded: Pressure Suppression Pressure Limit.

Reference: None

Cog Level: High

Explanation: The Brunswick plant does not utilize suppression chamber airspace temperatures in EOPs, therefore this correlation is utilized to meet the K/A. Suppression pool spray is used to attempt to reverse a rising trend of primary containment pressure and is initiated before reaching 11.5 psig in the suppression chamber (PCCP PC/P-06). Pressure suppression pressure is a function of suppression pool level and suppression chamber pressure. The limit assumes no steam in the airspace. The question stem provides a crack in the downcomer in the suppression chamber area thereby bypassing the pressure suppression function of the suppression pool, introducing steam into the air space. Use of Suppression Pool sprays will mitigate this condition by condensing the steam and removing heat by evaporative and convective cooling. Loss of Suppression pool spray will result in uncondensed steam in the suppression chamber airspace, exceeding pressure suppression pressure. Rising pressure in the air space can be attributed to steam in the airspace which can be correlated to air space temperature.

Distractor Analysis:

Choice A: Plausible because drywell pressure is utilized in the same PCCP leg when determining actions for securing H2/O2 monitors and for determining approach to Primary Containment Pressure Limit A. Vortex limits are to be considered but are a function of suppression pool level and pump flow, not pool or airspace temperature.

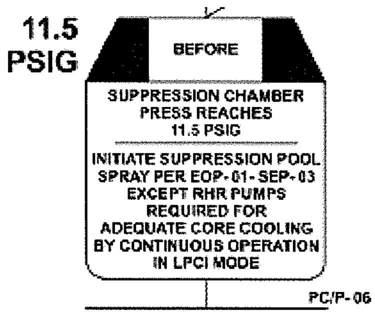
Choice B: Plausible because drywell pressure is utilized in the same PCCP leg when determining actions for securing H2/O2 monitors and for determining approach to Primary Containment Pressure Limit A.

Choice C: Plausible because vortex limits are to be considered but are a function of suppression pool level and pump flow, not pool or airspace temperature.

Choice D: Correct answer, see explanation.

SRO Basis: N/A

EOP-02-PCCP



0OI-37.8

STEP BASES:

Operation of suppression pool sprays reduces primary containment pressure by condensing steam that may be present in the suppression chamber airspace, and by absorbing heat energy from the enclosed atmosphere through the processes of evaporative and convective cooling.

0EOP-01-UG

PRESSURE SUPPRESSION PRESSURE

The lesser of either (1) the highest suppression chamber pressure which can occur without steam in the suppression chamber air space or (2) the highest suppression chamber pressure at which initiation of reactor depressurization will not result in exceeding Primary Containment Pressure Limit A before reactor pressure drops to the Minimum Reactor Flooding Pressure (MRFP), or (3) the highest suppression chamber pressure which can be maintained without exceeding the suppression chamber boundary design load if SRVs are opened. This pressure is a function of primary containment water level, and is utilized to assure the pressure suppression function of the containment is maintained while the reactor is at pressure (Figure 7).

19. 239001 1

During Unit Two power operation, a power supply loss results in a reactor scram. The operator notes the following MSIV indications immediately after the scram:

Inboard DC solenoid white light	OUT
Inboard AC solenoid white light	LIT
Outboard DC solenoid white light	OUT
Outboard AC solenoid white light	OUT

Which one of the following identifies the power supply that has been lost?

A. Division I AC Power

B. Division I DC Power

C. Division II AC Power

D. Division II DC Power

Answer: D

K/A: 239001 Main and Reheat Steam System
K2.01 - Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids (CFR: 41.7)

RO/SRO Rating: 3.2/3.3

Objective: LOI-CLS-LP-025, Obj. 5 - List the power supplies (division and voltage) for the MSIV Solenoids

Reference: None

Cog Level: Low

Explanation: Each MSIV operator contains two AC solenoids and one DC solenoid. One of the AC solenoids is used for valve stroke testing at power and is called the Slow Closure Test Solenoid. The other two solenoids (one AC and one DC) determine the position of the MSIV by porting or venting the pneumatic source to or from the operator. Both of these solenoids must be deenergized for the MSIV to be closed. The AC solenoids are powered from the Reactor Protection System and the DC solenoids are powered from the Station Battery System

Distractor Analysis:

Choice A: Plausible because this is a power supply to the MSIV solenoids but not the one that is de-energized for this example.

Choice B: Plausible because this is a power supply to the MSIV solenoids just but the one that is de-energized for this example.

Choice C: Plausible because this is a power supply to the MSIV solenoids but not the one that is de-energized for this example.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

TABLE 25-3, MSIV ISOLATION SIGNAL STATUS

Light	INBD DC	INBD AC	OUTBD DC	OUTBD AC
Solenoid Power	125 VDC "A"	RPS "A"	125 VDC "B"	RPS "B"
PCIS Logic	B	A	A	B

X LIT X Y

20. 239002 1

Which one of the following describes the effect that a loss of MCC 1XDB will have on the Unit One SRVs, if needed for pressure control operations from the RSDP?

SRVs B, E, and ____(1)____ will lose ____(2)____.

A. (1) F
   (2) position indication ONLY

B. (1) F
   (2) ALL control and indications

C. (1) G
   (2) position indication ONLY

D. (1) G
   (2) ALL control and indications

Answer: D

K/A: 239002 Relief/Safety Valves
K3.01 - Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on following: Reactor pressure control (CFR: 41.7 / 45.4)

RO/SRO Rating: 3.9/4.0

Objective: CLS-LP-204. State which SRVs may be controlled from the Remote Shutdown Panel.
15d. Given plant conditions, predict how ADS/SRVs will be affected by the following: Loss of DC power

Reference: None

Cog Level:High

Explanation: SRVs B,E,& G require 125 VDC to operate manually from the Remote Shutdown Panel. Unit One RSDP control & indication for these SRVs is powered from MCC 1XDB (in RB). This a Unit difference in that Unit Two RSDP SRV control & indication is powered form DP 2B (in DG Bldg).

Distractor Analysis:

Choice A: Plausible because SRV F is a non-ADS valve which has an alternate ASSD power supply but can only be controlled from the main control room.

Choice B: Plausible because SRV F is a non-ADS valve which has an alternate ASSD power supply but can only be controlled from the main control room.

Choice C: Plausible because SRV G has control on the RSDP, but unlike the control room, only one DC power supply is available to provide both control and indication.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

From OI-50

ATTACHMENT 1B
Page 26 of 28

MCC 1XDB Reference Drawing: F-30008	LOCATION: Unit 1 Reactor Building 20' Southeast	NORMAL SUPPLY: Switchboard 1B	ALTERNATE SUPPLY: N/A

BREAKER	LOAD	EFFECT
B53	Remote Shutdown Panel Instrumentation	1. Loss of power to 1-E51-FIC-3325, RCIC RSDP Flow Controller. 2. Loss of 1-E51-FT-N003, RCIC RSDP flow indication. 3. Loss of 1-B21-LSH-N017D-3, RCIC RSDP high-level trip. 4. Loss of 1-E11-FI-3338, RSDP RHR system flow. 5. Loss of 1-C32-PT-3732, RSDP Reactor Pressure indication. 6. Loss of 1-B21-LI-6977, RSDP Rx water level indicator. 7. Loss of 1-CAC-LI-3342, RSDP torus water level indication. 8. Loss of 1-CAC-PI-3341, RSDP Drywell pressure indication. 9. Loss of 1-B21-LI-R604-BX, RSDP Rx water level indication. 10. Loss of 1-CAC-TR-778, RSDP temperature recorder. 11. Loss of 1-E51-F045 position input to the EGM for the ramp function initiation and reset. 12. RSDP SRV control and indication. (SRV's B, E&G) 13. Loss of ASSD function. 14. Loss of 1-RCC-TR-778, RSDP Penetration Cooling temperature recorder.

ATTACHMENT 2A
Page 7 of 28

PANEL 4A Reference Drawing: LL-03024-6	LOCATION: Control Building 49' East	NORMAL SUPPLY: Switchboard 2A	ALTERNATE SUPPLY: N/A

CKT #	LOAD	EFFECT
4 (cont'd)	RHR Div I Low Pressure Permissive (410#)	1. RHR Div I Low Pressure Permissive INOP. (Channels A1-B1) 2. Either Loop can function with Div. II Low Pressure Instrumentation.
5	Recirc Pump A Auxiliary Equipment Alternate Control Power	1. Loss of alternate control power to ATWS Trip Logic A. 2. Normal power is from Panel 10A, Ckt. 3.
6	Backup Scram valve, 2-C12-F110A	1. Backup Scram valve will not reposition on a full scram, Div II Backup Scram valve will still function.
	Div I Backup Scram Logic	1. Scram Discharge Volume Vent and Drain Valves will not receive close signal from Div I, all valves will still function from Div II. 2. DFWLCS will not receive auto set down from Div I Digital Feedwater will not set down to 170". 3. Ten second time delay prior to scram reset, will not function for A RPS.
7	Spare	Spare
8	Spare	Spare
9	Reactor Water High Level Trip "C"	1. De-energizes for trip function. 2. ½ of high level trip is sealed in, requires 2 out of 3 to get full trip. 3. C high level amber light will not illuminate.
10	Spare	Spare
11	ADS Logic B alternate power	1. Loss of alternate power, B logic will still function with normal power.
	ADS/ SRV alternate solenoid power	1. Loss of alternate power, all valves will fully function with normal power. 2. Loss of ASSD power to SRV F.

21. 239002 2

Unit Two is operating at rated power with no activities in progress.

A leaking SRV has resulted in slowly rising Suppression Pool temperature

Which one of the following identifies the required actions to be performed IAW 0AOP-30, Safety/Relief Valve Failures?

A. Place Suppression Pool Cooling in service. When Suppression Pool temperature exceeds 95°F, enter PCCP.

B. Place Suppression Pool Cooling in service. When Suppression Pool temperature exceeds 105°F, enter PCCP.

n.p. C. When Suppression Pool temperature exceeds 95°F, SCRAM the reactor and enter PCCP.

n.p. D. When Suppression Pool temperature exceeds 105°F, SCRAM the reactor and enter PCCP.

As soon as [illegible] temp exceeds 95°F ________,

As soon as [illegible] temp exceeds 105°F ________,

Answer: A

T2 G1

K/A: 239002 Relief/Safety Valves
G2.04.02 - Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8)

RO/SRO Rating: 4.5/4.6

Objective: LOI-CLS-LP-300-L, Obj. 02 - Given plant conditions, determine if the Primary Containment Control Procedure should be entered.

Reference: None

Cog Level: High

Explanation: A leaking SRV requires entry into 0AOP-30, Safety/Relief Valve Failures, which directs placing Suppression Pool Cooling in service if pool temperatures are increasing due to a leaking SRV. If Suppression Pool temperature reaches 95°F and no testing which could add heat to the torus is in progress, (105°F if testing is being performed) then EOP-02-PCCP would be entered. A reactor SCRAM is not required in this condition until Suppression Pool Temperature reaches 110°F (ref EOP-02-PCCP, SP/T-07)

Distractor Analysis:

Choice A: Correct answer, see explanation

Choice B: Plausible because 105°F is a valid entry value for PCCP entry if testing is in progress which would add heat to the suppression chamber.

Choice C: Plausible because 95° is a valid entry value for PCCP if no testing in progress which adds heat to the torus but SCRAM is not required until 110°F.

Choice D: Plausible because105° is a valid entry value for PCCP if no testing in progress which adds heat to the torus but SCRAM is not required until 110°F

SRO Basis: N/A

AOP-30

3.2.4 **IF** a safety/relief valve is leaking, **THEN PERFORM** the following:

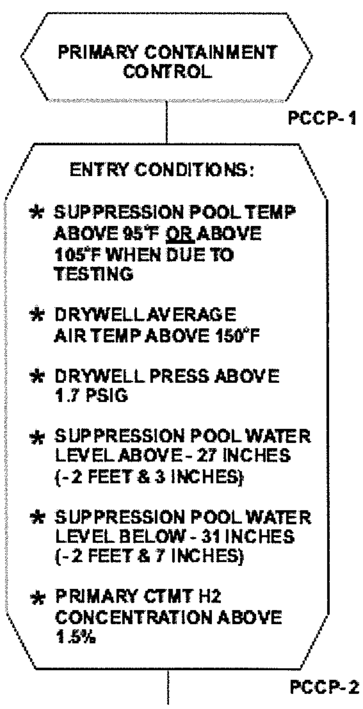
1. **MONITOR** tailpipe temperatures. ☐
2. **MONITOR** primary containment parameters. ☐
3. **REFER** to 0AOP-14.0. ☐
4. **IF** leakage is causing the suppression pool temperature to increase, **THEN PLACE** Suppression Pool Cooling in service in accordance with 1(2)OP-17, as necessary. ☐

3.2.1 **IF** a safety/relief valve is stuck open, **THEN REDUCE** reactor power in anticipation of a reactor scram. ☐

3.2.2 **IF** suppression pool temperature increases to 110°F, **THEN PERFORM** the following:

1. **INSERT** a manual reactor SCRAM. ☐

0EOP-02-PCCP



22. 245000 1

Unit Two is at 20% power with main turbine roll in progress IAW 2OP-26, Turbine System Operating Procedure. The following turbine journal bearing vibration readings are observed on TSI-XR-640:

Bearing #1	5 mils	Bearing #6	10 mils
Bearing #2	5 mils	Bearing #7	11 mils
Bearing #3	6 mils	Bearing #8	13 mils
Bearing #4	7 mils	Bearing #9	11 mils
Bearing #5	8 mils	Bearing #10	10 mils

Turbine speed is 900 RPM and rising.
Which one of the following identifies the impact of the vibration readings ~~on turbine operation and what~~ operator action is ~~required~~ IAW 2OP-26?

A. The turbine should have automatically tripped.
Trip the main turbine ONLY.

B. The turbine should have automatically tripped.
Scram the reactor and then trip the main turbine.

C. The turbine should NOT have automatically tripped.
Trip the main turbine ONLY.

D. The turbine should NOT have automatically tripped.
Scram the reactor and then trip the main turbine.

Answer: A

K/A: 245000 Main Turbine Generator and Auxiliary Systems
A2.09 - Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine vibration
(CFR: 41.5 / 45.6)

RO/SRO Rating: 2.5/2.8

Objective: LOI-CLS-LP-026, Obj. 28n - Given plant conditions, predict the effect that the following will have on the Main Turbine, Gland Seal, and Moisture Reheater System: Main Turbine Hi Vibration

Reference: None

Cog Level: High

Explanation: TSI is normally disarmed during turbine operation (>23% power with the turbine online), but is armed for turbine roll. When turbine RPM is between 801 - 1400 RPM, the trip setpoints are 12 mils for bearings 1-8 and 10 mils for 9 & 10. This requires an immediate turbine trip per OP-26, Section 5.4.2. If reactor power is greater than 26% then a reactor scram will occur when the turbine is tripped. Operator actions call for scramming the reactor first, then tripping the turbine. In the described conditions, a scram is not required since power is below 26%.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because if power was greater than 26% then a reactor scram would be required.

Choice C: Plausible because TSI is normally bypassed (but in this case it is not for the startup) and a turbine trip only is required.

Choice D: Plausible because TSI is normally bypassed (but in this case it is not for the startup) and a scram is not required at less than 26% power.

SRO Basis: N/A

From OP-26, Turbine Startup:

17. **IF** any of the following conditions occur while turbine speed is increasing, **THEN DEPRESS** the *EMERGENCY TRIP SYSTEM* push button to trip the turbine **AND ENSURE** the following turbine valves close: ☐

All four Stop Valves ☐

All four Control Valves ☐

All four Intermediate Stop Valves ☐

All four Intercept Valves ☐

1OP-26	Rev. 84	Page 41 of 153

5.4.2 Procedural Steps

a. Turbine shell to rotor differential expansion indicating in red band, as indicated on *TSI-TXR-638* Point 1. ☐

b. Turbine journal bearing high vibration, as indicated on *TSI-XR-640*: ☐

Turbine Speed	vs.	Vibration
Less than or equal to 800 rpm		Greater than 8 mils.
801 to 1400 rpm		Greater than 12 mils for bearings V1-V8, or greater than 10 mils for bearings V9 and V10.

From the APP:

ACTIONS

MAIN TURBINE:

NOTE:	The main turbine high vibration trip is disabled when operating at or above 23% rated thermal power with the generator online.

b. **IF** vibration is at or above 12 mils on bearings 1-8 or 10 mils on bearing 9 & 10 **AND** an adjacent bearing has also exhibited a significant increase in vibration, **THEN PERFORM** the following:

(1) Scram the reactor

(2) Trip the turbine

(3) **IF** directed by the Unit SCO, **THEN BREAK** condenser vacuum.

(4) **ENTER** 1EOP-01-RSP **AND EXIT** this procedure.

23. 256000 1

Unit One is operating at 30% power with the following plant conditions:

Hotwell temperature	118°F
Condensate Pump A	Standby
Condensate Pump B/C	Running

Debris in the Off-Gas System begins to plug the Off-Gas Filter.

Which one of the following choices completes the statements below if debris continues to build up on the Off-Gas Filter?

The plant effect is that condenser vacuum will lower and ____(1)____.

IAW 0AOP-37.0, Loss of Condenser Vacuum, efficiency of the operating Steam Jet Air Ejector (SJAE) can be improved by throttling the SJAE Condensate Recirculation Valve, CO-FV-49, open as long as condensate flow does NOT exceed ____(2)____ gpm.

A. (1) Hotwell temperatures will increase
   (2) 14,400

B. (1) Hotwell temperatures will increase
   (2) 16,000

C. (1) Off Gas system will auto bypass
   (2) 14,400

D. (1) Off Gas system will auto bypass
   (2) 16,000

Answer: A

K/A: 256000 Reactor Condensate System
K6.09 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CONDENSATE SYSTEM: Offgas system (CFR: 41.7 / 45.7)

RO/SRO Rating: 2.6/2.6

Objective: LOI-CLS-LP-032-A, Obj. 20 - Given plant conditions predict the effects that a loss or malfunction of the following will have on the Feed and/or Condensate System: Off-Gas System.

Reference: None

Cog Level: High

Explanation: Lowering condenser vacuum is from a malfunction of the Off Gas system. As condenser vacuum decrease hotwell temperature will increase. AOP-37 provides guidance for maintaining the efficiency of the SJAE by throttling of the CO-FV-49 valve open and flow is limited by the table according to the pump arrangement. With B&C pump flow is limited to 14400 gpm.

Distractor Analysis:

Choice A: Correct answer, see explanation

Choice B: Plausible because hotwell temperatures will increase and if a different pump arrangement was in affect then 16000 gpm would be correct.

Choice C: Plausible because 14400 is correct but off gas will bypass on high flow conditions not low flow.

Choice D: Plausible because 16000 would be correct for a different pump arrangement and off gas will bypass on high flow conditions not low flow

SRO Basis: N/A

AOP-37

NOTE:	SJAE efficiency at low reactor power may be improved by throttling open *SJAE CONDENSATE RECIRCULATION VALVE, CO-FV-49*, while maintaining condensate pump discharge header pressure greater than 190 psig.

3.2.6 **IF** desired, **THEN PERFORM** the following:

1. **ENSURE** *VALVE CO-FV-49 INLET ISOLATION VALVE, CO-V110*, is open. ☐

3.0 OPERATOR ACTIONS

2. **THROTTLE OPEN** *SJAE CONDENSATE RECIRCULATION VALVE, CO-FV-49*, **NOT** to exceed the flow limits in the following table. ☐

Condensate Pumps Operating	Average Hotwell Temperature**(Note 1)**	Condensate Flow Limit (GPM) **(Note 2)**
B & C	N/A	14,400
A & B **or** A & C	≥ 115°F	16,000
A & B **or** A & C	< 115 °F	17,400
A, B, & C	N/A	18,200

NOTE 1: Average hotwell temperature is available from process computer point *U1(U2) CO_C099 AVG HOTWELLTEMP U1(U2)*. **IF** *U1(U2) CO_C099* is **NOT** available, **THEN** use the highest functional hotwell temperature process computer point.

NOTE 2: Condensate flow through each of the *CONDENSATE PUMP RECIRC VLVs, CO-FV-147, (CO-FV-148), (CO-FV-149)* of 1500 gpm will **NOT** be indicated on *AIR EJECTOR COND FLOW RECIRC CTL, CO-FIC-49,* and must be considered in the following system flow limits. In addition, total condensate flow may be influenced by hotwell reject flow, RFP or condensate booster pump minimum flow valves opening, heater drain tank level controller positions, feedwater heaters out of service, etc.

24. 259002 1

Unit Two is operating at 65% power with Reactor Feed Pump (RFP) 2A running and RFP 2B unavailable. The operator observes the following:

RFP A Control Trouble alarm is received
RFP A Manual/DFCS selector switch is in DFCS
DFCS Control light for RFP A on XU-1 is out

Which one of the following choices completes the statements below?

RFP 2A speed will ____(1)____.
The operator can control RFP speed by ____(2)____.

A. (1) drop to the idle speed setpoint
    (2) operating the RFP Raise/Lower control switch on XU-1

B. (1) drop to the idle speed setpoint
    (2) placing the RFP A Speed Controller in Manual and adjusting the output demand

C. (1) remain at the last known demand
    (2) operating the RFP Raise/Lower control switch on XU-1

D. (1) remain at the last known demand
    (2) placing the RFP A Speed Controller in Manual and adjusting the output demand

Answer: C

K/A: 259002 Reactor Water Level Control System
A3.10 - Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: TDRFP lockup: TDRFP (CFR: 41.7 / 45.7)

RO/SRO Rating: 3.1/3.0

Objective: LOI-CLS-LP-032.2, Obj. 08j - Given the following plant conditions, predict the response of the RFPT and Speed Control System: Loss of signal from the DFCS

Reference: None

Cog Level: High

Explanation: For Brunswick a TDRFP lockup is being met by the loss of signal locking the RFP controls at the last known signal. IF RFPT B(A) *MAN/DFCS* selector switch is in *DFCS,* AND the DFCS control signal subsequently drops below 2450 rpm, OR increases to greater than 5450 rpm, THEN Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current speed. In this condition, the RFPT will only respond to *LOWER/RAISE* speed control switch commands

Distractor Analysis:

Choice A: Plausible because the Woodward manual control signal automatically tracks the DFCS output signal. Although this failure mechanism no longer exists, a drop in speed to the idle speed setpoint is a failure that used to be associated with a loss of hydraulic oil pressure.

Choice B: Plausible because the DFCS control signal has failed. With the DFCS Control light out, the RFP is under manual control of the Woodward governor and adjusting the output of the individual RFP Speed Controller will have no effect. A drop in speed to the idle speed setpoint is a failure that used to be associated with a loss of hydraulic oil pressure. This failure mechanism no longer exists.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the Woodward manual control signal automatically tracks the DFCS output signal. With the DFCS Control light out, the RFP is under manual control of the Woodward governor and adjusting the output of the individual RFP Speed Controller will have no effect.

SRO Basis: N/A

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

3.1.1 **IF** automatic level control will **NOT** restore normal reactor vessel level, **THEN MANUALLY CONTROL** reactor feed pumps to restore normal level. ☐

0AOP-23.0	Rev. 37	Page 3 of 10

NOTE: **IF** RFPT B(A) *MAN/DFCS* selector switch is in *DFCS*, and the DFCS control signal subsequently drops below 2450 rpm, or increases to greater than 5450 rpm, Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current speed. In this condition, the RFPT will only respond to *LOWER/RAISE* speed control switch commands until the *MAN/DFCS* selector switch is placed in *MAN*, DFCS *CTRL RESET* pushbutton is depressed, and the *MAN/DFCS* selector switch returned to *DFCS*.

2OP-32	Rev. 180	Page 49 of 330

4.2.4 DFCS Control Signal Failure

If the 5009 control system detects that the Remote Speed Setpoint (RSS) from the DFCS is outside the failure limits, an RSS signal failure condition is set and, if the 5009 control system was in the DFCS mode, an automatic transfer to the manual mode will occur. The RFPT speed setpoint (and hence RFPT speed) will be maintained at the last "good" value and can be controlled using the Panel XU-1 RAISE / LOWER switch (Figure 32.3-14). The MANUAL / DFCS switch should be placed in the MANUAL position.

SD-32.3	Rev. 5	Page 66 of 123

25. 261000 1

Venting of the suppression chamber is being performed IAW 1OP-10, Standby Gas Treatment System Operating Procedure. SBGT system valve status:

1-CAC-V172, Supp Pool Purge Exh Vlv	Open
1-CAC-V22, Torus Purge Exh Vlv	Open
1-VA-1D-BFV-RB, Reactor Building SBGT Train 1A Inlet Valve	Closed
1-VA-1H-BFV-RB, Reactor Building SBGT Train 1B Inlet Valve	Closed

A transient occurs which causes Drywell pressure to rise to 1.5 psig and Reactor water level to lower to 160 inches before being recovered.

Which one of the following choices completes the statements below concerning the expected response, if any, of Suppression Chamber Purge and SBGT system valves?

The 1-CAC-V172 and 1-CAC-V22 ___(1)___.
The 1-VA-1D-BFV-RB and 1-VA-1H-BFV-RB ___(2)___.

A. (1) close
(2) remain closed

B. (1) close
(2) open

C. (1) remain open
(2) remain closed

D. (1) remain open
(2) open

Answer: A

K/A: 261000 Standby Gas Treatment System
K1.03 - Knowledge of the physical connections and/or cause effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Suppression pool (CFR: 41.2 to 41.9 / 45.7 to 45.8)

RO/SRO Rating: 2.9/3.1

Objective: LOI-CLS-LP-004.1, Obj. 5 - List the signals and setpoints that will cause a Secondary Containment isolation

LOI-CLS-LP-012, Obj. 06 - Given plant conditions, determine if a Group Isolation should occur.

Reference: None

Cog Level: High

Explanation: The suppression pool vent valves close with a group 6 isolation signal while the SBGT RB suction valves open on a Secondary Containment Isolation signal. Based on the level indication a LL1 signal is present which would actuate a Group 6 isolation. A SCI signal occurs at LL2 (105 inches). Both actuate with DW pressure greater than 1.7 psig. DW pressure of 1.5 psig actuates an alarm only.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because there is a group 6 isolation due to low water level. If level were below LL2 (105") then this would be correct.

Choice C: Plausible because if level was above 166 inches then this would be correct. There is no isolation signal from DW pressure.

Choice D: Plausible because there is a group 6 signal but not a SCI signal. If drywell pressure were greater than 1.7 psig or level were below 105", an SCI would occur.

SRO Basis: N/A

TABLE 12-2
Primary Containment Isolation System Group Isolation Instrumentation Setpoints

ISOLATION GROUP	ISOLATION SIGNAL	TRIP SETPOINT		NOTES
		Tech Spec. Allowable	Actual (Note 1)	
Group 5	High Steam Flow	≤ 275%	220%	Note 5
	Low Steam Pressure	≥ 53psig	70 psig	
	High Turb Exh Pressure	≤ 6 psig	5 psig	
	Steam Line Area Hi Temp	≤ 175°F	165°F	Note 4
	Steam Line Tunnel High Amb Temp	≤ 200°F	165°F/190°F	Note 4
	Steam Line Tunnel dT High	≤ 50°F	47°F	Note 4
	Equip Area High Temp	≤ 175°F	165°F	
	Equip Area dT High	≤ 50°F	47°F	
Group 6	Low Level #1	≥ 153"	166"	
	High Drywell Pressure	≤ 1.8 psig	1.7 psig	
	Rx Bldg Exhaust Hi Rad	≤ 16 mR/hr	4 mR/hr	
	Rx Bldg Exhaust Hi Temp	N/A	135°F*	Note 6
	High Main Stack Rad	ODCM	ODCM	Note 2

The following CAC valves (Unit 1/2 power supply listed in parentheses) close on a Group 6 Isolation (Figure 12-11):

Division I

AC Powered

CAC-V5	Suppression Pool Nitrogen Inlet	31AB/32AB
CAC-V6	Drywell Nitrogen Inlet	31AB/32AB
CAC-V7	Suppression Pool Purge Exhaust	31AB/32AB
CAC-V9	Drywell Purge Exhaust	31B/32B
CAC-V162	Suppression Pool CAD N_2 Inlet	31AB/32AB
CAC-V163	Drywell Nitrogen Inlet	31AB/32AB
CAC-V172	Suppression Pool Purge Exhaust	31AB/32AB

DC Powered

CAC-V49	Drywell Head Purge Exhaust	11A/12A
CAC-V160	Suppression Pool CAD N2 Inlet	11A/12A
CAC-V161	Drywell Nitrogen Inlet	11A/12A

Division II

AC Powered

CAC-V4	CAC Nitrogen Inlet	31AB/32AB
CAC-V8	Suppression Pool Purge Exh Byp	31B /32B
CAC-V10	Drywell Purge Exh Backup	31B/ 32B
CAC-V15	Prim Cont Air Purge Inlet	31AB/32AB
CAC-V22	Suppression Pool Exh	MCC 1XD/2XD
CAC-V23	DW Purge Exh Backup Byp	MCC 1XF/2XF

2.6 Standby Gas Treatment

SBGT provides a means of minimizing the release of radioactivity from Secondary Containment by filtering and exhausting containment air. Two trains are provided, each consisting of a fan and filtration devices which remove particulate and iodine prior to exhausting to the main plant stack. The units draw a suction on the 50' elevation of the Reactor Building into which all areas of the Reactor Building communicate. This system can be placed into service manually or will enter service automatically under the same conditions that RB HVAC will isolate. The parameters monitored for SBGT initiation and RB isolation are:

PARAMETER

Low Reactor Water Level #2

High Drywell Pressure

Main Stack High Radiation

Reactor Building Ventilation
Exhaust High Temperature

Reactor Building Ventilation
Exhaust High Radiation

Any of these signals initiate the following sequence of events:

- Closes the isolation dampers which stops the fans in the Reactor Building Ventilation System.
- Starts both SBGT fans simultaneously.
- Closes the isolation valves and stops the fans in the Purge System.
- Opens the SBGT inlet and outlet isolation valves (U2 only).
- Opens the SBGT Reactor Building suction valves.
- Closes the SBGT Primary Containment suction valves.

26. 261000 2

Which one of the following requires the SBGT Train A to be manually reset in order to restore auto start capability?

A. When the fan electrical power is lost and then restored.

B. When the control switch is repositioned from STBY to SYST A PREF.

C. ONLY when the inlet temperature (TS-7) reaches 180°F ~~and subsequently~~ lowers ~~to <180°F~~.

D. ONLY when the prefilter temperature (TS-1) reaches 210°F and subsequently lowers to <210°F.

Answer: D

K/A: 261000 Standby Gas Treatment System
K4.04 Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Radioactive particulate filtration (CFR: 41.7)

RO/SRO Rating: 2.7/2.9

Objective: LOI-CLS-LP-010, Obj. 06 - Describe the function of the temperature switches under abnormal operations.

Reference: None

Cog Level: Low

Explanation: If the 210 degree prefilter temperature is reached the SBGT Fan will shutdown and in order to reset the temperature must have reduced below the setpoint.

Distractor Analysis:

Choice A: Plausible because logic power failure may require a reset but the fan itself does not.

Choice B: Plausible because this action will place the system in auto initiation mode, but no manual reset is required.

Choice C: Plausible because inlet temperature greater than 180 would initiate the emergency operation logic for the SBGT train.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

2. **Prefilter Compartment**

(TS-1)
The Prefilter compartment is equipped with two switches per train. Switches VA-TS-5301 (VA-TS-5296) actuate at 210°F to secure the Fan and Heater if the train is operating. Local and remote lights indicate actuation of the temperature switches.

The system starter circuit must be manually reset after actuation of these switches and after the temperature has dropped to less than the set point by momentarily placing the train selector switch to the RESET position.

(TS-2)
Switches VA-TS-5300 (VA-TS 5295) actuate at 180°F to control the heater to regulate the train inlet temperature when the Fan is running. Local and remote lights indicate temperature switch actuation.

3.2.1 Control Room: RTGB XU-51

1. The four-position, RESET / SYST A(B) PREF / STBY / ON, selector switch controls the status of the SBGT train. The RESET position is a momentary contact position with spring-return to the SYST A (B) PREF position. The other three positions are maintained contact positions. The RESET position allows resetting the Starter Circuit when a manual reset is required following actuation of the Prefilter Compartment high temperature switches at 210°F.

   The SBGT A (B) PREF position enables the train to start automatically if an initiation signal is present. Placing the switch in the ON position will start the Fan provided all other start permissives are met.

   The STANDBY position is misleading in that, with the switch in this position, the train is <u>NOT</u> placed in the condition where it is ready to automatically start. Instead, the automatic start feature is defeated.

27. 262001 1

Unit Two is operating at rated power when a Transformer Bus Lockout occurs.

Which one of the following identifies the equipment that will remain de-energized with no operator actions?

A. Recirc VFDs 2A and 2B.

B. Demin Water Xfer Pmps 2A, 2B, and 2C.

C. Circulating Water Discharge (CWOD) Pumps 2C and 2D.

D. ISFSI Drain Collection Pumps (2-DST-ISFSI-P1-PMP and 2-DST-ISFSI-P2-PMP).

Answer: A

K/A: 262001 A.C. Electrical Distribution
K2.01 - Knowledge of electrical power supplies to the following: Off-site sources of power (CFR: 41.7)

RO/SRO Rating: 3.3/3.6

Objective: LOI-CLS-LP-050-A, Obj. 08c - Given plant conditions predict the changes in Unit 1 and/or 2 parameters associated with the operation of the following equipment: Transformer bus lockout relay

Reference: None

Cog Level: Low

Explanation: A Transformer Bus overcurrent trip will result in the SAT being de-energized and the Feed to Caswell Beach Bus B. The SAT feeds Bus 2B and Common Bus B. Common Bus B would auto crosstie to Common Bus A and remain energized. Caswell Beach Bus B can be transferred to Caswell Beach Bus A, but this is a dead bus transfer (manual, not automatic). Bus 2B feeds the VFDs which would require the plant to insert a manual scram due to no recirc pumps.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because these are powered from the SAT feed but have auto-crosstie feature to U1. Even though these are U2 designated pumps they are normally fed from Common A through Common C to WTA.

Choice C: Plausible because these are powered from an offsite power source (U1 Transformer Bus to Caswell Beach Bus A).

Choice D: Plausible because this is an offsite power source from the Southport Feeder.

SRO Basis: N/A

1.3.1 **System Components and Configuration** (Figures 1, 2, and 3)

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance Of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II). (Se Figure 50.1-1)

The BOP Common Buses A/B are powered from the respective unit's SAT and have auto dead bus crosstie capability should one lose its normal source. The Caswell Beach Buses A/B are powered from the respective unit's Caswell Beach transformers and have manual dead bus crosstie capability. The B/C/D Buses can be powered from the respective unit's UAT or SAT. Buses B, C and D have their source breakers interlocked to prevent parallel operation of the power sources. Buses B, C and D are provided with manual bus transfer schemes that allow momentary parallel operation of the power sources while transferring power supplies. Buses C/D are normally

powered from the respective unit's UAT during power operations and have an auto dead bus fast transfer to the SAT on loss of the UAT (any generator lockout). The B Bus has no auto transfer capability, therefore is normally supplied from the respective unit's SAT.

Load: **480V Motor Control Center WTA**
Location: Water Treatment Building 20'
Drawing Reference: F-3052
Upstream Power Source: **480V Substation Common C**

COMPT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
C73	Paint and Sandblasting Shop (OP-52.1)	Loss of power to 480V panel HA in Sandblasting Shop.
C78	Demin. Water XFER Pump 2A (OP-31.2)	Loss of load.
C79	Demin. Water XFER Pump 2B (OP-31.2)	Loss of load.
C80	Demin. Water XFER Pump 2C (OP-31.2)	Loss of load.

Load: **Caswell Beach 4160V Bus A**
Location: Caswell Beach Pumping Station
Drawing Reference: F-03025
Upstream Power Source: **Unit 1, Caswell Beach Transformer No 1**

COMPT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
AN1	Circulating Water Discharge Pump 2C (2OP-29)	Loss of load.
AN2	Circulating Water Discharge Pump 2D (2OP-29)	Loss of load.

The affects of a loss of the Southport Feeder:

6.10 Loss of power to the ISFSI Drain Collection Pump Station

6.10.1 Loss of power to ISFSI Drain Collection Pumps (2-DST-ISFSI-P1-PMP and 2-DST-ISFSI-P2-PMP)

28. 262002 1

Unit Two is operating at rated power when a Primary UPS Inverter malfunction results in an overvoltage on the inverter output.

The following annunciators are in alarm:

UPS Primary Power Convtr Trouble
UPS Transfer to Reserve

Which one of the following choices completes the statement below?

IAW the above APPs, the required operator action is to transfer UPS loads from ___(1)___ to ___(2)___.

A. (1) the standby UPS inverter
   (2) MCC 2CA

B. (1) the standby UPS inverter
   (2) MCC 2CB

C. (1) MCC 2CA
   (2) the standby UPS inverter

D. (1) MCC 2CB
   (2) the standby UPS inverter

Answer: D

K/A: 262002 Uninterruptable Power Supply (A.C./D.C.)
A2.02 - 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: Over voltage
(CFR: 41.5 / 45.6)

RO/SRO Rating: 2.5/2.7

Objective: LOI-CLS-LP-052-A, Obj. 7a - Predict the impact(s) of the following on the UPS System: Over Voltage.

Reference: None

Modified question from the NRC 10-2 exam. (262002_14) The previous question asked the power supply while this question asks the action required.

Cog Level: High

Explanation: Overvoltage or undervoltage on the UPS inverter output will cause an automatic transfer of the UPS loads. In normal alignment, the primary UPS will be in service supplying loads and the standby UPS will be energized. However, the standby UPS alignment is such that it will not automatically pick up the loads if the primary unit fails. This protects the standby unit from automatically tying onto a faulted bus. The transfer scheme instead automatically transfers the loads to the Division II AC source (hard source). The standby inverter must be manually placed in service. This configuration protects the standby UPS but results in vulnerability in that if the Division II AC source is lost, the UPS loads would be deenergized. APP-UA-06 6-9, UPS Transfer to reserve, provides direction to place the UPS loads on an operable inverter. Since the primary inverter is inoperable, the standby inverter would be placed in service.

Distractor Analysis:

Choice A: Plausible because the original design of the system was to have the standby inverter automatically pick up the loads. OP-52 provides a procedure section to transfer UPS loads to the alternate source.

Choice B: Plausible because the original design of the system was to have the standby inverter automatically pick up the loads. OP-52 provides a procedure section to transfer UPS loads to the alternate source. Inverter power supplies are from 2CA and 2CB.

Choice C: Plausible because MCC 2CA is the normal (and only) AC power supply to the primary inverter. However, there is no hard connection or procedure instructions to align 2CA directly to the loads. APP-UA-06 6-9 directs transferring UPS loads to an operable inverter.

Choice D: Correct answer, see explanation

SRO Basis: N/A

SD-52

1.3.3. Uninterruptible Power Supplies

Each unit's Vital UPS System consists of a vital bus (Distribution Panel 1A/2A), a primary unit, and a standby unit. The primary and standby units are each capable of supplying the total UPS load. Only one UPS unit will be in service at a time with the other unit being available as a backup. If the primary unit fails, the UPS loads are automatically transferred to the alternate AC source. If desired, the loads can then be manually transferred to the standby UPS unit.

The UPS system is normally aligned as follows:

The primary unit is in service with its output connected to the UPS distribution system. Its rectifier receives 480 VAC power from a Division I emergency distribution panel. A 250 VDC from DC Switchboard 1A (2A) is supplied in parallel with the rectifier output to power the inverter should the normal AC source be lost. The alternate AC source from the standby unit is available at the static transfer switch to pick up the loads if the inverter output is lost.

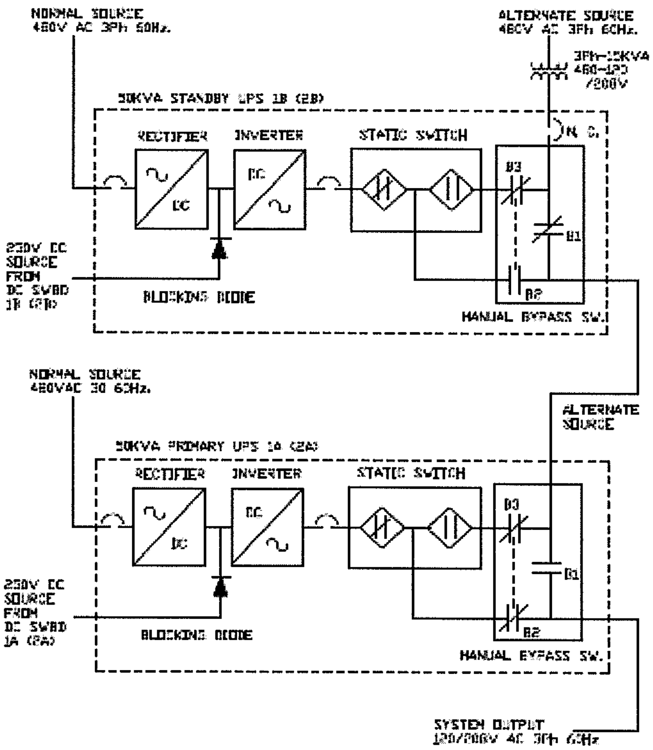
The standby unit is also energized with its 480 VAC input supplied from a Division II emergency distribution panel and its 250 VDC supplied from DC Switchboard 1B (2B); but, its output is bypassed by its manual bypass switch and its alternate AC input is being supplied directly to the primary unit. The standby unit receives its alternate AC source of power from the same Division II distribution panel as its rectifier AC input through a 480-120/208 VAC transformer. The standby units alternate AC input is also referred to as the hard source.

TABLE 52-1
Vital UPS Power Supplies

	UNIT 1	UNIT 2
480 VAC SUPPLY PRIMARY UNIT STANDBY UNIT	MCC 1CA (E5) MCC 1CB (E6)	MCC 2CA (E7) MCC 2CB (E8)
250 VDC SUPPLY PRIMARY UNIT STANDBY UNIT	DC SWBD 1A DC SWBD 1B	DC SWBD 2A DC SWBD 2B
ALTERNATE AC	MCC 1CB (E6)	MCC 2CB (E8)

The static transfer switch provides a means of switching either manually or automatically between two, three-phase, four wire power sources without an interruption of the load power. The switch performs a transfer between the power sources using solid state components; allowing a faster transfer than could be accomplished by a mechanical switch. The switching action itself is practically instantaneous, and the time involved in the operation is mainly the sensing time required to determine that a transfer is necessary, normally a small fraction of a cycle. The switch receives an inverter input through the inverter output breaker, CB102, and a input from a alternate AC source. The switch is normally aligned such that the inverter output is supplying the system loads. Should the inverter output voltage drop below or increase above a preset level the static transfer switch will automatically transfer to the alternate AC source. Both a high or a low output voltage could affect equipment operating characteristics and lead to equipment damage.

FIGURE 52-7
Basic Vital UPS System



2APP-UA-06

UPS TRANSFER TO RESERVE

ACTIONS

1. If UPS loads have automatically transferred to the alternate source, transfer UPS loads back to an operable UPS unit per 2OP-52, Section 8, Infrequent Operations.

29. 263000 1

Which one of the following completes the statements below regarding 125/250 VDC Station Distribution?

During an equalize charge, the charger output to the battery will be at a ___(1)___ voltage than when in the float mode.

The 125 VDC batteries are sized to supply emergency power at a 150 amp rate for ___(2)___ hours.

A. (1) lower
   (2) 8

B. (1) lower
   (2) 10

C. (1) higher
   (2) 8

D. (1) higher
   (2) 10

Answer: C

K/A: 263000 D.C. Electrical Distribution
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate
(CFR: 41.5 / 45.5)

RO/SRO Rating: 2.5/2.8

Objective: LOI-CLS-LP-051, Obj. 13 - Describe the location and operation of Battery Chargers 1B-1, 1B-2, 2B-1, and 2B-2 AC Power Transfer Switches.

Reference: None

Cog Level: Low

Explanation: The float mode voltage for the 125 VDC battery charger is ~135 volts while in equalize the charger output is ~140 volts. The design of the batteries is for 150 amps for 8 hours.

Distractor Analysis:

Choice A: Plausible because the student may have a knowledge deficiency on which (float vs equalize) value is for equalize and the 8 hours is correct.

Choice B: Plausible because the Caswell Beach batteries are rated for 10 hours and the student may have a knowledge deficiency on which (float vs equalize) value is for equalize.

Choice C: Correct answer, see explanation.

Choice D: Plausible because higher is correct and the Caswell Beach batteries are rated for 10 hours.

SRO Basis: N/A

2.0 COMPONENT DESCRIPTION/DESIGN DATA

2.1 Battery Capacity Ratings

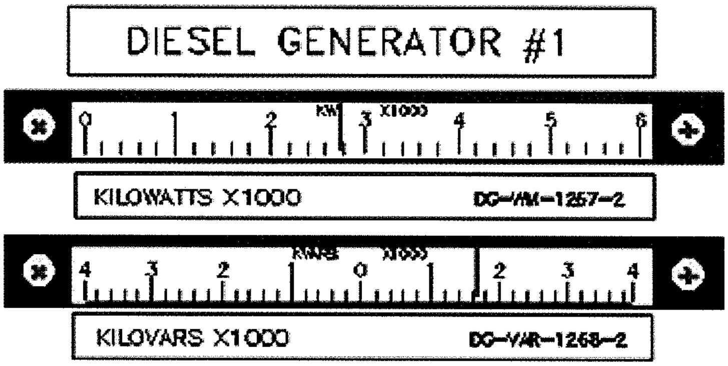
All of the battery systems (with the exception of the Caswell Beach Microwave) have a design Ampere-Hour capacity rating which defines the batteries expected lifetime, in hours, based upon a given continuous loading, in amperes. It should be noted that this is merely a reference number and that battery lifetime is shortened if it is discharged at a higher rate or lengthened if discharged at a lower rate. The individual battery capacities are:

BATTERY SYSTEM	AMP-HOUR RATING
125/250 VDC Station (each division)	1200 AMP-HOURS at a 150 amp rate for 8 hours
24/48 VDC Station (each division)	600 AMP-HOURS at a 75 amp rate for 8 hours
125 VDC Caswell Beach	200 AMP-HOURS at a 20 amp rate for 10 hours

There is no direct indication of the status of the battery charger; i.e., whether it is in the float charge or equalizer charge mode. If in the float charge mode the volt meter should read approximately 135 VDC. If in the equalizer charge mode the meter should read approximately 140 VDC.

30. 264000 1

DG1 has been paralleled to 4160 kv Bus 1D for the performance of 0PT-12.2A, No. 1 Diesel Generator Monthly Load Test. The current DG parameters are:



Which one of the following choices identifies the result of momentarily placing the DG1 Voltage Adjusting Rheostat to Raise? (Assume system load remains constant)

Real load will ____(1)____ and reactive load will ____(2)____.

A. (1) rise
   (2) remain the same

B. (1) rise
   (2) rise

C. (1) remain the same
   (2) rise

D. (1) rise
   (2) lower

Answer: C

K/A: 264000 Emergency Generators (Diesel/Jet)
K5.05 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Paralleling A.C. power sources (CFR: 41.5 / 45.3)

RO/SRO Rating: 3.4/3.4

Objective: AOI-CLS-LP-39, Obj. 03e - Describe the operation of the below listed EDG components: Voltage adjust rheostat.

Reference: None

Cog Level: Low

Explanation: Diesel Generator KW and KVAR is controlled locally using the Governor Motor Control switch and the Voltage Adjusting Rheostat switch, respectively. Once the DG is paralleled with AC sources, the Voltage Adjusting Rheostat switch will affect generator excitation which will in turn control KVAR output. Generator load is adjusted using the Governor Motor Control switch which, prior to synchronization, would be used to adjust generator speed and frequency.

Distractor Analysis:

Choice A: Plausible because raising voltage adjustment would raise KVAR but would have no effect on real load (KW).

Choice B: Plausible because raising voltage adjustment would raise KVAR but would have no effect on real load (KW)

Choice C: Correct answer, see explanation

Choice D: Plausible because raising voltage adjustment would raise KVAR but would have no effect on real load (KW)

SRO Basis: N/A

0PT-12.2A

1. **ADJUST** diesel generator load, at the allowed load rate per Table 4:

   – **RAISE** KW to greater than or equal to 2800 KW and less than or equal to 3000 KW with *GOVERNOR MOTOR CONTROL*

   – **RAISE** kvars to between +1700 and +1900 with *VOLTAGE ADJUSTING RHEOSTAT.*

0OP-50.1 prior to synchronization:

8.7.2 Procedural Steps

5. **ADJUST** generator stator voltage to equal to or slightly greater than emergency bus voltage with *VOLTAGE ADJUSTING RHEOSTAT.* ☐

NOTE:	Normal emergency bus voltage is 4160V.

6. **PLACE** the *SYNCHRONIZING GENERATOR* synchroscope switch in *ON* at the appropriate generator gaugeboard. ☐

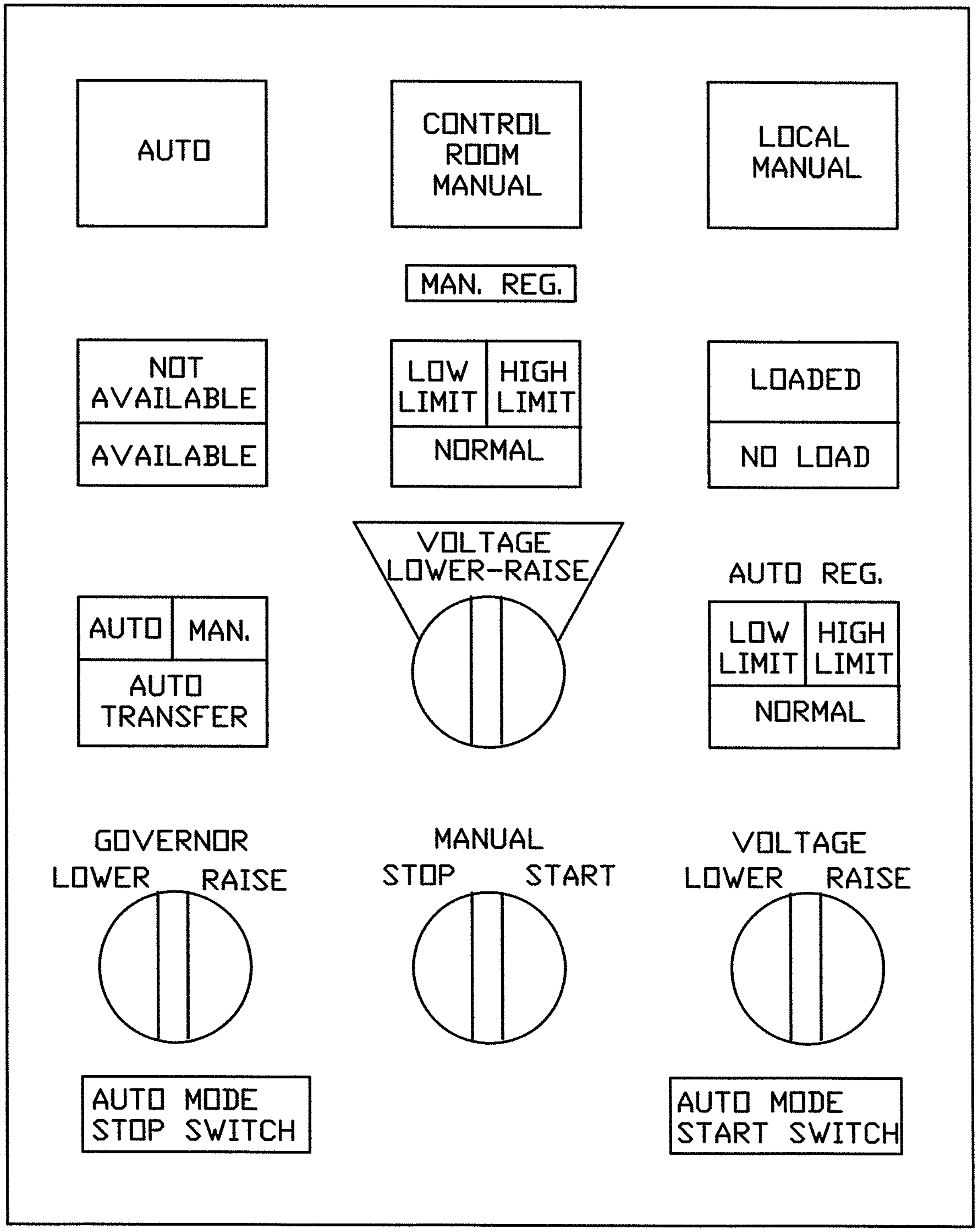
7. **ADJUST** speed of the diesel generator with the *GOVERNOR MOTOR CONTROL* switch until the synchroscope is rotating slowly in the FAST direction (clockwise). ☐

0OP-50.1 following synchronization:

12. **MAINTAIN** generator kvars approximately one-half the KW load, with *VOLTAGE ADJUSTING RHEOSTAT,* while raising diesel generator load. ☐

13. **RAISE** diesel generator load by momentarily placing the *GOVERNOR MOTOR CONTROL* switch in *RAISE* thus decreasing the normal supply amperage as reported by the Auxiliary Operator. ☐

FIGURE 39-9
Diesel Engine RTGB Controls



31. 264000 2

Unit Two is operating at rated power when a loss of offsite power occurs. Subsequently a phase overcurrent occurs on Bus E3. The CRS has directed cross-tying electrical buses IAW 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses.

Which one of the following indicates the status of DG3 and the actions required by 0AOP-36.1?

DG3 is ___(1)___ and actions to cross-tie bus ___(2)___ must be performed.

A. (1) tripped
   (2) E1 to E3

B. (1) tripped
   (2) E8 to E7

C. (1) running unloaded
   (2) E1 to E3

D. (1) running unloaded
   (2) E8 to E7

Answer: B

K/A: 264000 Emergency Generators (Diesel/Jet)
A2.06 - Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Opening normal and/or alternate power to emergency bus (CFR: 41.5 / 45.6)

RO/SRO Rating: 3.4/3.4

Objective: LOI-CLS-LP-050-B, Obj. 13c - Given plant conditions, determine if the following breakrs could be closed: E1 to E3 (or E2 - E4) cross-tie breakers.

Reference: None

Cog Level: High

Explanation: A phase overcurrent condition on the E-bus will result in tripping of the associated DG output breaker and will also result in a trip of the DG. IAW 0AOP-36.1, cross-tying of electrical buses should occur at the 4160V level unless evidence of a fault on the 4160V bus exists as determined by Attachment 5 of the procedure. In the conditions cited, cross-tying at the 480V (E8 - E7) level would be required.

Distractor Analysis:

Choice A: Plausible because a trip of the DG would occur. However, cross-tying of E-buses must occur at the 480V (E8 to E7) level under these conditions.

Choice B: Correct answer, see explanation.

Choice C: Plausible because the DG output breaker would trip on the fault condition. During normal shutdown sequence, the output breaker would be tripped, then the DG manually secured. However, in this condition a trip of the DG would result. E1 to E3 crosstie not allowed due to the fault condition.

Choice D: Plausible because the DG output breaker would trip on the fault condition. During normal shutdown sequence, the output breaker would be tripped, then the DG manually secured. However, in this condition a trip of the DG would result. E8 to E7 would be the desired cross tie performed.

SRO Basis: N/A

SD-39, Section 3.2.7

An automatic trip of the Diesel output breaker will result from any of the following electrical faults:

- phase differential overcurrent
- phase overcurrent
- reverse power
- Loss of field

These trips will also shut down and lockout the engine controls. The breaker will also trip if the Diesel Engine trips. (Figure 39-17)

3.2.11 Emergency Bus Cross-Tie

NOTE:	**IF** cross-tying E-buses **THEN** Technical Specifications 3.8.7 **OR** 3.8.8 should be consulted.

1. **IF** desired to cross-tie a 4160V E bus, **THEN PERFORM** the following:

   a. **CONFIRM** indications of a lockout **OR** phase overcurrent trip do **NOT** exist as identified on Attachment 5. ☐

ATTACHMENT 5
Page 1 of 1
4160V E Bus Lockout Indications

NOTE:	If a lockout indication exists, a cross-tie for the associated bus should **NOT** be performed until the fault is analyzed.

2. **IF** 4160V E buses are **NOT** cross-tied **AND** it is desired to cross-tie 480V E buses, **THEN PERFORM** the following:

   a. **OBTAIN** permission from both Units' CRS to close the 480V E bus cross-tie breakers. ☐

32. 271000 1

Unit One was operating at rated power with AOG-HCV-102, AOG System Bypass Valve, control switch in AUTO.

Subsequently, several annunciators began alarming, including:

UA-03 4-2, *Process Off-Gas Rad Hi-Hi*
UA-03 5-4, *Process OG Vent Pipe Rad Hi-Hi*
UA-48 5-2, *AOG System Disch Rad High*
UA-48 6-2, *AOG Building Radiation High*

Which one of following annunciators is also triggered by the same radiation monitor that will prevent the AOG System Bypass Valve from being manually opened?

A. UA-03 4-2

B. UA-03 5-4

C. UA-48 5-2

D. UA-48 6-2

Answer: A

K/A: 271000 Offgas System
K1.11 - Knowledge of the physical connections and/or cause effect relationships between OFFGAS SYSTEM and the following: †Station radioactive release rate (CFR: 41.2 to 41.9 / 45.7 to 45.8)

RO/SRO Rating: 3.1/3.6

Objective: LOI-CLS-LP-030-A, Obj. 7c - Describe the interrelationships between the following systems and the Condenser Air Removal/Augmented Offgas system: Process Radiation Monitoring.

Reference: None

Modified question from 10-1 NRC exam. (272000_1) The previous question focused on the time delay while this question focuses on the signal.

Cog Level: Low

Explanation: Steam Jet Air Ejector offgas is monitored for radiation levels and the output is provided to the off-gas system logic for alarms and automatic actuation signals. A number of signals, including off-gas radiation levels, will cause off-gas system actuations in the form of bed bypass, bypass closure, and bed isolation. If SJAE offgas radiation reaches a preset value, a 15 minute timer is initiated. If the condition still exists after the timer times out, the AOG Bypass Valve (HCV-102) will receive a closed signal to ensure all offgas flow is being routed through the AOG filters and charcoal beds. This ensures proper filtration of the off-gas and prevents the release of radioactive materials to the environment.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this is a radiation alarm associated with the Off Gas system.

Choice C: Plausible because this is a radiation alarm associated with the Off Gas system.

Choice D: Plausible because this is a radiation alarm associated with the Off Gas system.

SRO Basis: N/A

SD-33, Sect 3.1.4

HCV-102 will automatically close on:

- Any combination of Hi-Hi, downscale, or inop on both of the Off-Gas line radiation monitors (15-minute Time Delay)

SD-11

2.2 Condenser Off-Gas Radiation Monitoring System, also called the Steam Jet Air Ejector Radiation Monitoring System (Figure 11-2)

The log radiation monitors (D12-RM-K601A, B, and K602) are located in the process radiation monitoring instrument rack (H12-P604) in the electronic equipment room. The instrument provides personnel with a front panel display indication of the detected radiation level and supplies this radiation level signal to the system recorder for permanent record. The log radiation monitor also drives four trip circuits which actuate annunciators, initiates the off-gas timer which, after 15 minutes, initiates closure of 1(2)AOG-HCV-102 if the radiation level becomes excessively high.

Unit 1
APP UA-03 4-2
Page 1 of 2

PROCESS OFF-GAS RAD HI-HI

AUTO ACTIONS

1. Process off-gas timer is initiated if both channels are affected
2. **WHEN** process off-gas timer has timed out (15 minutes), if open, the following valves close:
   - *AOG SYSTEM BYPASS VALVE, AOG-HCV-102*
   - *OFF-GAS FILTER HOUSE LOOP SEAL RESERVOIR DRAIN VALVE, 1-OG-SV-4907*

Unit 1
APP UA-03 5-4
Page 1 of 2

PROCESS OG VENT PIPE RAD HI-HI

AUTO ACTIONS

1. Reactor Building Ventilation System trips and isolates
2. Standby gas treatment trains start
3. Group 6 isolation valves close

Unit 1
APP UA-48 5-2
Page 1 of 2

AOG SYSTEM DISCH RAD HIGH

NOTE:	Inoperability of this annunciator will result in an ODCM Required Compensatory Measure.

AUTO ACTIONS

NONE

Unit 1
APP UA-48 6-2
Page 1 of 2

AOG BUILDING RADIATION HIGH

AUTO ACTIONS

NONE

33. 272000 1

Which one of the following identifies the power supply to the Main Stack Radiation Monitor?

A. Powered from Unit One UPS ONLY.

B. Powered from Unit Two UPS ONLY.

C. Normally powered from Unit One UPS, can be transferred to Unit Two UPS.

D. Normally powered from Unit Two UPS, can be transferred to Unit One UPS.

Answer: D

K/A: 272000 Radiation Monitoring System
K2.03 - Knowledge of electrical power supplies to the following: Stack gas radiation monitoring system (CFR: 41.7)

RO/SRO Rating: 2.5/2.8

Objective: CLS-LP-11.0, Obj. 2e - State the purpose of the following major components of the Process Radiation Monitoring System: Main Stack Radiation Monitor

Reference: None

Cog Level: Low

Explanation: From OI-50.5, U2 UPS is the normal power supply while U1 UPS is the alternate power supply

Distractor Analysis:

Choice A: Plausible because U1 is the alternate power supply to the Main Stack Radiation Monitor.

Choice B: Plausible because U2 is the normal power supply to the Main Stack Radiation Monitor.

Choice C: Plausible because U1 is the alternate power supply and U2 is the normal.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

From OI-50.5

ATTACHMENT 3
Page 2 of 8

120V UPS Distribution Panel 2-2A Load Summary

Load: 120V UPS Distribution Panel 2-2A-UPS (HG4)
Location: Control Building 23' SW
Drawing Reference: F-03027
Upstream Power Source: **Primary 50KVA UPS 2A Power Supply**
DC Source: 250VDC Distribution Swbd 2A, Compt GJ5
AC Source: 480V Motor Control Center 2CA, Compt C07
Alt Source: Standby 50KVA UPS 2B Power Supply
Standby 50KVA UPS 2B Power Supply
DC Source: 250VDC Distribution Swbd 2B, Compt GL5
AC Source: 480V Motor Control Center 2CB, Compt C55
Alt Source: 480V Motor Control Center 2CB, Compt C79

CKT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
6	Main Stack Radiation Monitor 2-D12-RM-80S and Sample Detection Skid Normal Power Supply (TS 3.3.6.1, TRM 3.4 ODCM 7.3.2)	**Will receive a full Group 6 isolation on Unit 1 and Unit 2** (all Group 6 valves will close, reactor building will isolate, SBGT trains will start due to stack rad monitor trip signal). 1/2-UA-03-6-3 and 1/2-UA-05-6-10 will alarm. The stack rad monitor trip signals can be overridden for each Unit by taking 1/2-CAC-CS-5519 to OVERRIDE. Transfer switches 2-UPS-TRF-L0S (Control Building U-2 Cable Spread area) and 2-UPS-TRF-L0T (Stack Rad Monitor Building) may be used to restore power to the stack rad monitor from UPS Distribution Panel 1-1A-UPS, circuit #6, if desired.

ATTACHMENT 2
Page 1 of 7

120V UPS Distribution Panel 1-1A Load Summary

Load: 120V UPS Distribution Panel 1-1A-UPS (HG4)
Location: Control Building 23' NW
Drawing Reference: F-90098
Upstream Power Source: **Primary 50KVA UPS 1A Power Supply**
DC Source: 250VDC Distribution Swbd 1A, Compt GJ5
AC Source: 480V Motor Control Center 1CA, Compt C07
Alt Source: Standby 50KVA UPS 1B Power Supply
Standby 50KVA UPS 1B Power Supply
DC Source: 250VDC Distribution Swbd 1B, Compt GL5
AC Source: 480V Motor Control Center 1CB, Compt C55
Alt Source: 480V Motor Control Center 1CB, Compt C69

CKT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
6	Main Stack Radiation Monitor 2-D12-RM-80S and Sample Detection Skid Alternate Power Supply	No effect if transfer switches 2-UPS-TRF-L0S (Control Building U-2 Cable Spread area) and 2-UPS-TRF-L0T (Stack Rad Monitor Building) are selected to Normal. See Unit 2 UPS Distribution Panel 2-2A-UPS, circuit #6 for effects of loss of normal power to the stack radiation monitor

34. 288000 1

Unit Two is operating at rated power when an unisolable steam leak occurs in the turbine building.

Which one of the following choices completes the statement below?

A required action IAW RRCP is to ___(1)___ to ___(2)___.

A. (1) place Turbine Bldg Ventilation in the recirculation line-up
   (2) maintain the Turbine Building at a negative pressure

B. (1) place Turbine Bldg Ventilation in the recirculation line-up
   (2) provide filtration of the release

C. (1) start an additional Turbine Building Ventilation Exhaust Fan
   (2) maintain the Turbine Building at a negative pressure

D. (1) start an additional Turbine Building Ventilation Exhaust Fan
   (2) provide filtration of the release

Answer: B

K/A: 288000 Plant Ventilation Systems
K5.01 - Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS: Airborne contamination control (CFR: 41.7 / 45.4)

RO/SRO Rating: 3.1/3.2

Objective: LOI-CLS-LP-300N, Obj. 19 - Given plant conditions and 0EOP-04-RRCP, determine the following: Required actions to be taken

Reference: None

Modified question from the 10-2 NRC exam (Radiation Control_20) The previous question asked if the AOP and EOP were performed together and if once through needed to be secured. This question asks the required action and why.

Cog Level: Low

Explanation: IAW RRCP if once through is in service TB Vent should be placed in Recirc mode to be able to monitor the release. Starting an additional fan may help with keeping the TB at a negative pressure to limit the release but is not an action in the procedure.

Distractor Analysis:

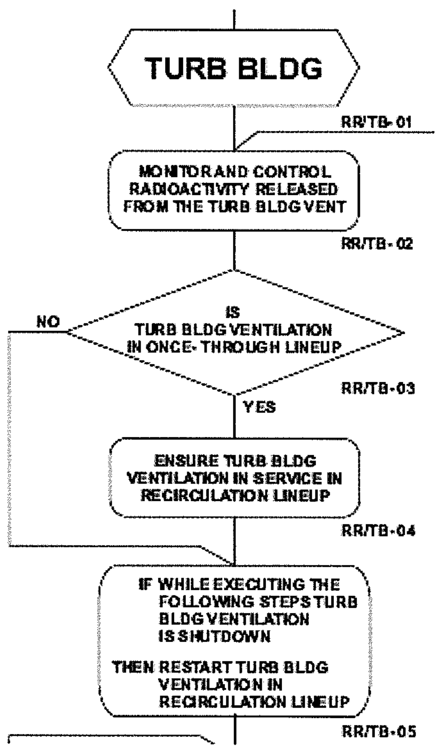
Choice A: Plausible because this is an operating mode of TB Ventilation and although the recirc mode will keep the TB at a negative pressure during normal operation it is not assured under these conditions.

Choice B: Correct Answer, see explanation

Choice C: Plausible because starting an additional exhaust fan would increase the negative pressure in the TB but would also increase the amount of unfiltered exhaust from the building.

Choice D: Plausible because starting an additional exhaust fan would increase the negative pressure in the TB. In once through ventilation, the exhaust is not filtered.

SRO Basis: N/A



Operation of the Turbine Building Ventilation in the recirculation lineup helps to improve Turbine Building accessibility. In addition, since both units share a common Turbine Building airspace, if the building is intact, removing Turbine Building Ventilation from the once through lineup will terminate a large unfiltered volume discharge flow path for a leak on either unit. Due to the normal Turbine Building Air Filtration Unit and WRGM operational requirements when in once through lineup, at least one Turbine Building Air Filtration Unit and WRGM will be in service providing a monitored and filtered discharge flowpath.

35. 290001 1

A microburst thunderstorm with high wind speeds has caused Reactor Building static pressure to lower.
The control room receives annunciator *Rx Bldg Static Press Diff-Low*.

Which one of the following identifies the response and/or actions necessary to maintain normal Reactor Building differential pressure?

Reactor Building ventilation:

A. supply and exhaust fans will trip requiring a manual start of the SBGT system.

B. supply and exhaust fans will trip and the SBGT system will AUTO start.

C. exhaust fan vortex dampers will throttle open.

D. supply fan vortex dampers will throttle closed.

Answer: D

K/A: 290001 Secondary Containment
A3.02 - Ability to monitor automatic operations of the SECONDARY CONTAINMENT including: Normal building differential pressure: Plant-Specific (CFR: 41.7 / 45.7)

RO/SRO Rating: 3.5/3.5

Objective: LOI-CLS-LP-037.1, Obj. 7 - Describe how the Reactor Building Ventilation System maintains a negative differential pressure between the Reactor Building and outside atmosphere.

Reference: None

Question is modified from an 04 NRC exam. (288000_2)

Cog Level: Low

Explanation: With high wind conditions d/p can drop, the supply fans have vortex vanes that will throttle to maintain d/p. The exhaust fans have the vortex vanes disabled. The alarm point for diff-low is 0.1 inches of water. If static pressure increased to 4 inches of water the RB fans would trip. Several of the RB trip signals are also auto start signals to SBGT but building static pressure is not one.

Distractor Analysis:

Choice A: Plausible because if the static pressure increased to four inches water, this would be correct.

Choice B: Plausible because if the static pressure was lower the fans would trip and several of the RB trip signals are also auto start signals to SBGT.

Choice C: Plausible because the exhaust fans do have vortex dampers but they are disabled.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

The fan discharge dampers are operated by the fan control switch and are a permissive for fan start. The fan intake vortex vanes are automatically positioned as required by a differential pressure controller to maintain a preset negative pressure between Reactor Building and outside atmospheric pressure. To start the supply or exhaust fans the Reactor Building supply and exhaust isolation dampers must be full open. The following damper

The fan discharge dampers are operated pneumatically by the fan control switch and are a permissive for fan start. The exhaust fans have intake vortex vanes but they have been disabled at the full open position.

RX BLDG STATIC PRESS DIFF-LOW

AUTO ACTIONS

NONE

CAUSE

1. Low negative pressure differential in the Reactor Building.
2. High wind speeds
3. Circuit malfunction.

DEVICE/SETPOINTS

Pressure Differential Switch YA-PDS-1508	0.1 inches water

36. 295001 1

Unit One is at rated power.
Unit Two is at 48% power in single recirculation loop operation.

Which one the following choices completes the statement below concerning the Minimum Critical Power Ratio (MCPR) safety limit for Unit One and Unit Two?

MCPR shall be greater than or equal to ____(1)____ for Unit One and ____(2)____ for Unit Two.

A. (1) 1.11
   (2) 1.12

B. (1) 1.11
   (2) 1.13

C. (1) 1.12
   (2) 1.12

D. (1) 1.12
   (2) 1.13

Answer: B

K/A: 295001 Partial or Complete Loss of Forced Core Flow Circulation
G2.02.22 - Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5/43.2/45.2)

RO/SRO Rating: 4.0/4.7

Objective: LOI-CLS-LP-200-B, Obj. 3 - State each TS Safety Limit and discuss the basis for each of the Safety Limits

Reference: None

Cog Level: Low

Explanation: The MCPR safety limit for both Units in two loop operation is 1.11. The MCPR safety limits for Single Loop Operations (SLO) for U1 is 1.12 and for U2 is 1.13.

Distractor Analysis:

Choice A: Plausible because 1.12 is the SLO safety limit for Unit 1.

Choice B: Correct Answer, see explanation

Choice C: Plausible because 1.12 is the SLO safety limit for Unit 1 and the TLO limit is the same for both Units.

Choice D: Plausible because 1.12 is the SLO safety limit for Unit 1 and 1.13 is the SLO safety limit for Unit 2

SRO Basis: N/A

Unit 1 TS:

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.11 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

Unit 2 TS:

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.11 for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

37. 295002 1

Unit Two is at rated power with the following plant conditions:

SJAE Train A is in full load operation
CWIPs A, B, and D are in operation
Circ Isol Valve Mode Selector Switch is in D position

The following alarms are received:

CW Screen A Diff-High or Stopped
CW Screen Diff Hi-Hi
CW Pump A Trip
Exhaust Hood A Vacuum Low
Exhaust Hood B Vacuum Low

Which one of the following identifies the action that is required IAW 0AOP-37.0, Low Condenser Vacuum?

A. Start the Mechanical Vacuum Pumps.

B. Place SJAE A and B Trains in half load operation.

C. Starting of CWIP C and it is limited to two consecutive attempts

D. Restarting of CWIP A and it is limited to two consecutive attempts

Answer: C

K/A: 295002 Loss of Main Condenser Vacuum
G2.04.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

RO/SRO Rating: 4.1/4.3

Objective: LOI-CLS-LP-026, Obj. 4j - Given the plant conditions and one of the following events use plant procedures to determine actions required to control and/or mitigate the consequences of the event: Loss of Vacuum.

Reference: None

Cog Level: High

Explanation: A trip of a running CWIP would result in degraded vacuum. AOP-37.0 directs start of an available pump. Standby pumps (at ambient) are allowed two consecutive starts.

Distractor Analysis:

Choice A: Plausible because mechanical vacuum pumps are used to establish condenser vacuum during startup.

Choice B: Plausible because placing additional steam jets in operation could help with restoration of vacuum if the cause were a malfunctioning steam jet. AOP-37

Choice C: Correct answer, see explanation.

Choice D: Plausible because a restart of a tripped CWIP could be performed but a running pump is limited to one restart.

SRO Basis: N/A

0AOP-37.0

CAUTION

Only two consecutive starts with no time interval between starts is permitted for each CWIP with the pump motor at ambient temperature. With the pump motor at rated temperature, only one consecutive start is allowed.

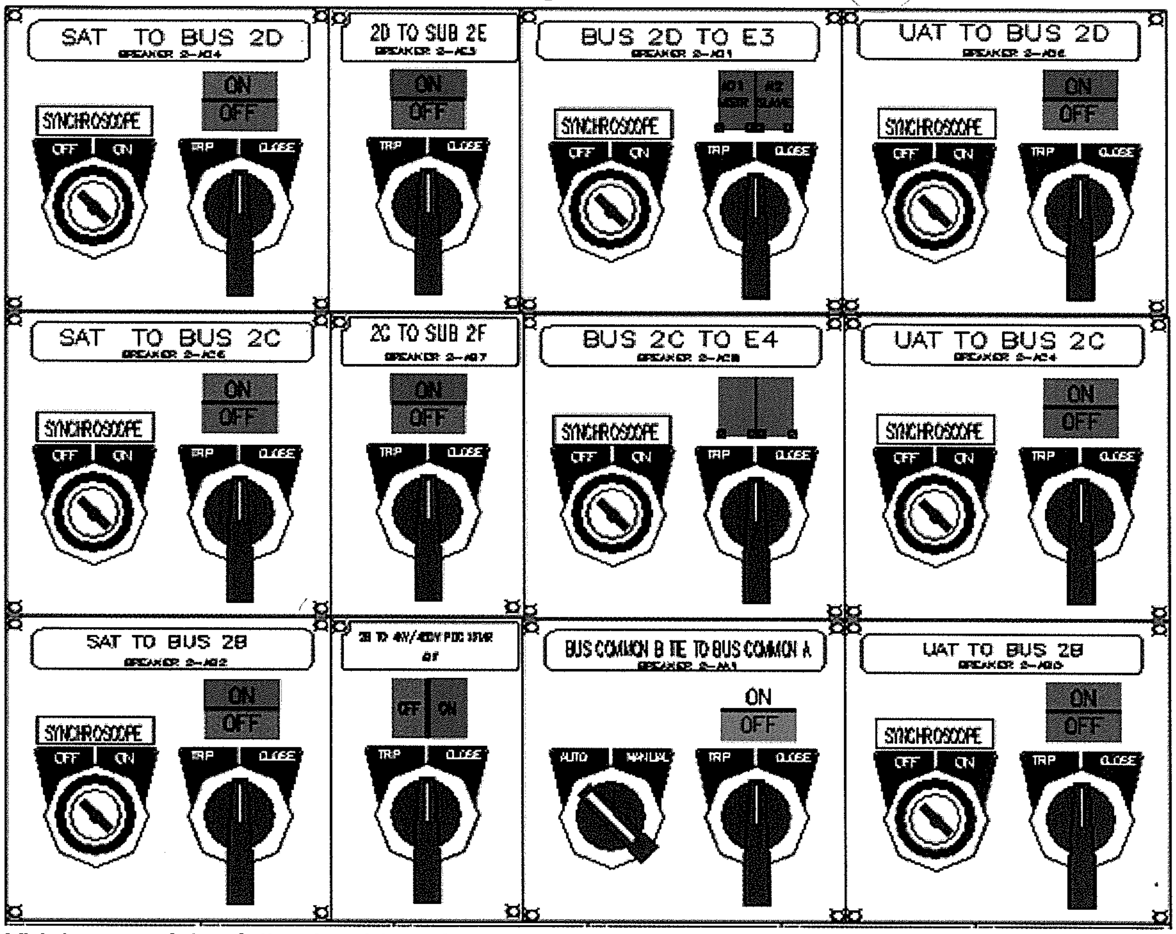
3.2.3 **IF** a CWIP has tripped with reactor power greater than 80%, **AND** a pump can be started within 5 minutes, **THEN START** an available CWIP as needed to maintain condenser vacuum. ☐

3.2.11 **IF** one SJAE train is in service in *FULL LOAD* **AND** the train appears to have malfunctioned, **THEN PLACE** standby SJAE train in service in *FULL LOAD* **AND SHUTDOWN** malfunctioning train in accordance with 1(2)OP-30. ☐

3.2.12 **IF** two SJAE trains are in service in *HALF LOAD* **AND** one train appears to have malfunctioned, **THEN TRANSFER** the other SJAE train to *FULL LOAD* **AND SHUTDOWN** malfunctioning train in accordance with 1(2)OP-30. ☐

38. 295003 1

Unit Two was operating at 50% power when an electrical transient occurred.
The BOP operator observes the following electrical indications after the transient:



Which one of the following identifies the cause of the Unit Two electrical transient?

A. A UAT Lockout

B. A SAT Lockout

C. A Generator Lockout

D. A 2C BOP Bus Lockout

Answer: D

K/A: 295003 Partial or Complete Loss of A.C. Power
AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Cause of partial or complete loss of A.C. power
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.4/3.7

Objective: LOI-CLS-LP-050-B, Obj. 20 - Given plant conditions, determine if lockouts will occur for the following supply breakers:

Reference: None

Cog Level: High

Explanation: While operating, the typical lineup would be for the UAT to supply 2C and 2D busses while the SAT will supply 2B Bus. The 2C and 2D busses normally would transfer to the SAT feed on a scram/loss of the UAT. On a 2C bus lockout, the bus will not auto transfer to the SAT feed.

Distractor Analysis:

Choice A: Plausible because the normal feeder breaker from the UAT to 2C bus is open.

Choice B: Plausible because the SAT feeder breaker to 2C/2D Busses are open.

Choice C: Plausible because the normal feeder breaker from the UAT to 2C bus is open.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

SD-50.1

Figure 10 shows the basic protective relaying scheme for the C/D Buses. Each incoming line from the SAT and UAT is monitored for overcurrent and undervoltage. Actuation of the overcurrent relay will trip the associated bus incoming supply breaker by actuation of a lockout relay. The lockout relay will also prevent closure of the other bus supply breaker in the event of a bus overcurrent. Actuation of the undervoltage relay will trip the associated bus incoming feeder breaker but does not lock out the bus from being supplied from the other feeder.

39. 295004 1

Unit One is operating at rated power when a loss of 125 VDC distribution panel 1B occurs. No operator action has been taken.

Which one of the following choices completes the statement below?

The 1B CRD pump can be shutdown ___(1)___.
If the CRD pump has an overcurrent condition, the 1B CRD pump breaker ___(2)___ trip.

A. (1) from the RTGB or locally at the breaker
   (2) will

B. (1) from the RTGB or locally at the breaker
   (2) will NOT

C. (1) locally at the breaker ONLY
   (2) will

D. (1) locally at the breaker ONLY
   (2) will NOT

Answer: D

K/A: 295004 Partial or Complete Loss of D.C. Power
AA1.03 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: A.C. electrical distribution (CFR: 41.7 / 45.6)

RO/SRO Rating: 3.4/3.6

Objective: LOI-CLS-LP-051, Obj. 7b - Given plant conditions, determine the effect that a loss of DC power will have on the following: AC Electrical Distribution System

Reference: None

Cog Level: High

still

Explanation: DC provides the control power source for 4kV. Electrically operated 480V emergency bus breakers cannot be remotely operated without control power, but will trip on overcurrent. 4kV breakers cannot be operated except manually, and will not trip even on fault conditions without control power.

Distractor Analysis:

Choice A: Plausible because DC control power is required for operation of 4KV breakers from the RTGB and it can be operated locally. This is different from 480V emergency bus breakers which would still trip on overcurrent if DC is lost.

Choice B: Plausible because DC control power is required for operation of 4KV breakers from the RTGB and it can be operated locally.

Choice C: Plausible because with a loss of DC control power, the breaker can only be operated locally. Only the 480V emergency Bus breakers will trip on an overcurrent condition without control power.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

SD-51

2. AC Electrical Distribution

DC provides the control power source for 480 VAC Substations and larger switchgear. Electrically operated 480 VAC breakers cannot be remotely operated without control power, but will trip on overcurrent. 4160 VAC and larger breakers cannot be operated except manually, and will not trip even on fault conditions without control power. Several buses have an alternate control power source, but require manual transfer.

0OI-50

ATTACHMENT 1B
Page 3 of 28

PANEL 1B Reference Drawing: LL-30024-5	LOCATION: Diesel Building, 50', E-2 Cell	NORMAL SUPPLY: Switchboard 1B	ALTERNATE SUPPLY: N/A

CKT #	LOAD	EFFECT
1	Switchgear E2 Breaker Test Cabinet	1. Loss of testing capabilities.
2	Spare	Spare
3	480 VAC Switchgear 'E6' Control Power	1. Loss of normal control power to E6, main and cross tie breakers. Alternate is from Panel 25, Ckt. 15. Control power is manually transferred. 2. Loss of breaker indication and control from both unit's RTGB's. 3. Loss of local ASSD control. 4. Over-current trips will still function. 5. Loss of the main breaker trip function when the associated 4KV breaker opens or for a transformer high temperature trip. 6. Operation of breakers can be performed manually.

ATTACHMENT 1B
Page 4 of 28

PANEL 1B Reference Drawing: LL-30024-5	LOCATION: Diesel Building, 50', E-2 Cell	NORMAL SUPPLY: Switchboard 1B	ALTERNATE SUPPLY: N/A

CKT #	LOAD	EFFECT
17	Switchgear E2 Control Power	1. Loss of normal control power. Alternate is from Panel 25, ckt. 1. 2. Alternate power must be manually aligned. 3. Loss of open, close and trip functions of all affected 4KV breakers. (See 0OI-50.2 for loads) 4. Loss of local and remote breaker position indications. (Mechanical position indicators are available) 5. ASSD requirements for affected loads will not function.

40. 295005 1

Unit Two is operating at rated power.

At 12:15:00 the following annunciators are received:

Stat Coolant Inlet Flow-Low
Loss of Stat Coolant Trip Ckt Ener

Reactor power has been lowered IAW 0ENP-24.5, Reactivity Control Planning.

At 12:16:00 Main Generator amperes are 17,814 amps.

Assuming no further operator action, which one of the following indicates the expected plant response?

The Main Generator will trip at:

A. 12:17:00.

B. 12:18:00.

C. 12:18:30.

20
D. 12:1~~9~~:30.

Answer: C T1, G1

K/A: 295005 Main Turbine generator Trip
AK2.04 - Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Main generator protection (CFR: 41.7 / 45.8)

RO/SRO Rating: 3.3/3.3

Objective: LOI-CLS-LP-027.2 Obj. 05 - Given plant conditions, determine if the "Loss of Stator Cooling" trip circuitry should be energized.

Reference: None

Cog Level: High

Explanation: Generator amps above 6018 amps arms the 3.5 minute timer. With generator amps above 19, 814 amps, both timers (2.0 minute and 3.5 minute) are armed and will initiate at 12:15:00 on the low flow condition and timer energized annunciator. Since both timers start at the same time, the elapsed times are not cumulative (i.e. total time will be 3.5 minutes NOT 5.5 minutes). Lowering generator amps to below 18,823 amps will drop out the two minute timer before it can actuate. The 3.5 minute timer continues to run unless amperes are lowered below 5717 amps. With the conditions described and no further operator action, the 3.5 minute timer will initiate a generator trip at 12:18:30.

Distractor Analysis:

Choice A: Plausible because the 2.0 minute timer would initiate a generator trip at 12:17:00 if generator amps were above the timer dropout setpoint of 18,823 amps.

Choice B: Plausible because the calculated trip time from the 2.0 minute timer would be at 12:18:00 if generator amps were above the timer dropout setpoint of 18,823 amps AND the candidate used 12:16:00 as the calculation start time.

Choice C: Correct answer, see explanation.

Choice D: Plausible because the calculated trip time from the 3.5 minute timer would be at 12:19:30 if generator amps were above the timer dropout setpoint of 5717 amps AND the candidate used 12:16:00 as the calculation start time.

SRO Basis: N/A

SD-27.2

A trip circuit is installed on both units to automatically protect the generator on a loss of stator cooling. Additionally stator cooling water conductivity limits are provided for continued operation of the generator. On a loss of stator water cooling flow, the main turbine must be tripped after 60 minutes if initial conductivity was less than 0.5 µmhos/cm. If initial conductivity was greater than 0.5 µmhos/cm, then the turbine must be tripped within 3 minutes.

The automatic Stator Coolant Trip Circuit initiates whenever generator armature amperage is greater than 6018 amps, and any of the following conditions occur:

- System flow is less than 423 – 444 gpm
- System pressure is less than 37.6 (U-1), 36.3 (U-2) $\pm$ 2 psig
- Coolant temperature exceeds 82-84°C

Once actuated, the trip circuit relays will energize causing actuation of a 2 minute timer if amps are above 19814 and a 3 1/2 minute timer if amps are above 6018. The trip circuit relays, when energized, will also actuate an annunciator, UA-02 1-9, "LOSS OF STATOR COOLANT TRIP CIRCUIT ENER", and open a contact in the EHC load reference motor's increase circuitry to prevent load increase.

The 2 minute timer will trip the turbine when timed out if stator amps are not <18823 amps. The 3.5 minute timer will trip the turbine when timed out if stator amps are not <5717 amps. The trip circuit, once energized, will remain energized until the timer(s) actuate, until the amperage reduction is achieved, or until the degraded condition is corrected.

LOSS OF STAT COOLANT TRIP CKT ENER

AUTO ACTIONS

1. Two timers are actuated which will initiate a turbine trip.

CAUSE

1. Stator coolant outlet temperature is high when armature current is above 6018 amps.
2. Stator coolant flow is low.
3. Stator coolant pressure is low.
4. Circuit malfunction.

ACTIONS (Continued)

CAUTION

If stator current is not less than 18,823 amps after 2.0 minutes or less than 5,717 amps after 3.5 minutes, the turbine will trip.

3. Reduce generator load per 0ENP-24.5 and 0GP-12. If it is expected that the load reduction rate will not satisfy the 2.0 and 3.5 minute timers, then scram the reactor and trip the turbine. Enter 2EOP-RSP, Reactor Scram Procedure.

41. 295006 1

Following a scram, which one of the following conditions meet the definition of Shutdown Under All Conditions Without Boron? In A W

~~All rods inserted except:~~

A. nine rods at position 02.

B. two rods which are at position 04.

C. one rod at position 02 and one rod at position 24.

D. ten rods at position 02 and one rod at position 48.

Answer: A

K/A: 295006 SCRAM
AK1.02 - Knowledge of the operational implications of the following concepts as they apply to SCRAM: Shutdown margin (CFR: 41.8 to 41.10)

RO/SRO Rating: 3.4/3.7

Objective: LOI-CLS-LP-300-C, Obj. 07 - Given plant conditions and the Reactor Scram Procedure, determine if branching into the Level/Power Control Procedure is required.

Reference: None

Cog Level: Low

Explanation: From 0OI-37.3, Reactor Scram Procedure Basis Document: Positive confirmation that the reactor will remain shut down under all conditions is best obtained by determining that no control rod is withdrawn beyond the Maximum Subcritical Bank Withdrawal Position, of position 00. Table 1 has been added to provide a listing of those conditions for the reactor being shutdown under all conditions without boron. This was added specifically for condition where 10 control rods could be withdrawn to position 02 as long as no control rod is withdrawn beyond position 02.

Distractor Analysis:

Choice A: Correct answer, see explanation

Choice B: Plausible because choice provides a possible configuration of control rods following a reactor scram.

Choice C: Plausible because choice provides a combination of allowable circumstances but together they do not satisfy the criteria.

Choice D: Plausible because choice provides a combination of allowable circumstances but together they do not satisfy the criteria.

SRO Basis: N/A

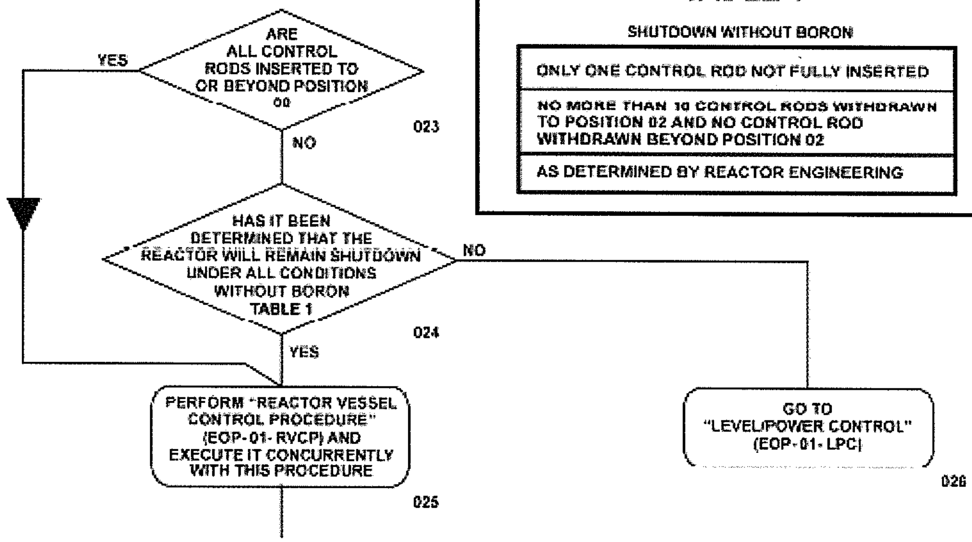


TABLE 1

SHUTDOWN WITHOUT BORON

ONLY ONE CONTROL ROD NOT FULLY INSERTED
NO MORE THAN 10 CONTROL RODS WITHDRAWN TO POSITION 02 AND NO CONTROL ROD WITHDRAWN BEYOND POSITION 02
AS DETERMINED BY REACTOR ENGINEERING

From 0OI-37.3, Reactor Scram Procedure Basis Document:

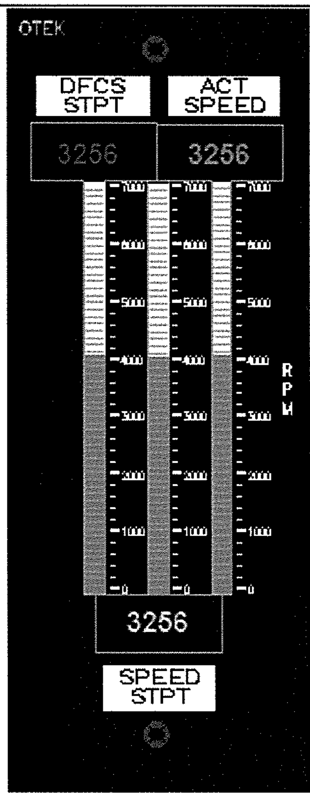
Positive confirmation that the reactor will remain shut down under all conditions is best obtained by determining that no control rod is withdrawn beyond the Maximum Subcritical Bank Withdrawal Position, of position 00. Table 1 has been added to provide a listing of those conditions for the reactor being shutdown under all conditions without boron. This was added specifically for condition where 10 control rods could be withdrawn to position 02 as long as no control rod is withdrawn beyond position 02.

42. 295009 1

Unit Two is at rated power when the following conditions are observed:

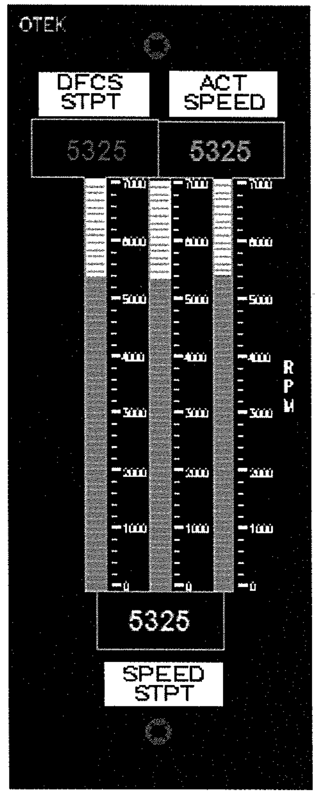
Reactor water level is 180 inches
Steam Flow indicates 12.76 Mlbs/hr
Feed Flow indicates 12.05 Mlbs/hr
RFPT speed indications exist at P603 are:

RFP A Speed
2—RFA—SI—7325



RFP A

RFP B Speed
2—RFB—SI—7326



RFP B

With the conditions continuing to degrade, which one of the following actions is required to restore and maintain level in-band IAW 0AOP-23.0, Condensate/Feedwater System Failures?

A. The master control 2-C32-SIC-R600 has failed in auto, place the Master Feedwater Controller in MANUAL and restore normal level.

B. Place the "A" RFP Feedwater Controller in MANUAL and restore normal level.

C. Place the "B" RFP MANUAL/DFCS switch in MANUAL and restore normal level.

D. Place both RFP Feedwater Controllers in MANUAL and balance the RFP speed.

Answer: B

K/A: 295009 Low Reactor Water Level
AA2.02 - Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Steam flow/feed flow mismatch (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.6/3.7

Objective: LOI-CLS-LP-032.2, Obj. 11 - Describe how feed flow is balanced if a flow mismatch exists and one RFPT will not respond to electrical control

Reference: None

Modified question from 10-2 NRC Exam. (259002_13). this question was modified to provide pictures of the indications and had a failure downward in speed of a RFP instead of upward.

Cog Level: High

Explanation: With the given conditions steam flow is greater than feed flow resulting in a lowering level. The A RFP has failed in automatic operation causing the low level condition and the B RFP is trying to raise flow due to compensate. Manual control of the A RFP is necessary to arrest the trend and restore level. If B RFP had failed high and A RFP was compensating, the level would high, not low.

Distractor Analysis:

Choice A: Plausible because Master level control is in control now, placing it in MANUAL will not correct the failure of A RFP to control.

Choice B: Correct Answer, see explanation

Choice C: Plausible because "B" RFP is operating correctly attempting to raise flow to compensate for the low level.

Choice D: Plausible because placing both RFPs in Manual and balancing rpms will NOT control water level.

SRO Basis: N/A

From AOP-23:

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

3.1.1 **IF** automatic level control will **NOT** restore normal reactor vessel level, **THEN MANUALLY CONTROL** reactor feed pumps to restore normal level. ☐

4.0 GENERAL DISCUSSION

High or low water level during power operation is an abnormal condition that could result in major damage to the HPCI, RCIC, RFP, and main turbines, jet pumps, and recirculation pumps.

High water level may result in excessive moisture carryover, causing erosion of turbine blading. Low water level may cause steam carryunder that can lead to cavitation in recirculation and jet pumps and excessive core internal vibration.

Automatic level control is preferred. Manual control should be taken only if operation in automatic is unsafe or would cause unnecessary transients. If two feed pumps are in operation and the feed pumps have opposite demand signals, take manual control of the pump whose demand signal coincides with the direction of the level change and attempt to control water level.

43. 295010 1

A LOOP has occurred on Unit Two with the following plant conditions:

HPCI	Failed
RCIC	Under Clearance
SLC	Injecting with Demin Water
CRD	Flow maximized IAW SEP-09
ADS	Inhibited
Reactor Water Level	36 inches and stable
Reactor Pressure	950 psig and stable
Suppression Pool Level	-26 inches
Drywell Pressure	2 psig and rising slowly
Drywell Temperature	180°F and rising slowly

Which one of the following choices identifies the required action IAW PCCP?

A. Initiate Drywell Sprays ONLY IAW SEP-02.

B. Start all available DW Coolers IAW SEP-10.

C. Perform Emergency Depressurization IAW RVCP.

D. Initiate Suppression Pool Sprays ONLY IAW SEP-03.

Answer: B

K/A: 295010 High Drywell Pressure
AK1.03 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Temperature increases (CFR: 41.8 to 41.10)

RO/SRO Rating: 3.2/3.4

Objective: LOI-CLS-LP-300L, Obj. 11 - Given PCCP, which steps have been completed and plant parameters, determine the required operator actions. (LOCT)

Reference: None

bank question last used on 07 NRC Exam.

Cog Level: High

Explanation: IAW SEP-10 if a low water condition is present and no actual loca is present then DW coolers can be jumpered and restarted. PCCP also has direction to initiate sprays but under these conditions it cannot be procedurally performed. ED would be required if cannot maintain less than 300 degrees in the DW.

Distractor Analysis:

Choice A: Plausible because DW sprays can be performed in the temperature leg of PCCP, although with pressure at 2 psig it can not be performed..

Choice B: Correct answer, see explanation

Choice C: Plausible because ED can be performed if temperature can not be restored and mantained less than 300 degrees.

Choice D: Plausible because the pressure leg of PCCP directs torus sprays prior to exceeding 11.5 psig although the conditions will not allow the procedure to be performed.

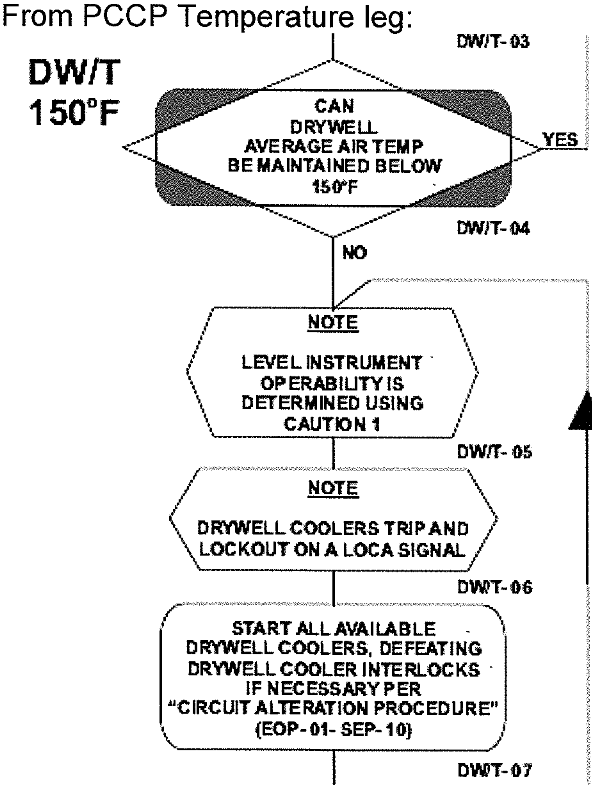
SRO Basis: N/A

From SEP-10

2.4 **IF:**

RO:	-	Directed to defeat the drywell cooler LOCA lockout logic due to low reactor water level, **AND**	☐
RO:	-	Actual LOCA conditions do **NOT** exist in the drywell, **AND**	☐
RO:	-	RBCCW is operating and supplying the drywell, **THEN**	☐
RO:	-	**PERFORM** Section 4 on page 8 of this procedure.	☐

From PCCP Temperature leg:



44. 295016 1

Which one of the following is the reason for inserting a Scram during the performance of 0AOP-32.0, Plant Shutdown From Outside Control Room?

A Scram is inserted prior to ___(1)___ to ensure ___(2)___.

A. (1) evacuating the Control Room ONLY
   (2) the reactor is placed in a hot shutdown condition

B. (1) evacuating the Control Room ONLY
   (2) control of engineered safeguards systems from a backup control center can be executed

C. (1) OR following Control Room evacuation
   (2) the reactor is placed in a hot shutdown condition

D. (1) OR following Control Room evacuation
   (2) control of engineered safeguards systems from a backup control center can be executed

Answer: C

K/A: 295016 Control Room Abandonment
AK3.01 - Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Reactor SCRAM (CFR: 41.5 / 45.6)

RO/SRO Rating: 4.1/4.2

Objective: LOI-CLS-LP-302-E, Obj. 7 - Given plant conditions and entry into 0AOP-32.0, Plant Shutdown From Outside Control Room, explain the basis for a specific caution, note, or series of procedure steps.

Reference: None

Cog Level: Low

Explanation: The reason for a reactor scram during a control room abandonment if not performed prior to evacuating is part of Control Room design which incorporates the capability for prompt hot shutdown of the reactor per Criterion 19 of UFSAR.

Distractor Analysis:

Choice A: Plausible because AOP-32 immediate actions require inserting a manual Scram prior to evacuation, but also provides additional actions for opening RPS EPA breakers to Scram the reactor if not performed from the control room and hot shutdown is correct.

Choice B: Plausible because AOP-32 immediate actions require inserting a manual Scram prior to evacuation, but also provides additional actions for opening RPS EPA breakers to Scram the reactor if not performed from the control room and control of engineered safeguards equipment from the remote shutdown panel will NOT be required is an initial condition assumption and not the reason for inserting a Scram.

Choice C: Correct Answer, see explanation

Choice D: Plausible because a scram is inserted prior to or following evacuation and control of engineered safeguards equipment from the remote shutdown panel will NOT be required is an initial condition assumption and not the reason for inserting a Scram

SRO Basis: N/A

4.0 GENERAL DISCUSSION

The following conditions are assumed to exist as the main Control Room becomes uninhabitable:

1. The event causing the main Control Room to become uninhabitable is assumed to be such that all operators are able to leave the Control Room and assemble at the remote shutdown panel to receive necessary keys, headsets, and procedures, which are maintained in the shutdown panel. (Security will maintain an extra set of keys), and to assist in the remote shutdown.

2. The Shift Manager will order a manual reactor scram, turbine trip, and main steam isolation valves closure prior to Control Room evacuation or go to the battery room and open EPA breakers for RPS MGs and alternate power supply.

3. The emergency AC buses are energized by the preferred off-site power supply. No loss of off-site power is considered after the start of the emergency in this procedure.

4. No accident occurred concurrent with the event which required evacuation of the Control Room, so that control of engineered safeguards systems from a backup control center will not be required.

5. DC services are supplied from at least one plant DC power system for each essential system or equipment required for remote shutdown.

6. All support systems are in normal lineups, operating normally, and continue to operate normally throughout the emergency.

7. Nothing abnormal in the plant between the time of Control Room evacuation and reporting to assigned stations occurs which would deteriorate conditions of the reactor to the point that immediate local action is required on reaching the station.

4.0 GENERAL DISCUSSION

This procedure provides instructions to place the plant in cold shutdown condition from outside the Control Room. The manual reactor scram and main steam isolation will automatically bring the RCIC, HPCI, and reactor relief valves into operation. This will place the reactor in hot shutdown condition. During this phase of shutdown, the suppression pool will be cooled as required by manually placing the RHR system in the suppression pool cooling mode. Reactor pressure will be controlled and core decay and sensible heat rejected to the suppression pool by steam flow to HPCI and RCIC and by dumping steam through the relief valves. Reactor water inventory will be maintained initially by both RCIC and HPCI systems and later by the RCIC system during the pressure reduction period. This procedure will cool the reactor and reduce its pressure at a controlled rate until reactor pressure becomes so low that the relief valves will close or RCIC system will discontinue operation. This condition will be reached at 50-100 psig reactor pressure. One control rod drive pump will be used to maintain rod drives cool and augment the RCIC system, and will be available to maintain level after the RCIC system has been secured. At this time, RHR is manually placed in shutdown cooling to bring the reactor to a cold shutdown condition. The manual control of the equipment required for the above operation (refer to Table 1 for Units 1 and 2) is achieved by operating the control switches located on the individual breaker compartment doors. The key locked NORMAL/LOCAL selector switch is placed in LOCAL position and its START/STOP or OPEN/CLOSE local control switch is operated to control equipment required for remote shutdown.

45. 295017 1

On SPDS Screen 500, Radioactivity Release Control, the Off-Site Whole Body Dose Rate limit will **first** be exceeded (a red Alarm condition) at which one of the following values?

A. 450 mRem / Yr

B. 500 mRem / Yr

C. 2700 mRem / Yr

D. 3000 mRem / Yr

Answer: B

K/A: 295017 High Off-Site Release Rate
AK2.08 - Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: SPDS/ERIS/CRIDS/GDS (CFR: 41.7 / 45.8)

RO/SRO Rating: 2.8/3.3

Objective: LOI-CLS-LP-060, Obj. 06 - Given plant conditions, determine expected ERFIS/SPDS indications.

Reference: None

Cog Level: Low

Explanation: The Off-Site Whole Body Dose Rate goes into prealarm (yellow) at 450 mR/Yr and goes into alarm (red) state at 500 mR/Yr. The other values are the prealarm and alarm conditions for Off-Site Skin Dose Rate.

Distractor Analysis:

Choice A: Plausible because this is the prealarm condition for Off-Site Whole Body Dose Rate

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is the prealarm condition for Off-Site Skin Dose Rate

Choice D: Plausible because this is the alarm condition for Off-Site Skin Dose Rate

SRO Basis: N/A

49. **OFF-SITE SKIN DOSE RATE [500,826]**

This graph incorporates the Radioactive Gaseous Release Calculation which determines off-site dose rates. If any of the values used in the equation are not available, the computer will not substitute values as recommended in 0E&RC-2020, Section 10.3, Noble Gas Instantaneous Release Rate Determination.

The trend plot consists of the latest history, a bar graph reflecting the current value, a current digital readout, and limit tags. The limit tags and available scales are listed below.

Limit Tag	Prealarm	Alarm
TECH SPEC	2700 mrem/yr	3000 mrem/yr

50. **OFF-SITE WHOLE BODY DOSE RATE [500,825]**

This graph incorporates the Radioactive Gaseous Release Calculation which determines off-site levels. If any of the values used in the equation are not available, the computer will not substitute values as recommended in 0E&RC-2020, Section 10.3, Noble Gas Instantaneous Release Rate Determination.

The trend plot consists of the latest history, a bar graph reflecting the current value, a current digital readout, and limit tags. The limit tags and available scales are listed below.

Limit Tag	Prealarm	Alarm
TECH SPEC	450 mrem/yr	500 mrem/yr

46. 295018 1

Unit One is performing a reactor startup.

The following events occur prior to rolling the main turbine:

Bus 1D experiences a fault and trips
Unit One NSW header ruptures in the Service Water Building

Unit One Service Water pumps supplying the NSW Header are manually tripped IAW 0AOP-18.0, Nuclear Service Water System Failure.

Which one of the following identifies the status of the Diesel Generators?

A. ONLY DG1 is running with cooling water supplied from the Unit Two NSW header.

B. ONLY DG1 is running with cooling water supplied from the Unit One CSW header.

C. DGs 1 & 3 are running with cooling water supplied to both DGs 1 & 3 from the Unit Two NSW header.

D. DGs 1 & 3 are running with cooling water supplied to DG1 from the Unit One CSW header and to DG3 from the Unit Two NSW header.

Answer: C

K/A: 295018 Partial or Complete Loss of Component Cooling Water
AA1.01 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Backup systems (CFR: 41.7 / 45.6)

RO/SRO Rating: 3.3/3.4

Objective: AOI-CLS-LP-043, Obj. 6c - Discuss the automatic functions/interlocks associated with the Service Water System: Diesel Generator Cooling Water Supply Valves

Reference: None

Cog Level: High

Explanation: Divisional start signal will auto start both DG 1 and 3. If service water pressure upstream of the jacket water heater exchanger remains below 5.6 psig for 30 seconds then the alternate unit supply valve (in this case Unit 2) will open and the normal supply valve will close. Since DG 3 service water is normally from Unit 2, and Unit 2 service water system is intact, the normal supply from Unit 2 will remain in service. For DG 1, the Unit 1 service water header is depressurized due to the rupture, therefore cooling water for DG 1 will align to the Unit Two Nuclear Service Water header.

Distractor Analysis:

Choice A: Plausible because if 1D 4160 deenergizes while UAT is energized (Unit online or in UAT backfeed), then only DG 1 would start on loss of E-Bus voltage. Unit two nuclear service water will automatically supply the diesel due to loss of Unit one nuclear service water.

Choice B: Plausible because if 1D 4160 deenergizes while UAT is energized (Unit online or in UAT backfeed), then only DG 1 would start on loss of E-Bus voltage. Without a casualty, conventional service water may be available to supply the nuclear header if aligned manually or aligned for auto start on the nuclear header.

Choice C: Correct answer, see explanation

Choice D: Plausible because UAT is deenergized during startup prior to synchronizing the generator to the grid. If 1D 4160 deenergizes while UAT is deenergized then a divisional DG start would result (DG 1 & 3).

SRO Basis: N/A

From SD-39.0, Emergency Diesel Generators

2.7 Diesel Generator Service Water (Figure 39-7)

Two service water supply lines provide service water to the tube side of each EDG set jacket water cooler. Each unit's Nuclear Service Water (NSW) System provides an independent source to all four Diesels. Diesel generator start and speed increase above 500 rpm opens the valve from the respective unit's NSW header. Should the service water pressure upstream of the jacket water heat exchanger remain below 5.6 psig for 30 seconds when the valve is open the alternate unit supply valve will open, then the normal supply valve will close. When the engine is shutdown and speed drops below 500 rpm the open valve will close. This switching sequence is initiated any time service water flow is lost when an EDG set is operating. Return flow of service water from all four jacket water coolers is routed to a common return line which discharges to SW Outfall Collection Tank via an 18" CPVC line.

3.2.4 Automatic Start

3. A EDG auto start signal will be generated for **EDGs 1 and 3 (2 and 4)** if any one of the following conditions exists (Figure 39-13):

- Loss of 1C or 2C 4160 BUS will cause EDGs. 2 & 4 to start
- Loss of 1D or 2D 4160V BUS will cause EDGs 1 & 3 to start

The loss of BOP bus is sensed by undervoltage on the secondary side of the UAT with the UAT to D(C) bus breaker open **OR** undervoltage on the secondary side of the SAT with the SAT to D(C) bus breaker open **AND** BOP bus undervoltage.

From 0AOP-18.0, Nuclear Service Water System Failure

3.0 OPERATOR ACTIONS

3.2.5 **IF** NSW Header pressure is low from a suspected leak or pipe rupture at an unknown location **OR** decreased service water pump availability, **THEN PERFORM** the following:

4. **IF** isolation of the above listed components did **NOT** increase NSW Header to greater than or equal to 40 psig, **THEN PERFORM** the following:

   a. **ASSUME** NSW Header rupture has occurred. ☐

   b. **STOP all** service water pumps supplying the NSW Header. ☐

47. 295019 1

Which one of the following identifies an alarm signal that will initiate the Backup Nitrogen System, including the reason for initiating Backup Nitrogen System?

A. *Instr Air Press-Low*
   Ensures operability ~~of ADS valves~~ and Inboard MSIV's.

B. *RB Instr Air Receiver 1A Press Low*
   Ensures operability ~~of ADS valves~~ and Inboard MSIV's.

C. *Instr Air Press-Low*
   Ensures operability of ADS valves and the Hardened Wetwell Vent Valves.

D. *RB Instr Air Receiver 1A Press Low*
   Ensures operability of ADS valves and the Hardened Wetwell Vent Valves.

Answer: D

K/A: 295019 Partial or Complete Loss of Instrument Air
G2.01.32 - Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

RO/SRO Rating: 3.8/4.0

Objective: LOI-CLS-LP-046-A, Obj. 7d - Given plant conditions, determine if the following automatic actions should occur: Nitrogen Backup Initiation

LOI-CLS-LP-046-A, Obj. 14 - Predict the effect that a loss or malfunction of the Pneumatic System would have on plant operation.

Reference: None

Modified from a Bank question that was used on 08 NRC Exam. (295018_9) modified to provide alarms instead of conditions.

Cog Level: High

Explanation: The Backup Nitrogen System would supply pneumatics to SRV Accumulators, the Reactor Building to Suppression Chamber Vacuum Breaker Isolation Valves, and the Hardened Wetwell Vent Isolation Valves. RB Instr Air Receiver pressure low is received at 95 psig which is the isolation setpoint for backup nitrogen.

Distractor Analysis:

Choice A: Plausible because alarm indicates a low air pressure condition but not the setpoint for BU Nitrogen. The inboard MSIVs are supplied from PNS during full power operations and from non-interruptible instrument air, BU Nitrogen does not supply the inboard MSIV's.

Choice B: Plausible because alarm does initiate BU Nitrogen. The inboard MSIVs are supplied from PNS during full power operations and from non-interruptible instrument air, BU Nitrogen does not supply the inboard MSIV's.

Choice C: Plausible because alarm indicates a low air pressure condition but not the setpoint for BU Nitrogen.

Choice D: Correct answer, see explanation.

SRO Basis: N/A

Unit 1
APP UA-01 1-1
Page 1 of 2

RB INSTR AIR RECEIVER 1A PRESS LOW

AUTO ACTIONS

1. RNA-SV-5482, High Pressure Bottle Rack Isolation Valve, opens, supplying SRVs and CAC-16 with a pneumatic source.

95 psig

Unit 1
APP UA-01 4-4
Page 1 of 2

INSTR AIR PRESS-LOW

AUTO ACTIONS

NONE

48. 295021 1

Unit Two had just been placed in Cold Shutdown when offsite power is lost.
All group isolations occur as expected.

The operators are executing 0AOP-15.0, Loss of Shutdown Cooling, but are having difficulty opening inboard suction isolation valve (E11-F009).
Reactor water level is being maintained between 200-220 inches.

Which one of the following parameters must be monitored for determination of a mode change to Hot Shutdown?

A. Reactor vessel pressure.

B. Reactor bottom head temperature.

C. Reactor recirculation loop temperature.

D. RHR heat exchanger inlet temperature.

Answer: A

K/A: 295021 Loss of Shutdown Cooling
AA2.04 - Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water temperature (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.6/3.6

Objective: LOI-CLS-LP-307-B, Obj. 1g - GP-05, Unit Shutdown: Given plant conditions, monitor cooldown rate per PT-01.7.

LOI-CLS-LP-302-L, Obj. 03 - Given plant conditions and AOP-15.0, Loss of Shutdown Cooling, determine the required supplementary actions.

Reference: None

Cog Level: Low

Explanation: Natural circulation cannot be depended on to provide adequate flow through the bottom head region or the recirculation loops. The recirculation loop suction temperatures and bottom head temperatures therefore cannot be utilized for vessel coolant temperature monitoring for indication of boiling. Under natural circulation conditions, reactor vessel pressure must be monitored for coolant temperature determination. (AOP-15.0)

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because bottom head temperature is used in PT-01.7, but is inaccurate for these conditions.

Choice C: Plausible because recirc loop temperatures are used in PT-01.7, but is inaccurate for these conditions.

Choice D: Plausible because if SDC was in service this would be an option but the group isolation has occurred.

SRO Basis: N/A

0AOP-15

3.0 OPERATOR ACTIONS

CAUTION
Natural circulation can **NOT** be depended on to provide adequate flow through the bottom head region or the recirculation loops. The recirculation loop suction temperatures and bottom head temperatures therefore can **NOT** be utilized for vessel coolant temperature monitoring for indication of boiling. Under natural circulation conditions, reactor vessel pressure must be monitored for coolant temperature determination. If coolant temperature was initially less than 212°F, pressure must be closely monitored for indications of a trend of rising pressure. If this trend is established, it must be assumed that 212°F has been exceeded, boiling is occurring, and a mode change has taken place.

3.2.4 **MONITOR** reactor coolant heatup/cooldown in accordance with 1(2)PT-01.7 for any unexpected trends. ☐

49. 295022 1

Which one of the following choices completes the statement below IAW 0AOP-02.0, Control Rod Malfunction/Misposition, for a loss of CRD pumps?

If reactor pressure is ___(1)___ with charging header pressure less than 940 psig, **immediately** insert a manual reactor scram upon the receipt of the ___(2)___ HCU low pressure alarm.

A. (1) less than 950 psig
(2) first

B. (1) less than 950 psig
(2) second

C. (1) greater than or equal to 950 psig
(2) first

D. (1) greater than or equal to 950 psig
(2) second

Answer: A

K/A: 295022 Loss of Control Rod Drive Pumps
AK1.02 - Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS: Reactivity control (CFR: 41.8 to 41.10)

RO/SRO Rating: 3.6/3.7

Objective: LOI-CLS-LP-008-B, Obj. 10 - Given plant conditions, determine proper operator actions if no CRD pumps are operating.

Reference: None

Modified question from 10-1 NRC exam. (Equipment control_33). Question was modified to ask the conditions instead of given the conditions.

Cog Level: High

Explanation: The CRD system is designed so that primary coolant pressure assists in driving control rods into the core upon receipt of a reactor scram. If low reactor pressure exists , then accumulator pressure alone may not be sufficient to ensure control rods will insert. HCU low pressure alarms in the control room indicate either low accumulator pressure or high water level in the accumulator. AOP-02 conservatively assumes low pressure and directs a scram upon receipt of the alarm if reactor pressure is below 950 psig and CRD pressure cannot be restored to 940 psig or greater with either CRD pump.

Distractor Analysis:

Choice A: Correct answer, see explanation

Choice B: Plausible because if reactor pressure was greater than 950 psig and two accumulator alarm were received, action to restore pressure is required.

Choice C: Plausible because if reactor pressure was greater than 950 psig and two accumulator alarm were received, action to restore pressure is required.

Choice D: Plausible because if reactor pressure was greater than 950 psig and two accumulator alarm were received, action to restore pressure is required.

SRO Basis: N/A

SD-08

4.5 **Abnormal Operation**

4.5.1 **Control Rod Malfunction/Misposition (AOP-02.0)**

A malfunction in the Control Rod Drive System may result in the inability to move individual control rods. The system is designed so that primary coolant system pressure or accumulator pressure is available to scram the reactor. AOP-02.0 should be consulted for specific actions to be performed. The AOP requires a manual scram if no CRD pump is operating, reactor pressure is less than 950 psig, and an HCU accumulator has low pressure. This is to ensure that the control rods can be inserted before accumulators depressurize since a scram on reactor pressure alone cannot be assured if reactor pressure is low.

AOP-02

2. **IF** reactor pressure is less than 950 psig (e.g., during startup **OR** shutdown evolutions), **AND** CRD pressure **CANNOT** be restored to greater than **OR** equal to 940 psig with either CRD pump, **THEN** upon receipt of the first HCU low pressure alarm (A-07, 6-1, confirmed by amber light on Full Core Display), **IMMEDIATELY INSERT** a manual reactor SCRAM. ☐

3.0 OPERATOR ACTIONS

R1 3.2.4 **CONTACT** the Reactor Engineer for further control rod movement instructions. ☐

3.2.5 **MONITOR** off-gas radiation **AND NOTIFY** E&RC to take coolant samples if fuel element failure is suspected. ☐

3.2.6 **IF** the CRD Hydraulic system has malfunctioned, **THEN PERFORM** the following:

1. **IF** the operating CRD Pump has failed, **THEN RESTART** the CRD Hydraulic system following loss of a CRD pump in accordance with1(2)OP-08 Section 8.17. ☐

2. **IF** reactor pressure is less than 950 psig (e.g., during startup **OR** shutdown evolutions), **AND** CRD pressure **CANNOT** be restored to greater than **OR** equal to 940 psig with either CRD pump, **THEN** upon receipt of the first HCU low pressure alarm (A-07, 6-1, confirmed by amber light on Full Core Display), **IMMEDIATELY INSERT** a manual reactor SCRAM. ☐

3. **IF** reactor pressure is greater than **OR** equal to 950 psig, **AND** two **OR** more HCU low pressure alarms (A-07, 6-1, confirmed by amber light on Full Core Display), **THEN ENSURE** CRD pressure is restored to greater than **OR** equal to 940 psig within 20 minutes. ☐

4. **REFER** to Tech Spec 3.1.5 for any control rod scram accumulator required actions. ☐

5. **CHECK** CRD pump suction **AND** drive water filters for high differential pressure. ☐

6. **MONITOR** the following CRD system parameters for possible system leakage **OR** flow control valve failures:

   - CRD Drive Water Pressure, *C11(C12)-PDI-R602.* ☐
   - CRD Cooling Water Pressure, *C11(C12)-PDI-R603.* ☐
   - CRD Drive Temperature, *C11(C12)-TR-R018.* ☐
   - CRD Charging Water Header Pressure, *C11(C12)-PI-R601.* ☐

50. 295023 1

Which one of the following choices completes the statement below IAW the Technical Specifications and Bases for LCO 3.9.1, Refueling Equipment Interlocks?

Refueling Equipment Interlocks ____(1)____ with the mode switch in the ____(2)____ position.

A. (1) prevents an iodine gas release
   (2) REFUEL

B. (1) prevents an iodine gas release
   (2) SHUTDOWN

C. (1) prevent inadvertent criticality
   (2) REFUEL

D. (1) prevent inadvertent criticality
   (2) SHUTDOWN

Answer: C

K/A: 295023 Refueling Accidents
AK3.02 - Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: Interlocks associated with fuel handling equipment (CFR: 41.5 / 45.6)

RO/SRO Rating: 3.4/3.8

Objective: LOI-CLS-LP-058.1, Obj. 21 - State the function of the refueling interlocks

Reference: None

Cog Level: Low

Explanation: In accordance with the TS / Bases/ UFSAR, the purpose of the interlocks is to prevent an inadvertent criticality and the LCO and Applicability is when the mode switch is in Refuel. Iodine gas would be released if the fuel was damaged. The interlock will prevent entry into an AOP or EOP.

Distractor Analysis:

Choice A: Plausible because the mode switch is correct but the reason is incorrect in that the interlocks do not prevent damage to the fuel which would cause a release of the iodine gas.

Choice B: Plausible because the interlocks do not prevent damage to the fuel which would cause a release of the iodine gas. For refueling the mode switch can be in Shutdown or refuel position in accordance with TS Table 1.1-1

Choice C: Correct Answer, see explanation

Choice D: Plausible because for refueling the mode switch can be in Shutdown or refuel position in accordance with TS Table 1.1-1

SRO Basis: N/A

3.9 REFUELING OPERATIONS

3.9.1 Refueling Equipment Interlocks

LCO 3.9.1 The refueling equipment interlocks associated with the refuel position of the reactor mode switch shall be OPERABLE.

APPLICABILITY: During in-vessel fuel movement with equipment associated with the interlocks when the reactor mode switch is in the refuel position.

Bases:

B 3.9 REFUELING OPERATIONS

B 3.9.1 Refueling Equipment Interlocks

BASES

BACKGROUND Refueling equipment interlocks restrict the operation of the refueling equipment or the withdrawal of control rods to reinforce unit procedures that prevent the reactor from achieving criticality during refueling. The refueling interlock circuitry senses the conditions of the refueling equipment and the control rods. Depending on the sensed conditions, interlocks are actuated to prevent the operation of the refueling equipment or the withdrawal of control rods.

1.2.2.6.4 Refueling Interlocks

A system of interlocks is provided to prevent an inadvertent criticality during refueling operations by restricting the movements of refueling equipment and control rods when the reactor is in the refuel mode. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling bridge, the refueling bridge hoists, the fuel grapple, the control rods. (Refueling interlocks do not affect the Asea-Brown Boveri hoist.)

51. 295024 1

During an accident, Unit One plant conditions are:

Reactor pressure	500 psig
Drywell pressure	20 psig
Suppression chamber pressure	19 psig
Suppression pool level	-42 inches
Suppression pool temperature	160°F

(Reference provided)

Which one of the following is the reason emergency depressurization is required?

A. Steam exists in the suppression chamber air space.

B. Prevent exceeding suppression chamber design temperature.

C. Prevent exceeding suppression chamber boundary design load.

D. Suppression chamber level is at the elevation of the downcomers.

Answer: A

K/A: 295024 High Drywell Pressure
EK3.04 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: †Emergency depressurization (CFR: 41.5 / 45.6)

RO/SRO Rating: 3.7/4.1

Objective: CLS-LP-300-L, Obj. 3E, Define the following terms: Pressure Suppression Pressure Limit

CLS-LP-300-L, Obj. 4E, State the effect on Primary Containment if the following limits are exceeded: Pressure Suppression Pressure Limit

Reference: Pressure Suppression Pressure graph (0EOP-01-UG Attachment 5 Figure 7)

Bank question last used on 08 NRC exam.

Cog Level: High

Explanation: The PSP curve contains 4 segments. The bottom horizontal line is the elevation of the downcomers. The top horizontal line is the elevation of the bottom of the ring header. The diagonal line sloping up to the right starting at low suppression pool level is the highest pressure that could be in the suppression chamber air space without steam in the air space. At the current suppression pool level this segment of the curve applies and is being exceeded. The last segment toward the high suppression pool level limit is for design loading of the suppression chamber. Suppression chamber design temperature is the reason for emergency depressurization if Heat Capacity Temperature Limit is Unsafe, which it is not.

Distractor Analysis:

Choice A: Correct Answer, see explanation

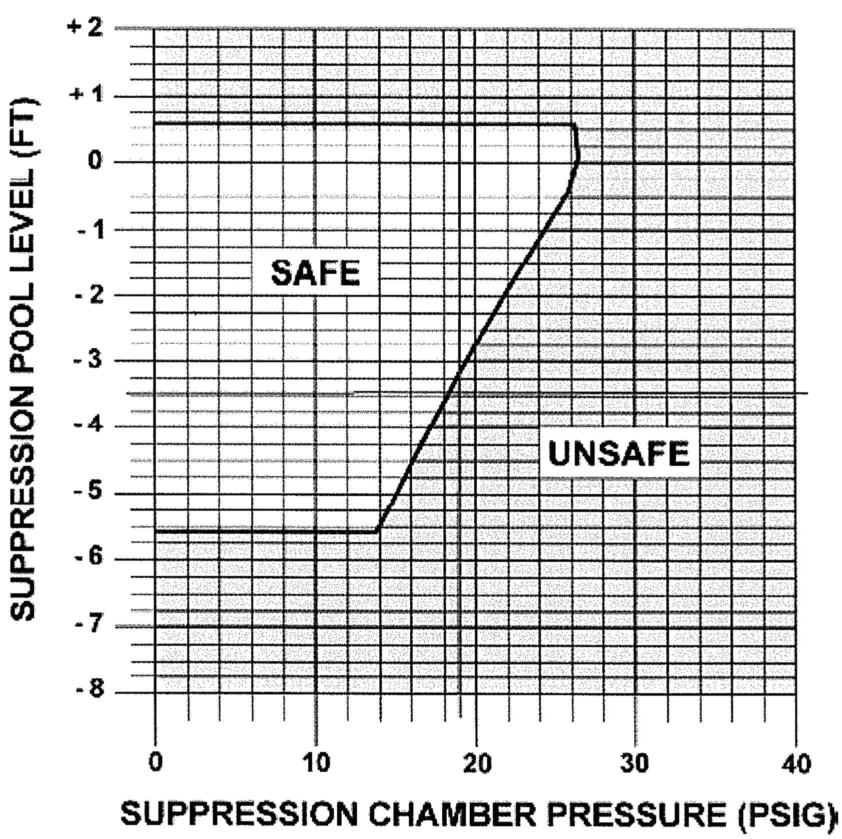
Choice B: Plausible since this would be correct for exceeding HCTL and suppression pool temperature is very elevated (would be correct at a higher reactor pressure)

Choice C: Plausible since this is part bases for the Pressure suppression Pressure limit (based on a higher suppression pool level and pressure)

Choice D: Plausible since this is part bases for the Pressure suppression Pressure limit (based on a lower suppression pool level)

SRO Basis: N/A

ATTACHMENT 5
Page 22 of 27
FIGURE 7
Pressure Suppression Pressure



52. 295025 1

Given the following plant conditions with RCIC in pressure control mode:

RCIC controller output	70%
Bypass to CST Vlv, E51-F022	Throttled
RCIC Flow	300 gpm
RPV pressure	990 psig, slowly rising
RCIC controller	Automatic set @ 300 gpm

Which one of the following identifies two independent actions that will stabilize RPV pressure?

Throttle the E51-F022 in the ___(1)___ direction, or by ___(2)___ the RCIC Flow Controller auto setpoint.

A. (1) open
(2) lowering

B. (1) open
(2) raising

C. (1) closed
(2) lowering

D. (1) closed
(2) raising

Answer: D

K/A: 295025 High Reactor Pressure
EA1.05 - Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: RCIC: Plant-Specific (CFR: 41.7 / 45.6)

RO/SRO Rating: 3.7/3.7

Objective: CLS-LP-016-A Obj. 17b - Describe how the following evolutions are performed during operation of the RCIC system: Adjusting RCIC flow in the reactor pressure control mode.

Reference: None

Cog Level: High

Explanation: There are two ways to reduce the RPV pressure with the conditions given. One way is to close the E51-F022 valve, thereby decreasing the size of the hole and forcing the turbine to work harder to deliver the same flowrate. The second is to raise the controller setpoint thereby causing the turbine to work harder by forcing more flow through the same size hole.

Distractor Analysis:

Choice A: Plausible because these are the opposite of the actual answers and if the operator was trying to raise RPV pressure this would be correct.

Choice B: Plausible because raising is correct and the operator could have a misconception about the F022 valve.

Choice C: Plausible because closing the F022 is correct and the operator could have a misconception about the flow controller.

Choice D: Correct Answer, see explanation.

SRO Basis: N/A

From RCIC Hard Card:

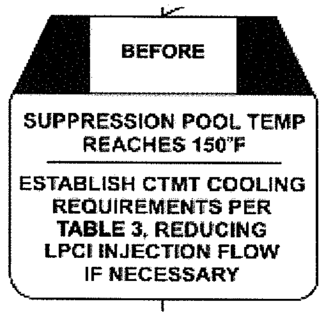
RCIC PRESSURE CONTROL
(1OP-16 SECTION 8.2)

1.	**ENSURE** THE FOLLOWING VALVES ARE OPEN: E51-V8 (VALVE POSITION), E51-V8 (ACTUATOR POSITION), AND E51-V9.	☐
2.	**OPEN** E51-F046	☐
3.	**START** VACUUM PUMP AND LEAVE SWITCH IN START.	☐
4.	**ENSURE** E51-F013 IS CLOSED	☐
5.	**ENSURE** E41-F011 IS OPEN	☐
6.	**THROTTLE OPEN** E51-F022 UNTIL DUAL INDICATION IS OBTAINED	☐
7.	**OPEN** E51-F045	☐
8.	**THROTTLE OPEN** E51-F022 OR **ADJUST** RCIC FLOW CONTROL, E51-FIC-R600, TO OBTAIN DESIRED SYSTEM PARAMETERS AND REACTOR PRESSURE.	☐
9.	**ENSURE** E51-F019 IS CLOSED WITH FLOW GREATER THAN 80 GPM.	☐
10.	**ENSURE** THE FOLLOWING VALVES ARE CLOSED: E51-F025, E51-F026, E51-F004, AND E51-F005.	☐
11.	**START** SBGT (1OP-10)	☐
12.	**ENSURE** BAROMETRIC CNDSR CONDENSATE PUMP OPERATES	☐

FOR SHUTDOWN: REFER TO 1OP-16
FOR TRANSFER BETWEEN PRESSURE AND LEVEL CONTROL: REFER TO 1OP-16

2 1/1086

53. 295026 1



IAW OI-37.4, Reactor Vessel Control Procedure Basis Document, which one of the following identifies why the step above is performed?

A. To prevent exceeding the Heat Capacity Temperature Limit.

B. To prevent exceeding primary containment design temperature.

C. To maintain long term operation of the Core Spray and RHR Pumps.

D. To minimize off-site releases per Alternative Source Term calculations.

Answer: C

K/A: 295026 Suppression Pool High Water Temperature
EK2.01 - Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Suppression pool cooling (CFR: 41.7 / 45.8)

RO/SRO Rating: 3.9/4.0

Objective:

Reference: None

Cog Level: Low

Explanation: The calculation for the NPSH to the Core Spray and RHR pumps specify establishing cooling at ten minutes into the design basis LOCA. The calculation also assumes that the temperature of the suppression pool will be at approximately 169°F at ten minutes. If containment cooling is not established, then it is possible that the Core Spray or RHR pumps will be lost due to inadequate NPSH. To prevent having a ten minute action statement the temperature of 150°F was chosen.

Distractor Analysis:

Choice A: Plausible because high torus temperature does affect HCTL but at this point in RVCP injection to the vessel is irrespective of NPSH or Vortex Limits

Choice B: Plausible because long term containment cooling will limit primary containment temperature rise.

Choice C: Correct Answer, see explanation definition.

Choice D: Plausible because table 5 of RVCP has actions concerning Alternative Source Term.

SRO Basis: N/A

STEPS RC/L-44

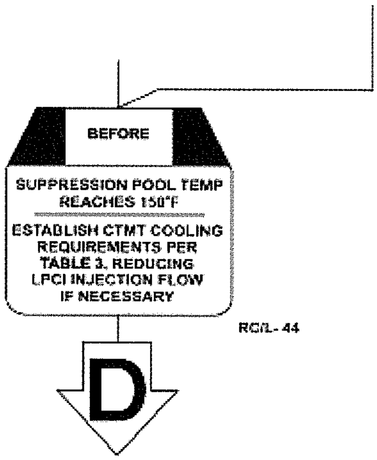


TABLE 3

MINIMUM CONTAINMENT COOLING REQUIREMENTS

MINIMUM PUMP REQUIREMENTS	FLOW (GPM)
1 - RHR PUMP PER LOOP WITH HX BYPASS VALVE CLOSED	7700/LOOP
1 - RHR SW PUMP PER LOOP	4000/LOOP
2 - RHR PUMPS WITH HX BYPASS VALVE CLOSED	11,500
2 - RHR SW PUMPS	8000

STEP BASES:

This step provides guidance on establishing cooling to the suppression pool during a design basis LOCA. The calculation for the NPSH to the Core Spray and RHR pumps specify establishing cooling at ten minutes into the design basis LOCA. The calculation also assumes that the temperature of the suppression pool will be at approximately 169°F at ten minutes. If containment cooling is not established at a suppression pool temperature of 169°F, then it is possible that the Core Spray or RHR pumps will be lost due to inadequate NPSH. The EPGs specify injecting irrespective of NPSH and vortex limits if reactor vessel water level is below Minimum Steam Cooling Reactor Water Level. This step provides guidance to reduce injection into the reactor vessel to establish the desired cooling for the containment. A value of 150°F has been selected for use in the step. This provides a margin of at least 19°F, to the limit used in the calculation for NPSH. These actions are incorporated to provide assurance that the unit can remain in the EOPs and not be required to go to primary containment flooding prematurely. The selection of a suppression pool temperature limit precludes establishing a specific time limit in the procedure.

54. 295028 1

Conditions on Unit Two have degraded to where the Drywell Air Temperature is 340°F.

Which one of the following identifies the components whose environmental qualification is affected by this temperature IAW 0OI-37.8, Primary Containment Control Procedure Basis Document?

A. SRV solenoids

B. Inboard MSIV solenoids

C. Torus to Drywell Vacuum Breakers

D. CAC 4409 and 4410 Hydrogen Analyzers

Answer: A

K/A: 295028 High Drywell Temperature
EK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification (CFR: 41.8 to 41.10)

RO/SRO Rating: 2.9/3.1

Objective: CLS-LP-300L Obj. 4h, State the effect on Primary Containment if the following limits are exceeded: Drywell Design Temperature Limit.

Reference: None

Cog Level: Low

Explanation: From the Bases document: Temperature should not be allowed to exceed the SRV maximum qualification temperature of 340°F.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this is a piece of equipment in the Drywell.

Choice C: Plausible because this is a piece of equipment in the Drywell that has a water level limit of +6 inches in the suppression pool.

Choice D: Plausible because the H2/O2 analyzers were designed for pressures up to 30 psig. To preclude damage to these sample pumps and the subsequent radioactive release to secondary containment, the sample pumps are isolated when drywell pressure exceeds 30 psig.

SRO Basis: N/A

Consistent with the definition of "restore," emergency depressurization is not required until it has been determined that drywell sprays (initiated in Step DW/T-16) are ineffective in reducing drywell temperature. This determination may be made when, before, or after the temperature actually reaches 300°F. It is not expected that either the containment integrity (300°F) or SRV operability (340°F) will be immediately challenged when the respective temperature limits are reached. If drywell temperature is already above 300°F when Step DW/T-16 and DW/T-19 are reached, drywell sprays may still be used, if available, in preference to emergency depressurization. If sprays are effective in reducing the drywell temperature, emergency depressurization need not be performed. Extended operation above 300°F is not permitted and the temperature should not be allowed to exceed the SRV maximum qualification temperature of 340°F.

55. 295029 1

A LOCA has occurred concurrently with a LOOP on Unit Two.
The following conditions exist:

Drywell Pressure	4.5 psig
Reactor Water Level	95 inches and rising
Reactor Pressure	280 psig
Suppression Chamber level	5.5 inches
CST level	20 feet

Which one of the following identifies an injection source that must be secured IAW PCCP? (Assume no circuit alterations have been performed)

A. CRD

B. RCIC

C. HPCI

D. Core Spray

Answer: B

K/A: 295029 High Suppression Pool Water Level
EA1.04 - Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: RCIC: Plant-Specific (CFR: 41.7 / 45.6)

RO/SRO Rating: 3.4/3.5

Objective: LOI-CLS-LP-016-A Obj. 15g - Given plant conditions, predict the RCIC system response to the following conditions: High/low Suppression Pool water level.

Reference: None

Cog Level: High

Explanation: IAW PCCP, if torus level cannot be maintained below 6 inches and adequate core cooling is assured, injection from sources external to containment are terminated (SP/L-10). Reactor water level of 185 inches rising assures adequate core cooling. HPCI suction path will automatically align to the torus IF a low CST level exists OR a high torus level exists. In the conditions described, HPCI suction will have aligned to the torus so its injection is from an internal source. RCIC suction path will automatically align to the torus IF a low CST level exists but will NOT align to the torus on a high torus level so its injection supply is from external sources (CST) and must be secured IAW SP/L-10. Drywell spray and torus spray are running and would not be secured by procedure until drywell pressure lowers to 2.5 psig. Drywell spray would be secured if suppression pool level cannot be maintained below 21 inches.

Distractor Analysis:

Choice A: Plausible because SP/L-15 of PCCP requires securing injection sources external to containment. Systems being used for boron injection and CRD exception to this.

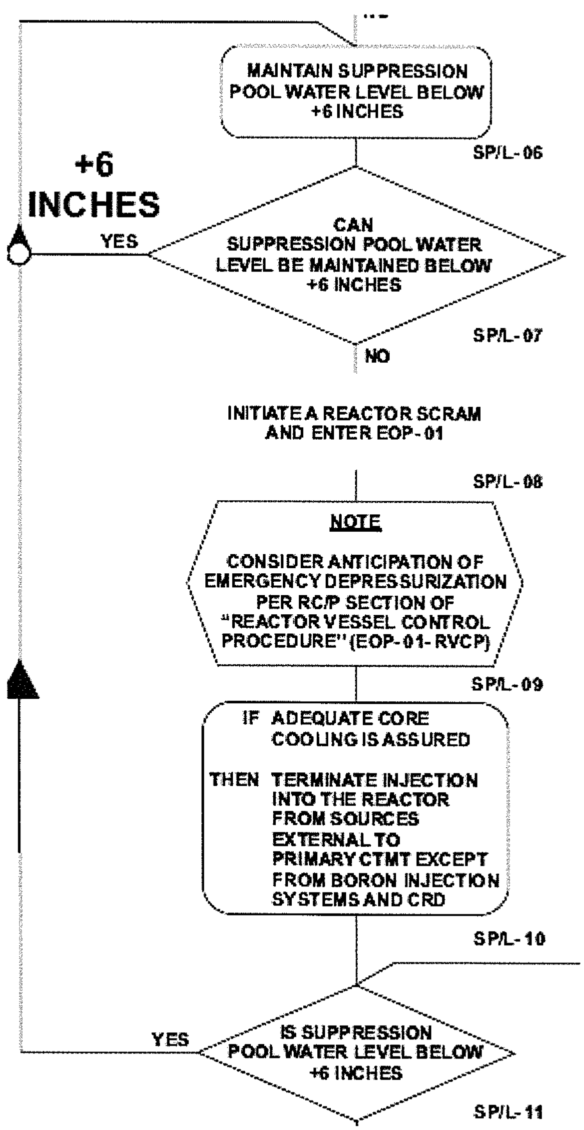
Choice B: Correct Answer, see explanation

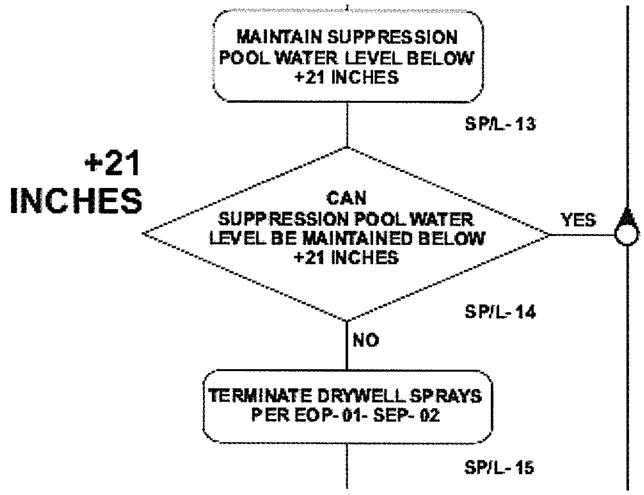
Choice C: Plausible because HPCI injection is one of the sources of level control. SP/L-28 requires securing of HPCI if torus level cannot be maintained above minus 6.5 feet.

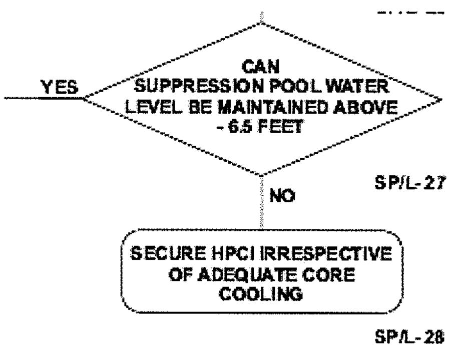
Choice D: Plausible because these systems are providing input to containment and the vessel.

SRO Basis: N/A

0EOP-02-PCCP







0EOP-01-SEP-02

2.7 **WHEN** drywell pressure drops below 2.5 psig **OR IF** directed to terminate drywell spray, **THEN PERFORM** the following:

RO: 2.7.1 **CLOSE** Loop A(B) *DRYWELL SPRAY OTBD ISOL VLV, E11-F016A(F016B).* ☐

RO: 2.7.2 **CLOSE** Loop A(B) *DRYWELL SPRAY INBD ISOL VLV, E11-F021A(F021B).* ☐

SD-16

1.3.1 Water Flow Path (Figure 16-2)

The primary water supply to the RCIC System is the condensate storage tank (CST) through the normally open Condensate Storage Tank Suction Valve, E51-F010. In the event the CST level decreases to a predetermined level, the RCIC suction will automatically transfer to the suppression pool through normally closed Suppression Pool Suction Valves, E51-F029 and E51-F031.

SD-19

HPCI Pump suction is established from the Suppression Pool by opening the normally closed Suppression Pool Suction Valves, E41-F042 and E41-F041. These valves automatically open, if no HPCI System isolation signal is present, on a low CST level or high Suppression Pool level. Upon receiving an isolation signal (PCIS Group 4), these valves automatically close.

Table 19-6 - HPCI Suppression Pool Suction Transfer Signals

Signal	Setpoint	Tech Spec
CST Level Low	23'5" elev. (3'5" tank level)	≥23'4" elev. (≥3'4" tank level)
Suppression Pool Level	-25"	≤-2'

2APP-A-01

Unit 2
APP A-01 1-5
Page 1 of 2

SUPPRESSION CHAMBER LVL HI-HI

AUTO ACTIONS

1. If closed, Torus Suction Vlv, E41-F042, opens
2. If closed, Torus Suction Vlv, E41-F041, opens
3. If open, CST Suction Vlv, E41-F004, closes

CAUSE

1. Suppression pool water level high (-25 inches)
2. Circuit malfunction

Unit 2
APP A-01 6-4
Page 1 of 2

HPCI COND STORAGE TNK WTR LVL LO

AUTO ACTIONS

1. If closed, Torus Suction Vlv, E41-F041, opens
2. If closed, Torus Suction Vlv, E41-F042, opens
3. If open, CST Suction Vlv, E41-F004, closes

CAUSES

1. Low level in CST due to usage or leaks (23 feet, 5 inches)
2. Loss of power to 125V DC Distribution Panel 4A
3. Circuit malfunction

WHITE

RCIC SUCT XFR CST LO LVL

1.0 OPERATOR ACTIONS:

1.1 **CONFIRM** CST level approximately 3 feet or less.

1.2 **OBSERVE** Automatic Functions:

- **IF** closed, **THEN** *TORUS SUCTION VLV, E51-F029*, opens (if no RCIC isolation signal is present)
- **IF** closed, **THEN** *TORUS SUCTION VLV, E51-F031*, opens (if no RCIC isolation signal is present)
- **IF** open, **THEN** *CST SUCTION VLV, E51-F010*, closes

56. 295030 1

Following a DBA LOCA on Unit Two, plant conditions are as follows:

Reactor water level	55 inches and rising
Reactor pressure	150 psig
Torus temperature	220°F
Suppression Chamber pressure	10.5 psig
Torus level	-43 inches
2A Core Spray pump flow	5000 gpm
2B Core Spray pump flow	2000 gpm
2A RHR pump flow	8000 gpm
2B RHR pump flow	6000 gpm

(reference provided)

Which one of the following identifies the ECCS pump(s) that is/are operating within the associated NPSH limit(s)?

A. 2B CS Pump ONLY

B. 2A CS and 2A RHR pumps ONLY

C. 2B CS and 2B RHR pumps ONLY

D. 2A CS, 2A RHR and 2B RHR ONLY

Answer: A

K/A: 295030 Low Suppression Pool Water Level
EA2.02 - Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.9/3.9

Objective: CLS-LP-300B, Obj. 17. Given plant condition and the NPSH and vortex limit graphs for the RHR and CS, determine if the NPSH and/or vortex limits have been exceeded for either of the two systems.

Reference: 0EOP-01-UG, Attachment 5, Figures 5 & 6, Core Spray & RHR NPSH Limit.

Bank question that was last used on the 08 NRC exam.

Cog Level: High

EXPLANATION: The student will need to plot each point on NPSH limit graph.
Torus pressure must be corrected down 0.5 psig to obtain the proper restriction line.
The correct torus pressure is 10.5 psig - 0.5 psig = 10 psig.
This correction must be performed for both the RHR and CS graphs.

Distractor Analysis:

Choice A: Correct answer, see explanation.

Choice B: Plausible because if the student fails to adjust torus pressure on RHR and CS graphs and thinks that above the line is the safe region then this answer would be correct.

Choice C: Plausible because if the student fails to adjust torus pressure on RHR and CS graphs, this answer would be correct.

Choice D: Plausible because if the student believes that above the line is the safe region then this answer would be correct.

SRO Basis: N/A

0EOP-01-UG

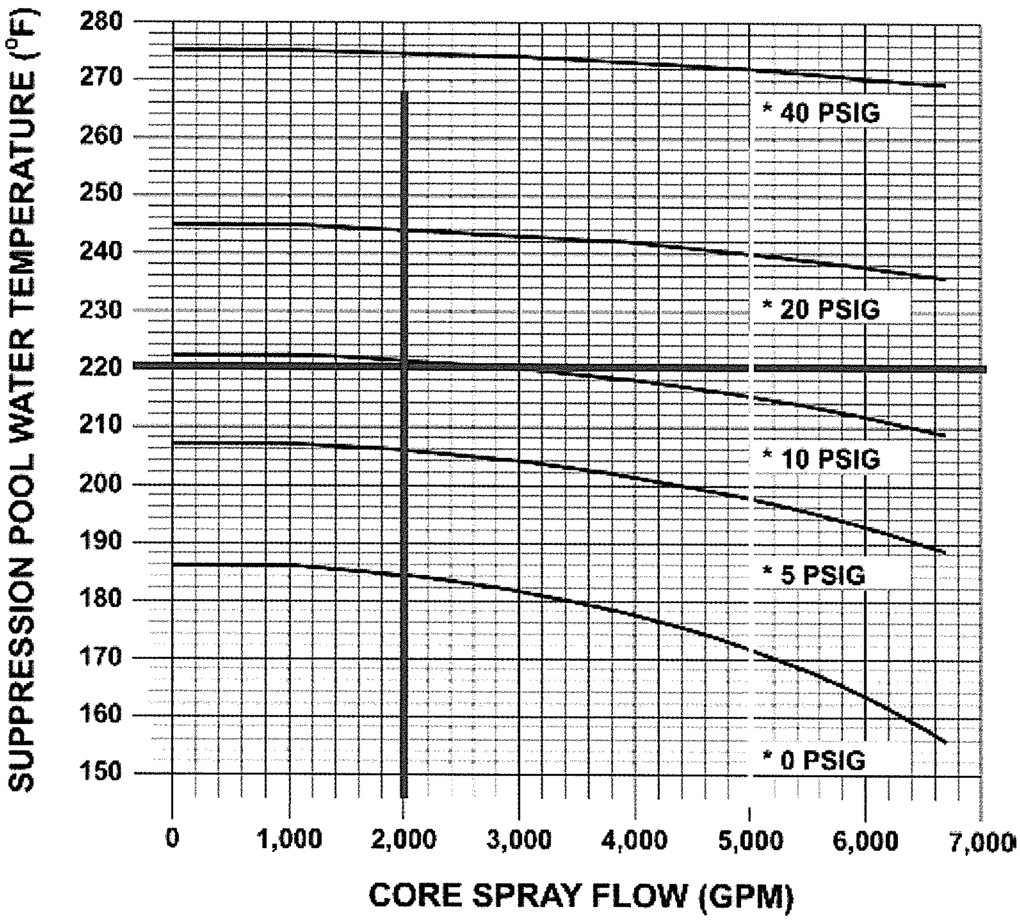
ATTACHMENT 5
Page 14 of 28
Definitions

For the NPSH graphs, adequate net positive suction head is available when the flow rate and Suppression Pool temperature combination is below the adjusted Suppression Pool pressure curve. The indicated Suppression Pool pressure must be reduced by 0.5 psig for every foot of water level less than -2.6 feet to determine the correct Suppression Pool pressure curve to be used in evaluating NPSH.

Loop A flow in yellow, Loop b flow in Blue, Temp in red, safe below the 10 psig line.

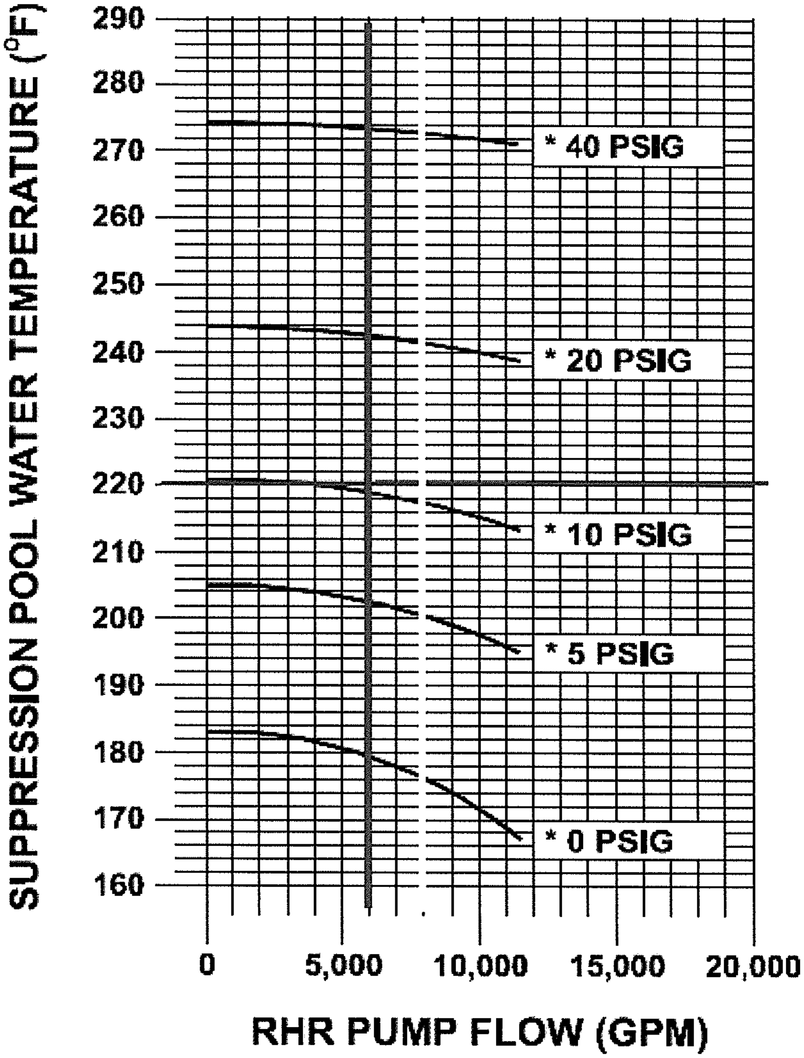
FIGURE 5

Core Spray NPSH Limit



SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

FIGURE 6
RHR NPSH Limit



SUBTRACT 0.5 PSIG FROM INDICATED SUPPRESSION CHAMBER PRESSURE FOR EACH FOOT OF WATER LEVEL BELOW A SUPPRESSION POOL WATER LEVEL OF -31 INCHES (-2.6 FEET).

*SUPPRESSION CHAMBER PRESSURE (CAC-PI-1257-2A OR CAC-PI-1257-2B)

57. 295031 1

The plant has experienced a level transient which has caused the Reactor Recirc Pumps to trip. (Level stabilized when the Reactor Recirc Pumps tripped.)

Which one of the following choices completes the statement below?

Under these conditions, primary loop natural circulation ___(1)___ occurring because reactor water level is ___(2)___.

A. (1) is
   (2) above the jet pump suction

B. (1) is
   (2) above the steam separator return to the downcomer region

C. (1) is NOT
   (2) below the jet pump suction

D. (1) is NOT
   (2) below the steam separator return to the downcomer region

Answer: D

K/A: 295031 Reactor Low Water Level
EK1.02 - Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Natural circulation: Plant-Specific (CFR: 41.8 to 41.10)

RO/SRO Rating: 3.8/4.1

Objective: LOI-CLS-LP-002, Obj. 16 - Given plant conditions, determine if the conditions for natural circulation to occur in the reactor vessel are met.

Reference: None

Cog Level: Low

Explanation: In order to establish natural circulation water must reach the turnaround point in the steam separators (200 - 220 inches in the core). The student must equate the RR pump trip to a Low Level Two condition (105 inches).

Distractor Analysis:

Choice A: Plausible because level is above the jet pump suction which is where the water would travel through to the bottom head region (part of the flowpath) but this does not establish natural circulation conditions.

Choice B: Plausible because the student must know the relationship between LL2 (105 inches) and steam separator turnaround point (200-220 inches). If it is not recognized that a recirc pump trip occurs below the steam separator turnaround point, this choice may be picked.

Choice C: Plausible because natural circulation conditions are not established but jet pump suction level is not a requirement for natural circulation.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

2. Natural Circulation

Natural circulation is important to core cooling following transient or accident conditions when forced circulation by way of the reactor recirculation pumps is not available. Natural circulation not only occurs following transients and accidents, but is a major contributor to the flow rate through the core during startup, full power operation and shutdown.

In order for natural circulation to occur, there must be communication between the heat source and the heat sink. The reactor internals are designed to ensure communication at all times. Several flow paths exist for natural circulation and a loss of one will tend to promote the others.

Primary Loop

Water flow through the core removes heat from the fuel and boils to produce steam. The water and steam mixture passes through the separator and dryer for moisture removal. Steam leaves the vessel and passes to the turbine, while the water stays behind and is recirculated to the downcomer region to mix with the cooler feedwater. This flow continues downward through the jet pumps to pass back up through the core to repeat the cycle.

58. 295032 1

Which one of the following statements identifies the reason SCCP directs emergency depressurization based on temperature IAW 0OI-37.9, Secondary Containment Control Procedure Basis Document?

A. To prevent an unmonitored release.

B. To ensure ODCM site boundary dose limits are not exceeded.

C. To prevent damage to equipment required for safe shutdown.

D. To preserve personnel access into the reactor building.

Answer: C

K/A: 295032 High Secondary Containment Area Temperature
EK3.01 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Emergency/normal depressurization (CFR: 41.5 / 45.6)

RO/SRO Rating: 3.5/3.8

Objective: LOI-CLS-LP-300-M Obj. 13a, Given plant conditions and the SCCP, determine the required actions if the following limits are exceeded: Maximum Safe operating values with a primary system discharging into secondary containment.

Reference: None

Bank questions that was last used on the 08 NRC exam.

Cog Level: Low

Explanation: The Maximum Safe Operating Temperature values are the area temperatures above which equipment necessary for the safe shutdown of the plant will fail. These area temperatures are utilized in establishing the conditions which reactor depressurization is required. The criteria of more than one area specified in this step identifies the rise in reactor building parameters as a wide spread problem which may pose a direct and immediate threat to secondary containment integrity, equipment located in the RB, and continued safe operation of the plant.

Distractor Analysis:

Choice A: Plausible because not mitigating rad release.

Choice B: Plausible because other things, RRCP, deal with rad release concerns.

Choice C: Correct answer, see explanation.

Choice D: Plausible because this is the reason for Radiation limitations.

SRO Basis: N/A

The Maximum Safe Operating Values are the area temperatures above which equipment necessary for the safe shutdown of the plant will fail. These area temperatures are utilized in establishing the conditions under which reactor depressurization is required. Separate temperatures are provided for each Secondary Containment area.

59. 295037 1

LPC is being executed on Unit One due to an ATWS. Plant conditions are:

Reactor power is 21%
Reactor pressure is 940 psig, controlled by EHC
Reactor water level is 170 inches, controlled by feedwater

Which one of the following identifies why LPC directs deliberately lowering reactor water level to 90 inches?

A. Raise feedwater subcooling.

B. Increase the resulting boron concentration in the core.

C. Reduce heat input to primary containment if the MSIVs close.

D. Minimize the possibility of large scale core oscillations.

Answer: D

K/A: 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
EK1.02 - Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor water level effects on reactor power (CFR: 41.8 to 41.10)

RO/SRO Rating: 4.1/4.3

Objective: LOI-CLS-LP-300-E, Obj. 06 - Explain the reason for lowering reactor water level while performing the Level/Power Control Procedure.

Reference: None

Cog Level: Low

Explanation: (IAW OI-37.5) To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, reactor water level is lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, the initiation and growth of oscillations is principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude.

Distractor Analysis:

Choice A: Plausible because lowering level will reduce (not raise) feedwater subcooling by introducing the feedwater into the steam space below the feedwater spargers.

Choice B: Plausible because reducing the core water inventory would result in a higher boron concentration (per unit volume).

Choice C: Plausible because LPC would continue to lower level below 90 inches if conditions of Table 3 are met. The purpose of lowering level below 90 inches is to reduce heat input to containment.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

STEP BASES:

To prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities, reactor water level is initially lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, the initiation and growth of oscillations is principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude.

If reactor power is at or below the APRM downscale trip setpoint (2%), it is highly unlikely that the core bulk boiling boundary would be below that which provides suitable stability margin for operation at high powers and low flows. (A minimum boiling boundary of 4 ft above the bottom of active fuel has been shown to be effective as a stability control because a relatively long two-phase column is required to develop a coupled neutronic/ thermal-hydraulic instability.) Furthermore, flow/density variations would be limited with reactor power this low since the core has a relatively low average void content.

The combination of high reactor power (above the APRM downscale trip), high Suppression Pool temperature (above the Boron Injection Initiation Temperature), and an open SRV or high drywell pressure (above the scram setpoint), are symptomatic of heat being rejected to the Suppression Pool at a rate in excess of that which can be removed by the Suppression Pool Cooling system. Unless mitigated, these conditions ultimately result in loss of NPSH for ECCS pumps taking suction on the Suppression Pool, containment overpressurization, and (ultimately) loss of Primary Containment integrity, which in turn could lead to a loss of adequate core cooling and uncontrolled release of radioactivity to the environment.

If the Suppression Pool heatup continues and heat being rejected to the Suppression Pool remains above decay heat levels, the reactor power reduction achieved by lowering reactor water level to TAF retards Suppression Pool heatup, thus preserving Suppression Pool heat capacity and delaying Emergency Depressurization as long as possible.

60. 295038 1

Which one of the following alarms is an entry condition into RRCP?

A. *Service Wtr Effluent Rad High.*

B. *Area Rad Turbine Building High.*

C. *RBCCW Liquid Process Rad High.*

D. *Process Reactor Building Vent Rad High.*

Answer: A

K/A: 295038 High Off-Site Release Rate
EK2.06 - Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Process liquid radiation monitoring system (CFR: 41.7 / 45.8)

RO/SRO Rating: 3.4/3.7

Objective: LOI-CLS-LP-300-N, Obj. 2 - Given plant conditions, determine if 0EOP-04-RRCP should be entered.
Reference: None

Modified question from the 10-1 NRC exam. (Emergency Procedure_30) Changed the distractors to make a different answer correct.

Cog Level: Low

Explanation: A number of conditions associated with pathways outside containment for the transport of radioactive materials are listed for entry to RRCP. Distractors are monitored areas and processes with valid alarms but which are enclosed processes or area rads which are not used for determination of off-site doses.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because Turbine Building Vent Rad High annunciator is an entry condition for RRCP. Area Rad is not but is similar in description.

Choice C: Plausible because this alarm requires actions up to and including entry into AOPs. However, EOP entry is not required directly from the alarm.

Choice D: Plausible because Reactor Building Roof Vent Hi and Process Off-Gas Vent Pipe Rad Hi alarms are both entry conditions for RRCP. Process Reactor Building Vent Rad High is not but is similar in description.

SRO Basis: N/A

RRCP

RADIOACTIVITY RELEASE CONTROL

RR-1

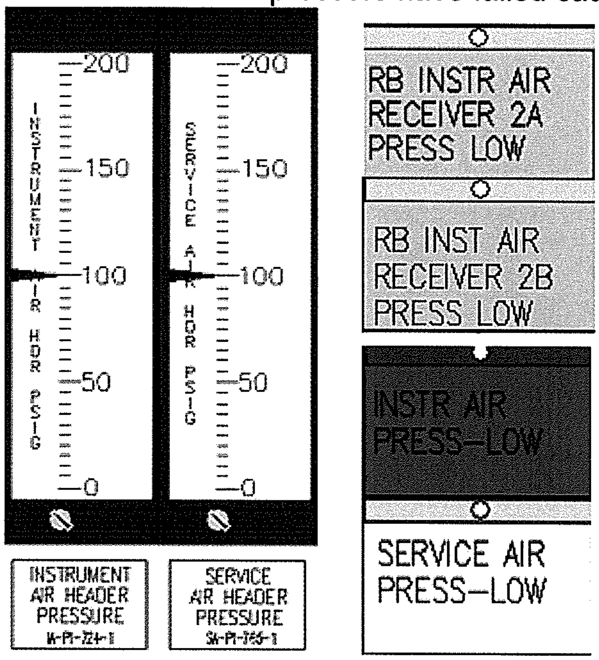
ENTRY CONDITIONS:

* MAIN STEAM LINE RAD HI ANNUN SETPOINT EXCEEDED (UA-23,2-6)
* PROCESS OFF-GAS RAD HI ANNUN SETPOINT EXCEEDED (UA-03,5-2) (SJAE)
* RX BLDG ROOF VENT RAD HIGH ANNUN SETPOINT EXCEEDED (UA-03,2-3)
* TURB BLDG VENT RAD HIGH ANNUN SETPOINT EXCEEDED (UA-03,3-3)
* PROCESS OG VENT PIPE RAD HI ANNUN SETPOINT EXCEEDED (UA-03,6-4) (STACK)
* SERVICE WTR EFFLUENT RAD HIGH ANNUN SETPOINT EXCEEDED (UA-03,5-5)
* ANY UNMONITORED OFF-SITE RADIOACTIVITY RELEASE
* CALCULATED DOSE RATE LIMIT OF "NOBLE GAS INSTANTANEOUS RELEASE RATE DETERMINATION" (E&RC-2020) EXCEEDED

RR-2

61. 300000 1

The service air compressors have failed causing the following plant conditions:



Not in Alarm

Not in Alarm

In Alarm

In Alarm

Which one of the following identifies the operator action that must be taken IAW 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures?

A. Insert a manual reactor scram

B. Close Service Air Isol Vlvs, SA-PV-706-1 & 706-2

C. Open Div I(II) Backup N2 Rack Isol Vlvs, RNA-SV-5482(5481)

D. Open Serv Air Dryer 1A Bypass Pressure Control Valve, 1-SA-PV-5067

Answer: B

K/A: 300000 Instrument Air System
G2.02.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
(CFR: 41.5 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.4

Objective: LOI-CLS-LP-046, Obj. 11b - State the reason(s) for the following actions occurring during operation of the Pneumatic Systems: Service Air Header is isolated at 105 psig

LOI-CLS-LP-046, Obj. 07a - Given plant conditions, determine if the following automatic actions should occur:

Reference: None

Modified question from the 10-2 NRC exam. (300000_14) Previous question asked what auto action should occur while this question has the student determine what auto action that has failed.

Cog Level: High

Explanation: The service air and instrument air systems use the same air compressors and are essentially the same air system. As pressure decreases the service air system will auto isolate at 105 psig. When the isolation occurs the pressure indication will go to zero because the isolation valves are upstream of the tap off for the indication. With system leakage it goes to zero fairly quickly. The alarm for low service air comes in at 105 psig and low instrument air at 100 psig. This is an auto action that should have occurred so the operator will make it happen per our procedures.

Distractor Analysis:

Choice A: Plausible because if Unable to maintain at least one division noninterruptible instrument air pressure above 95 psig then this is an operator action.

Choice B: Correct Answer, see explanation

Choice C: Plausible because if RB air pressure alarm is in then this is an operator action

Choice D: Plausible because if pressure was <98 psig this would be an operator action.

SRO Basis: N/A

INSTR AIR PRESS-LOW

AUTO ACTIONS

NONE

CAUSE

1. Abnormal supply to system due to:
   a. Loss of power to compressor(s).
   b. Failure of the Air Compressor 1B or 2B to start or load automatically on decreasing pressure.
   c. Service Air Isolation Valves, SA-PV-706-1 and SA-PV-706-2, did not close on decreasing header pressure (105 psig).
   d. Heavy leak in the supply system.
   e. Idle compressor check valve leaking.
   f. Service air dryer high differential pressure or malfunction.
2. Abnormal distribution system due to:
   a. Heavy usage of service air or instrument air.
   b. Leaks in the service air or instrument air piping.
3. Circuit malfunction.

OBSERVATIONS

1. Instrument air header pressure decreases as shown by Control Room and Local Indicators IA-PI-724-1 and IA-PI-724-2.
2. Service air header pressure decreases as shown by Control Room and Local Indicators SA-PI-765-1 and SA-PI-765-2.
3. Air Compressor 1B and 2B running when pressure falls to approximately 110 psig.
4. SERVICE AIR PRESS-LOW (UA-01 5-4) alarms.
5. AIR DRYER TROUBLE (1/2 APP-UA-01 5-3, 1APP-UA-01 6-3).
6. Service air header is isolated at less than 105 psig.
7. Overall unit operation for any abnormality due to low instrument air system pressure.

7. Written procedures are not necessary for situations where:

   a. Prompt action is necessary to minimize personnel injury, damage to the facility, and to protect the health and safety of the public.

   b. Prompt action is necessary to prevent the deterioration of plant conditions or components to a possibly unsafe or unstable level. If time permits, approval from the SM/CRS shall be obtained. Factors to consider included: complexity of action, potential for damage from common cause, etc.

   c. Conditions exist which may require timely actions due to failure of automatic control systems or uncertain equipment status (i.e., taking manual control of hand/auto stations, position of selector switches, etc.).

62. 400000 1

Which one of the following choices completes the statements below?

The Service Water to TBCCW MOVs (SW-V3 and V4) are powered from ____(1)____.
The Service Water to RBCCW MOVs (SW-V103 and V106) are powered from ____(2)____
.

A. (1) BOP distribution
   (2) BOP distribution

B. (1) BOP distribution
   (2) E Bus distribution

C. (1) E Bus distribution
   (2) BOP distribution

D. (1) E Bus distribution
   (2) E Bus distribution

Answer: D

K/A: 400000 Component Cooling Water System
K2.02 - Knowledge of electrical power supplies to the following: CCW valves (CFR: 41.7)

RO/SRO Rating: 2.9/2.9

Objective: LOI-CLS-LP-043, Obj. 8d - State the power supply (bus and voltage) for the following Service Water System components: MOVs

Reference: None

Cog Level: Low

Explanation: Conventional Header to TBCCW Heat Exchanger Outboard Valve SW-V4 1XA (2XA)
Conventional Header to TBCCW Heat Exchanger Inboard Valve SW-V3 1XB (2XB)
SW Header to RBCCW Heat Exchangers Isolation Valve SW-V106 1XA (2XA)
SW Header to RBCCW Heat Exchangers Isolation Valve SW-V103 1XB (2XB)

Distractor Analysis:

Choice A: Plausible because other SW system valves are powered from BOP.

Choice B: Plausible because other SW system valves are powered from BOP.

Choice C: Plausible because other SW system valves are powered from BOP.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

Example of SW valves powered from BOP sources (From SD-43):

TBCCW Heat Exchanger 1B (2B) SW Outlet Valve SW-V6	1TG (2TG)	EG3 (EG3)

63. 400000 2

Unit Two is at rated power with the following alignment for RBCCW:

2A RBCCW	control switch in ON
2B RBCCW	control switch in ON
2C RBCCW	control switch in AUTO

The 2D to E3 master breaker tripped spuriously and DG3 failed to auto start.

Which one of the following identifies the expected response of the RBCCW pumps?

A. No RBCCW pumps are running.

B. ONLY the 2B RBCCW pump is running.

C. ONLY the 2C RBCCW pump is running.

D. 2B and 2C RBCCW pumps are running.

Answer: B

K/A: 400000 Component Cooling Water System
K6.07 - Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Breakers, relays, and disconnects (CFR: 41.7 / 45.7)

RO/SRO Rating: 2.7/2.8

Objective: LOI-CLS-LP-021, Obj. 9a - Given plant conditions, determine how RBCCW is affected by the following: Loss of AC Power

Reference: None

Cog Level: High

Explanation: Failure of power to E3 renders A and C pumps without power. (Plausibility of no pumps running) Failure of power to E3 also disables the auto start pressure switch, RCC-PS-672 (DP-2A 120V) therefore B RCC pump would not receive an AUTO start signal but it is already running.

Distractor Analysis:

Choice A: Plausible because the A and C pumps lose power and if the B pump was not running then it would not get a start signal.

Choice B: Correct Answer, see explanation

Choice C: Plausible if the student has the power supplies wrong as A and B from E3 instead of A and C.

Choice D: Plausible if the student thinks the power supply is lost only to A pump and considers the low pressure starting the C pump.

SRO Basis: N/A

Load: 480V Motor Control Center 2-2XE
Location: Reactor Building 50' N
Drawing Reference: F-03049
Upstream Power Source: **480V Substation E7**

COMPT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
EA1	RBCCW Pump 2A RCC-2A-PMP	Loss of load
EA7	RBCCW Pump 2C RCC-2C-PMP	Loss of load

Load: 120V Distribution Panel 2-2A-120V (H06)
Location: Control Building 23' SE
Drawing Reference: LL-09341-4
Upstream Power Source: **120V Emergency Distribution Panel 2E7**

CKT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
24	RBCCW Discharge Header Pressure Switch RCC-PS-672	Auto start of RBCCW pumps will be defeated for low discharge header pressure

From SD-21:

Power supplies are as follows:

PUMP	POWER SUPPLY
RBCCW Pump 1A	MCC 1XE
RBCCW Pump 1B	MCC 1XF
RBCCW Pump 1C	MCC 1XE
RBCCW Pump 2A	MCC 2XE
RBCCW Pump 2B	MCC 2XF
RBCCW Pump 2C	MCC 2XE

64. 600000 1

The CRS has declared the Control Room Envelope Boundary inoperable.
A fire external to the control room has resulted in smoke intrusion into the control room. The CRS has determined that control room evacuation is not necessary and has ordered the donning of SCBAs by control room personnel.

Which one of the following indicates the approximate length of time each SCBA bottle will last IAW 0OI-01.01, BNP Conduct of Operations Supplement?

A. 10 minutes

B. One hour

C. Two hours

D. Four hours

Answer: B

K/A: 600000 Plant Fire On Site
AA2.10 - Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Time limit of long-term-breathing air system for control room (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 2.9/3.1

Objective: None

Reference: None

Cog Level: Low

Explanation: Smoke in the control room from outside sources is addressed in 0OI-01.01. If control room evacuation is not required, SCBAs may be used for long term habitability. There are no installed breathing air systems for the control room. IAW E&RC-0292, SCBA Use and Maintenance, and 0OI-01.01, the approximate duration of the SCBAs is one hour.

Distractor Analysis:

Choice A: Plausible because E&RC 0292 states expected duration of the Ska-Pak is five to ten minutes.

Choice B: Correct Answer, see explanation

Choice C: Plausible because two hours is a period of time that may be reasonable to the candidate.

Choice D: Plausible because four hours is a period of time that may be reasonable to the candidate.

SRO Basis: N/A

0ERC-0292

6.16 The air supply in a Scott Air-Pak 4.5 SCBA is rated at approximately 1 hour.

6.17 The air supply in the Ska-Pak with escape bottle is from 5 to 10 minutes.

0OI-01.01

a. **General Mitigating Actions**

(3) In the case of an actual radiological **OR** chemical event, **OR** a challenge from smoke where the use of SCBA air packs has been chosen as the mitigating action, Fire Brigade **OR** FIN team personnel will be directed to deliver additional/replacement SCBA bottles as necessary to CRE occupants. SCBA bottles contain approximately 1 hour worth of air.

b. **Smoke in the Control Room from Outside Sources**

(1) Place CREV in service.

(2) Review 0AOP-32.0.

NOTE:	One of the small, two of the large, **AND** one of the Xlarge SCBA masks are stored with the two SCBA air packs stored in the CAS.

(3) Issue SCBA air packs **AND** masks from the Control Room emergency inventory to CRE occupants if Control Room evacuation is not necessary.

(8) Attempt to determine estimate of duration of SCBA use for review of continued CRE occupancy plans. For extended SCBA use, the availability of full SCBA bottles **AND** masks may become a limiting factor for continued CRE occupancy.

65. 700000 1

Unit Two is operating at rated power when small oscillations due to unstable voltage regulation is observed.

Which one of the following choices completes the statements below IAW 2OP-27, Generator and Exciter System Operating Procedure?

The **first** action required is to ____(1)____.

This action must be reported to the ____(2)____.

A. (1) place the generator voltage regulator in Manual
(2) System Load Dispatcher

B. (1) place the generator voltage regulator in Manual
(2) Plant Transmission Activities Coordinator (PTAC)

C. (1) disable the Power System Stabilizer (PSS)
(2) System Load Dispatcher

D. (1) disable the Power System Stabilizer (PSS)
(2) Plant Transmission Activities Coordinator (PTAC)

Answer: C

K/A: 700000 Generator Voltage and Electric Grid Disturbances
G2.04.30 - 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. (CFR: 41.10 / 43.5 / 45.11)

RO/SRO Rating: 2.7/4.1

Objective: LOI-CLS-LP-027-A, Obj. 18h - Given the plant conditions and one of the following events use plant procedures to determine the actions required to control and/or mitigate the consequences of the event: Unstable Voltage Regulator Operation

Reference: None

Cog Level: High

Explanation: The Power System Stabilizer (PSS) provides for dampening of power oscillations resulting from unstable grid perturbations or improper voltage regulator operation. In the event unstable voltage regulation is observed, the first steps required IAW 2OP-27 are to disable the PSS to determine if a malfunction of the PSS is the cause of the condition. If the PSS is not the cause and oscillations continue, the next action will be to take the voltage regulator to Manual. The System Load Dispatcher is required to be notified if either of these actions are performed. The PTAC is required to be notified for certain events as specified in 0OI-01.01, Attachment 30, but not for the actions for the conditions described in the question.

Distractor Analysis:

Choice A: Plausible because OP-27 provides instructions for placing the voltage regulator in Manual but only after attempts to stabilize voltage oscillations by disabling PSS are attempted. System Load Dispatcher notification is required in either case.

Choice B: Plausible because OP-27 provides instructions for placing the voltage regulator in Manual but only after attempts to stabilize voltage oscillations by disabling PSS are attempted. PTAC notification is required for numerous conditions specified in 0OI-01.01.

Choice C: Correct answer, see explanation.

Choice D: Plausible because disabling of the PSS is the first action specified in OP-27. Notification of System Load Dispatcher is required. PTAC notification is required for numerous conditions specified in 0OI-01.01.

SRO Basis: N/A

SD-27

2.17.7 Power System Stabilizer (PSS)

This circuit is designed to modulate the excitation system to provide supplementary damping for low frequency power oscillations that may exist due to system transmission configurations and plant loading. Modulation of the generator excitation produces transient changes in the generator's electrical output power which counteracts this situation.

A Power System Stabilizer has been added to prevent these oscillations from occurring. Once the unit is above 325MVA load the Power System Stabilizer will be enabled and will automatically dampen possible low frequency oscillations. OP-27 should be followed in the event that power oscillations occur while the PSS is ACTIVE or in the event that the PSS is DISABLED.

8.6 Unstable Voltage Regulator Operation

C
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Us

8.6.1 Initial Conditions

1. The Generator and Exciter System is in operation in accordance with Section 5.1. ☐

NOTE:	Voltage may be oscillating as a result of the Power System Stabilizer modulating exciter field voltage to provide positive dampening of power oscillations.

2. Unstable voltage regulation is indicated (Panel XU-1) by oscillations on *GEN VOLT REG DIFF VOLT, GEN-VM-3495* and oscillations on any of the following: ☐

   - *GEN-VM-733*
   - *GEN-AM-735*
   - *GEN-MVAR-728*
   - *GEN-AM-729*

8.6.2 Procedural Steps

24 1. **NOTIFY** System Load Dispatcher prior to disabling PSS **OR** removing voltage regulator from automatic control. ☐

2. **PLACE** *PSS CONTROL, PSSCS1*, to *DISABLE*, at Power System Stabilizer control panel, *2-GEN-PSS*. ☐

3. **PLACE** *PSS ALARM BYPASS, PSSCS3*, in *BYPASS*. ☐

R18

ATTACHMENT 30
Page 3 of 3

Off Normal and Infrequent Grid Operations Matrix

Condition	Request ECC provide time & duration estimate	Notify the Plant PTAC	Evaluate plant risk & need to curtail activities	Curtail activities that risk loss of generation	Notify Plant Mgmt	Prepare to reduce power generation	Reference	Reduce power generation as directed by Ops Management	Refer to Tech. Spec. 3.8.1	Verify FSS in service
Tier 1 Trans. Lines Out	X	X	X	X	X	X	1(2)OP-50 OGP-12	X		X
Tier 2 Trans. Lines Out	X	X	X							X
Minimum Load Emergency	X				X	X	OGP-12	X		
System Reliability Alert	X	X	X	X			EAP-025			
Energy Emergency Alert (EEA Levels 1 – 3)	X	X	X	X						
Enable Part Time Load Shedding	X		X	X			1(2)OP-50			
<u>Anticipated</u> LOCA Voltage Support Problem	X	X	X	X			1(2)OP-50			
<u>Actual</u> LOCA Voltage Support Problem	X	X	X	X	X		1(2)OP-50 OAOP-22		X	
Grid Frequency Problem	X	X	X	X	X		OAOP-22		X	
<u>Other</u> Grid Related Problems: - ECC Predictive Tool/Computer Problems (EMS, SCADA, RTCA) - Substation Problems - Neighboring Utility Problem	X	X	X							
Sabotage/Terrorism					X		OAOP-22			
Grid Cyber Attack OR Notification	Open Comm. to ECC	X	X	X	X	X	OAOP-22	X	X	X

66. G2.01.05 1

Following a seven day break in work schedule an operator is assigned to an operating unit with the following work schedule.

Day 1	12	hours
Day 2	12	hours
Day 3	12	hours
Day 4	12	hours
Day 5	0	hours
Day 6	12	hours
Day 7	13	hours
Day 8	14	hours
Day 9	0	hours

Which one of the following choices will ensure compliance IAW ADM-NGGC-0206, Managing Fatigue and Work Hour Limits, without requiring a waiver?

A. Take day 1 off.

B. Take day 7 off.

C. Work only 12 hours on day 7.

D. Work only 13 hours on day 8.

Answer: B

K/A: G2.01.05 - Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 2.9/3.9

Objective: AOI-CLS-LP-201-D, Obj. 19 - Discuss working hour limitations per ADM-NGGC-0206.

Reference: None

Cog Level: High

Explanation: ADM-NGGC-0206 calculates work week limits on rolling time periods of 24 hours, 48 hours and 7 days (168 hours). These time periods do not reset following a day off but continue to roll. The limitations specified in the procedure are: 16 hours in any 24 hrs, 26 hours in any 48 hours, and 72 hours in any seven days. Turnover time is excluded. A difference exists in the 72 hour calculation if the work is being performed during an outage. In that case, the 72 hour limit may be considered on a calendar day period instead of rolling.

In the example given, work hours are exceeded on day 8 (27 hours in 48), on the seven days from day 1 through day 7 (73 hours in one week), and on the seven days from day 2 through day 8 (75 hours in one week). Taking day seven off would clear the excessive hours from day 8, and both rolling weeks.

Distractor Analysis:

Choice A: Plausible because taking day one off would clear the excessive hours from the seven day period from day 1 through day 7. However, the second seven day period from day 2 through day 8 and day 8 hours are still excessive.

Choice B: Correct Answer, see explanation

Choice C: Plausible because working 12 hours on day 7 would clear the excessive hours on day 8 and would also clear the excessive hours on the seven day period from day 1 through day 7. However, the hours during the seven day period from day 2 through day 8 would still be excessive.

Choice D: Plausible because working 13 hours on day 8 would clear the excessive hours on day 8 and the seven day period from day 2 through day 8. However, the hours during seven day period from day 1 through day 7 would still be excessive.

SRO Basis: N/A

ADM-NGGC-0206

9.2 Covered Worker Work Hour Controls

9.2.1 On-Line Work Hour Limits

1. A worker's work hours shall not exceed the following limits unless a waiver is issued.

   a) 16 work hours in any 24-hour period (MWH16/24)

   b) 26 work hours in any 48-hour period (MWH26/48)

   c) 72 work hours in any 7-day or 168 hour period (MWH72/168)

9.2.2 Outage Work Hour Limits

1. Evaluating work hours to outage rules versus on-line rules is optional. Should a plant enter an unplanned outage, it is not necessary to reassign personnel to an outage schedule. Nor is it necessary to assign an outage schedule to personnel working on outage activities.

2. A worker's work hours shall not exceed the following limits unless a waiver is issued.

   a) 16 work hours in any 24-hour period (MWH16/24)

   b) 26 work hours in any 48-hour period (MWH26/48)

   c) 72 work hours in any 7-day or 168 hour period (MWH72/168)

NOTE:	EmpCenter evaluates this rule on a rolling 168 hour window. The regulatory limit is actually 72 hours in any 7 calendar day period. If EmpCenter indicates a violation based on 168 hours, but the worker has not exceeded 72 hours in the 7 calendar day period, the violation may be overridden.

8.2 Workers shall not exceed the work hour limits defined in the procedure unless authorized by a waiver. If it is determined that a worker has violated the requirements, an NCR must be written and, if applicable, the NCR number added to the comment fields in EmpCenter.

9.1.2 Items To Be Excluded From Work Hour Calculation

4. All turnover time can be excluded from the total hours worked however only one period of turnover time per shift can be excluded for determining if the minimum break requirements are met.

67. G2.01.32 1

Which one of the following identifies the reason 1OP-10, Standby Gas Treatment System Operating Procedure, prohibits venting the drywell and the suppression pool chamber simultaneously with the reactor at power?

This would cause:

A. the operation of torus to drywell vacuum breakers

B. the pressure suppression function to be bypassed.

C. the operation of reactor building to torus vacuum breakers.

D. excessive release of radioactivity through the main stack.

Answer: B

K/A: G2.01.32 - Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

RO/SRO Rating: 3.8/4.0

Objective:

Reference: None

Bank question that was last used on the 07 NRC exam.

Cog Level: Low

Explanation: Per OP-10, torus and drywell cannot be vented at the same time in Modes 1, 2 or 3. per the LER reference, this could result in bypassing pressure suppression function.

Distractor Analysis:

Choice A: Plausible because this lineup equalizes pressure between the drywell and the suppression pool free air space since the vacuum breakers operate on a d/p between the spaces this would bypass them, not open them.

Choice B: Correct Answer, see explanation

Choice C: Plausible because these vacuum breakers prevent drawing a negative pressure in the suppression pool. Cross connecting the drywell and the suppression pool free air space will not cause a negative pressure in the suppression pool.

Choice D: Plausible because this action would only be done when LOCA conditions do NOT exist

SRO Basis: N/A

3.0 PRECAUTIONS AND LIMITATIONS

3.4 The Standby Gas Treatment System will **NOT** automatically start if the control switch is in *STBY*.

R16 3.5 Venting of the Drywell and Suppression Pool simultaneously shall **NOT** be performed when the plant is in Mode 1, 2, or 3.

2.0 REFERENCES

R16 2.16 LER 1-97-011, Drywell and Torus Inerting/Deinerting Lineup Results in Unanalyzed Suppression Pool Bypass Path

68. G2.01.44 1

During a core reload the initial loading of fuel bundles around each SRM centered 4-bundle cell was completed with all four SRMs fully inserted and reading 50 cps.

After some additional fuel assemblies were loaded (none of which were adjacent to any SRM) the RO observes that all SRMs have increased steadily. Current readings are as follows:

SRM A	120 cps	SRM C	260 cps
SRM B	130 cps	SRM D	140 cps

Which one of the following identifies the required action, if any, at this time IAW 0FH-11, Refueling?

A. Immediately suspend the movement of fuel.

B. Continue monitoring SRMs during fuel movement as readings are not unusual.

C. Report to the CRS that SRM C is INOPERABLE and recommend bypassing SRM C until I&C can investigate. Fuel movements may continue.

D. Record a new baseline count rate for the SRMs on 0FH-11, Attachment 6, Documentation for SRM Baseline. Fuel movements may continue.

Answer: A

K/A: G2.01.44 - Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation. (CFR: 41.10 / 43.7 / 45.12)

RO/SRO Rating: 3.9/3.8

Objective: LOI-CLS-LP-305, Obj. 18 - Given the conditions during a refueling outage state the operator actions for rising SRM count rates and/or inadvertent criticality.

Reference: None

Bank question that was last used on the 04 NRC exam.

Cog Level: High

Explanation: FH-11 requires suspending fuel movement if SRM increase by factor of 5 relative to initial base-line SRM reading (or doubles with any single bundle). Also, malfunctioning of any SRM channel shall be reason to terminate refueling operations until TS compliance can be determined.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because factor of five (from 50 to 250 cps) can be confused with five doublings (from 50 to 800 cps).

Choice C: Plausible because fuel movements could continue on a failed SRM if TS actions are completed.

Choice D: Plausible because if a bundle is placed next to an SRM new baseline data is recorded. The question states that all adjacent bundles are loaded.

SRO Basis: N/A

4.0 PRECAUTIONS AND LIMITATIONS

4.42 Fuel movement shall be suspended and the Reactor Engineer contacted if either of the following occur:

4.42.1 An SRM reading increases by a factor of two upon insertion of any single bundle. During a spiral reload, this restriction applies only after the initial loading of fuel bundles around each SRM is complete. During a Core Shuffle, this restriction does **NOT** apply to the SRM that is having an adjacent fuel bundle inserted or removed.

4.42.2 An SRM increases by a factor of five relative to the SRM baseline count rate recorded on Attachment 6.

1. **OBSERVE AND LOG** on Form 0ENP-24.12-3, the SRM readings as each bundle is lowered in place. _____

2. **IF** a fuel bundle is inserted or removed adjacent to an SRM, **THEN RECORD** new baseline count rate for that SRM on Attachment 6. _____

3. **FOLLOW** fuel location on maps by stepping through core component sequence in ShuffleWorks as each fuel move is performed. _____

69. G2.02.20 1

Which one of the following time periods are generally part of the MAX/SAFE/GEN operational periods IAW 0AI-147, Systematic Approach to Trouble Shooting?

A. January and February ONLY

B. June and July ONLY

C. July and August ONLY

D. January, February, June, July and August

Answer: D

K/A: G2.02.20 - Knowledge of the process for managing troubleshooting activities. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 2.6/3.8

Objective:

Reference: None

Cog Level: Low

Explanation: AI-147provides specific dates for MAX/SAFE/GEN operational periods. Each of the available choices present options that a student may conclude reasonable, therefore plausible.

Distractor Analysis:

Choice A: Plausible because student may conclude this is reasonable.

Choice B: Correct Answer, see explanation

Choice C: Plausible because student may conclude this is reasonable.

Choice D: Plausible because student may conclude this is reasonable.

SRO Basis: N/A

0AI-147

3.10 **Max/Safe/Gen**: The Max/Safe/Gen period is generally January, February, June, July and August. Within this period, troubleshooting activities with potential of loss generation (Medium or High Risk activities) require Plant General Manager approval

70. G2.02.35 1

Which one of the following meets the conditions required to be in MODE 4 IAW Technical Specifications?

The Reactor Mode Switch must be in:

A. either Shutdown or Refuel and the first reactor head bolt fully tensioned.

B. either Shutdown or Refuel and all reactor head bolts fully tensioned.

C. Shutdown and the first reactor head bolt fully tensioned.

D. Shutdown and all reactor head bolts fully tensioned.

Answer: D

K/A: G2.02.35 - Ability to determine Technical Specification Mode of Operation.
(CFR: 41.7 / 41.10 / 43.2 / 45.13)

RO/SRO Rating: 3.6/4.5

Objective: CLS-LP-200-B, Obj. 5 - Given a set of plant conditions, determine the plant MODE

Reference: None

Bank question that was last used on the 08 NRC exam.

Cog Level: Low

Explanation: With the reactor head bolts less than fully tensioned the plant is in Mode 5 provided the Mode switch is in Shutdown or Refuel. As head bolts are being tensioned the plant is still in Mode 5. When the last head bolt is fully tensioned the plant will transition from Mode 5 to Mode 4 provided the Mode switch is in Shutdown. If the Mode switch is left in Refuel this would transition the plant from Mode 5 to Mode 2 (and most likely constitute a violation of LCO 3.0.4)

Distractor Analysis:

Choice A: Plausible since mode transition from Mode 5 to Mode 4 occurs during head bolt tensioning and the Mode switch can be in Shutdown or Refuel for Mode 5 (if in Refuel when the last bolt tensioned, transition is from 5 to 2, not 5 to 4)

Choice B: Plausible since mode transition from Mode 5 to Mode 4 occurs when the last head bolt is tensioned and the Mode switch can be in Shutdown or Refuel for Mode 5 (if in Refuel when the last bolt tensioned, transition is from 5 to 2, not 5 to 4)

Choice C: Plausible since the Mode Switch position is correct and the transition occurs during tensioning of head bolts

Choice D: Correct Answer, see explanation

SRO Basis: N/A

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[a] or Startup/Hot Standby	NA
3	Hot Shutdown[a]	Shutdown	> 212
4	Cold Shutdown[a]	Shutdown	≤ 212
5	Refueling[b]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

71. G2.02.42 1

Unit Two is operating at rated power. Drywell leakage calculations are being performed IAW the CO DSR.

The 08:00 drywell leakage calculations were:

Floor Drain Leakage	1.3 gpm
Equipment Drain Leakage	3.5 gpm

These leakage values have been constant for several days.

At 1200, the difference in integrator readings is:

Floor Drain	816 gallons
Equipment Drain	960 gallons

Which one of the following choices is correct IAW RCS Operational Leakage Technical Specifications?

A. RCS Operational Leakage is within limits.

B. Unidentified leakage increase is NOT within limits.

C. Average unidentified leakage is NOT within limits.

D. Average total leakage is NOT within limits.

Answer: B

K/A: G2.02.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

RO/SRO Rating: 3.9/4.6

Objective: LOI-CLS-LP-200-B, Obj. 10d - Explain the following terms as they apply to the Technical Specifications: Condition

Reference: None

Bank question that was last used on the 08 NRC exam.

Cog Level: High

Explanation: Floor drain leakage is 3.4 gpm (816 gal/240 minutes). Equipment drain leakage is 4.0 gpm (960 gallons/240 minutes). Total leakage is 7.4 gpm. Unidentified (floor drain) is <5 gpm and total leakage is <25 gpm, but unidentified leakage increased by 2.1 gpm within the previous 24 hours which does not meet LCO 3.4.4.d (Unidentified leakage increase $\leq$ 2 gpm within previous 24 hours).

Distractor Analysis:

Choice A: Plausible because calculation must be for a four hour period (240 minutes) If one hour (60 minutes) were used, the resulting total leakage would be 29.6 gpm (13.6 floor drains, 16 equipment drains).

Choice B: Plausible because calculation errors could occur resulting in greater than 5 gpm if a four hour period is not used or if candidate associates total floor drain leakage with unidentified leakage.

Choice C: Correct answer, see explanation

Choice D: Plausible because calculation errors could occur or candidate could improperly associate identified and unidentified drains.

SRO Basis: N/A

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE averaged over the previous 24 hour period;

c. ≤ 25 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

72. G2.03.11 1

Following a large line break in the drywell, H2/O2 monitors have been placed in service. Plant conditions:

Drywell hydrogen	2.5% (ERFIS)
Drywell oxygen	3.5% (ERFIS)
Torus hydrogen	1.4% (ERFIS)
Torus oxygen	3.5% (ERFIS)
Torus level	-36 inches

Which one of the following choices describes the actions directed by PCCP?

Vent and purge primary containment ___(1)___ release rate limits.
Venting from the ___(2)___ is preferred.

A. (1) within
   (2) torus

B. (1) within
   (2) drywell

C. (1) irrespective of
   (2) torus

D. (1) irrespective of
   (2) drywell

Answer: A

K/A: G2.03.11 - Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

RO/SRO Rating: 3.8/4.3

Objective: LOI-CLS-LP-300-L, Obj. 8c - Given the Primary Containment Control Procedure and plant conditions, determine if the following actions are required: Venting the primary containment while staying within radioactivity release rate limits.

LOI-CLS-LP-300-L, Obj. 8d - Given the Primary Containment Control Procedure and plant conditions, determine if the following actions are required: Venting the primary containment IRRESPECTIVE of radioactivity release rate limits.

Reference: None

Bank question that was last used on the 04 NRC exam.

Cog Level: High

Explanation: PCCP hydrogen leg directs venting and purging primary containment only within ODCM limits when H2 concentration is above 1% provided O2 concentration remains below 4%. Venting from the torus is preferred due to scrubbing of iodine by the pool.

Distractor Analysis:

Choice A: Correct answer, see explanation

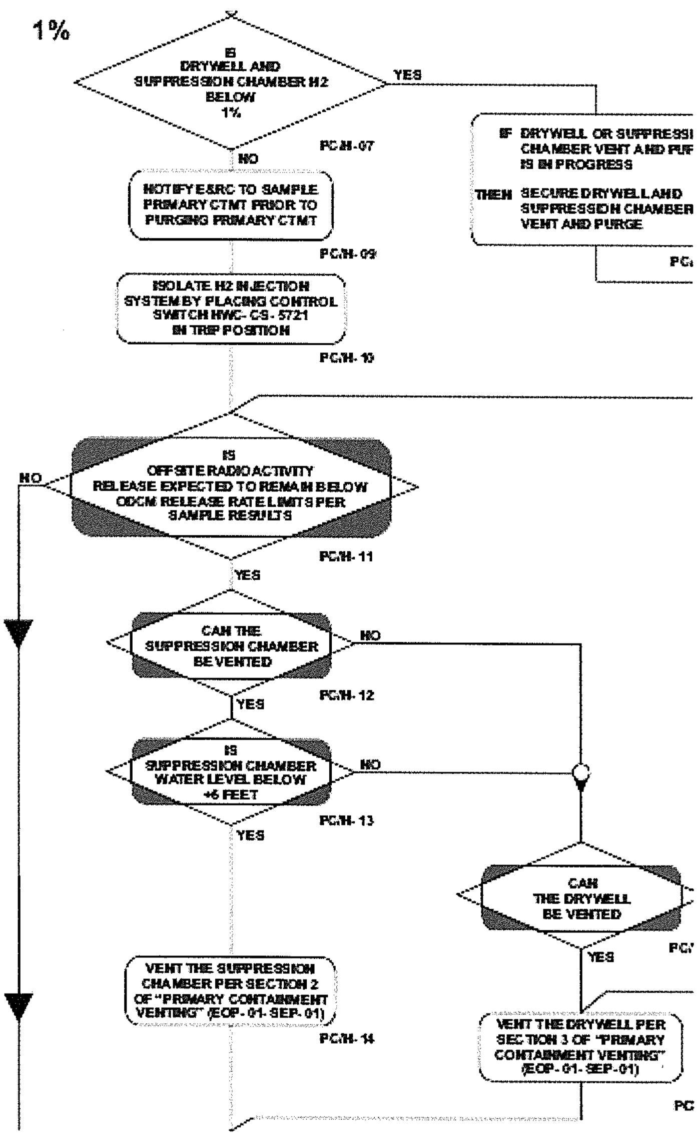
Choice B: Plausible because O2 concentration is below 4%, so venting would be within limits. Venting of drywell would be appropriate if torus water level was higher (+6 ft).

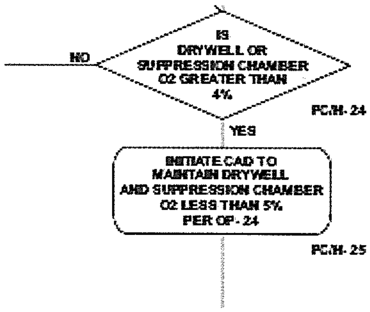
Choice C: Plausible because if O2 concentration was 4% or greater PCCP directs CAD initiation and venting IAW OP-24 which vents irrespective of release limits.

Choice D: Plausible because if O2 concentration was 4% or greater PCCP directs CAD initiation and venting IAW OP-24 which vents irrespective of release limits. Venting of drywell would be appropriate if torus water level was higher (+6 ft).

SRO Basis: N/A

EOP-02-PCCP





2OP-24

16. **THROTTLE** *DW N_2 INLET VLV, CAC-V163* **OR** *CAC-V161,* **AND** *SUPP POOL N_2 INL VLV, CAC-V162* **OR** *CAC-V160*, as necessary, to maintain drywell **AND** suppression pool oxygen concentrations less than 5%. ☐

NOTE:	Primary containment venting should be initiated to prevent exceeding 30 psig in the drywell and secured when drywell pressure lowers to 25 psig. The preferred vent path is through the suppression chamber to allow scrubbing of fission products.

18. **IF** directed by the Site Emergency Coordinator **OR** Unit CRS to maintain primary containment pressure less than 30 psig by venting the suppression chamber, **THEN VENT** as follows:

73. G2.03.13 1

Access is required to a Unit One plant area for inspection of a suspected steam leak. E&RC surveys indicate radiation levels in the area of 1100 Mrem/hr at 30 cm and 510 Rads/hr at one meter.

Which one of the following choices completes the statements below IAW 0E&RC-0040, Administrative Controls for High Radiation Areas, Locked High Radiation Areas, and Very High Radiation Areas?

This area meets the radiological posting criteria for a ___(1)___.

Entrance into this area must be approved by E&RC manager or designee, RP Supervisor, and ___(2)___.

A. (1) Very High Radiation Area
   (2) Unit One CRS

B. (1) Very High Radiation Area
   (2) Shift Manager

C. (1) Locked High Radiation Area
   (2) Unit One CRS

D. (1) Locked High Radiation Area
   (2) Shift Manager

Answer: B

K/A: G2.03.13 - Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
(CFR: 41.12 / 43.4 / 45.9 / 45.10)

RO/SRO Rating: 3.4/3.8

Objective: LOI-CLS-LP-201-F, Obj. 10 - Explain the requirement regarding control of High Radiation Areas per E&RC-0040.

Reference: None

Cog Level: High

Explanation: Locked high radiation area (LHRA) criteria is > 1000 mrem/hr at 30 cm but < 500 Rads/hr at one meter. Very high radiation area (VHRA) criteria is 1000 mrem/hr at 30 cm and > 500 Rads/hr at one meter. This question provides criteria for a VHRA which requires RP Supervisor, E&RC Manager and Shift Manager approval IAW 0E&RC-0040.

Distractor Analysis:

Choice A: Plausible because changing either mrem or rad would result in the area being a LHRA. Shift Manager not CRS approval is required for VHRA entry.

Choice B: Correct answer, see explanation

Choice C: Plausible because changing either mrem or rad would result in the area being a VHRA.

Choice D: Plausible because changing either mrem or rad would result in the area being a VHRA.

SRO Basis: N/A

0ERC-0040

LOCKED HIGH RADIATION AREA (LHRA)

9.2 High Radiation Area (HRA) With Dose Rates Exceeding 1,000 Mrem/Hour at 30 Centimeters From the Radiation Source or From Any Surface Penetrated by the Radiation, But Less Than 500 Rads/Hour at 1 Meter From the Radiation Source or From Any Surface Penetrated by the Radiation

2.2 LHRAs

2.2.1 Requirements for entering LHRAs with documented stable radiological conditions:

1. Appropriate RWP
2. Radiation Protection pre-job briefing (ref. HPS-NGGC-0019)
3. Knowledge of current work area radiological conditions
4. Electronic dosimeter with pre-set dose/dose rate alarms
5. Consider use of the telemetry system.
6. Qualified HP Technician with dose rate monitoring instrument to provide intermittent coverage during evolutions with potential to change radiological conditions (e.g., system breaches, draining process piping, start up of systems using main steam, during HWC injection increases, etc.)
7. Positive control of LHRA entrance to prevent inadvertent entry

VERY HIGH RADIATION AREA

9.3 High Radiation Areas with Dose Rates Exceeding 1,000 Mrem/Hour at 30 Centimeters From the Radiation Source or From Any Surface Penetrated by the Radiation and Greater Than 500 Rads/Hour At 1 Meter From the Radiation Source or From Any Surface Penetrated by the Radiation

9.9 Access to Very High Radiation Areas

9.9.1 **WHEN** a key to a VHRA is needed, **COMPLETE** applicable portions of Attachment 1, Part 1.

9.9.2 **OBTAIN** the signature of the RP Supervisor and the Manager - E&RC or designee.

9.9.3 **NOTIFY** the affected unit's Operations Shift Manager for approval to enter into a VHRA.

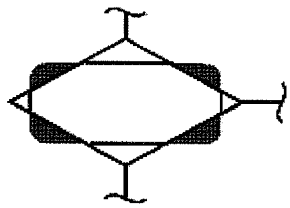
9.9.4 **WHEN** an entry into a VHRA is determined to be completed, **ENSURE** that the entrance is closed and locked, **AND COMPLETE** applicable portions of Attachment 1, Part 2, for First Physical Verification Signature.

Approvals: (Required for Very High Radiation Area)

______________________ / __________ ______________________ / __________
RP Supervisor (Signature) Date Manager - E&RC or Designee (Signature)/Date

Operations Shift Manager Notified ______________________ / __________
Shift Manager's Name (Print) Date

74. G2.04.19 1



Which one of the following identifies the EOP flowchart symbol above IAW 0EOP-01-UG, EOP Users Guide?

A. Action Step

B. Caution Step

C. Critical Step

D. Decision Step

Answer: C

K/A: G2.04.19 - Knowledge of EOP layout, symbols, and icons. (CFR: 41.10 / 45.13)

RO/SRO Rating: 3.4/4.1

Objective: LOI-CLS-LP-300-B, Obj. 2d - Given the following Emergency Operating Procedure (EOP) flowchart symbols, correctly identify each: Critical Step

Reference: None

Cog Level: Low

Explanation: As listed in the Users Guide this is a critical step.

Distractor Analysis:

Choice A: Plausible because an Action step is listed in the Users Guide

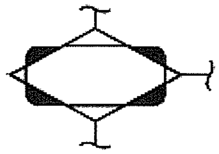
Choice B: Plausible because a Caution Step is listed in the Users Guide

Choice C: Correct Answer, see explanation.

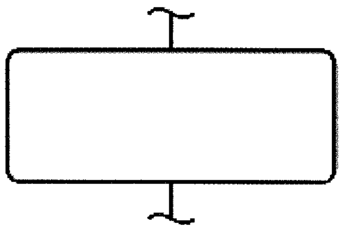
Choice D: Plausible because a Decision Step is listed in the Users Guide

SRO Basis: N/A

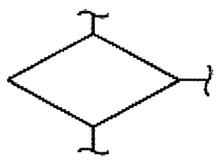
3.2.3 Critical Step



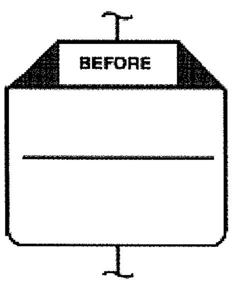
3.2.1 Action Step



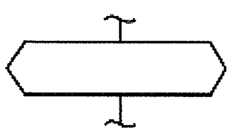
3.2.2 Decision Step



3.2.4 Before Step



3.2.5 Caution Step



75. G2.04.25 1

Which one of the following choices completes the following statements IAW 0FPP-031, Fire Brigade Staffing Roster and Equipment Requirements?

The Fire Brigade Advisor position is filled by ___(1)___.

The Fire Brigade Advisor ___(2)___ considered available to support activities other than the Fire Brigade during 0ASSD-02 implementation.

A. (1) an SRO ONLY
   (2) is

B. (1) any licensed operator
   (2) is

C. (1) an SRO ONLY
   (2) is NOT

D. (1) any licensed operator
   (2) is NOT

Answer: B

K/A: G2.04.25 - Knowledge of fire protection procedures. (CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 3.3/3.7

Objective: CLS-LP-013, Obj. 4d - Describe the requirements of the General Fire Plan (PFP-013) as they relate to the following: Fire Brigade Advisor - FBA.

CLS-LP-013, Obj. 9 - Describe the responsibilities of qualified fire brigade members.

Reference: None

Cog Level: Low

Explanation: 0FPP-031 defines who may fill the Fire Brigade Advisor position and the availability of that position for other tasks. Minimum fire brigade member staffing is specified in the procedure. The shift incident commander and individual fire brigade members, with the exception of the fire brigade advisor, may not have any other fire duties (i.e. ASSD).

Distractor Analysis:

Choice A: Plausible because there are SRO only positions during emergencies such as Remote Shutdown Panel operator in ASSD procedures.

Choice B: Correct Answer, see explanation

Choice C: Plausible because there are SRO only positions during emergencies such as Remote Shutdown Panel operator in ASSD procedures. The shift incident commander and individual fire brigade members, with the exception of the fire brigade advisor, may not have any other fire duties (i.e. ASSD).

Choice D: Plausible because The shift incident commander and individual fire brigade members, with the exception of the fire brigade advisor, may not have any other fire duties (i.e. ASSD).

SRO Basis: N/A

0FPP-031

3.0 RESPONSIBILITIES

3.5 The Fire Brigade Advisor is responsible for evaluating the operational impact on plant systems and equipment during a fire or other event requiring a response by the Shift Fire Brigade. The Fire Brigade Advisor position is filled by a licensed operator (i.e., RO or SRO). The Fire Brigade Advisor is considered available to support activities other than the Fire Brigade during an ASSD event if required for 0ASSD-02 or 2ASSD-05, or to ensure control room minimum staffing requirements (1 SRO and 1 RO per unit) are maintained for any unit that is not evacuating its control room.

ATTACHMENT 1
Page 1 of 1
Incident Command Fire Brigade Roster

Circle one: Dayshift / Nightshift

POSITION	NAME	INITIALS
Shift Incident Commander		
Fire Brigade Member		
Fire Brigade Member		
Fire Brigade Member		
Fire Brigade Member		
Fire Brigade Advisor*		
Maintenance Contact		N/A
Security Contact		N/A
E&RC Contact		N/A

* The Fire Brigade Advisor is considered available to support activities other than the Fire Brigade during an ASSD event if required for 0ASSD-02 or 2ASSD-05, or to ensure control room minimum staffing requirements (1 SRO and 1 RO per unit) are maintained for any unit that is not evacuating its control room.

0AP-033

c. The Shift Fire Brigade shall not include members of the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency.