≥ 25% RTP

QUESTION 1

Which ONE of the following completes the statements?

Minimum Critical Power Ratio

A single recirculation pump trip from rated power will cause the value of Critical Power (CP) to ___(1)___.

Thermal Limits ___(2)___ required to be adjusted in accordance with Technical Specification ~~3.4.1,~~ Reactor Coolant System (RCS), for continued power operation.
(Assume operation greater than 24 hours)

COLR limits for – must be applied.

A. (1) lower
   (2) are

B. (1) rise
   (2) are

C. (1) rise
   (2) are NOT

D. (1) lower
   (2) are NOT

3.2.1 APLHGR —
3.2.2. MCPR —
3.2.3 LHGR

Answer: **A**

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

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

Explanation: **A** CORRECT – Critical Power is equal to the bundle power at which the Onset of Transition Boiling (OTB) occurs. As actual power goes down, void fraction increases. This causes critical power to lower, although not as significantly as actual power. Since actual power drops further than critical power, the Critical Power Ratio gets larger. CPR= CP/AP

B Incorrect –First Part: Incorrect. Plausible if examinee has a misconception of Critical Power and the effects of lower core flow on Critical Power. Second Part: Correct.

C Incorrect –First Part: Incorrect. Plausible if examinee has a misconception of Critical Power and the effects of lower core flow on Critical Power. Second Part: Incorrect. Plausible if the examinee is not aware of the requirements of TS 3.4.1 or is not familiar with the time requirement for the applicable action statement.

D Incorrect – First Part: Correct. Second Part: Incorrect. Plausible if the examinee is not aware of the requirements of TS 3.4.1 or is not familiar with the time requirement for the applicable action statement.

Technical Reference(s): TS 3.4.1, Reactor Coolant System (RCS), COLR Unit 2 C17

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New
Question History:	Previous NRC: Brunswick NRC 2010 #37
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	24 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> No recirculation loops in operation.	B.1	Be in MODE 3.	12 hours

Brunswick NRC 2010 #37

37. 295001 K1.03 002

Which one of the following completes the statement below?

A single recirculation pump trip from rated power will cause the value of Critical Power to __(1)__ and Thermal Limits __(2)__ required to be adjusted IAW Technical Specification 3.4.1, Reactor Coolant System (RCS), for continued power operation. (Assume operation greater than 24 hours)

A. (1) rise
(2) are

B. (1) rise
(2) are not

C✓ (1) lower
(2) are

D. (1) lower
(2) are not

QUESTION 2

A complete loss of off site power has occurred.

Unit 2 RCIC is operating and HPCI is inoperable. Reactor pressure is being controlled 800 to 1000 psig by one fully open SRV and manually cycling a second SRV.

Based on these conditions, which ONE of the following completes the statement?

Reactor Power is ___(1)____ five percent (5%) power and Reactor Water Level is ___(2)___.

A. (1) below (2) rising

B. (1) below (2) lowering

C. (1) above (2) rising

D. (1) above (2) lowering

Answer: **D**

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

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

Explanation: D CORRECT – Each SRV is approximately 900000 lbm/hr or about 6% power. With 1+ SRVs open, power is greater than 5% and RCIC supplies about 300000 lbm/hr so this is less than the inventory lost through the SRVs.

A Incorrect –Plausible if the candidate believes that 2 or more SRVs are needed for 5% power and RCIC can supply enough water.

B Incorrect – Plausible if the candidate believes that 2 or more SRVs are needed for 5% power.

C Incorrect - Plausible if the candidate believes that RCIC can supply enough water.

Technical Reference(s): OPL171.009 & OPL171.040

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.040 V.B.1

Question Source:	Bank:	
	Modified Bank:	
	New:	X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

(6) The worst over pressure transient is:

(a) 3-second closure of all MSIVs neglecting the direct scram (valve position scram).

(b) Results in a maximum vessel pressure which, if a neutron flux scram is assumed and 12 valves are operable, results in adequate margin to the code allowable over pressure limit of 1375 psig bottom head pressure.

(7) To meet operational design, the analysis of the plant isolation transient (generator load reject without bypass valves) shows that 12 of the 13 valves limit peak pressure to a value well below the limit of 1375 psig.

b. The total safety / relief valve capacity has been established to meet the over pressure protection criteria of the ASME code.

(1) There are 13 Safety / Relief valves.

(a) Each SRV has a capacity of 905,000lb/hr @ 1135psig. This gives a total capacity ~ 84.1% (79.5% EPU) design steam flow at the reference pressure.

(b) Valve leakage is detected by a temperature element and an acoustic monitor on each tailpipe. However, only the acoustic monitor will generate an alarm on panel 9-3. Obj. V.B.6 Obj. V.C.4

c. Pump

TP-1 & TP-4

(1) Will deliver 600 gpm (rated flow) at speeds 2130 to 4500 rpm

Obj.V.B.1.
Remember SER 3-05

(2) Bearings submerged in oil

Discuss thumb rule 500 X Indicated GPM ≈ XX lbm/hr

(3) Located low in Reactor Building (Elev. 519) for adequate NPSH when taking suction from suppression pool.

Discuss level control strategies when using RCIC

600gpm X 500 ≈ 300,000 lbm/hr

QUESTION 3

The 250V RMOV Board 2A Normal Supply breaker tripped. Operators have re-energized the RMOV on the Alternate Supply.

Which ONE of the following completes the statement?

In accordance with 0-OI-57D, DC Electrical System, NO Motor Operated Valve (MOV) powered from 250V RMOV Board 2A may be operated ________.

A. except as required to perform testing or mitigate accident conditions

B. until the load restrictions on the Alternate Supply are verified

C. until the board is placed back on the Normal Supply

D. except as required to mitigate accident conditions or comply with Technical Specifications

Answer: **D**

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

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D C POWER:
AA1.01 D.C electrical distribution systems

Explanation: **D** CORRECT – No DC MOV may be operated except as required to mitigate accident conditions, to obtain safe shutdown or to comply with Technical Specifications

A Incorrect – Testing is NOT allowed. Plausible if the candidate believes that testing is an acceptable operation while on the alternate supply.

B Incorrect – This is plausible because 0-OI-57D has precautions and limitations which address load restrictions for the alternate battery while the board is on the alternate supply.

C Incorrect – Plausible if the candidate believes there are NO MOV operations allowed while on the alternate supply.

Technical Reference(s): 0-OI-57D

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous: None
Question Cognitive Level:	Memory or Fundamental Knowledge : X Comprehension or Analysis
10 CFR Part 55 Content:	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

3.0 PRECAUTIONS AND LIMITATIONS

A. In the event a Unit Battery System is removed from service or a 250VDC RMOV Board is transferred to the alternate supply, one or more of the limitations below may apply. If time permits, a Caution Order should be placed on the affected MOV hand switches prior to transfer of board to alternate to prevent violation of these safe shutdown restrictions.

1. In the event any 250VDC RMOV Board is on its alternate supply, the following restrictions apply to DC motor operated valves that are supplied from a battery that is feeding any RMOV board alternate supply:

   a. No DC MOV may be operated except as required to mitigate accident conditions, to obtain safe shutdown or to comply with Technical Specifications (i.e. to comply with LCO ACTIONS statements only).

   b. Testing (including SI/SRs) that requires DC motor operated valve operation is NOT allowed. [Ref. Dwgs. 1-45E701-3, 2-45E702-4, 3-45E703-3]

DC MOVs that may NOT be operated except as required to mitigate accident conditions or to obtain safe shutdown or to comply with Technical Specifications (i.e. to comply with LCO ACTIONS statements only) with RMOV boards on alternate supply.

RMOV BOARD ON ALTERNATE	NORMAL SUPPLY BATTERY	ALTERNATE SUPPLY BATTERY	MAY NOT OPERATE MOVs SUPPLIED FROM RMOV BD (i.e. supplied from the alternate battery)
1A	1	2	1C, 2A, 3C, 1A
1B	3	1	1A, 2C, 3B, 1B
1C	2	1	1A, 2C, 3B, 1C
2A	2	3	1B, 2B, 3A, 2A
2B	3	1	1A, 2C, 3B, 2B
2C	1	2	1C, 2A, 3C, 2C
3A	3	2	1C, 2A, 3C, 3A
3B	1	3	1B, 2B, 3A, 3B
3C	2	3	1B, 2B, 3A, 3C

3.0 PRECAUTIONS AND LIMITATIONS (continued)

M. The 250 V DC RMOV boards have alternate power supplies from another 250 V Unit DC board. For a unit in MODES 1, 2, or 3, the boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

1. The alternate battery that has been loaded due to the transfer may be considered operable if the controlled drawing restrictions as referenced in P&L 3.0AA are met.

2. Transfer of individual loads required by the Technical Specifications on the Unit Batteries such as the RPT Logic should be considered inoperable if divisional separation cannot be proven. If transfer of such loads is performed solely due to an inoperable distribution board or source, then Technical Specification LCO 3.0.6 can apply to the loads, however, a distribution LCO must be entered.

3. For a unit in MODE 4 or 5, the DC boards can be placed on their alternate feeder breakers and considered OPERABLE as long as the restrictions on the associated drawings are met.

N. A 250V DC unit battery charger should NOT be considered operable if its safety related supply is NOT available. If normal power (safety related supply) is available but the charger is on its alternate supply it is still considered operable.

O. When a 250V RMOV board is transferred to the alternate supply (except for 2B 250V DC RMOV Bd), both divisions (I and II) will be supplied from the same source.

P. Testing the Alternate Feeder breakers on the 250V Turbine Distribution boards can only be performed when transferring the Board or when the board is de-energized. Due to the wiring scheme, the breakers cannot be tested in the board. Also, due to the loads on the board, it is NOT desirable to transfer the board with the Unit on-line. This only applies to the 250V Turbine Distribution Boards.
[PER 03-017377-000]

Q. Battery Boards should be unloaded before removing Battery or Battery Charger from service, unless the evolution is of short duration (i.e. transferring battery chargers) or plant conditions warrant otherwise.

R. A critical voltage for any cell is 2.13 volts. Prolonged operation of a cell below 2.13 volts will reduce its life expectancy. However it is NOT unusual for a replacement cell to measure 2.07 volts (on float charge) and to slowly rise in voltage over a 3 month period to normal float voltage ranges.

QUESTION 4

A Unit 1 startup was in progress when the A RPS bus was lost ~~due to a trip of the A RPS MG~~.

Current conditions are

- Reactor power: 29%
- Main Generator load: 295MWe
- Main Turbine 1st Stage Pressure: 145 psig
- Recirculation EOC/RPT is in service
- Feedwater Level Control system is Three Element Control
- RPS bus A is deenergized

Assuming NO operator actions are taken, which ONE of the following describes the IMMEDIATE effect if the Main Turbine were to trip?

A. BOTH Reactor Recirculation Pumps would trip

B. ONLY the "A" Reactor Recirculation Pump would trip

C. BOTH Reactor Recirculation Pumps would run back due to the 28% limiter

D. The Reactor Recirculation Pumps would continue to operate at the current speed

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295005 AA1.01	
	Importance Rating	3.1	

Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP:
AA1.01 Recirculation system

Explanation: **D** CORRECT – The EOC RPT has been defeated and power is low enough (base on first stage pressure) that it should not occur. There would ~~be no change in~~ reactor power, it would just be directed through the bypass valves and feedwater flow would not change thus no runback would be triggered.

A Incorrect –Power is low enough that EOC RPT would not occur.

B Incorrect – Loss of RPS A would affect EOC RPT Logic power for Recirc pump A.

C Incorrect – There would be no change in reactor power, it would just be directed through the bypass valves and feedwater flow would not change thus no runback would be triggered.

Technical Reference(s): 2-OI-68

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Hope Creek 2010 #50

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

TURB CV FAST CLOSURE TURB SV CLOSURE SCRAM/RPT TRIP LOGIC BYPASS
16

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Sensor/Trip Point:

Relays

5A-K9A	1-PIS-001-0091B	Turbine first stage pressure ≤ 147 psig.
5A-K9C	1-PIS-001-0081B	
5A-K9B	1-PIS-001-0091A	
5A-K9D	1-PIS-001-0081A	

Sensor Location: Elevation 586 1-PIS-001-0081A(B) on 1-LPNL-925-0111 j-T2
1-PIS-001-0091A(B) on 1-LPNL-925-0110 h-T5

Probable Cause:
A. Power is less than 30% (first stage turbine pressure is ≤ 147 psig).
B. SI (or SR) in progress.
C. Sensor malfunction.

Automatic Action:
A. The control valve fast closure scram and the turbine stop valve scram is bypassed.
B. RPT Trip is bypassed.

Operator Action:
A. **CHECK** first stage pressure with 1-PI-1-79 on Panel 1-9-7. ☐
B. **IF** pressure is above 147 psig, **THEN**
REFER TO Tech Spec Table 3.3.1.1-1, Section 3.3.4.1, TRM Section 3.3.1. ☐
C. **IF** alarm remains sealed in with reactor power greater than 30%, **THEN**

CONTACT Reactor Engineering to determine if it is necessary to apply a thermal limit penalty due to Power Load Unbalance trip being out of service.(PLOOS).

References: 1-45E620-6-2 1-730E915-9 and 10 1-47E610-1-2

3.3 Tech Specs

A. Both Recirculation Pumps Out of Service, REFER TO Tech Spec 3.4.1.

B. [NRC/C] Core Thermal-Hydraulic Stability.

1. The reactor is to be verified outside Regions I, II & III.

2. When OPRMS are INOP, REFER TO 3-SR-3.3.1.1.I. [NCO 940245010]

C. [NRC/C] Single Loop Operation: Per Technical Specifications, the reactor can be operated indefinitely with one Recirc loop out of service, provided the requirements of T.S. 3.4.1 are implemented within 24 hours of entering single loop operations.

D. TS BASES SR 3.5.1.5, If RECIRC PUMP 3A(3B) DISCHARGE VALVE 3-FCV-068-0003(0079) is declared inoperable while the valve is OPEN, the associated LPCI subsystem is to be declared INOPERABLE.

E. TS SR 3.4.2.1, 3-SR-3.4.2.1, Jet Pump Mismatch and Operability, is required to be performed 24 hours after reaching > 25% RTP and/or within 4 hours after returning an idle Recirc Pump to service.

3.4 Recirc Pump Controllers

A. For the purpose of personnel safety or prevent equipment damage, the EMERGENCY STOP pushbutton can be depressed any time it is needed.

B. [II/C] When initiating manual runbacks, the appropriate manual push-button is to be depressed until the backlight is blinking, then the push-button can be released.[PER 08 013657 000]

C. [NER/C] Raising Recirc flow or reactor power should ONLY be allowed when steam dome saturation temp and bottom head drain temperatures are within 145°F of each other. [GE SIL 251 and 430]

D. Recirc Pump controller limits are as follows:

1. When any individual RFP flow is less than 19% and reactor water level is below 27 inches, or if a reactor scram occurs, speed limit is set to 75%(~1130 RPM speed) and if speed is greater than 75%(~1130 RPM speed), Recirc speed will run back to 75%(~1130 RPM speed).

2. When total feed water flow is less than 19% (15 sec TD) or Recirc Pump discharge valve is less than 90% open, speed limit is set to 28% (~480 RPM speed) and if speed is greater than 28%(~480 RPM speed), Recirc speed will run back to 28%(~480 RPM speed).

Illustration 4
(Page 4 of 4)

Recirc System Trips/Interlocks

E. RPT Breakers Open (Trips the feeder breakers)

1. EOC-RPT

   a. Turbine Stop Valve Closure, Power > 30% by Turbine 1st Stage Pressure

   b. Turbine Control Valve Fast Closure, Power > 30% by Turbine 1st Stage

2. ATWS-RPT

   a. RPV Pressure > 1148 psig (2 of 2 pressure switches in Logic A or B)

   b. RPV Water Level < -45 inches (2 of 2 level switches in Logic A or B)

3. Local Trip

F. Feeder Breaker Shut and No Voltage 0n Recirc Board

Hope Creek 2010 Question: RO#50

A plant startup was in progress when the 'A' RPS bus was lost due to the inadvertent tripping of the 'A' RPS MG output breaker.

Current conditions are:

- Reactor power: 23%
- Main Generator load: 260 MWe
- Main Turbine 1st Stage Pressure 80 psig
- DFCS is in Single Element Control on the Master Level Controller
- 'A' RPS bus de-energized

Assuming NO operator actions are taken, what would be the IMMEDIATE effect if the Main Turbine were to trip under these conditions?

A. Trip of both Reactor Recirculation Pumps.

B. ONLY an Intermediate runback of both Reactor Recirculation Pumps.

C. Trip of the 'A' Recirc pump ONLY.

D. ONLY a Full runback of both Reactor Recirculation Pumps.

Proposed Answer: A

QUESTION 5

Which ONE of the following completes the statement related to the Core Operating Limits Report (COLR)?

In MODE 2, the basis for the Shutdown Margin (SDM) requirement is to ensure that ______________________.

A. the assumptions for the analyzed Control Rod Drop Accident are met

B. the assumptions for the analysis of inadvertent Group 1 Isolation are met

C. an automatic scram will shutdown the reactor before fuel damage occurs in the event of an inadvertent criticality

D. an automatic scram will shutdown the reactor before fuel damage occurs in the event of a reactor period of less than 30 seconds

ANSWER: **A**

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

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

Explanation: **A** CORRECT – Technical Specification 3.1.1, Shutdown Margin (SDM) states that the SDM will be within the limits of the COLR. The bases for the limit is stated in the Tech Spec bases as a Control Drop Accident.

B Incorrect – Plausible if the candidate confuses the analysis for SDM with the analysis of abnormal operating transient listed in the FSAR, in this case the closure of all MSIVs.

C Incorrect – Plausible if the candidate confuses the analysis for SDM with the analysis of a Recirculation Flow Controller Failure - Increasing Flow.

D Incorrect Plausible if the candidate confuses the analysis for SDM with the analysis of a Continuous Rod Withdrawal During Reactor Startup.

Technical Reference(s): Unit 3 COLR, TS 3.1.1 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

EDMS: L32 120425 800

TVA NPG

Nuclear Fuel Engineering - BWRFE
1101 Market Street, Chattanooga TN 37402

Date: April 24, 2012

7 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2, 9, and 10) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more

(continued)

14.10.6.1 Recirculation Flow Controller Failure - Increasing Flow

Several possibilities exist for an unplanned increase in core coolant flow resulting from a recirculation flow control system malfunction. Failure of the master controller can result in a speed increase for both recirculation pumps. On Unit 1, the maximum output signal of the master controller is provided with rate limits which are adjusted in such a way that a master controller failure is less severe than a failure of one of the MG set speed controllers. The most severe case of increasing coolant

flow results when the MG set fluid coupler for one recirculation pump attempts to achieve full speed at maximum acceleration. The maximum acceleration for this failure is 25 percent of full speed per second. The most severe transient results when reactor power is initially at 68 percent of rated, which is at the lower end of the automatic flow control range. These conditions correspond to the lowest power and flow conditions on the automatic flow control characteristic curve for the reactor.

Figure 14.10-17 shows typical results of the transient. The changes in nuclear system pressure are not significant with regard to overpressure. The pressure decreases over most of the transient. The rapid increase in core coolant flow causes an increase in neutron flux, which initiates a reactor scram. The transient fuel surface heat flux reaches 83 percent of rated heat flux, but it barely exceeds the steady state power-flow control curve. No fuel damage occurs.

14.10.3.2 <u>Continuous Rod Withdrawal During Reactor Startup</u>

Control rod withdrawal errors are considered when the reactor is at power levels below the power range. The most severe case occurs when the reactor is just critical at room temperature and an out-of-sequence rod is continuously withdrawn. The rod worth minimizer would normally prevent withdrawal of such a rod. It is assumed that the Intermediate Range Neutron Monitoring (IRM) channels are in the worst conditions of allowed bypass. The scaling arrangement of the IRMs is such that for unbypassed IRM channels a scram signal is generated before the detected neutron flux has increased by more than a factor of ten. In addition a high neutron flux scram is generated by the APRMs at 15 percent and at 120 percent of rated power.

The analysis was performed for a 2.5 percent Δk control rod withdrawal at the rod drive speed of 3 in./sec starting from an average moderator temperature of 82°F.

The results of these analyses indicate a maximum fuel temperature well below the melting point of UO_2 and a maximum fuel clad temperature which is less than the normal operating temperature of the clad. The possible failure of the fuel clad due to strain was analyzed using the following conservative assumptions:

1. The total volume expansion of UO_2 is in the radial direction,
2. There is no thermal expansion of the fuel cladding, and
3. The fuel is assumed to be incompressible.

The results of these analyses indicate a maximum radial strain analogous to a radial expansion of 0.6 mils, which is much less than the postulated cladding damage limit of approximately 1 percent plastic strain, which corresponds to approximately 3.3 mils radial expansion.

Thus, no fuel damage will occur due to a continuous rod withdrawal during reactor startup.

QUESTION 6

Unit 2 was operating at 100% power when the MSIVs closed on high steam flow and all control rods inserted.

The Unit 2 Control Room has been abandoned and the operators are carrying out the actions of Control Room Abandonment, 2-AOI-100-2, at Panel 2-25-32.

Which ONE of the following completes the statement?

If the TRANSFER SWITCH, 2-XS-1-14, is placed in EMERG with 2-HS-1-14C handswitch on panel 2-25-32 positioned as shown above, the A INBOARD MAIN STEAM LINE ISOLATION VALVE, 2-FCV-1-14, will ______.

A. remain CLOSED since a PCIS isolation is present

B. OPEN, since ALL PCIS isolations are bypassed in emergency

C. OPEN, then reclose since high steam flow will immediately reoccur

D. remain CLOSED since the control room switch 2-HS-1-14 on Panel 9-3 is in the Close position

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295016 G2.1.32	
	Importance Rating	3.8	

295016 Control Room Abandonment
Ability to explain and apply system precautions and limitations

Explanation: **B** CORRECT – As noted in the caution in 2-AOI-100-2, the control switch on 2-25-32 must be in the desired position prior to placing the transfer switch in EMERG. Otherwise the valve will go to the position the handswitch on the Backup Control Panel is positioned.

A Incorrect – Plausible if the candidate doesn't know that the PCIS isolations are bypassed when the switch is placed in EMERG.

C Incorrect –Plausible that the valve will OPEN but with the PCIS isolations bypassed, and no flow through the outboard valve, the inboard valve will not see high steam flow and not reclose.

D Incorrect –Plausible if the candidate doesn't know that the function of the EMERG position is to bypass the control room switch.

Technical Reference(s): 2-AOI-100-2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

4.2 **Unit 2 Subsequent Actions (continued)**

> **CAUTION**
>
> Failure to place control switch in desired position prior to transferring to emergency position may result in inadvertent actuation of the component.

[6] **CLOSE** MSIVs using the following switch sequence at Panel 2-25-32:

[6.1] **PLACE** control switch in CLOSE. ☐

[6.2] **PLACE** transfer switch in EMERG. ☐

MSIV LINE	Control Switch	Required Position		Transfer Switch	Required Position	
A INBOARD	2-HS-1-14C	CLOSE	☐	2-XS-1-14	EMERG	☐
B INBOARD	2-HS-1-26C	CLOSE	☐	2-XS-1-26	EMERG	☐
C INBOARD	2-HS-1-37C	CLOSE	☐	2-XS-1-37	EMERG	☐
D INBOARD	2-HS-1-51C	CLOSE	☐	2-XS-1-51	EMERG	☐
A OUTBOARD	2-HS-1-15C	CLOSE	☐	2-XS-1-15	EMERG	☐
B OUTBOARD	2-HS-1-27C	CLOSE	☐	2-XS-1-27	EMERG	☐
C OUTBOARD	2-HS-1-38C	CLOSE	☐	2-XS-1-38	EMERG	☐
D OUTBOARD	2-HS-1-52C	CLOSE	☐	2-XS-1-52	EMERG	☐

QUESTION 7

Unit 1 is operating at 100% Reactor Power, when RBCCW Pump 1A trips resulting in the following:

- RBCCW Pump discharge header pressure is 48 psig
- RBCCW PUMP DISCH HDR PRESS LOW, (1-9-4C, window 12), in alarm

Which ONE of the following system loads is isolated from RBCCW cooling?

A. Drywell Blowers

B. Drywell equipment drain sump heat exchanger

C. Reactor Recirculation Pump seal coolers

D. Reactor Building equipment drain sump heat exchanger

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295018 AA1.02	
	Importance Rating	3.3	

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

Explanation: D CORRECT – Reactor Building Equipment Drain Sump Heat Exchanger is on the Non-essential loop.

A Incorrect – Drywell blowers are a load on the essential loop of RBCCW. Plausible if candidate does not know the loads on the RBCCW essential and non-essential loops.

B Incorrect – Drywell equipment drain sump heat exchanger is a load on the essential loop of RBCCW. Plausible if candidate does not know the loads on the RBCCW essential and non-essential loops.

C Incorrect – Reactor Recirculation Pump seal coolers are loads on the essential loop of RBCCW. Plausible if candidate does not know the loads on the RBCCW essential and non-essential loops.

Technical Reference(s): 1-AOI-70, OPL171.047

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (4) Secondary coolant and auxiliary systems that affect the facility.

1.0 PURPOSE

This instruction provides symptoms, automatic actions, and operator actions for a partial or complete loss of RBCCW System.

2.0 SYMPTOMS

A. Annunciator in alarm:

1. RBCCW PUMP DISCH HDR PRESS LOW (1-XA-55-4C, Window 12)
2. RBCCW PUMP SUCT HDR TEMP HIGH (1-XA-55-4C, Window 5)
3. PNL 9-47 RHRSW TEMP ABNORMAL (1-XA-55-4C, Window 7)
4. RBCCW 1-FCV-70-48 CLOSED (1-XA-55-4C, Window 19)
5. RECIRC PUMP 1A COOLING WATER FLOW LOW (1-XA-55-4A, Window 34)
6. RECIRC PUMP 1B COOLING WATER FLOW LOW (1-XA-55-4B, Window 34)
7. RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (1-XA-55-4B, Window 17)
8. RWCU RECIRC PUMP CLG WATER TEMP HIGH (1-XA-55-4B, Window 9)
9. DRYWELL EQPT DR SUMP TEMP HIGH (1-XA-55-4C, Window 16)
10. DRYWELL TEMP HIGH (1-XA-55-3B, Window 16)
11. RBCCW SURGE TANK LEVEL LOW (1-XA-55-4C, Window 13)
12. DRYWELL PRESSURE ABNORMAL (1-XA-55-5B, Window 31)

3.0 AUTOMATIC ACTIONS

RBCCW SECTIONALIZING VLV, 1-FCV-70-48, closes automatically on RBCCW Pump discharge header pressure at or below 57 psig.

d. Proper system flow operation is assured by monitoring the system DP (pump discharge minus pump suction). — Done Each Shift

2. RBCCW Heat Loads

a. Essential loop loads — Obj. V.B.2, Obj. V.D.2

- Drywell Blowers(10)
- Reactor recirculation pump motor coolers (2)
- Reactor recirculation pump seal coolers (2)
- Drywell equipment drain sump heat exchanger (1)

b. Non-essential loop loads — Obj. V.B.3, Obj. V.D.3

- Reactor Building equipment drain sump heat exchanger (1)
- Reactor water cleanup pump seal water coolers and bearing oil coolers (2)
- RWCU Non-regenerative heat exchangers (2)
- Fuel pool cooling heat exchangers (2)
- Reactor recirculation pump discharge sample cooler (1)

QUESTION 8

All three Units are operating at 100% power.

The following conditions exist on the B Emergency Diesel Generator:

- The Left Bank Compressor breaker has just tripped due to overload
- The Left Bank Air Receiver pressure is 190 psig
- The Right Bank Compressor is running continuously
- The Right Bank Air Receiver pressure is 157 psig and dropping slowly

Which ONE of the following completes the statements?

In accordance with Technical Specifications, the B Emergency Diesel Generator ___(1)___.

___(2)___ Technical Specification(s) must be entered.

A. (1) is OPERABLE
(2) NO

B. (1) must be declared INOPERABLE IMMEDIATELY
(2) 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air, AND 3.8.1 AC Sources – Operating

C. (1) must be declared INOPERABLE IMMEDIATELY
(2) ONLY Technical Specification 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

D. (1) must be declared INOPERABLE WITHIN ONE HOUR
(2) 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air, AND 3.8.1 AC Sources – Operating

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295019 G2.2.39	
	Importance Rating	3.9	

Knowledge of less than or equal to one hour Technical Specification action statements for systems related to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR

Explanation: A CORRECT – One starting air system with at least 165 psig in the receiver is required to be operable for the diesel to be operable. With 165 psig in the receiver, one system is operable.

B Incorrect – Plausible if the diesel air system is believed to be inop, the DG must be immediately declared inoperable and the technical specification for an inoperable DG entered.

C Incorrect - Plausible if the candidate thinks that one should not "cascade" in tech specs after declaring the air system inop

D Incorrect -Plausible if the candidate believes that the operator has one hour to take action as in TS 3.0.3.

Technical Reference(s): TS 3.8.3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 X
10 CFR Part 55 Content:	55.41 (8) Components, capacity, and functions of emergency systems.	

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each DG.

Diesel Fuel Oil, Lube Oil, and Starting Air
3.8.3

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One or more DGs with the required starting air receiver unit pressure < 165 psig	D.1	Declare associated DG inoperable.	Immediately
E. Required Action and associated Completion Time not met. <u>OR</u> One or more DGs with diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than Condition A, B, C or D.	E.1	Declare associated DG inoperable.	Immediately

BASES

ACTIONS

C.1 (continued)

of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, since particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and since proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, re-sampling, and re-analysis of the DG fuel oil.

D.1

Only one of the two redundant air starting systems is required to support associated DG operability. With the starting air receiver pressure < 165 psig in the required starting air system, sufficient capacity to start the associated DG may not exist. The associated DG may be incapable of performing its intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions for an inoperable DG, LCO 3.8.1, "AC Sources - Operating."

The

QUESTION 9

Unit 2 is shutdown for refueling and is currently in Mode 4.

- RHR Pump 2A is in shutdown cooling
- Reactor Recirculation Pump 2A is in service
- RPV Level is +60 inches
- RPV Water Temperature is 125° F

A loss of shutdown cooling occurs and shutdown cooling CANNOT be restored.

Which ONE of the following is required by 2-AOI-74-1, LOSS OF SHUTDOWN COOLING and the reason for the action?

A. Raise RPV to +80 inches and maintain level +70 to +90 inches to promote vessel circulation.

B. Raise RPV to +80 inches and maintain level +70 to +90 inches to remove decay heat.

C. Verify the Recirculation Pump is still in service to promote vessel circulation.

D. Verify the Recirculation Pump is still in service to remove decay heat.

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295021 AA2.07	
	Importance Rating	2.9	

Ability to determine and /or interpret the following as they apply to LOSS OF SHUTDOWN COOLING:
AA2.07 Reactor recirculation flow

Explanation: **C** CORRECT –This is directed by step 4.2 [19]; If no RHR pumps can be placed in service, THEN VERIFY a Recirculation Pump in Service.

A Incorrect. Plausible because this is directed by step [16]; IF forced circulation has been lost AND vessel cavity is less than 80 inches.

B Incorrect – Plausible because this is directed by step [16]; IF forced circulation has been lost AND vessel cavity is less than 80 inches. However, this is to promote circulation and does not remove decay heat.

D Incorrect – Plausible because this is the right action but for the wrong reason.

Technical Reference(s): 2-AOI-74-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC None	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 X
10 CFR Part 55 Content: procedures for the facility.	55.41 (10) Administrative, normal, abnormal, and emergency operating	

4.2 Subsequent Actions (continued)

CAUTION
Accurate coolant temperatures will **NOT** be available if all forced circulation is lost.

[16] [NER/C] **IF** forced circulation has been lost **AND** vessel cavity is less than 80 inches, **THEN** (Otherwise N/A)

PERFORM the following: ☐

[16.1] **RAISE** RPV water level to 80 inches as indicated on RX WTR LEVEL FLOOD-UP, 2-LI-3-55 ☐

[16.2] **MAINTAIN** RPV water level between +70 inches to +90 inches as indicated on RX WTR LEVEL FLOOD-UP, 2-LI-3-55. ☐

[16.3] **INCREASE** frequency of monitoring reactor coolant temperature and pressure, using multiple indications. ☐

[17] **IF** the affected loop of RHR cannot be placed back in Shutdown Cooling, **THEN** (Otherwise N/A)

PLACE the alternate loop of RHR in Shutdown Cooling. **REFER TO** 2-OI-74. ☐

[18] **IF** no Unit 2 RHR loop can be placed in Shutdown Cooling, **THEN** (Otherwise N/A)

OBTAIN Shift Manager approval and **PLACE** Unit 1 RHR loop in service, CROSS-TIED with Unit 2, for Shutdown Cooling. **REFER TO** 2-OI-74. ☐

[19] **IF** no RHR loops can be placed in service, **THEN** (Otherwise N/A)

VERIFY a Recirculation Pump in service. **REFER TO** 2-OI-68. ☐

[20] **IF** the Reactor is in a Cold Shutdown Condition (Mode 4 or Mode 5) **AND** the reactor vessel head studs are tensioned or head tensioning is in progress, **THEN** (Otherwise N/A)

PERFORM 2-SR-3.4.9.5-7, RPV Temperature Monitoring with Head Tensioned. ☐

QUESTION 10

As part of Unit 2 refueling, a Subcriticality Check is in progress in accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling, Attachment 7.

- 8 Shorting Links are attached to the Unit 2 Rx Mode Switch

As the analytically determined strongest control rod is being withdrawn, control rod withdrawal is STOPPED at position 40 and the following is observed:

- All 4 SRM Period Meters indicate POSITIVE, CONSTANT period
- Indicated count rate on all 4 SRM's is RISING

Which ONE of the following will mitigate this transient?

A. IRM High Rod Block

B. IRM High-High Scram

C. SRM High Rod Block

D. SRM High-High Scram

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295023 AK3.04	
	Importance Rating	3.0	

Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS:
AK3.04 Non-coincident scram function

Explanation: **D** CORRECT – This subcriticality test of the strongest worth control rod is performed in Non-coincident mode to demonstrate effective Shutdown Margin. Once demonstrated, the SRM are placed back in Coincident. Removing the SRM RPS shorting links in Attachment 2 will place them in Non-coincident scram mode. The shorting links will be placed on the MODE switch for visibility per attachment 2. The conditions in the stem indicate criticality and the SRM scram will be the first protective action to mitigate the transient and correct the inadvertent criticality.

A- Incorrect –The IRM rod block is plausible because an IRM rod block will be received and it will prevent the withdrawn rod from being further withdrawn. It will not correct the inadvertent criticality.

B Incorrect – Plausible because an IRM scram would occur if the SRM scram function was disabled, shorting links INSTALLED.

C- Incorrect – The SRM rod block is plausible because an SRM rod block will be received and it will prevent the withdrawn rod from being further withdrawn. It will not correct the inadvertent criticality.

Technical Reference(s): OPL171.028, 0-GOI-100-3C Attachments 2 and 7

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New	 X
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis:	X

10 CFR Part 55 Content: 55.41 (6) Design, components, and functions of reactivity control mechanisms and instrumentation.

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
3) During certain plant evolutions (normally associated with refueling) these shorting links can be removed to place the SRMs in the coincidence (one out of two taken twice) SCRAM mode or the non-coincidence (only one detector must trip) SCRAM mode. The non-coincidence mode is established during fuel movement inside the core.	Objective 12i, 13a
4) Since SRMs are quadrant specific, an excessive rapid rise in neutron counts caused by fuel movement in a localized area would not be detected by more than one SRM. To ensure protection from this type of event, the RPS system is placed into a configuration such that a single SRM reaching its trip set point would SCRAM the reactor.	Fundamentals: Discuss how loosely coupled the core is at low neutron flux levels.
a) There are 8 shorting links total, colored green, yellow, red, and blue. These links are installed in the 9-15 and 9-17 panels in the auxiliary instrument room.	A local criticality event resulting from a fuel movement error is a significant concern.
b) If all of the green, yellow and red links are removed the SRMs will SCRAM the reactor if either A or C and B or D detectors reach the Hi-Hi trip point.	

Attachment 2
(Page 1 of 2)

SRM RPS Shorting Links

CAUTION

During installation and removal of the RPS shorting links, the possibility exists for a momentary circuit interruption resulting in a reactor half scram.

NOTES

1) Shorting links may be installed or removed in 1-, 2-, or 3-GOI-100-3A, -3B, or -3C.
2) When fuel is not in the vessel, RPS neutron monitoring shorting links may be reinstalled to prevent erroneous neutron monitoring related spikes or faults from causing ESF actuations.
3) Removing both green (SRM A, SRM B) AND both yellow (SRM C, SRM D) links along with both red links (6 total links) will allow the SRM Hi-Hi (2×10^5 cps) scram to operate in one-out-of-two, taken-twice logic (coincident logic).
4) Removing both green (SRM A, SRM B) AND both yellow (SRM C, SRM D) links along with both red AND blue links (8 total links) will allow the Neutron Monitoring System (SRM, IRM, and APRM) scrams to operate in non-coincident logic.

   If the blue shorting links (two total links) are removed, the Voters and IRMs are placed in non-coincident trip logic where any <u>one</u> channel, if tripped, will produce a full Reactor scram.
5) Links not removed or installed are N/A.
6) After removal of any shorting links, links will be placed on reactor mode switch for visibility.
7) The preferred method of installing or removing shorting links, to minimize ESF risks, are as follows:
   - Removing: FIRST - Greens/Yellows, SECOND - Reds/Blues
   - Installing: FIRST - Reds/Blues, SECOND - Yellows/Greens

Attachment 7
(Page 1 of 2)

Subcriticality Check

Date ________

NOTE

If this Attachment is being performed in conjunction with Section 5.5, Steps 1.0[1] and 1.0[2] may be N/A'd. The analytically determined strongest rod should be recorded in Step 1.0[3].

1.0 PERFORMING SUBCRITICALITY CHECK

[1] **OBTAIN** Reactor Engineering Supervisor concurrence of the necessity for performing a subcriticality check. ________

[2] **OBTAIN** Plant Manager permission prior to performing a subcriticality check. ________

[3] **CHOOSE** a control rod that is surrounded by fuel and in the vicinity of the cell to be loaded.

(Cell to be loaded should be marked N/A if all fuel is loaded).

Control Rod ___ - ___
XX YY

Cell to be Loaded ___ - ___
XX YY ________
RE

[4] **VERIFY** that the bundles located in a 3 X 3 control cell array containing the control rod to be withdrawn are loaded properly. The 3 X 3 array shall be centered on the specified control rod unless the control rod is located near the periphery of the fueled region. (This step is considered completed if core verification has been performed.) ________

[5] **REQUEST** the Unit Supervisor place the neutron monitoring system in coincident mode, non-coincident mode, or **VERIFY** the neutron monitoring system is in the coincident mode with all four SRMs or FLCs operable, and **VERIFY** fuel loading operations have halted and all personnel are off of the refuel floor roof and out of line of sight of the reactor core for the duration of this subcriticality check. ________

QUESTION 11

Unit 1 was operating at 100% power.

Drywell temperature and pressure began to slowly rise. Operators entered 1-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage Into Drywell.

When drywell pressure reached 2.0 psig, the operators inserted a manual scram.

After ONLY the immediate operator actions for the scram have been completed, the following conditions exist:

- Reactor water level is (-)10 inches with level slowly rising
- Reactor pressure is steady at 960 psig
- All control rods indicate green double dashes
- Drywell pressure is 2.3 psig and slowly rising
- Drywell temperature is 150°F and slowly rising

Which ONE of the following completes the statement?

Based on these conditions, the operating crew shall _______________ .

A. continue in 1-AOI-64-1, no EOI entry conditions have been met

B. enter EOI-1, RPV Control, and exit 1-AOI-64-1

C. enter EOI-1, RPV Control, and continue in 1-AOI-64-1

D. enter EOI-1, RPV Control, and EOI-2, Primary Containment Control, and continue in 1-AOI-64-1

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295024 G2.4.8	
	Importance Rating	3.8	

Knowledge of how abnormal operating procedures are used in conjunction with EOPs for HIGH DRYWELL PRESSURE
G2.4.8

Explanation: **C** CORRECT – EOP-1 is entered on RPV low level. Scram response has activated, based on the lowest level during the transient was -25 inches. There is no entry condition for EOI-2, so the AOI is the only procedure currently in use for controlling the drywell parameters.

A Incorrect –. Plausible because the candidate may not recognize that EOI-1 should be entered on low RPV level below +2 inches.

B Incorrect – Plausible because Drywell pressure is high, however it is not high enough to enter EOI-2 and there is no direction or reason to exit the AOI (as it is attempting to control the drywell pressure issue).

D Incorrect – Plausible if the candidate confuses the high drywell pressure entry condition. There is no EOI-2 entry condition currently.

Technical Reference(s): EOIPM Section 8 pg. 48

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New X

Question History: Previous NRC None

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

10 CFR Part 55 Content: 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility

I. EOI Flowchart Use With Other Plant Procedures

The EOIs are entered, based upon specific conditions symptomatic of emergencies, or conditions that could degrade into emergencies. Therefore the operator actions, provided within the EOIs, allow the operator to mitigate the consequences of a broad range of accidents and multiple equipment failures.

Other procedures, such as AOIs, ARPs, EPIPs, etc., have event specific entry conditions and may be used to supplement EOIs. In some instances the EOIs will direct the operators to the unit operating procedures (OIs, GOIs, and AOIs) for completion of specific tasks. Usually, the EOIs direct the operators to specific EOI Appendices. The Appendices are specific task related procedures written to satisfy directives given within the EOIs.

Actions that contradict any direction given by the EOIs, or reduce the effectiveness of any directions given by the EOIs, WILL NOT be implemented for any reason.

QUESTION 12

With Unit 3 Reactor operating at 100% power, a failure in the EHC System causes reactor pressure to slowly rise.

Which ONE of the following places the below list of events in the proper sequence as pressure rises?

1. Reactor scram on high pressure
2. Technical Specification Safety Limit exceeded
3. First Group of Main Safety Relief Valves (MSRVs) opens
4. Technical Specification LCO 3.4.10 limit exceeded

A. 2, 4, 1, 3

B. 4, 1, 2, 3

C. 4, 1, 3, 2

D. 2, 4, 3, 1

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295025 EA2.01	
	Importance Rating	4.3	

Ability to determine and/or interpret the following concepts as they apply to HIGH REACTOR PRESSURE
EA2.01 Reactor pressure

Explanation: C CORRECT – Event 4 occurs at 1058 psig, Event 1 at 1073 psig and Event 3 at 1135 psig, event 2 occurs at 1325 psig

A Incorrect – Plausible if the candidate confuses the Tech Spec Safety Limit 1325psig with the TS 3.4.10 Reactor Steam Dome Pressure Limit of 1050psig.

B Incorrect – Plausible if the candidate confuses the Tech Spec Safety Limit 1325psig with the SRV lift setpoints.

D Incorrect – Plausible if the candidate believes that the SRVs open before the scram based on no scram required for just an open SRV.

Technical Reference(s): TS 3.4.3, TS 3.3.1.1, TS 3.4.10, TS 2.1.2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC None	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (3) Mechanical components and design features of the reactor primary system.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1050 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1	Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift settings of the required 12 S/RVs are within ± 3% of the setpoint as follows: Number of S/RVs — Setpoint (psig) 4 — 1135 4 — 1145 5 — 1155 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

2.0 SAFETY LIMITS (SLs)

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 ≤ 25% RTP.

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

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.11 for single 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.

QUESTION 13

Following a Unit 2 Main Steam Isolation closure from full power, BOTH loops of RHR are in Suppression Pool Cooling mode with the following conditions:

- Reactor power: ~~0%~~ (all control rods fully inserted)
- RPV Level (-)50 inches, slowly raising to normal band with both loops of Core Spray
- RPV Pressure 50 psig
- Drywell Pressure 1.8 psig and steady
- Suppression Pool Level 11.4 feet and steady
- Suppression Pool Temp 140°F and lowering
- RHR A and C in ~~Suppression Pool Cooling~~ @ 11500 gpm,
- RHRSW A flow @ 4500 gpm
- RHRSW C flow @ 4000 gpm
- RHR B and D in ~~Suppression Pool Cooling~~ @ 13200 gpm,
- RHRSW B flow @ 4400 gpm
- RHRSW D flow @ 4400 gpm

[REFERENCE PROVIDED]

identifies

Based on these conditions, which ONE of the following describes the required operator action and the reason?

A. Balance Loop I RHRSW flows ~~to meet 2-OI-23 operating limits~~

B. Reduce Loop II RHR pumps flows ~~to avoid cavitation due to insufficient NPSH~~

C. Reduce Loop II RHR system flow because it has exceeded 2-EOI APPENDIX-17A operating limits

D. Raise Suppression Pool level to prevent RHR Pumps from air entrainment due to vortex limit concerns

Answer: **C**

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

Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE
EA1.01 Suppression pool cooling

Explanation: C CORRECT – Loop flow should be at or below 13000 gpm for two pump operation.

A- Incorrect – There are limits on RHRSW operation (1350 to 4500 GPM) but there is no requirement to balance flows.

B- Incorrect – Based on NPSH curves, there is sufficient NPSH, and the pump indications do not support cavitation. The suppression pool temperature is elevated but not enough to cause a loss of NPSH.

D- Incorrect – Plausible because 11.4 feet in the suppression pool is low, but not low enough for vortex concerns yet. The vortex limit is 10 feet.

Technical Reference(s): 2-EOI-Appendix 17A, EOI Curve 2 NPSH Limits

Proposed references to be provided to applicants during examination: EOI Curve 2 NPSH Limits

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New	 X
Question History:	Previous NRC: River Bend 2010 #13	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 X

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

2-EOI APPENDIX-17A

RHR SYSTEM OPERATION SUPPRESSSION POOL COOLING

LOCATION. Unit 2 Control Room

ATTACHMENTS: 1. NPSH Monitoring (√)

NOTE: Placing a BYPASS SEL switch in BYPASS in step 1 below prevents automatic opening of the affected RHR loop's outboard injection valve. This makes LPCI mode of that RHR loop inoperable.

1. IF.............Adequate core cooling is assured,
   OR
   Directed to cool the Suppression Pool irrespective of adequate core cooling,
   THEN **BYPASS** LPCI injection valve open interlock AS NECESSARY:

   - **PLACE** 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS. ____
   - **PLACE** 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in BYPASS. ____

2. **PLACE** RHR SYSTEM I(II) in Suppression Pool Cooling as follows:

   a. **VERIFY** at least one RHRSW pump supplying each EECW header. ____

   b. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s). ____

   c. **THROTTLE** the following in-service RHRSW outlet valves to obtain between 1350 and 4500 gpm RHRSW flow:

   - 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV ____
   - 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV ____
   - 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV ____
   - 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV. ____

   d. IF .... Directed by SRO,
   THEN....... **PLACE** 2-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD in MANUAL OVERIDE ____

2. (Continued)

CAUTION

RHR System flows below 7,000 gpm or above 10,000 gpm for one-pump operation may result in excessive vibration and equipment damage.

i. **THROTTLE OPEN** 1-FCV-74-59(73), RHR SYS I(II) SUPPR POOL CLG/TEST VLV, to maintain EITHER of the following as indicated on 1-FI-74-50(64), RHR SYS I(II) FLOW:

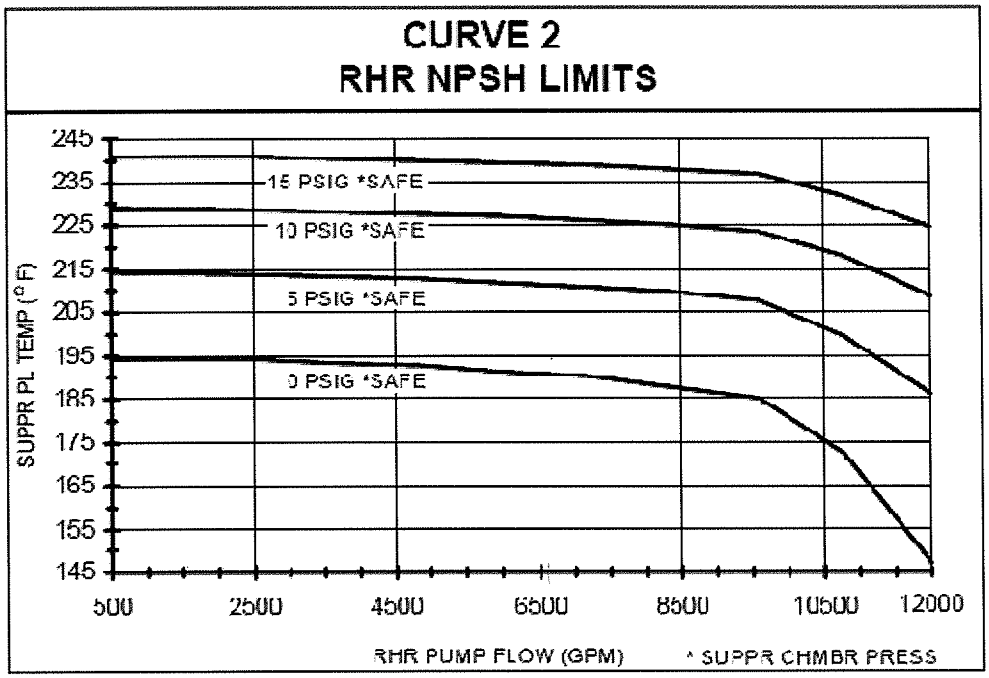
- Between 7,000 and 10,000 gpm for one-pump operation. _____

  OR

- At or below 13,000 gpm for two-pump operation. _____

NPSH MONITORING

Adequate NPSH is assured by maintaining pump flow rates below the curve for the applicable Suppression Chamber pressure. For Suppression Chamber pressures between the values on the curves extrapolation must be used



Other indications of inadequate NPSH are:

- Suppression pool level below 10.0 ft
- System flowrate decreasing with constant valve position
- System flowrate or discharge pressure less than expected for present system conditions
- Pump discharge pressure lower than expected or fluctuating excessively
- Pump motor amps lower than expected or fluctuating excessively
- Pump suction pressure low (local indication)

CAUTIONS

CAUTION #2

OPERATION OF RHR OR CS WITH SUCTION FROM THE SUPPR PL MAY RESULT IN EQUIPMENT DAMAGE IF:

- PUMP FLOW IS <u>ABOVE</u> THE NPSH LIMIT (CURVE 1 OR 2)

  OR

- SUPPR PL LVL IS <u>BELOW</u> THE VORTEX LIMIT (10 FT).

CAUTION #4

REDUCING PC PRESS WILL REDUCE THE AVAILABLE NPSH FOR PUMPS TAKING SUCTION FROM THE SUPPR PL

River Bend 2010 #13

QUESTION 13 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group # 1	
	K/A #	295026 EA1.01	IR 4.1	

Ability to operate and/or monitor suppression pool cooling as it applies to suppression pool high water temperature.

Proposed Question:

Following an ATWS, both loops of Residual Heat Removal are in the Suppression Pool Cooling mode with maximum flow through the heat exchangers.

- Reactor power <5%
- RPV Level -50 inches, slowly raising to normal band
- RPV Pressure 0 psig
- Suppression Pool Level 19 feet 11 inches
- Suppression Pool Temp 140°F
- RHR A in Sup Pool Cooling @ 5400 gpm, SWP flow @ 5500 gpm
- RHR B in Sup Pool Cooling @ 5600 gpm, SWP flow @ 5700 gpm

Both divisions of Standby Service Water are in service due to a loss of Normal Service Water.

Based on these conditions, which of the following should be of concern to the operator?

A. RHR B system flow has exceeded limits.

B. SWP flow has exceeded limits.

C. RHR pumps may experience air entrainment due to vortex limit concerns.

D. RHR pumps may experience cavitation due to NPSH concerns.

Proposed Answer: A.

Explanation:
A. Correct-RHR shell side flow limit is 5550 gpm.
B. SWP flow is less than 5800 gpm per loop limit.
C. Vortex limit concerns are at <10 feet SP Level
D. NPSH concerns begin at 160°F

Technical Reference(s): SOP-0031, EOP-0001

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-0511 Obj 6; RLP-STM-0204 Obj 8

Question Source: Modified Bank RBS 2008 NRC Exam #12Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3

10 CFR Part 55 Content: 55.41.b7

QUESTION 14

Unit 2 was operating at 100% power when a small steam leak caused the reactor to scram.

Currently, the following conditions exist:

- RPV Pressure is 900 psig and steady
- RPV Level is 25 inches and steady
- Drywell Pressure is 10 psig and slowly rising
- Suppression Chamber Pressure is 8.5 psig and slowly rising
- Containment venting is in progress
- Drywell Temperature is 165°F and slowly rising
- Suppression Pool Level is 15 feet
- Suppression Pool Temperature is 90°F

Which ONE of the following describes the NEXT action required by 2-EOI-2, PRIMARY CONTAINMENT CONTROL and the reason for that action?

A. Operate drywell sprays because the Drywell pressure is high

B. Operate all available drywell coolers because the Drywell temperature is high

C. EMERGENCY DEPRESSURIZE ~~the reactor~~ because the Suppression Chamber pressure is high

D. EMERGENCY DEPRESSURIZE ~~the reactor~~ because the Suppression Pool Level is low

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295028 EK3.04	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE
EK3.04 Increased drywell cooling

Explanation: B CORRECT – EOI-2 directs the use of all available DW cooling in step DW/T-3

A- Incorrect –Plausible because drywell pressure is high and spraying would lower temperature, however, drywell sprays are only directed after the Suppression Chamber Pressure is greater than 12 psig.

B- Incorrect – Emergency depressurization is only directed if drywell pressure is not within PSP curve 6.

D- Incorrect –Emergency depressurization is only directed if temperature cannot be maintained below 280° F.

Technical Reference(s): 2-EOI-2

Proposed references to be provided to applicants during examination: None

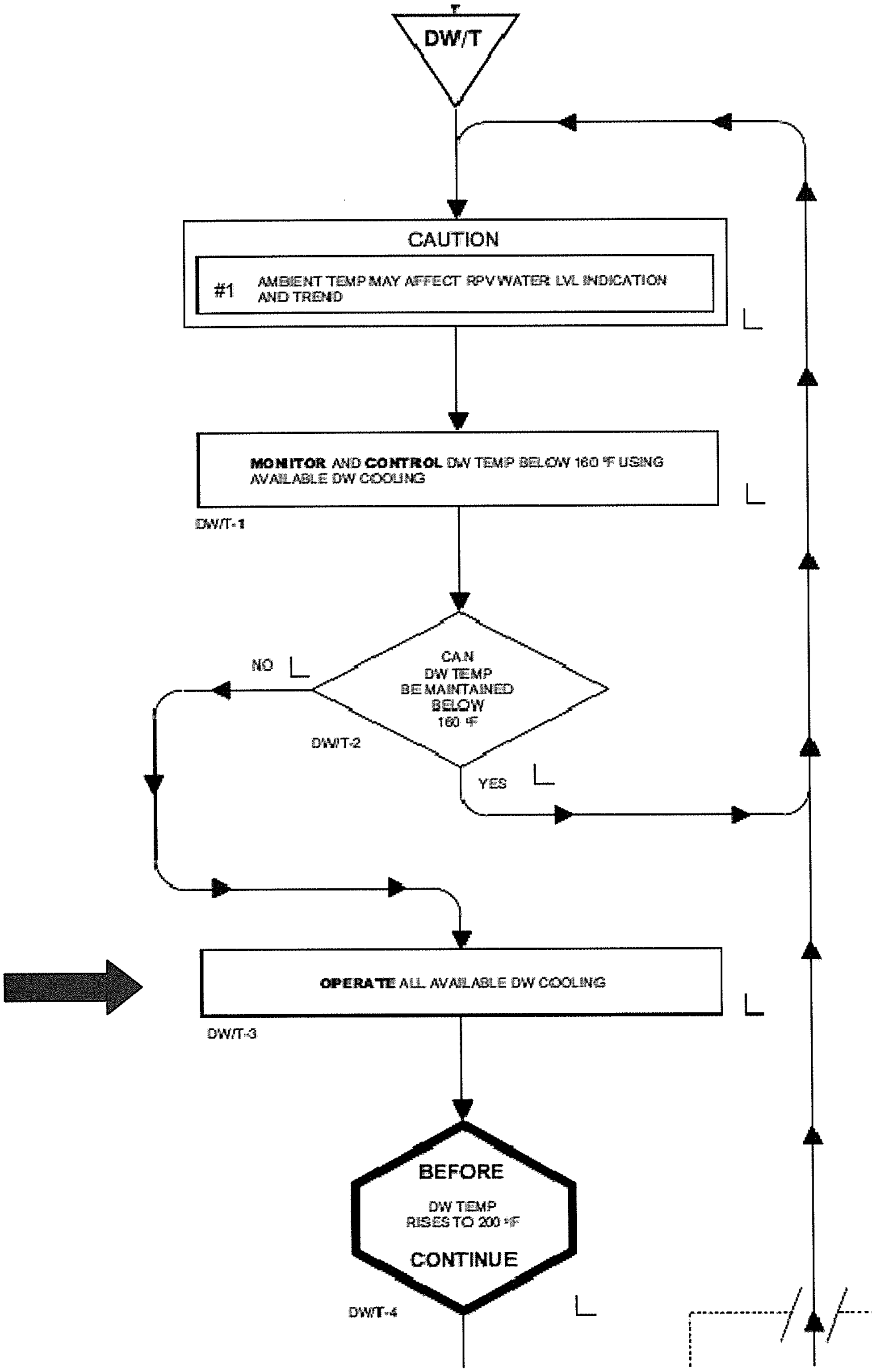
Learning Objective (As available):

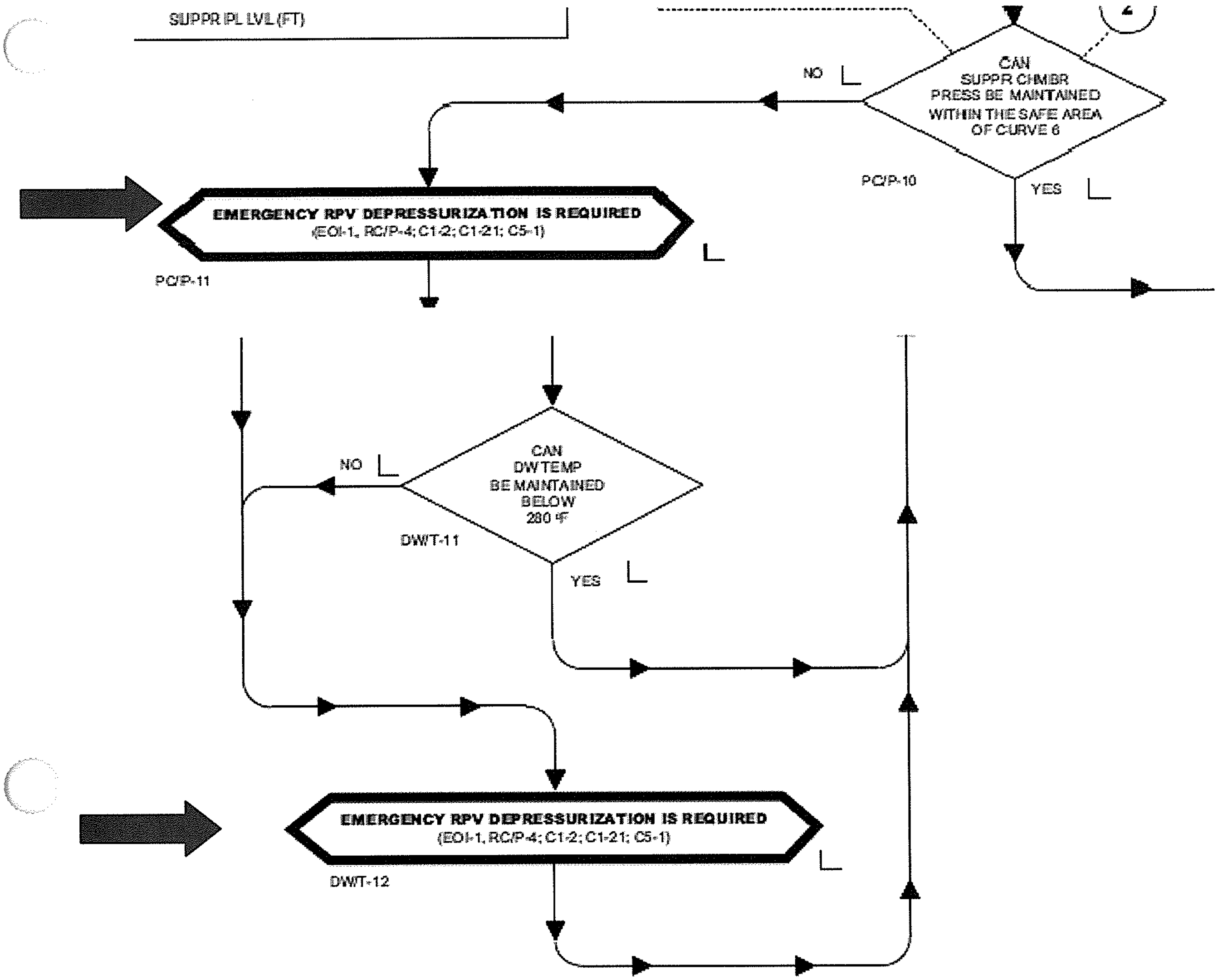
Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC None

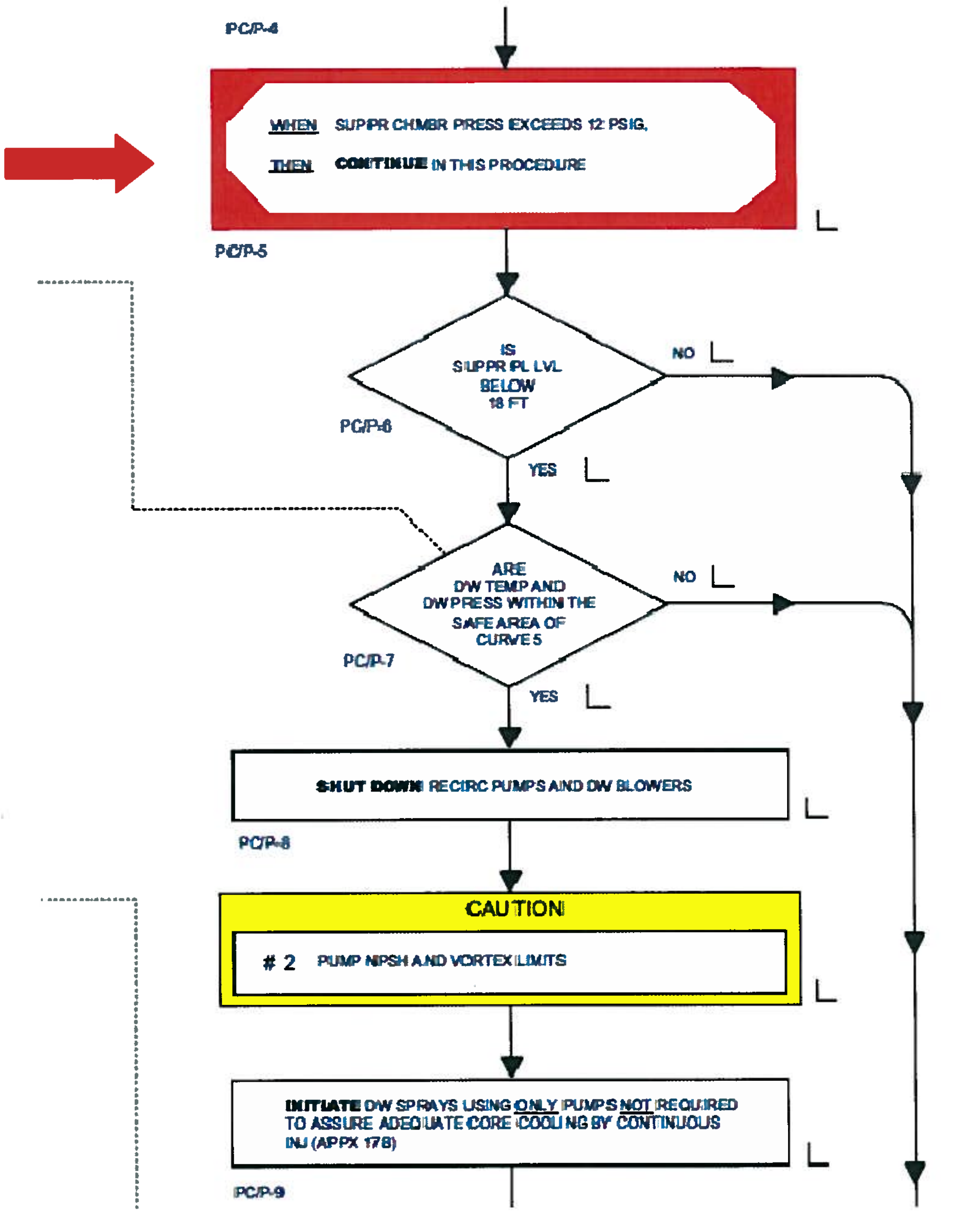
Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility


DW/T
CAUTION
#1 AMBIENT TEMP MAY AFFECT RPV WATER LVL INDICATION AND TREND
MONITOR AND CONTROL DW TEMP BELOW 160 °F USING AVAILABLE DW COOLING
DW/T-1
CAN DW TEMP BE MAINTAINED BELOW 160 °F
NO
YES
DW/T-2
OPERATE ALL AVAILABLE DW COOLING
DW/T-3
BEFORE
DW TEMP RISES TO 200 °F
CONTINUE
DW/T-4



SUPPR PL LVL (FT)
CAN SUPPR CHMBR PRESS BE MAINTAINED WITHIN THE SAFE AREA OF CURVE 6
NO
YES
PC/P-10
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-2; C1-21; C5-1)
PC/P-11
CAN DW TEMP BE MAINTAINED BELOW 280 °F
NO
YES
DW/T-11
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-2; C1-21; C5-1)
DW/T-12


PC/P-4
WHEN SUPPR CHMBR PRESS EXCEEDS 12 PSIG,
THEN CONTINUE IN THIS PROCEDURE
PC/P-5
IS SUPPR PL LVL BELOW 18 FT
NO
YES
PC/P-6
ARE DW TEMP AND DW PRESS WITHIN THE SAFE AREA OF CURVE 5
NO
YES
PC/P-7
SHUT DOWN RECIRC PUMPS AND DW BLOWERS
PC/P-8
CAUTION
2 PUMP NPSH AND VORTEX LIMITS
INITIATE DW SPRAYS USING ONLY PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING BY CONTINUOUS INJ (APPX 17B)
PC/P-9

QUESTION 15

Unit 1 is shutdown due to a lowering suppression pool level.

- The Unit supervisor is directing operation in 1-EOI-1, RPV CONTROL, and 1-EOI-2, PRIMARY CONTAINMENT CONTROL
- RPV level is being maintained (+)2 to (+)51 inches with HPCI
- Suppression Pool level is 13 feet and slowly lowering

Which ONE of the following completes the statement?

~~1-EOI-2~~ requires the operator to___(1)___ when Suppression Pool level CANNOT be MAINTAINED greater than ___(2)___.

A. (1) trip and lock out HPCI
(2) 12.75 feet

B. (1) trip and lock out HPCI
(2) 11.5 feet

C. (1) verify HPCI isolated
(2) 12.75 feet

D. (1) verify HPCI isolated
(2) 11.5 feet

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295030 EK2.01	
	Importance Rating	3.8	

Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following
EK2.01 HPCI

Explanation: **A** CORRECT – Per 1-EOI-2, operators are directed by step SP/L-4 to secure HPCI irrespective of adequate core cooling.

B Incorrect – First Part: Correct. Second Part: Incorrect. Plausible if the candidate confuses the Suppression Pool level at which HPCI is secured (12.75 ft) with the level at which emergency depressurization is required, (11.5 feet).

C Incorrect – First Part: Incorrect. Plausible if the candidate believes HPCI auto isolates on level as it auto swaps on CST level. Second Part: Correct.

D Incorrect – First Part: Incorrect. Plausible if the candidate believes HPCI auto isolates on level as it auto swaps on CST level. Second Part: Incorrect. Plausible if the candidate confuses the Suppression Pool level at which HPCI is secured (12.75 ft) with the level at which emergency depressurization is required, (11.5 feet).

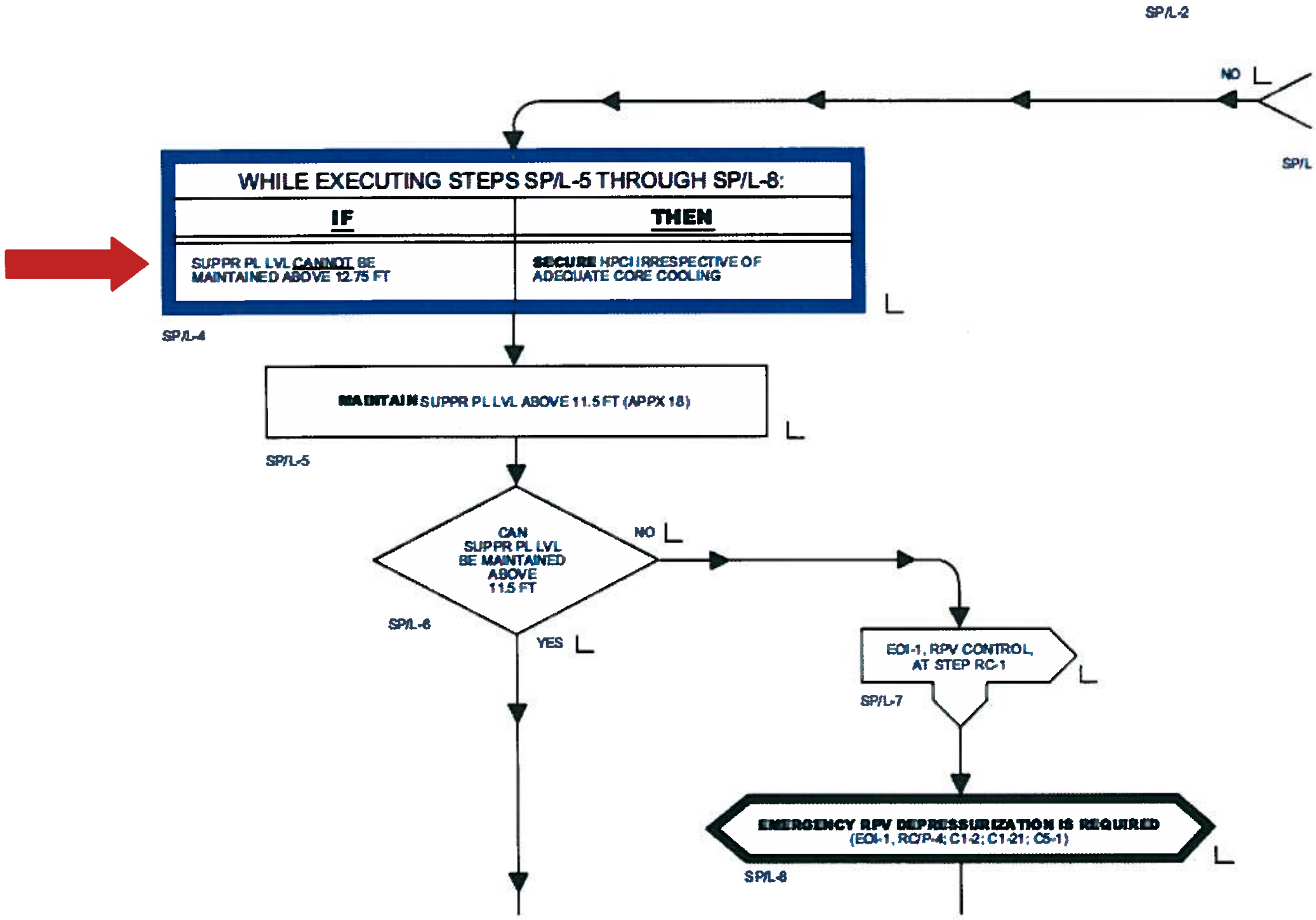
Technical Reference(s): 1-EOI-2 SP/L

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility


SP/L-2
NO
SP/L
WHILE EXECUTING STEPS SP/L-5 THROUGH SP/L-8:
IF
THEN
SUPPR PL LVL CANNOT BE MAINTAINED ABOVE 12.75 FT
SECURE HPCI IRRESPECTIVE OF ADEQUATE CORE COOLING
SP/L-4
MAINTAIN SUPPR PL LVL ABOVE 11.5 FT (APPX 18)
SP/L-5
CAN SUPPR PL LVL BE MAINTAINED ABOVE 11.5 FT
NO
YES
SP/L-6
EOI-1, RPV CONTROL, AT STEP RC-1
SP/L-7
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED (EOI-1, RC/P-4; C1-2; C1-21; C5-1)
SP/L-8

QUESTION 16

A Loss of Coolant Accident has occurred on Unit 3.

Which ONE of the following completes the following statement?

The core is adequately cooled if the MIMINUM RPV Water level is at or above the ___(1)___ and the Core Spray System is operating with a MINIMUM of___(2)___.

A (1) Two-thirds Core Height
(2) two loops injecting at 3125 gpm each

B. (1) the Top of Active Fuel
(2) two loops injecting at 3125 gpm each

C. (1) Two-thirds Core Height
(2) one loop injecting at 6250 gpm

D. (1) the Top of Active Fuel
(2) one loop injecting at 6250 gpm

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295031EK3.03	
	Importance Rating	4.1	

Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL:
EK3.03 Spray Cooling

Explanation: C CORRECT – To ensure adequate core cooling and not enter the SAMGs RPV level must be at least at or above two-thirds core height (-215 inches) and at least one Core Spray subsystem must be injecting at the subsystem design flow of 6250 gpm.

A Incorrect – First Part: Correct. Second Part: Incorrect. The total of the 2 loops is 6250, but neither subsystem is providing the minimum required flow.

B Incorrect – First Part: Incorrect. Second Part: Incorrect. The total of the 2 loops is 6250, but neither subsystem is providing the minimum required flow.

D Incorrect – First Part: Incorrect. Second Part: Correct.

Technical Reference(s): 3-EOI-C1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New
Question History:	Previous NRC: Duane Arnold 2007 NRC #55
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	55.41 8) Components, capacity, and functions of emergency systems.

WHILE EXECUTING STEPS C1-26 THROUGH C1-34:	
IF	**THEN**
ONE CS SUBSYSTEM CANNOT BE OPERATED WITH AT LEAST 6250 GPM TO THE RPV	CONTINUE AT STEP C1-35 C
RPV WATER LVL CANNOT BE DETERMINED TO BE ABOVE -215 IN.	CONTINUE AT STEP C1-35 C

C1-25

BFN Unit 0	CONTINGENCY #1 ALTERNATE LEVEL CONTROL	EOIPM SECTION 0-II-G Rev. 0001 Page 6 of 6

1.0 CONTINGENCY #1 ALTERNATE LEVEL CONTROL (continued)

(2) If either:

-215"

• RPV water level cannot be restored and maintained above **~~A.71~~** (Minimum Steam Cooling RPV Water Level) and no core spray subsystem flow can be restored and maintained equal to or greater than **A.48** (design ~~core spray~~ flow), or

6250 gpm

• RPV water level cannot be restored and maintained at or above **A.49** (elevation of jet pump suction),

then:

(a) Maximize injection into the RPV with available systems, injection subsystems, and alternate injection subsystems.

(b) If RPV water level cannot be restored and maintained above **A.71** (Minimum Steam Cooling RPV Water Level), PRIMARY CONTAINMENT FLOODING IS REQUIRED; enter the RPV and Primary Containment Flooding Severe Accident Guideline.

QUESTION 17

A reactor scram occurred on Unit 2 and control rods failed to fully insert.
Initial power level after the scram was 23%.

Currently:

- Standby Liquid Control System (SLC) is injecting into the reactor
- Standby Liquid Control System (SLC)Storage Tank Level is 63%
- RPV Pressure is 900 psig and steady
- RPV Level is -90 inches
- All APRMs are DOWNSCALE
- IRMS are inserted and indicating on Ranges 4 and 5
- SRMs are inserted and indicating UPSCALE

Which ONE of the following completes the statements?

At this time, the reactor ___(1)____ subcritical.

If RPV Water level is restored to +2 to +51 inches, the reactor will be ______(2)______.

A. (1) is NOT
   (2) critical

B. (1) is
   (2) critical

C. (1) is
   (2) subcritical

D. (1) is NOT
   (2) subcritical

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295037 EK1.03	
	Importance Rating	3.3	

Knowledge of the operation implications of the following concepts as the apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN
EK1.03 Boron effects on reactor power (SBLC)

Explanation: C CORRECT – Based on the Neutron monitoring indications, IRMs reading 4 to 5, the reactor is currently shutdown and since Hot Shutdown boron weight 67% tank level has been injected the reactor will remain subcritical when the level is restored.

A Incorrect – First Part: Incorrect. Plausible because the candidate may believe the reactor is still critical based on SRM and IRM readings. Second Part: Incorrect. Plausible because the candidate may believe it will return to criticality if reflooded.

B Incorrect – First Part: Correct. Second Part: Plausible because the candidate may believe it will return to criticality if reflooded.

D Incorrect – First Part: Incorrect. Plausible because the candidate may believe the reactor is still critical based on SRM and IRM readings. Second Part: Correct.

Technical Reference(s):EOI C-5, OPL171.202, bases

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:
	Modified Bank:
	New: X

Question History: Previous NRC None

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

10 CFR Part 55 Content: 55.41 (1) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

From OPL171.202

a. If the SLC System is successfully injecting into the RPV, then boron injection should continue as previously directed. The operator continues at Step RC/Q-18. If the SLC System is unable to inject into the RPV, then an alternate method of boron injection must be addressed. The operator continues at Step RC/Q-17.

2. Step RC/Q-17

Step RC/Q-17 directs the operator to inject boron into the RPV using the alternate method specified in EOI Appendix 3B. EOI Appendix 3B provides step-by-step guidance for cross-connecting one Unit's SLC tank to another Unit's CRD pump suction for an independent injection path of Boron into the RPV. Although the SLC System is highly reliable, a number of shared components (i.e., storage tank, injection line, etc.) make SLC susceptible to a single failure.

Unit difference: Unit 2 uses 1B CRD pump ,Unit 3 uses 3B CRD Pump and Unit 1 uses 1B CRD pump.

Unit 1 uses Unit 2 SLC Tank.

Unit 2 uses Unit 1 SLC Tank

Unit 3 uses Unit 2 SLC Tank

3. Step RC/Q-18

This contingent action step requires the operator to wait until the stated condition is met before continuing in this section of the procedure. This condition requires that the SLC System has injected into the RPV at least all but 43% of SLC tank level, which corresponds to Cold Shutdown Boron Weight of boron. Cold Shutdown Boron Weight is defined to be the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions.

Obj.V.B.9

4. Step RC/Q-19

Step RC/Q-19 is an action step which directs the operator to exit RC/Q Section of this procedure and enter AOI-100-1, Reactor Scram. With reactor subcriticality assured, appropriate instructions for maintaining shutdown conditions are provided in AOI-100-1, Reactor Scram.

WHEN SLC HAS INJECTED INTO THE RPV TO A TANK LVL OF 67%

THEN **CONTINUE**

C5-17

RESTORE AND MAINTAIN RPV WATER LVL BETWEEN +2 IN. AND +51 IN. USING SYSTEMS LISTED IN STEP C5-13 OR C5-15.

C5-18

WHEN SLC HAS INJECTED INTO THE RPV TO A TANK LVL OF 43%

THEN **CONTINUE**

RC/Q-18

EXIT RC/Q AND
ENTER AOI-100-1, REACTOR SCRAM

RC/Q-19

AOI-100-1,
REACTOR SCRAM

QUESTION 18

Unit 2 is performing a startup.
The following alarms/indications are received:

- STACK GAS RADIATION HIGH (2-9-3A, Window 13)
- OG POST-TREATMENT RADIATION HIGH, (2-9-4C, Window 33)

OG POST-TREATMENT CH A RAD MON RTMR, 2-RM-90-265A, is reading 6.5×10^4 cps
OG POST-TREATMENT CH B RAD MON RTMR, 2-RM-90-265B, is reading 6.3×10^4 cps

The Off Gas Treatment Select Switch, 2-XS-66-113, is in **BYPASS**.

Which ONE of the following identifies the AUTOMATIC actions (if any) of the Offgas system to limit the offsite release rate?

A. Adsorber Bypass Valve, 2-FCV-66-113B will CLOSE.
   Adsorber Inlet Valve, 2-FCV-66-113A will OPEN
   Charcoal Adsorber Train 2 Inlet Valve, 2-FCV-66-118 will OPEN

B. Adsorber Bypass Valve, 2-FCV-66-113B will CLOSE
   NO other valves will reposition

C. Adsorber Bypass Valve, 2-FCV-66-113B will CLOSE
   Adsorber Inlet Valve, 2-FCV-66-113A will OPEN
   NO other valves will reposition.

D. NO valves will reposition

Correct Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	295038 EK2.02	
	Importance Rating	3.6	

Knowledge of the interrelations between HIGH OFF_SITE RELEASE RATE and the following:
EK2.02 Offgas system

Explanation: **D** CORRECT: With the Off Gas Treatment Select Switch, 2-XS-66-113, is in BYPASS, No valves will reposition.

A. Incorrect: 2-FCV-66-113B will NOT close. Plausible in that this sounds correct, however if the adsorber trains were to go into service, 2-FCV-66-118 is the train 1 outlet valve.

B. Incorrect: Adsorber Bypass Valve, 2-FCV-66-113B closing and no other valves repositioning would shutoff offgas flow altogether.

C. Incorrect: This would be correct with the Offgas Selector switch in Auto.

Technical Reference(s): OPL171.030, 2-ARP9-4C

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New
Question History:	Previous NRC: BFN 1102 #18
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

4. Off-Gas Radiation High

   a. Two radiation elements in off-gas sample system

   b. Four alarms

      (1) Post Treatment Off-gas Radiation Hi-Hi-Hi

      DCN S17071A

      (2) Post Treatment Off-gas Radiation Hi-Hi

      (3) Post Treatment Off-gas Radiation Hi

      (4) Post Treatment Off-gas Radiation Downscale

   c. Closes off-gas isolation valve (28)

      (1) Both channels Hi Hi Hi or

      (2) One Channel Hi Hi Hi and the other channel downscale/inop or

      (3) Both channels downscale/inop

INSTRUCTOR NOTES

   d. Any Channel Hi

      Initiates charcoal adsorbers by opening adsorber inlet valves (113A) (117) and closing adsorber bypass valve (113B), provided HS 66-113 is in AUTO.

HS 66-113 is kept in TREAT to keep the adsorbers in service when the unit is at power. Major system flow changes would cause a radiation spike

5. Condenser Vacuum Low ($\leq$ 25" Hg Vacuum)

OG POST TRTMT RADIATION HIGH-HIGH 2-RA-90-265B	34

(Page 1 of 2)

Sensor/Trip Point:

2-RM-90-265A	3.1×10^5 cps
2-RM-90-266A	3.1×10^5 cps

Sensor Location: 2-RE-90-265 Panel 2-25-94 Off-Gas Building
2-RE-90-266 Elevation 538.5

Probable Cause:

A. Off-Gas flow change.
B. Adsorber lineup change.
C. Resin trap failure (RWCU or Condensate demins).
D. Fuel damage.

Automatic Action: None

Operator Action:

A. **VERIFY MONITOR** high activity on the following: ☐
- OFFGAS RADIATION recorder, 2-RR-90-266 on Panel 2-9-2. ☐
- OG POST-TREATMENT CHAN A RAD MON RTMR radiation monitor, 2-RM-90-266A on Panel 2-9-10. ☐
- OG POST-TREATMENT CHAN B RAD MON RTMR radiation monitor, 2-RM-90-265A on Panel 2-9-10. ☐

B. **VERIFY** Charcoal Adsorbers in service. ☐
C. **NOTIFY** Unit 1 and 3 operators of conditions and that verification of proper operation of Unit 1 and 3 Off-Gas system is required. ☐
D. **CHECK** STACK GAS/CONT RM RADIATION RECORDER, 0-RR-90-147 on Panel 1-9-2. ☐
E. **NOTIFY** Radiation Protection. ☐
F. **REQUEST** Chemistry perform radiochemical analysis to determine source. ☐
G. **REFER TO** 0-SI-4.8.b.1.a.1 and 0-SR-3.4.6.1-a for Technical Specification compliance and to determine if power level reduction is required. ☐
H. **IF** directed by Shift Manager or Unit Supervisor/SRO, **THEN REDUCE** reactor power to maintain off-gas radiation within ODCM limits. ☐

STACK GAS RADIATION HIGH 2-RA-90-147B 13

(Page 1 of 1)

Sensor/Trip Point:

	HI
RM-90-147B	11,948 CPS
RM-90-148B	11,948 CPS
0-RM-90-306	As listed in 2-AOI-90-2

Sensor Location: RE-90-147 and El 599'6", Pnl 25-39 RE-90-148 inside stack

Probable Cause:

A. Source check.
B. Resin trap failure (RWCU or Cond Demin).
C. Possible fuel element failure.
D. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** alarm condition on following:
   1. WIDE RANGE GASEOUS EFFLUENT RADIATION MONITOR, 0-RM-90-306 on Panel 2-9-10. ☐
   2. STACK GAS/CONT RM RADIATION, 0-RR-90-147 on Panel 1-9-2. ☐

B. **IF** alarm is from 0-RM-90-306, **THEN REFER TO** 2-AOI-90-2. ☐
C. **CHECK** following radiation recorder on Panel 2-9-2 and associated radiation monitors on Panel 2-9-10:
   1. OFFGAS RADIATION, 2-RR-90-266. ☐

D. **VERIFY** dilution fan running and damper open by checking red light illuminated above STACK DILUTION FAN 2A (2B), 2-HS-66-29A (31A) on Panel 2-9-8. ☐
E. **VERIFY** Charcoal Adsorbers in service. ☐
F. **NOTIFY** RAD PRO, Unit 1, Unit 3 and Unit Supervisor/SRO
G. **REQUEST** Chemistry perform radiochemical analysis to determine source. ☐

References: 2-47E620-3   0-47E610-90-4 & 20   GE-2-729E814RF-5   0-SIMI-90B

BFN 1102 NRC Exam #18

ILT 1102 Written Exam

18. 295038 EK2.10

Unit 2 is in Start Up. Off Gas Treatment Select Switch, 2-XS-66-113, is in BYPASS. The following alarms/indications are received:

- OG POST-TREATMENT RADIATION HIGH, (2-9-4C, Window 33)
- Offgas Post-Treatment Radiation is 6.5×10^4 cps

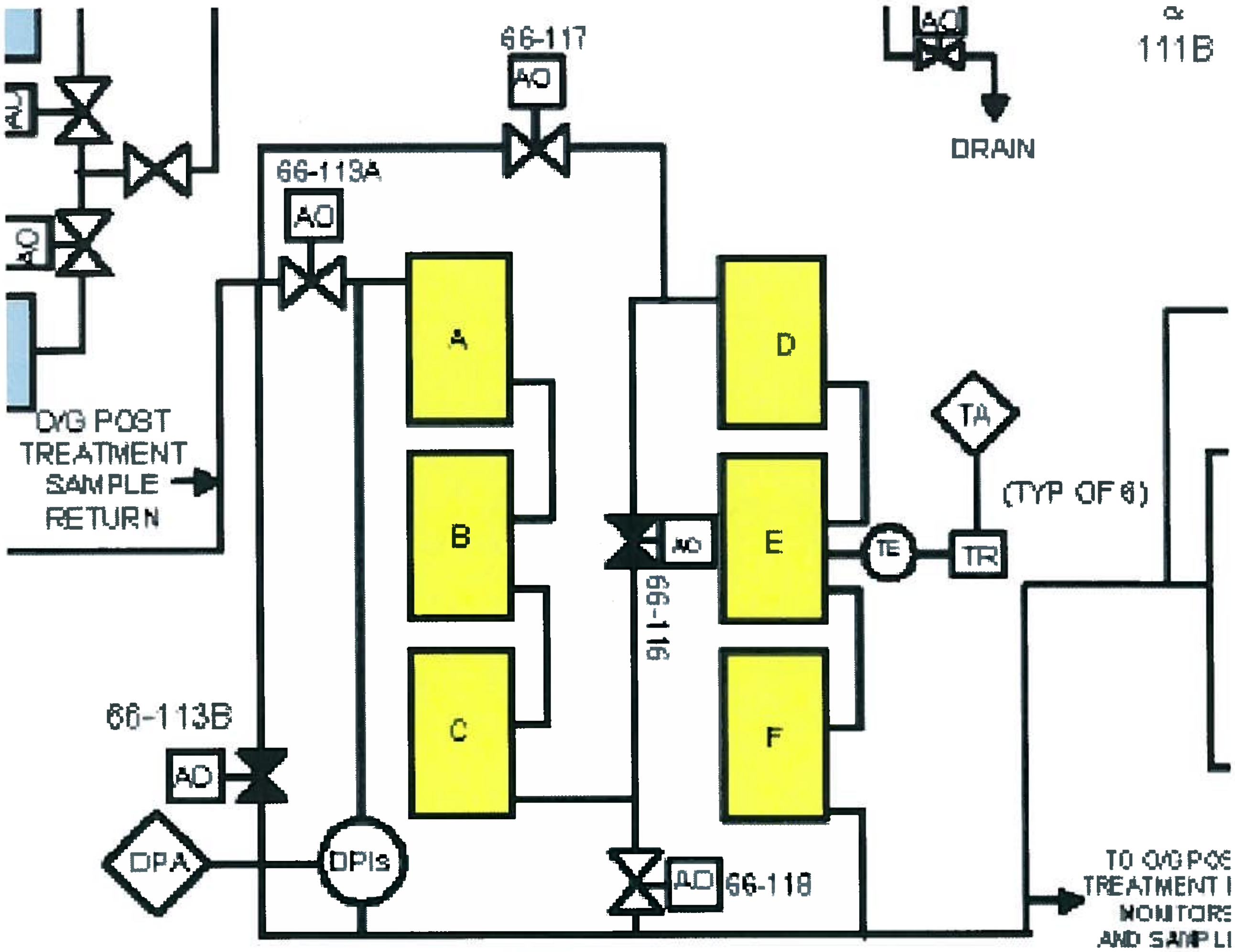
Which ONE of the following identifies the impact of this condition on the Offgas System?

A. **NO** valves will reposition

B. Adsorber Bypass Valve, 2-FCV-66-113B will close. **NO** other valves will reposition.

C. Adsorber Bypass Valve, 2-FCV-66-113B will close **AND** Adsorber Inlet Valve, 2-FCV-66-113A will open. **NO** other valves will reposition.

D. Adsorber Bypass Valve, 2-FCV-66-113B will close. Adsorber Inlet Valve, 2-FCV-66-113A **AND** Charcoal Adsorber Train 2 Inlet Valve, 2-FCV-66-118 will open.



Charcoal Adsorbers

QUESTION 19

The following conditions exist on Unit 1 due to a Main Bank Transformer 1A fault:

- Reactor Scram has occurred with ALL Control Rods inserted
- Reactor water level is (-)130 inches and stable
- Complete loss of Offsite power

Subsequently, an AUTOMATIC sprinkler actuation signal is received.

Which ONE of the following describes the status of the Electric Driven Fire Water Pumps?

A. ALL Electric Driven Fire Pumps are operating

B. ALL Electric Driven Fire Pumps are locked out for AUTOMATIC start

C. ONLY the "SELECTED" Electric Driven Fire Pump is operating

D. NONE of the Electric Driven Fire Pumps are locked out for AUTOMATIC start

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	600000 AK2.01	
	Importance Rating	2.6	

Knowledge of the interrelations between PLANT FIRE ON SITE and the following:
AK2.01Sensors/detectors and valves

Explanation: **B** CORRECT – The initiation signal is present with the sprinkler actuation, but prior to this , a CASA signal combined with the EDG's supplying their 4Kv SD Boards, generating a lockout of the auto start feature. A LOOP does establish this prerequisite. The pumps are normally aligned in a cascading start sequence with the first starting on a sprinkler actuation. The remaining will sequence on @< 120# for 15 seconds& <120# for 30 seconds. They may still be manually started.

A Incorrect- Plausible if the candidate does not recognize the auto start lockout signal AND thinks ALL of the pumps will start together vice sequencing on.

C Incorrect – Plausible if the candidate believes the lockout only affects the selected fire pump.

D Incorrect – Plausible if the candidate does not recognize the auto start lockout signal.

Technical Reference(s): OPL171.049, 0-OI-26

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Date ________

5.0 STARTUP

5.1 Automatic Start of a Fire Pump

[1] One or more of the following initiation signals are present:

- Transformers: temperature equal to or greater than 225°F. ☐
- Turbine Building protected areas; Oxygen, Acetylene and Oil storage rooms; Hydrogen trailer ports: temperature equal to or greater than 225°F (common) or temperature rise of 12°F/min (uncommon). ☐
- Transformer differential, overcurrent or sudden pressure relays actuated. ☐
- Any spray, sprinkler or fog system manual or automatic actuation. ☐
- EST-3 fire detection outputs to 0-RLY-26-101A through fire panel 0-LPNL-925-0555 and 0-RLY-26-101B through fire panel 0-LPNL-925-0556.

Date ________

5.1 Automatic Start of a Fire Pump (continued)

NOTE

Automatic start of Fire Pumps A, B, and C is locked out on the following (REFER to P&L 3.0T):

- Common Accident signal (CAS):
  High Drywell Pressure (2.45 psig) in conjunction with Low Reactor Pressure (450 psig) OR Low Low Low Reactor Water Level (-122 in.).

  AND

- Diesel Generators are supplying power to the U1/2 4160V Shutdown Boards (DGVA).

NOTES

1) <u>15 seconds</u> after the initiating signal, if system header pressure is less than 120 psig the second selected Fire Pump starts.
2) <u>30 seconds</u> after the initiating signal if, system header pressure is less than 120 psig the third selected Fire Pump starts.
3) <u>45 seconds</u> after the initiating signal if, system header pressure is less than 120 psig the Diesel Driven Fire Pump starts.

QUESTION 20

All units are operating at 100% RTP with lagging VARS
Current system voltage is 542KV
Current system frequency is 60.03 hz

Which ONE of the following completes the statements?

The operator is directed by 0-AOI-57-1E to lower____(1)___.

This will cause the power factor to ___(2)___.

A. (1) speed
   (2) rise (closer to 1.0)

B. (1) speed
   (2) drop (further from 1.0)

C. (1) voltage
   (2) rise (closer to 1.0)

D. (1) voltage
   (2) drop (further from 1.0)

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	700000 AK1.01	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND GRID DISTURBANCES:
AK1.01 Definition of terms: volts, watts, amps, VARs, power factor

Explanation: **C** CORRECT – Voltage is high and frequency is within the normal range of 60±.05. With voltage high 0-AOI-57-1E directs the operator to lower voltage. With lagging VARS, reducing voltage brings the Power Factor (MW/VA) closer to one (unity).

A Incorrect – First Part: Incorrect. Plausible if the candidate confuses MVARS and MW adjustments: Speed would be adjusted for a frequency change. Second Part: Incorrect. Speed adjustment would not affect the power factor. Plausible if the candidate does not understand reactive load and the relationship to real load.

B Incorrect – First Part: Incorrect. Plausible if the candidate confuses MVARS and MW adjustments: Speed would be adjusted for a frequency change. Second Part: Incorrect. Speed adjustment would not affect the power factor. Plausible if the candidate does not understand reactive load and the relationship to real load.

D Incorrect – Firs Part: Correct. Second Part: Incorrect. With lagging VARS, lowering voltage will bring the power factor closer to unity (rising power factor).

Technical Reference(s): 0-AOI-57-1E

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New
Question History:	Previous NRC: Perry 2009 #30
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

4.2 Subsequent Action (continued)

[6] **IF** grid instability is characterized by system voltage being maintained outside the normal limits of 520 + 10 KV, **THEN**

PERFORM the following steps:

[6.1] **IF** system voltage is greater than 540KV, **THEN**

[6.1.1] **LOWER** reactive power to system voltage returns to 530KV, **OR UNTIL** Generator Reactive power reaches -150 MVAR. ☐

[6.1.2] **CHECK** 161KV Cap Banks are Out of Service and **EVALUATE** conditions to determine appropriate actions. **REFER TO** 0-GOI-300-4. ☐

[6.2] **IF** system voltage is lower than 510KV, **THEN**

PERFORM the following:

[6.3] **RAISE** reactive power to system voltage returns to 510 KV **OR UNTIL** Generator Reactive Power reaches +300 MVAR, ☐

[6.4] **CHECK** 161KV Cap Banks are In Service and **EVALUATE** conditions to determine appropriate actions. **REFER TO** 0-GOI-300-4. ☐

[6.5] **EVALUATE** as applicable, entry into Technical Specifications 3.8.1, 3.8.2, 3.8.7 and 3.8.8. ☐

Illustration 7
(Page 1 of 1)

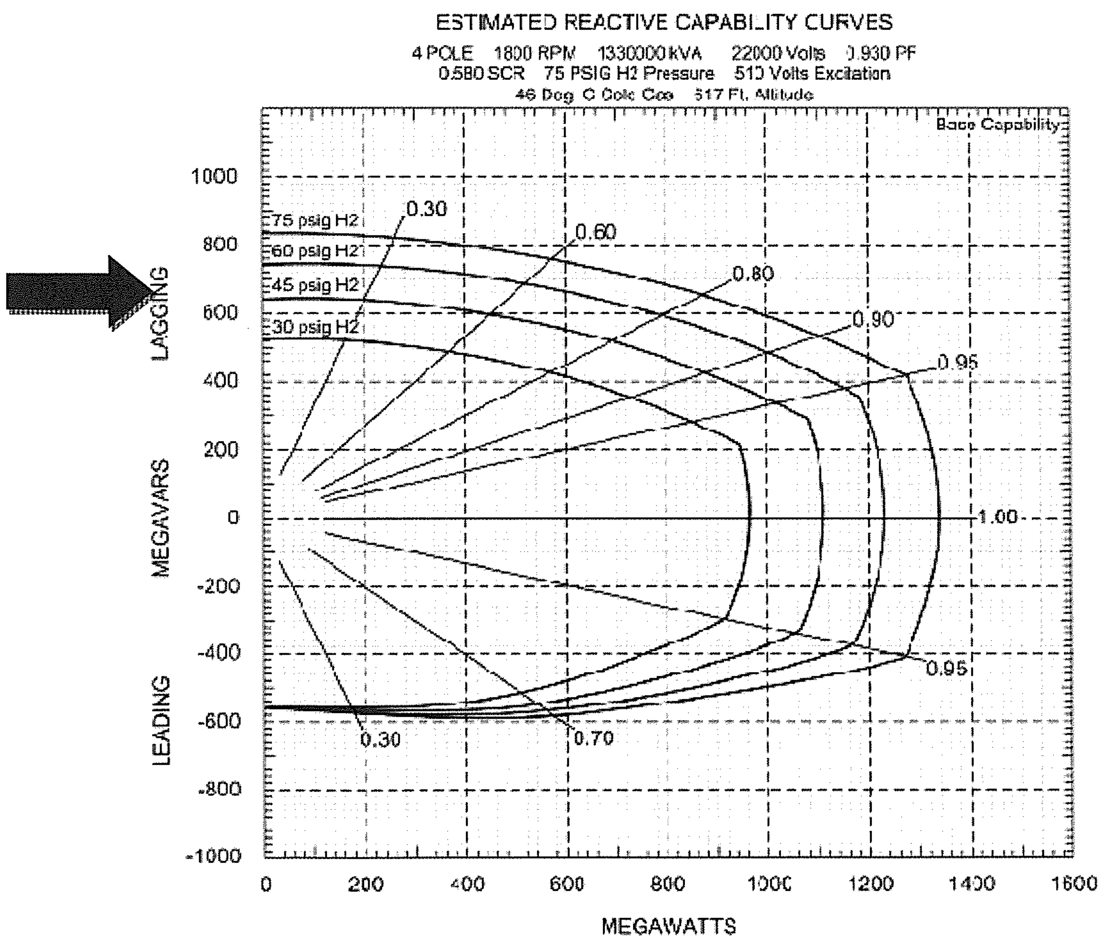
Generator Kilovar Limitations (Capability Curve)

NOTES

1) A 300 MVAR maximum outgoing (lagging) limit applies to all three units for both 500kV and 161kV offsite power source qualification (based on unit MVAR capability limits provided by BFN and used in grid studies). 0-GOI-300-4

2) When operated in automatic, the Generator Voltage Regulator has an electronic limit (URAL) which limits incoming reactive, at full load, to approximately 150 MVAR. When operated in manual, there is an administrative limit of 150 MVAR.

3) Operation with MVARs above 150 MVARs incoming is prohibited unless calibration or testing is being performed. Under no circumstances should the capability curve be exceeded.



Perry NRC 2009 #30

NRC EXAM - 2009

QUESTION RO 30

The plant is operating at 100% power. The following conditions exist:

- Unit Supervisor has entered ONI-S11 HI/LOW VOLTAGE
- Bus 1 voltage is 343 Kv
- Main Generator output is1290 Mwe
- Main Generator power factor is 0.98 lagging
- Main Generator is carrying 200 Mvar

System Control has asked Perry to raise Main Generator voltage to increase Bus 1 voltage to 345 Kv. Raising Main Generator voltage will cause __(1)__ and __(2)__.

	(1)	(2)
A.	VARs to decrease	power factor closer to unity
B.	VARs to decrease	power factor further from unity
C.	VARs to increase	power factor closer to unity
D.	VARs to increase	power factor further from unity

QUESTION 21

During a transient, the operators closed the Main Steam Line Isolation Valves (MSIVs) just before water reached the main steam lines.

According to 1-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low, which ONE of the following COULD the operators experience after the main steam lines flood during this transient?

Due to two phase flow ______.

A. the safety relief valves stick OPEN

B. the safety relief valves stick CLOSED

C. the feedwater turbine valves fail to operate

D. the HPCI turbine is damaged while operating

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295008 AK1.02	
	Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL:
AK1.02 Component erosion/damage

Explanation: **A** CORRECT – listed in 1-AOI -3-1 as a potential problem that may occur under these conditions.

B- Incorrect –Plausible because SRV problems are possible but a failure to OPEN is not.

C- Incorrect – Plausible because this listed in the procedure but not possible in this case since the MSIVs were closed before water reached the MSL.

D- Incorrect –Plausible, however, HPCI should have tripped already at high level.

Technical Reference(s):1-AOI-3-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC None	
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

4.2 Subsequent Actions (continued)

[13] **IF** a RFPT has tripped and is **NOT** required to maintain level, **THEN**

SECURE tripped RFPT. **REFER TO** 1-OI-3. ☐

[14] **IF** a Condensate Pump or Condensate Booster Pump has tripped, **THEN**

PLACE standby pump in service and **SECURE** tripped pump. **REFER TO** 1-OI-2.

TRIP Reactor Feedwater Pump(s), as necessary, and **SECURE** tripped pump. **REFER TO** 1-OI-3. ☐

[15] **IF** unit remains on-line, **THEN**

RETURN Reactor water level to normal operating level of 33" (normal range). ☐

CAUTION

The following could lead to a steamline break or other malfunction if water level is left unattended:

1) The added dead weight and potential seismic loads placed on the main steamline and its supports should the main steamline be flooded.
2) The loads placed on the main steamlines as a result of the potential for rapid collapse of steam voids resulting in water hammer.
3) The potential for safety relief valves sticking open following discharge of water or two-phase flow.
4) The potential inoperability of the main steamline isolation valves (MSIVs), main turbine stop or bypass valves, feedwater turbine valves, or atmospheric dump valves from the effects of water or two-phase flow.

[16] **IF** any of the following are inadvertently injecting into the Reactor, **AND** are not required for level control, **THEN**

STOP injection by tripping pump, closing discharge valves, or with flow reduction, as appropriate:

A. HPCI and/or RCIC with reactor pressure > 450 psi ☐

B. HPCI, RCIC, RHR, Core Spray, or feedwater/condensate with reactor pressure ≤ 450 psi ☐

QUESTION 22

Unit 3 is operating at 65% when a transient occurs which results in the following indications:

- Reactor Power is rising steadily
- RPV Water Level is 33 inches and stable
- Generator Output is rising steadily

Which ONE of the following describes the cause of these indications?

A. The reactor has just experienced a Control Rod Drop accident, raising Reactor Power.

B. The EHC system is malfunctioning, raising Reactor Pressure.

C. A malfunction in the Feedwater Level Control System (FWLCS) is causing Feedwater Flow, to lower.

D. One ~~or both~~ of the Recirc controllers is (are) failing, raising Reactor Recirculation Flow.

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295014 AA2.03	
	Importance Rating	4.0	

Ability to determine and /or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION
AA2.03 Cause of reactivity addition

Explanation: D CORRECT – A failure of reactor recirculation control resulting in a rising flow till cause reactor power to increase, feedwater will respond accordingly and keep level steady. EHC will maintain reactor pressure at the pressure setpoint and generator load will rise.

A Incorrect – Plausible because a rod drop causes a rise in reactor power, however, a rod drop accident causes a short term increase not a steady rise in power. Also the power rise in the core causes increased indicated RPV level as water is forced out of the core to the downcomer.

B Incorrect –Plausible because the EHC malfunction would raise power but would lower generator output as it would pass less steam to the turbine.

C Incorrect –Plausible, a rise in relatively cold feedwater flow could affect reactor power however, less cold feed water would cause reactor power and generator output to drop.

Technical Reference(s): 3-AOI-3-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	X
	New:	

Question History: Previous NRC Duane Arnold 2007 #61

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

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a control rod drop accident

2.0 SYMPTOMS

A. Reactor power rise.

B. LPRM HI status lights around dropping control rod in alarm on Full Core Display.

C. Reactor high neutron flux scram with any of the following annunciators on Panel 3-9-4 and 3-9-5 in alarm:

1. IRM CH A,C,E,G HI-HI/INOP (3-XA-55-5A, Window 33)
2. IRM CH B,D,F,H HI-HI/INOP (3-XA-55-5A, Window 34)
3. APRM HIGH/INOP OR OPRM TRIP(3-XA-55-5A, Window 25)
4. REACTOR CHANNEL A AUTO SCRAM (3-XA-55-5B, Window 1)
5. REACTOR CHANNEL B AUTO SCRAM (3-XA-55-5B, Window 2)
6. NEUTRON MONITORING SYS HALF SCRAM (3-XA-55-4A, Window 16)
7. CONTROL ROD WITHDRAWAL BLOCK (3-XA-55-5A, Window 7)

2.0 SYMPTOMS (continued)

F. Rising or lowering Reactor water level as indicated on:

1. RX VESSEL LEVEL (normal range), 3-LR-3-53 (3-9-5)
2. LEVEL A, 3-LI-3-53 (3-9-5)
3. LEVEL B, 3-LI-3-60 (3-9-5)
4. LEVEL C, 3-LI-3-206 (3-9-5)
5. LEVEL D, 3-LI-3-253 (3-9-5)
6. 3-LI-3-208A (3-9-5)
7. 3-LI-3-208B (3-9-3)
8. 3-LI-3-208C (3-9-3)
9. 3-LI-3-208D (3-9-5)

G. Annunciators in alarm on feedwater loss may include:

1. CNDS BSTR PUMP A SUCT PRESS LOW (3-XA-55-6A, Window 19)
2. CNDS BSTR PUMP B SUCT PRESS LOW (3-XA-55-6A, Window 20)
3. CNDS BSTR PUMP C SUCT PRESS LOW (3-XA-55-6A, Window 21)
4. CONDENSATE DEMIN ABNORMAL (3-XA-55-6B, Window 6)
5. RFPT TRIPPED (3-XA-55-6C, Window 29)
6. RFP DISCH FLOW LOW (3-XA-55-6C, Window 32)
7. RFPT A ABNORMAL (3-XA-55-6C, Window 1)
8. RFPT B ABNORMAL (3-XA-55-6C, Window 8)
9. RFPT C ABNORMAL (3-XA-55-6C, Window 15)
10. RFP A, B, OR C NPSH PRESS LOW (3-XA-55-6C, Window 19)
11. MOTOR TRIPOUT (3-XA-55-8C, Window 33)

H. Possible Reactor power reduction on reduced moderation for lowering reactor water level and/or Feedwater flow.

1.0 PURPOSE

This instruction provides the symptoms, automatic actions, and operator actions for a Recirculation Loop A or B Speed Control Failure.

2.0 SYMPTOMS

A. Annunciator Alarms

RECIRC FLOW SYSTEM TROUBLE ALARM annunciation (3-XA-55-4A, window 23) in alarm.

B. Unexplained raising or lowering in Recirc Pump speed.

C. Inability to change Recirc Pump speed.

3.0 AUTOMATIC ACTIONS

None

Duane Arnold 2007 #61

RO 61	K/A Number 295014	Statement AA2.03	IR 4.0	Origin B	Source Question 1999 DAEC NRC Exam
LOK H	10CFR55.41(b) 10	LOD (1-5)	Reference Documents AOP 255.2 Rev 28		
Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION : Cause of reactivity addition					

Plant conditions are as follows:

- Reactor Power is 65% and stable.
- RPV Water Level is 189 inches and stable.
- Generator Output is 350 Mwe and stable.

Due to a transient, the following indications are observed:

- Reactor Power is rising steadily.
- RPV Water Level is 180 inches, lowering slowly.
- Generator Output is rising steadily.

Which ONE of the following caused these indications?

a. The reactor has just experienced a Control Rod Drop accident.

b. The EHC system is malfunctioning, raising Reactor Pressure.

c. One or both of the Feed Regulating Valves is failing, raising Feed Flow.

d. One or both of the Recirc controllers is failing, raising Reactor Recirculation Flow.

Correct Answer: D One or both of the Recirc controllers is failing, raising Reactor Recirculation Flow.
Plausible Distractors: A is plausible: would be true for a step increase in Reactor Power, associated core voiding causes an increase in RPV Water Level. B is plausible: would cause a lowering Generator Output. C is plausible: would cause rising RPV Water Level.
Objective Link: None

QUESTION 23

A hydraulic ATWS has occurred on Unit 2. The following conditions exist:

- Reactor power is 58%
- All control rods inserted at least one notch on the scram

Which ONE of the following completes the statement?

For these plant conditions, the Rod Worth Minimizer ___(1)___ imposing an INSERT BLOCK ___(2)___.

A. (1) is
(2) due to the control rods being out of sequence

B. (1) is
(2) until the RWM is reinitialized

C. (1) is NOT
(2) at this time

D. (1) is NOT
(2) until reactor power drops to the LOW POWER ALARM POINT

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295015 AK3.01	
	Importance Rating	3.4	

Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM:
AK3.01 Bypassing rod insertion blocks

Explanation: **C** CORRECT – RWM does not begin to enforce the insert and withdraw limits until power reaches the LPSP (24%). Rods can be inserted at this time but must be bypassed to allow rod insertion once the LPSP is reached

A- Incorrect – Plausible if the candidate thinks that the RWM is enforcing at this time because the rods are out of sequence.

B- Incorrect – Plausible if the candidate believes the RWM is enforcing at this time but can be reset by initializing the RWM. Additionally the RWM will does produce a select error since the rods are chosen out of sequence.

D- Incorrect – Plausible because the RWM starts enforcing below the Low Power Setpoint NOT the Low Power Alarm Point.

Technical Reference(s): 2-OI-85, 2-EOI Appendix 1D

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (6) Design, components, and functions of reactivity control mechanisms and instrumentation

2-EOI APPENDIX-1D

INSERT CONTROL RODS USING REACTOR MANUAL CONTROL SYSTEM

LOCATION: Unit 2 Control Room, Panel 9-5

ATTACHMENTS: 1. Tools and Equipment
2. Core Position Map

(√)

NOTE: This EOI Appendix may be executed concurrently with EOI Appendix 1A or 1B at SRO's discretion when time and manpower permit.

1. **VERIFY** at least one CRD pump in service. ____

NOTE: Closing 2-85-586, CHARGING WATER ISOL, valve may reduce the effectiveness of EOI Appendix 1A or 1B.

2. IF .....Reactor Scram or ARI CANNOT be reset,
THEN ...**DISPATCH** personnel to close 2-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft). ____

3. **VERIFY** REACTOR MODE SWITCH in SHUTDOWN. ____

4. **BYPASS** Rod Worth Minimizer. ____

5. **REFER TO** Attachment 2 and **INSERT** control rods in the area of highest power as follows:

   a. **SELECT** control rod. ____

   b. **PLACE** CRD NOTCH OVERRIDE switch in EMERG ROD IN position UNTIL control rod is NOT moving inward. ____

   c. **REPEAT** Steps 5.a and 5.b for each control rod to be inserted. ____

NOTE: A ladder may be required to perform the following step. REFER TO Tools and Equipment, Attachment 1.

IF necessary, an alternate ladder is available at the HCU Modules, EAST and West banks. It is stored by the CRD Charging Cart.

6. WHEN ...NO further control rod movement is possible or desired,
THEN ...**DISPATCH** personnel to verify open 2-SHV-85-586, CHARGING WATER SHUTOFF (RB NE, El 565 ft). ____

END OF TEXT

8.17 Manual Bypass of the Rod Worth Minimizer

[1] **VERIFY** the following initial conditions are satisfied:

- The Shift Manager/Reactor Engineer has directed Rod Worth Minimizer to be bypassed. ☐
- A second licensed operator is available to verify control rod position. ☐

[2] **REVIEW** all Precautions and Limitations in Section 3.4. ☐

CAUTIONS

1) Step 8.17[3] will make the Rod Worth Minimizer inoperable and Technical Specifications Sections 3.1.6 and 3.3.2.1 will apply.

2) [QA/C] NPG-SPP-10.4 requires approval of the Plant Manager or his designee prior to any planned operation with the RWM bypassed unless bypassing of the RWM is specifically allowed within approved procedures. [ISE-NPS-92-R01]

[3] **PLACE** RWM SWITCH PANEL, 2-XS-85-9025, in BYPASS. ☐

[4] **CHECK** Manual Bypass light illuminated. ☐

[5] **CHECK** all other indications on Rod Worth Minimizer Operator's Panel extinguished. ☐

[6] **CHECK** Blue Rod Out Permit light above 2 HS 85 48 illuminated. ☐

[7] **RESET** CONTROL ROD WITHDRAWAL BLOCK annunciator, (2 XA 55 5A, Window 7). ☐

3.8 Rod Worth Minimizer (RWM) (continued)

I. A withdraw error occurs if:

1. A rod in the currently latched group is withdrawn past the withdraw limit for the group.

2. A rod in a group lower than the one currently latched is withdrawn past the withdraw limit for its group.

3. A rod in a group higher than the one currently latched is withdrawn past the insert limit for its group.

J. A select error occurs if:

1. With the reactor operating below the LPAP, a rod other than one contained in the currently latched group is selected, unless conditions for latching up or down are met.

2. With a rod block applied, any rod other than an error rod is selected.

3. When operating in the Sequence Control Mode, a rod is skipped.

K. An insert block occurs if:

1. With two insert errors existing, a rod is moved to cause a third insert error.

2. A withdraw error has been made, a withdraw block applied, and a rod other than the withdraw error rod is selected.

L. A withdraw block occurs if:

1. A withdraw error is made.

2. With three insert errors existing and an insert block present, a rod other than one of the insert errors is selected.

M. A select block occurs if:

1. The RWM Bypass Switch is in normal and the RWM program is <u>not</u> running; i.e., following return to normal from bypass and the program has <u>not</u> been initialized.

2. The RWM Bypass Switch is in normal and the program stops due to software error.

QUESTION 24

Unit 1 was operating at 100% power when a complete steam line rupture in the Turbine Building occurred.

During the transient the following parameters were reached:

- RPV Level (-)30 inches
- RPV Pressure 1135 psig
- RCIC STEAM LINE LEAK DETECTION TEMP HIGH 1-TA-71-41 in alarm
- RCIC room temperature on 1-TE-071-0041 at 167° F
- All other RCIC area temperatures reading between 130° F to 135° F (Peak values)

Currently, the following conditions exist

- Drywell temperature 143° F
- Drywell pressure 2.6 psig
- Suppression pool level (-)1.5 inches
- Suppression pool temperature is 107° F
- RPV Level is (+)30 inches with HPCI injecting to the vessel
- RPV Pressure is being controlled 800 to 1000 psig with the SRVs
- PCIS Groups 2,3,5, 6, and 8 are isolated

Which ONE of the following completes the statement?

Based on these conditions, ___(1)___ and EOIs 1,___(2)___ are being executed.

A. (1) the Group 5 isolation was spurious
(2) 2, and 3

B. (1) the Group 5 isolation was spurious
(2) and 2 ONLY

C. (1) all PCIS isolations occurred as expected
(2) 2, and 3

D. (1) all PCIS isolations occurred as expected
(2) and 2 ONLY

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295020 G2.4.4	
	Importance Rating	4.5	

INADVERTENT CONTAINMENT ISOLATION
G2.4.4 Ability to recognize abnormal conditions for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures

Explanation: **C** CORRECT – Group 5 is noted as 1 out 2 twice logic but this is actually for the TS not TE and there is a TS from each TE in each logic train. A single high TE should also see all 4 Group 5 channels produce an isolation signal. EOI-2 is entered on High SP Temp and Level. EOI-3 is entered on High RCIC area temp

A Incorrect – First Part: Incorrect, the Group 5 is noted as 1 out 2 twice logic but this is actually for the TS not TE and there is a TS from each TE in each logic train. A single high TE should also see all 4 Group 5 channels produce an isolation signal. Plausible because if the candidate believes that the 1 out of 2 twice logic refers to the TE then this will be chosen. Second Part: Correct.

B Incorrect – First Part: Incorrect, the Group 5 is noted as 1 out 2 twice logic but this is actually for the TS not TE and there is a TS from each TE in each logic train. A single high TE should also see all 4 Group 5 channels produce an isolation signal. Plausible because if the candidate believes that the 1 out of 2 twice logic refers to the TE then this will be chosen. Second Part: Incorrect, if the candidate believes RCIC area temp is spurious, then he/she will not enter EOI-3 or may believe it is not necessary after the isolation.

D Incorrect –First Part: Correct. Second Part: Incorrect, if the candidate believes RCIC area temp is spurious, then he/she will not enter EOI-3 or may believe it is not necessary after the isolation.

Technical Reference(s): 1-EOI-2, 1-EOI-3, 1-OI-71, OPL171.040

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.040 V.B.4.b

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

3.0 PRECAUTIONS AND LIMITATIONS

A. Turbine controls provide for automatic shut down of the RCIC turbine upon receiving any of the following signals: (**REFER TO** Section 8.4 for auto actions)

1. High RPV water level (+51 in.); 579 in. above vessel zero (RCIC TURB STM SUP VLV, 1-FCV-071-0008, and RCIC PMP MIN FLOW VLV, 1-FCV-071-0034, will close at +51 in. and will reopen when RCIC re-initiates at -45 in. RPV water level).
2. Turbine overspeed (Mechanical, 122.3% of rated speed).
3. Pump low suction pressure (10 inches HG vacuum).
4. Turbine high exhaust pressure (50 psig).
5. Any isolation signal.
6. Remote manual trip (RCIC TURBINE TRIP pushbutton, 1-HS-71-9A, depressed).

B. RCIC turbine steam supply will isolate from the following signals: (REFER TO 1-AOI-64-2c for auto actions)

1. RCIC steamline space torus area temperature at 165°F.
2. RCIC steamline space pump room temperature at 165°F.
3. RCIC turbine high steam flow (150% flow, 3-second time delay).
4. RCIC turbine steam line low pressure (approximately 60 psig).
5. RCIC turbine exhaust diaphragms ruptured (10 psig).
6. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION pushbutton, 1-HS-71-54, depressed, only if RCIC initiation signal is present).

C. The RCIC turbine will auto initiate on RPV Low-Low Water Level, -45 in. (**REFER TO** Section 5.1 for auto actions)

D. With a RCIC initiation signal present, RCIC PMP MIN FLOW VLV, 1-FCV-071-0034, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve does not auto open on low flow if an initiation signal is not present.

E. RCIC PMP MIN FLOW VLV, 1-FCV-071-0034, opens on receipt of an initiation signal even with RCIC turbine manually tripped, resulting in slowly draining CST to Suppression Chamber.

5. System Steam Supply Isolation

TP-10 & 11
Obj. V.D.5/V.E.7
Obj. V.B.4.b.
Obj. V.C.2.b.

a. Signals

(1) Low reactor pressure (U2 – 73 psig; U1&3 – 60 psig)

HPE TRAP! Pay attention to detail
UNIT DIFFERENCE!

Protects against gland seal leakage to RCIC Room when RCIC pump not capable of supplying cooling water to barometric condenser

DCN 62557A replaced 2-PS-71-1A/1B/1C/1D with new style PS that had greater inst errors
1 out of 2 twice

(2) Steam line high flow (150% of normal, 421" water gauge WP after 3-second time delay) Protects against a steam line break (equates to ~15.2 psid)

1 out of 2, In both logics

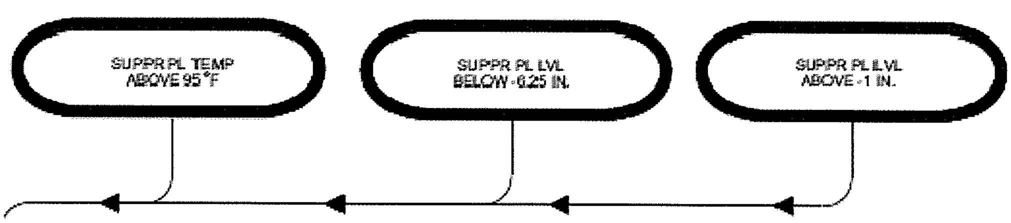
(3) High turbine exhaust diaphragm pressure (10 psig) Steam in RCIC Room from diaphragm rupture

1 out of 2 twice

(4) High temperature in steam space (147°F Torus Area, 160°F pump room) Protects against a steam line break.

1 out of 2 twice, In either logic

(5) Manual pushbutton provided on Panel 9-3 to allow operator to isolate RCIC with a valid initiation signal present.



ANY SECONDARY CNTMT AREA TEMP ABOVE MAX NORMAL VALUE OF TABLE 3

TABLE 3
SECONDARY CNTMT AREA TEMP

AREA	PANEL 9-3 ALARM WINDOW (UNLESS NOTED)	PANEL 9-21 TEMP ELEMENT (UNLESS NOTED)	MAX NORMAL VALUE °F	MAX SAFE VALUE °F	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	XA-55-3E-4	74-95A	ALARMED	215	FCV-74-47, 48
RHR SYS II PUMPS	XA-55-3E-4	74-95B	ALARMED	150	FCV-74-47, 48
HPCI ROOM	XA-55-3F-10	73-55A	ALARMED	275	FCV-73-2, 3, 44, 81
CS SYS I PUMPS RCIC ROOM	XA-55-3D-10	71-41A	ALARMED	190	FCV-71-2, 3, 39
TOP OF TORUS	XA-55-3D-10	71-41B, C, D	ALARMED	195	FCV-71-2, 3
	XA-55-3F-10	73-55B, C, D	ALARMED	245	FCV-73-2, 3, 81
	XA-55-3E-4	74-95G	ALARMED	190	FCV-74-47, 48
	XA-55-3E-4	74-95H	ALARMED	245	FCV-73-2, 3, 81
STEAM TUNNEL (RB)	XA-55-3D-24	1-60A (PANEL 9-3)	ALARMED	315	MSIVs FCV-71-2, 3, FCV-69-1, 2, 12
DW ACCESS	XA-55-3E-4	74-95E	ALARMED	175	FCV-74-47, 48

QUESTION 25

Which ONE of the following completes the statement?

If the RHR Loop II Pump Room exceeds the Maximum Safe Operating Temperature, action is required because _______________.

A. personnel access necessary for the continued safe OPERATION of the plant will be restricted

B. equipment necessary for the safe shutdown of the plant will exceed its environmental qualification and may fail to operate

C. spurious indications of a plant fire may be received and result in automatic actions that would further complicate accident mitigation

D. installed pump room cooling units necessary for heat removal will have exceeded their design heat removal capacity

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295032 EK2.08	
	Importance Rating	3.8	

Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following:
EK2.08 Systems required for safe shutdown

Explanation: **B** CORRECT – Maximum safe operating temperature is defined to be the highest temperature at which neither: 1) equipment necessary for the safe shutdown of the plant will fail, nor 2) personnel access necessary for safe shutdown of the plant will be prevented.

A Incorrect – Plausible if the candidate believes Max safe for continued personnel access for operation of the plant vice safe shutdown.

C Incorrect – Plausible given the emphasis on the SSIs the candidate may confuse the basis with the concern for a fire.

D Incorrect –Plausible because a concern over support systems could cause the supported system to fail to meet it design service time.

Technical Reference(s): EOI-3 bases

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New	
Question History:	Previous NRC Perry 2002 #37	
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

OPL171.204, EOI-3 SECONDARY CONTAINMENT CONTROL

Maximum safe operating temperature is defined to be the highest temperature at which neither: 1) equipment necessary for the safe shutdown of the plant will fail, nor 2) personnel access necessary for safe shutdown of the plant will be prevented. The maximum safe operating temperature value for all secondary containment areas is provided in Table 3, Secondary Containment Area Temperature.

This step is reached only when additional actions have been required to reverse a rising secondary containment area temperature trend. If all secondary containment area temperatures can be maintained below their respective maximum safe operating values, the operator returns to Step SC/T-1. If it is determined that all secondary containment area temperatures cannot be maintained below their respective maximum safe operating values, the operator continues at Step SC/T-7.

QUESTION Common 037

Following entry into PEI-N11, Containment Leakage Control, due to high temperature in the RWCU Pump Room, the room temperature exceeds its Maximum Safe Operating Value.

Which one of the following describes the operational implication of exceeding the Maximum Safe Operating Value in the RWCU Pump Room?

A. Personnel access necessary for the safe operation of the plant will be restricted.

B. Equipment necessary for the safe shutdown of the plant may fail to operate as required.

C. Installed pump room cooling units necessary for heat removal will have exceeded their design heat removal capacity.

D. Automatic isolation of the RWCU System due to RWCU Pump Room high temperature may fail to occur.

ANSWER: B.

QUESTION 26

Following an automatic reactor scram on Unit 2 due to a Loss of Offsite Power (LOOP) the following plant conditions exist:

- RX BLDG AREA RADIATION HIGH 2-RA-90-1D Panel 9-3A Window 22 in alarm
- RB EL 565E CRD-HCU EAST 2-RE-90-21A is reading 450 mR/hr and slowly rising

Which ONE of the following completes the statements?

Based on the above information, the leak is coming from the___(1)___.

EOI-3 directs the operators to attempt to manually isolate the ___(2)___.

A. (1) Scram Discharge Volume
(2) Scram Discharge Volume vents and drains, AND enter 2-EOI-1, RPV CONTROL

B. (1) Reactor Water Cleanup System
(2) Reactor Water Cleanup System ONLY

C. (1) Scram Discharge Volume
(2) Scram Discharge Volume vents and drains ONLY

D. (1) Reactor Water Cleanup System
(2) Reactor Water Cleanup System, AND enter 2-EOI-1, RPV CONTROL

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295033 EA1.05	
	Importance Rating	3.9	

Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS:
EA1.05 Affected system so as to be able to isolate damaged portions

Explanation: **C** CORRECT – EOI-3 will direct the Isolation of all systems discharging into the area except the systems required to fight a fire or directed by EOI's. Based on the annunciators in alarm and the location of the radiation elements associated, the discharging system can be determined to be the Scram Discharge Volume. EOI-1 will only be required to be entered if the Emergency Depressurization will reduce the discharge into secondary containment AND BEFORE ANY area reaches MAX SAFE. The radiation given in the system is steady and not approaching MAX SAFE, therefore entry into EOI-1 is NOT required.

A. Incorrect – First Part: Correct. Second Part: Incorrect, the radiation given in the system is steady and not approaching MAX SAFE, therefore entry into EOI-1 is NOT required. Plausible if the candidate does not know the MAX SAFE Value, or the criteria in EOI-3 flowchart for entry into EOI-1.

B. Incorrect – First Part: Incorrect. Plausible, because if the Rad alarm was for the west side with a high temp alarm then it could be from the RWCU system. Entering 2-EOI-1 is directed when one area reaches a MAX SAFE value.

D. Incorrect – First Part: Incorrect. Plausible, because if the Rad alarm was for the west side with a high temp alarm then it could be from the RWCU system. Second Part: Incorrect, the radiation given in the system is steady and not approaching MAX SAFE, therefore entry into EOI-1 is NOT required. Plausible if the candidate does not know the MAX SAFE Value, or the criteria in EOI-3 flowchart for entry into EOI-1.

Technical Reference(s): 2-ARP-9-3A, 2-EOI-3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New

Question History: Previous NRC Brunswick 2010 #58

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

10 CFR Part 55 Content: 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility

RX BLDG AREA RADIATION HIGH 2-RA-90-1D 22

(Page 1 of 2)

Sensor/Trip Point:

RI-90-4A	RI-90-24A	For setpoints REFER TO 2-SIMI-90B.
RI-90-9A	RI-90-25A	
RI-90-13A	RI-90-26A	
RI-90-14A	RI-90-27A	
RI-90-20A	RI-90-28A	
RI-90-21A	RI-90-30A	
RI-90-R22A	RI-90-29A	
RI-90-23A		

Sensor Location:

RE-90-4	MG set area	Rx Bldg El 639' R-10 S-LINE
RE-90-9	Clean-up System	Rx Bldg El 621' R-9 T-LINE
RE-90-13	North Clean-up Sys	Rx Bldg El 593' R-9 P-LINE
RE-90-14	South Clean-up Sys	Rx Bldg El 593' R-9 S-LINE
RE-90-20	CRD-HCU West	Rx Bldg El 565' R-9 R-LINE
RE-90-21	CRD-HCU East	Rx Bldg El 565' R-13 R-LINE
RE-90-22	TIP Room	Rx Bldg El 565' R-12 P-LINE
RE-90-23	TIP Drive	Rx Bldg El 565' R-12 P-LINE
RE-90-24	HPCI Room*	Rx Bldg El 519' R-14 U-LINE
RE-90-25	RHR West	Rx Bldg El 519' R-8 U-LINE
RE-90-26	Core Spray-RCIC	Rx Bldg El 519' R-9 N-LINE
RE-90-27	Core Spray	Rx Bldg El 519' R-14 N-LINE
RE-90-28	RHR East	Rx Bldg El 519' R-14 U-LINE
RE-90-30	Fuel Storage Pool	Rx Bldg El 664' R-12 P-LINE
RE-90-29	Suppression Pool	Rx Bldg El 519' R-14 U-LINE

Probable Cause:

A. Radiation levels have risen above alarm setpoint.

B. Dry Cask Storage activities in progress (activities could affect rad levels sensed by 2-RE-90-30)

> NOTE
>
> Due to the location of the Rad Monitor in relation to the Test line in the HPCI Quad, the HPCI Room Rad Alarm may be received when the HPCI Flow test is in progress.

C. HPCI Flow Rate Surveillance in Progress.

Automatic Action: None

Continued on Next Page

RX BLDG AREA RADIATION HIGH 2-RA-90-1D, Window 22
(Page 2 of 2)

Operator Action:

A. **DETERMINE** area with high radiation level on Panel 2-9-11. (Alarm on Panel 2-9-11 will automatically reset if radiation level lowers below setpoint.) ☐

B. IF Dry Cask storage activities are in progress, THEN **NOTIFY** CASK Supervisor. ☐

C. IFalarm is from the HPCI Room while Flow testing is performed, **NOTIFY** personnel at the HPCI Quad to validate conditions. ☐

D. **NOTIFY** RAD PRO. ☐

E. IF the TSC is NOT manned and a "VALID" radiological condition exists, **THEN**
USE public address system to evacuate area where high radiological conditions exist. ☐

F. IF TSC is manned and "VALID" radiological condition exists, **THEN** **NOTIFY** TSC to evacuate non-essential personnel from affected areas. ☐

G. **MONITOR** other parameters providing input to this annunciator frequently as these parameters will be masked from alarming while this alarm is sealed in. ☐

H. IF a CREV initiation is received, **THEN**
   1. **VERIFY** CREV A(B) Flow is ≥ 2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213) within 5 hours of the CREV initiation. [BFPER 03-017922] ☐
   2. IF CREV A(B) Flow is NOT ≥ 2700 CFM, and ≤ 3300 CFM as indicated on 0-FI-031-7214(7213), **THEN** **PERFORM** the following: (Otherwise N/A) [BFPER 03-017922]
      a. **STOP** the operating CREV per 0-OI-31. ☐
      b. **START** the standby CREV per 0-OI-31. ☐

I. IF alarm is due to malfunction, **REFER TO** 0-OI-55. ☐

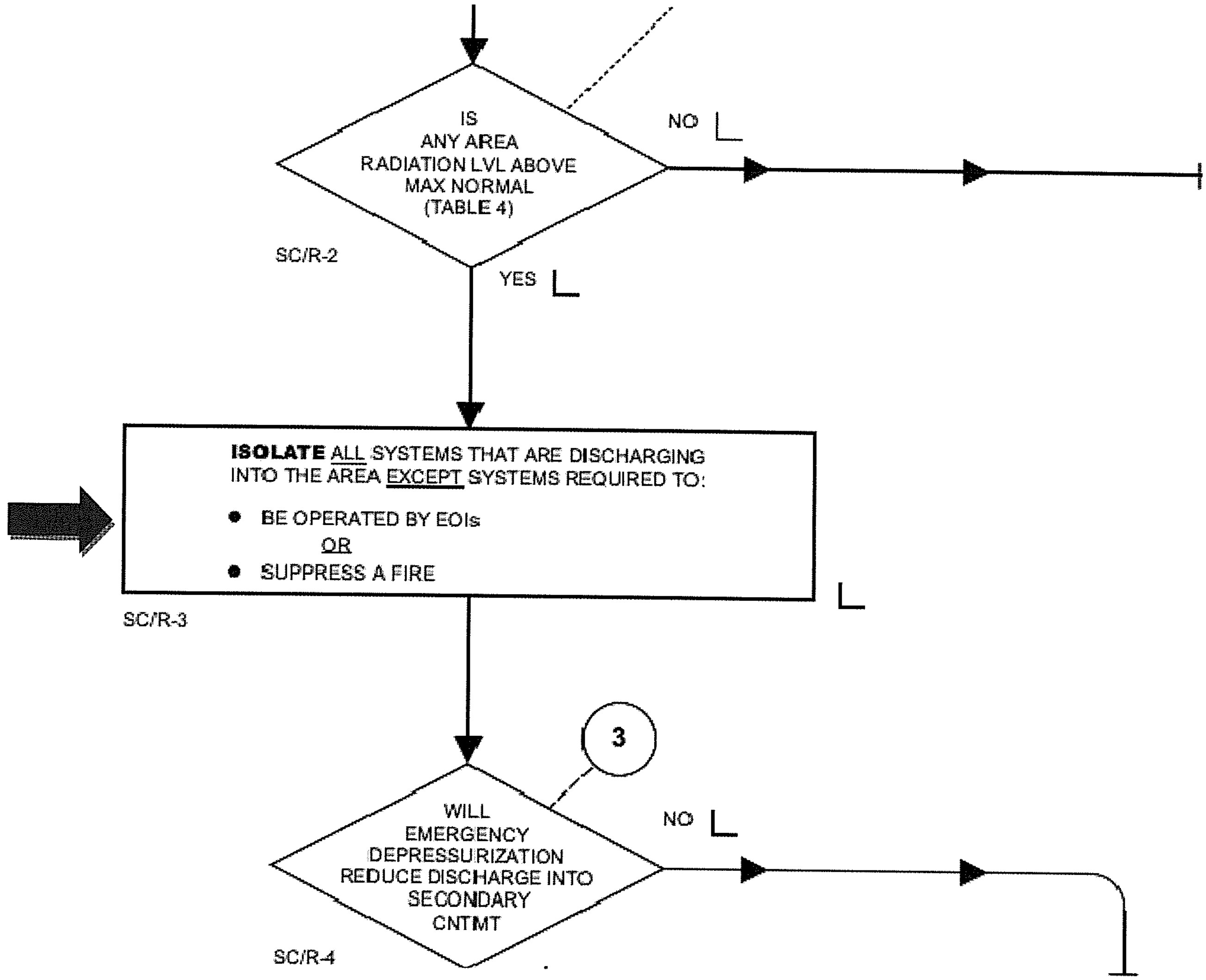
J. For all radiation indicators except FUEL STORAGE POOL radiation indicator, 2-RI-90-30, **ENTER** 2-EOI-3 Flowchart. ☐

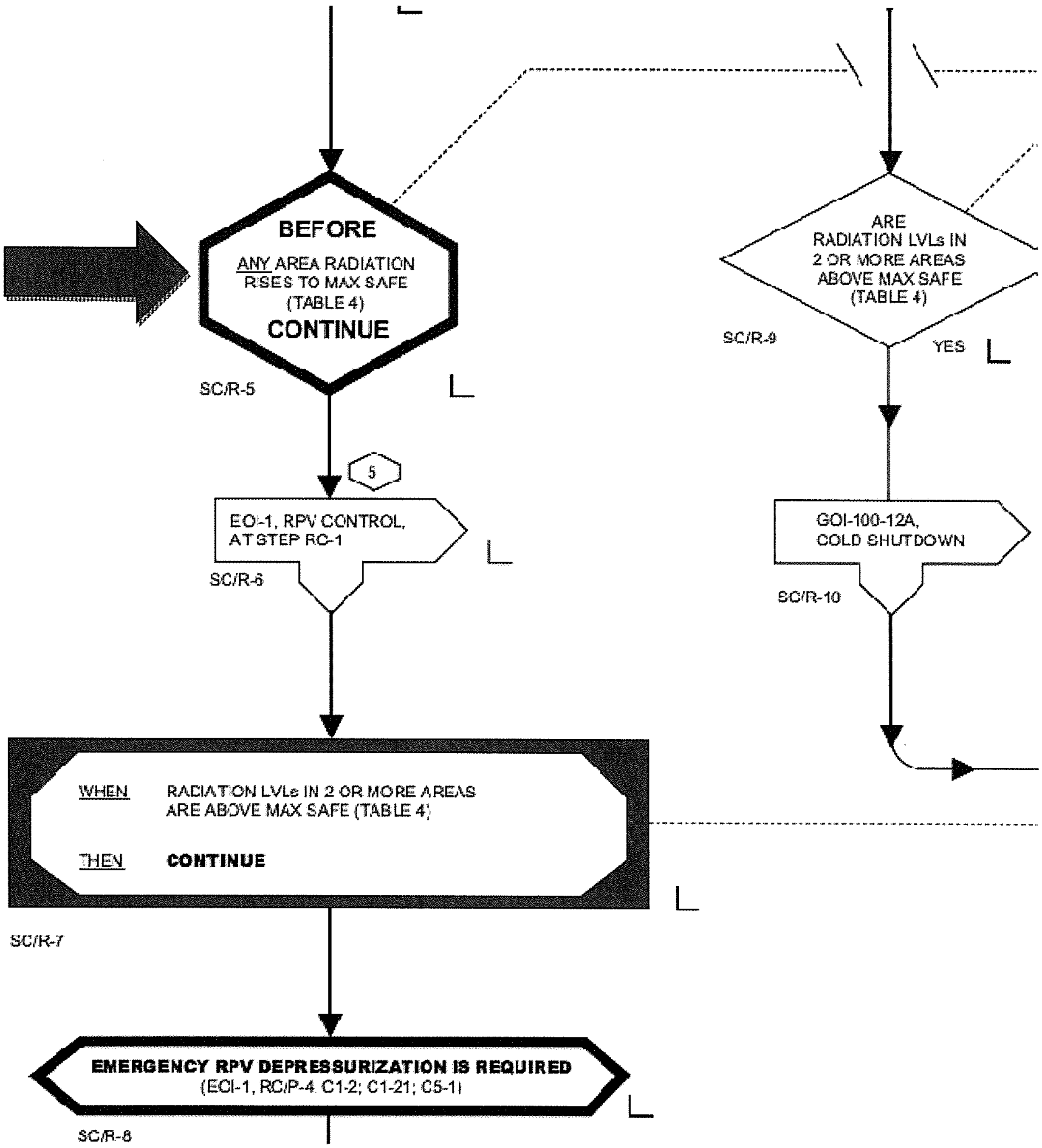
K. **REFER TO** 2-AOI-79-1 or 2-AOI-79-2 if applicable. ☐

References: 2-45E620-3   2-45E610-90-1   GE 0-730E356-1
TVA Calc NDQ00902005008/EDC63693

TABLE 4
SECONDARY CNTMT AREA RADIATION

AREA	APPLICABLE RADIATION INDICATORS	MAX NORMAL VALUE MR/HR	MAX SAFE VALUE MR/HR	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	90-25A	ALARMED	1000	FCV-74-47, 48
RHR SYS II PUMPS	90-28A	ALARMED	1000	FCV-74-47, 48
HPCI ROOM	90-24A	ALARMED	1000	FCV-73-2, 3, 44, 81
CS SYS I PUMPS RCIC ROOM	90-26A	ALARMED	1000	FCV-71-2, 3, 39
CS SYS II PUMPS	90-27A	ALARMED	1000	NONE
TOP OF TORUS GENERAL AREA	90-29A	ALARMED	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB EL 565 W	90-20A	ALARMED	1000	FCV-69-1, 2, 12 SDV VENTS & DRAINS
RB EL 565 E	90-21A	ALARMED	1000	SDV VENTS & DRAINS
RB EL 565 NE	90-23A	ALARMED	1000	NONE
TIP ROOM	90-22A	ALARMED	100,000	TIP BALL VALVE
RB EL 593	90-13A, 14A	ALARMED	1000	FCV-74-47, 48
RB EL 621	90-9A	ALARMED	1000	FCV-43-13, 14
RECIRC MG SETS	90-4A	ALARMED	1000	NONE
REFUEL FLOOR	90-1A, 2A, 3A	ALARMED	1000	NONE


IS
ANY AREA
RADIATION LVL ABOVE
MAX NORMAL
(TABLE 4)
NO
YES
SC/R-2
ISOLATE ALL SYSTEMS THAT ARE DISCHARGING INTO THE AREA EXCEPT SYSTEMS REQUIRED TO:
• BE OPERATED BY EOIs
OR
• SUPPRESS A FIRE
SC/R-3
3
WILL
EMERGENCY
DEPRESSURIZATION
REDUCE DISCHARGE INTO
SECONDARY
CNTMT
NO
SC/R-4


BEFORE
ANY AREA RADIATION RISES TO MAX SAFE (TABLE 4)
CONTINUE
SC/R-5
5
EO-1, RPV CONTROL, AT STEP RC-1
SC/R-6
WHEN RADIATION LVLs IN 2 OR MORE AREAS ARE ABOVE MAX SAFE (TABLE 4)
THEN CONTINUE
SC/R-7
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4 C1-2; C1-21; C5-1)
SC/R-8
ARE RADIATION LVLs IN 2 OR MORE AREAS ABOVE MAX SAFE (TABLE 4)
SC/R-9
YES
GOI-100-12A, COLD SHUTDOWN
SC/R-10

Brunswick 2010 #58

58. 295033 A1.05 002

Following a Reactor Scram on Unit Two due to a loss of Off-site power (LOOP) the following plant conditions exist:

AREA RAD RXBLDG HIGH	In alarm
SOUTH RHR RM FLOOD LEVEL HI	In alarm
SOUTH CS RM FLOOD LEVEL HI	In alarm
Reactor Building 20' Rad Level	Approaching Max Norm Operating Rad
Reactor Building 20' Temperature	Approaching Max Norm Operating Temp

Based on the conditions above which one of the following identifies:

(1)the source of the leak and

(2) the operator action required lAW SCCP?

A. (1)SDV

(2) Open seven ADS valves

B. (1)RBCCW

(2) Open seven ADS valves

C (1) SDV

(2) Cooldown within Technical Specification limits

D. (1)RBCCW

(2) Cooldown within Technical Specification limits

QUESTION 27

A LOCA has occurred on Unit 2 and the following conditions exist:

- RPV Level is (-)90 inches and steady
- RPV Pressure is 75 psig
- Core Spray Loop II is injecting at rated flow
- No other sources of injection are available
- Panel 9-4C Window 31 CORE SPRAY LOOP II PUMP ROOM FLOOD LEVEL HIGH 2-LA-77-25B is in alarm
- The leak is from the 2B Core Spray Pump discharge

Which ONE of the following completes the statement?

The operators are directed to ___(1)___ and ___(2)___.

A. (1) MANUALLY start the floor drain sump pumps
(2) SECURE 2B Core Spray Pump

B. (1) MANUALLY start the floor drain sump pumps
(2) CONTINUE to inject with both pumps to maintain RPV level

C. (1) VERIFY the floor drain sump pumps are running
(2) SECURE 2B Core Spray Pump

D. (1) VERIFY the floor drain sump pumps are running
(2) CONTINUE to inject with both pumps to maintain RPV level

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295036 EA1.01	
	Importance Rating	3.2	

Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:
EA1.01 Secondary containment equipment and floor drain systems

Explanation: **D** CORRECT –Since the PUMP ROOM FLOODED annunciator is in, the water level in the room is above 2 inches, the sump pumps should be running so the operator should verify they are running. While the 2B CS Pump is the source of the leak, since water level is constant and no other injection sources are available, pump operation should continue as required by the EOIs to maintain RPV level.

A Incorrect – Plausible if the candidate believes the sump pumps do not receive an auto start signal and that the CS pump should be secured as the source of the leak per step SC/L-9

B Incorrect – See A for Sump Pump operation

C Incorrect –See A for CS pump operation

Technical Reference(s): 2-ARP-9-4C, 2-EOI-3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New

Question History: Previous NRC: Nine Mile Point 2 2010 #62

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

10 CFR Part 55 Content: 55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility

CORE SPRAY LOOP II PUMP ROOM FLOOD LEVEL HIGH 2-LA-77-25B 31

(Page 1 of 1)

Sensor/Trip Point:

2-LS-77-25B ≥ 2 inches of water on the floor

Sensor Location: Sensor is located near the floor of the northeast Core Spray pump room.

Probable Cause: Greater than two inches of water on the floor.

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to visually check the northeast Core Spray pump room. ☐

B. **IF** alarm is valid, **THEN**
PERFORM the following: ☐
- **VERIFY** the floor drain sump pumps running. ☐
- **VERIFY** the floor drains for proper drainage. ☐
- **IF** possible, **THEN**
DETERMINE the source of the leak and the leak rate. ☐
- **ENTER** 2-EOI-3 FLOWCHART. ☐

IS
ANY AREA WATER LVL
ABOVE
2 IN.

NO

SC/L-6

YES

RESTORE AND **MAINTAIN** AREA WATER LVLs BELOW 2 IN.
USING ALL AVAILABLE SUMP PUMPS

SC/L-7

CAN ALL
AREA WATER LVLs BE
RESTORED AND MAINTAINED
BELOW
2 IN.

NO

SC/L-8

YES

SC/L-8

ISOLATE ALL SYSTEMS THAT ARE DISCHARGING
INTO THE AREA EXCEPT SYSTEMS REQUIRED TO:

- BE OPERATED BY EOIs

  OR
- SUPPRESS A FIRE

SC/L-9

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295036 EA1.01	
	Importance Rating	3.2	

Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Secondary containment equipment and floor drain systems

Question: RO #62

A LOCA has occurred with the following plant conditions:

- RPV Level is 20 inches and rising slowly
- Reactor Pressure is 40 psig and steady
- LPCS is injecting at design flow rate
- No other sources of injection are available
- The Reactor Building sump reaches the High-High setpoint
- The LPCS Pump is the source of the leak

Which one of the following is the required action regarding the LPCS Pump?

A. Continue to inject with the pump

B. Isolate the pump when annunciator 601411, LPCS PUMP ROOM FLOODING, alarms

C. Isolate the pump when LPCS area water level exceeds the Max Normal Operating Value

D. Isolate the pump when two area water levels exceed the Max Normal Operating Values

Answer: A

QUESTION 28

A LOCA has occurred on Unit 3. The lowest level reached during the transient was (-)190 inches.

Currently, the following conditions exist:

- RPV Level is 33 inches and slowly rising
- RPV Pressure is 200 psig
- Drywell Pressure is 8 psig
- RHR Loop I is currently injecting with two (2) pumps at 17,000
- No other system is injecting into the RPV

Which ONE of the following is (are) the MINIMUM operator action(s) necessary to reduce RHR flow and stabilize level per 3-EOI APPENDIX 6B, INJECTION SUBSYSTEMS LINEUP RHR SYSTEM I LPCI MODE?

A. Throttle 3-FCV-74-52, RHR SYS I LPCI OUTBD INJECT VALVE ONLY

B. Place 3-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS, and throttle 3-FCV-74-52, RHR SYS I LPCI OUTBD INJECT VALVE ONLY

C. Place 3-XS-74-122, RHR SYS I LPCI 2/3 CORE HEIGHT OVRD in MANUAL OVERRIDE, and throttle 3-FCV-74-52, RHR SYS I LPCI OUTBD INJECT VALVE ONLY

D. Place 3-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS, place 3-XS-74-122, RHR SYS I LPCI 2/3 CORE HEIGHT OVRD in MANUAL OVERRIDE, and throttle 3-FCV-74-52, RHR SYS I LPCI OUTBD INJECT VALVE

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	203000 A2.16	
	Importance Rating	4.4	

Ability(a) predict the impacts of the following on the RHR/LPCI INJECTION MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.16 Loss of coolant accident

Explanation: B CORRECT – Placing the select switch in bypass allows the operator to throttle flow

A Incorrect – Plausible if the candidate doesn't fully understand the valve logic. The LOCA initiation signal prevents the operator from taking control of the injection valve. The injection signal must be overridden.

C Incorrect – Plausible if the candidate doesn't fully understand the valve logic. The 2/3 core height override must be used to maintain RHR operation in other than LPCI mode when level is low. It is not necessary for throttling injection.

D Incorrect – Plausible if the candidate doesn't fully understand the valve logic. The 2/3 core height override must be used to maintain RHR operation in other than LPCI mode when level is low. It is not necessary for throttling injection.

Technical Reference(s): 3-EOI APPENDIX 6B

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis:	X

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

3-EOI APPENDIX-6B

INJECTION SUBSYSTEMS LINEUP
RHR SYSTEM I LPCI MODE

LOCATION: Unit 3 Control Room

ATTACHMENTS: 1. NPSH Monitoring

(√)

1. IF..............Adequate core cooling is assured,

   AND

   It becomes necessary to bypass the LPCI injection valve auto open signal to control injection,

   THEN.......**PLACE** 3-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in **BYPASS**. ____

QUESTION 29

Unit 2 has been scrammed and a cooldown to cold shutdown is in progress. The RPV DRAIN TO RWCU, 2-DRV-010-0505 is OPEN.

The following data has been recorded:

At 1600:

Reactor Pressure	640 psig
Rx Vessel Drain Temp ICS Point ID 56-8	460°F
Rx Vessel Head Temp ICS Point ID 56-1	531°F

At 1630:

Reactor Pressure	450 psig
Rx Vessel Drain Temp ICS Point ID 56-8	445°F
Rx Vessel Head Temp ICS Point ID 56-1	500°F

Which ONE of the following is the calculated cooldown RATE from 1600 to 1630?

[REFERENCE PROVIDED]

A. 30°F /HR

B. 36°F/HR

C. 62°F/HR

D. 72°F/HR

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	205000 A4.07	
	Importance Rating	3.7	

SHUTDOWN COOLING SYSTEM
Ability to manually operate and or monitor in the control room
A4.07 Reactor temperatures (moderator, vessel, flange)

Explanation: **D** CORRECT – 2-SR-3.4.9.1(1) requires recording the steam dome pressure and the temperature of the RPV D rain To RWCU data. The steam dome pressure is converted to temperature and the change in temperatures are both checked to ensure that the data has not changed by >100° F /HR

A Incorrect – Plausible if the operator believes that only the RPV Drain to RWCU is used to calculate the cooldown rate

B Incorrect – Plausible if the operator converts the pressures to temperature but does not multiply by 2 to get the hourly rather than half hour cooldown rate.

C Incorrect –Plausible if the operator uses the change in the vessel head temperature

Technical Reference(s): 2-SR-3.4.9.1(1)

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

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	X
	New	

Question History: Previous NRC: Duane Arnold 2007 #2

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

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

2-SR-3.4.9.1(1)

[5] RECORD data on Attachment 2, Reactor Heatup AND Cooldown Rate Monitoring in Section 8.0 beginning 15 minutes prior to control rod withdrawal for the purpose of achieving criticality as applicable, (applicable only during Mode 2 operation) AND every 30 minutes during RCS Heatup AND Cooldown, using the guidelines below:

[5.1] RECORD the time of day (military time preferred) on the top data line. ________

[5.2] CONVERT steam dome pressure to equivalent saturation temperature using Table 2 or use ICS. ________

[5.3] IF 2-DRV-010-0505, RPV DRAIN TO RWCU, is OPEN, THEN RECORD the Reactor Vessel Drain Line Temperaturefrom the following, OTHERWISE N/A this step: ________ REACTOR VESSEL DRAIN TEMP, Integrated Computer System (ICS) Point ID 56-8

OR

At Panel 2-9-47, 2-TR-56-4, RX VESSEL FLANGE DRAIN LINE, Point 17 (2-TE-56-8)

[5.4] After each set of readings,

REVIEW the data to ensure that the temperature for each data point has NOT changed by >100°F per hour,

Table 2
(Page 2 of 2)

Saturation Temperature vs Saturation Pressure for Demineralized Water

NOTE

Enter this table with the closest pressure greater than **OR** equal to the steam dome pressure for which the conversion is desired (for example, P = 532 psig use 478°F).

PSIG	TEMP. °F	PSIG	TEMP. °F	PSIG	TEMP. °F
225	397	530	476	1020	549
230	399	540	478	1040	551
235	400	550	480	1060	553
240	403	560	482	1080	556
245	404	570	483	1100	558
250	406	580	485	1120	560
255	408	590	487	1140	562
260	409	600	489	1160	565
265	411	610	491	1180	566
270	413	620	492		
275	414	630	494		
280	416	640	496		
290	419	650	497		
300	422	660	499		
310	425	670	501		
320	428	680	502		
330	430	690	504		
340	433	700	505		
350	436	710	507		
360	438	720	509		
370	441	730	510		
380	443	740	511		
390	446	750	513		
400	448	760	514		
410	450	780	517		
420	453	800	520		
430	455	820	523		
440	457	840	526		
450	460	860	529		
460	461	880	531		
470	463	900	534		
480	466	920	536		
490	468	940	539		
500	470	960	542		
510	472	980	544		
520	474	1000	546		

QUESTION 30

Which ONE of the following completes the statement?

HPCI is not required by Technical Specifications to be OPERABLE when RPV Pressure is___(1)___ because ____(2)___.

A. (1) ≤150 psig
(2) low pressure ECCS can maintain adequate core cooling

B. (1) ≤105 psig
(2) low pressure ECCS can maintain adequate core cooling

C. (1) ≤150 psig
(2) HPCI stall flow can result in water hammer and equipment damage

D. (1) ≤105psig
(2) HPCI stall flow can result in water hammer and equipment damage

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	206000 G2.2.38	
	Importance Rating	3.6	

HPCI
G2.2.38 Knowledge of conditions and limitations in the facility license

Explanation: **A** CORRECT – At 150 psig neither the SRVs nor HPCI are required due to the capability of the low pressure systems

B Incorrect – First Part: Incorrect. Plausible since HPCI isolates at 105 psig and the candidate may confuse the setpoints. Second Part: Correct.

C Incorrect – First Part: Correct. Second Part: Incorrect. Plausible since many operational concerns are based on water hammer.

D Incorrect – First Part: Incorrect. Plausible since HPCI isolates at 105 psig and the candidate may confuse the setpoints. Second Part: Incorrect. Plausible since many operational concerns are based on water hammer.

Technical Reference(s): TS 3.5.1 and basis

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge : X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

APPLICABILITY All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3, when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, when reactor steam dome pressure is ≤ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS - Shutdown."

QUESTION 31

Unit 1 was operating at 80%. Core Spray pump 1A is running for 1-SR-3.5.1.6, Core Spray Flow Rate Loop I, with the following parameters:

- System flow 3200 gpm
- Discharge pressure 240 psig

During the surveillance, a LOCA occurs on Unit 1 resulting in the following plant conditions:

- Drywell pressure 11 psig
- Reactor water level (-)65 inches
- Reactor pressure 400 psig

Which ONE of the following completes the statement?

The Core Spray Pump 1A Discharge Pressure will ___(1)____.

A. REMAIN THE SAME, Core Spray Pump continues to discharge to the Suppression Pool

B. RISE to just below pump shutoff head, Core Spray Pump flow is through the minimum flow valve

C. RISE to just above reactor pressure, Core Spray Pump is injecting into the Reactor

D. RISE to pump shutoff head, a flowpath is NOT available for the Core Spray Pump

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001 A3.03	
	Importance Rating	3.6	

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: A3.03 System pressure

Explanation: **D CORRECT** – 1-SR-3.5.1.6, Core Spray Flow Rate Loop I, testing establishes 3200 gpm at approximately 240 psig. On an initiation signal the Core Spray Sys I Test Valve, 1-FCV-75-22, goes closed. When the loop flow drops below to <2200 gpm, the minimum flow starts to open. The 1A Core Spray pump runs in this mode until RPV pressure drops below 450 psig when the RPV inject valve opens.

A- Incorrect – Plausible because this would be the correct answer if Core Spray Sys I Test valve did not close on an initiation.

B- Incorrect – Plausible because this would be the correct answer if the Core Spray Sys I Inbd Injection Valve, 1-FCV-75-25, were open and Core Spray was injecting.

C- Incorrect –Plausible because this would be the correct answer with no minimum flow.

Technical Reference(s): 1-OI-75, 1-SR-3.5.1.6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Quad Cities 2009 #31

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

5.1 Automatic Initiation

NOTES

1) ECCS SYS I(II) HI DW PRESS TEST/INHIBIT keylock switch, 1-HS-75-59(60), on Panel 1-9-3 must be in AUTO to allow automatic initiation on high Drywell pressure concurrent with low RPV pressure.
2) Core Spray will auto initiate on any of the following signals:
   - RPV vessel Low-Low-Low water level -122 inches
   - High Drywell pressure, at or above 2.45 psig with low RPV vessel pressure, at or below 450 psig
3) If only one pump in a loop is running, the injection valve for that loop must be throttled to achieve 3125 gpm pump flow to avoid pump run-out.
4) Upon an automatic Core Spray initiation with normal power available, Core Spray Pump 1A will start immediately and 1B, 1C, 1D will then sequentially start at 7-second intervals. Otherwise, all Core Spray pumps will start 7 seconds after diesel power is available.
5) Core Spray Room Coolers auto start on a Core Spray initiation.

[2] **WHEN** Reactor Pressure is less than 450 PSIG, **THEN:**

- RX PRESS LOW CORE SPRAY/RHR PERMISSIVE (1-XA-55-3C, Window 35) alarms. ☐
- If closed, CORE SPRAY SYS I and II OUTBD INJECT VALVE, 1-FCV-75-23 and 1-FCV-75-51, opens. ☐
- CORE SPRAY SYS I and II INBD INJECT VALVE, 1-FCV-75-25 and 1-FCV-75-53, opens. ☐

31 **ID: QDC.ILT.15519** **Points: 1.00**

Unit 1 was operating at power with the "A" Core Spray subsystem operating for its Quarterly System Flow Rate Test per QCOS 1400-01 with the following parameters:

- System flow 4500 gpm
- Discharge pressure 216 psig

During the surveillance, a LOCA occurs on Unit 1 resulting in the following plant conditions:

- Drywell pressure 11 psig
- Reactor water level -65"
- Reactor pressure 400 psig

Which of the following correctly describes the response of "A" Core Spray Pump Discharge Pressure from 216 psig to this point in the accident?

"A" Core Spray Pump Discharge Pressure will...

A. REMAIN THE SAME because the Core Spray Pump continues to discharge to the CCST.

B. INCREASE to pump shutoff head because a flowpath is NOT available for the Core Spray Pump.

C. INCREASE to just above reactor pressure because the Core Spray Pump is injecting into the Reactor.

D. INCREASE to just below pump shutoff head because Core Spray Pump flow is through the minimum flow valve.

Answer: D

QUESTION 32

Which ONE of the following completes the statement regarding the relationship between the Standby Liquid Control system and Core Spray Line Break Detection differential pressure (D/P) instrument?

The ___(1)___ leg of this D/P instrument senses ___(2)___ core plate pressure via the SLC/Core Differential Pressure penetration.

A. (1) low pressure
(2) below

B. (1) low pressure
(2) above

C. (1) high pressure
(2) above

D. (1) high pressure
(2) below

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000 K1.01	
	Importance Rating	3.0	

Knowledge of the physical connections and /or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: K1.01 Core spray line break detection

Explanation: **B CORRECT**- The low side of the detector senses above-core plate pressure plus the pressure due to the height of water in the vessel. Under normal conditions the high side of the detector senses core exit pressure plus pressure due to the height of water in the sensing leg. With the plant operating at rated conditions the detector reads 3.5 psid. If the Core Spray piping breaks between the reactor vessel and the shroud, and the high-side pressure at the detector would decrease , and the sensed low-side pressure will remain the same. This would cause the ΔP to decrease, causing an alarm to sound at 2 psid decreasing (following a 15-sec time delay).

A. Incorrect – First Part: Correct. Second Part: Incorrect. Plausible because the candidate may confuse the pipe-in-a-pipe configuration.

C. Incorrect – First Part: Incorrect. Plausible because the candidate may confuse the low pressure and high pressure legs. Second Part: Correct.

D. Incorrect – First Part: Incorrect. Plausible because the candidate may confuse the low pressure and high pressure legs. Second Part: Incorrect. Plausible because the candidate may confuse the pipe-in-a-pipe configuration.

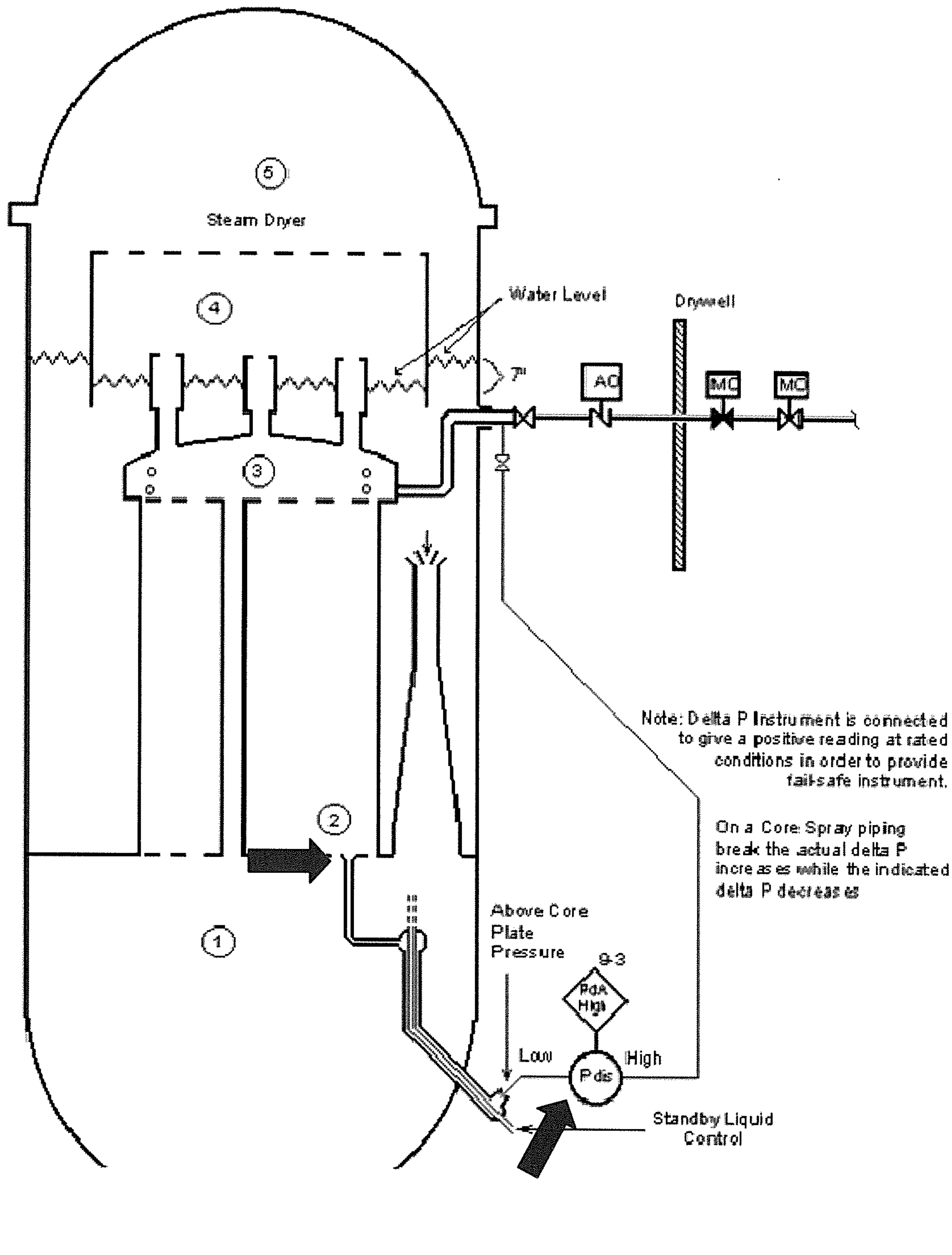
Technical Reference(s): OPL171.045, 1-ARP-9-3C window 14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Brunswick 2010 #9
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (6) Design, components, and function of reactivity control mechanisms and instrumentation.



TP-3: CORE SPRAY PIPE BREAK DETECTION INSTRUMENTATION

CORE SPRAY SYS I SPARGER BREAK 1-PDA-75-28 (14)	Sensor/Trip Point:	
	1-PDIS-075-0028	2 psig lowering ΔP (15 second time delay)

(Page 1 of 1)

Sensor Location: 1-LPNL-925-0057
Rx Bldg, El 565', R-3 S-LINE

Probable Cause:
A. Indication of Core Spray piping break inside primary containment.
B. Low core flow.
C. Sensor malfunction.

Automatic Action: None

Operator Action:
A. **DISPATCH** personnel to 1-LPNL-925-0057 to check CSS SYS I Hi ΔP, 1-PDIS-075-0028. **COMPARE** with CSS SYS II Hi ΔP, 1-PDIS-075-0056, on same panel. (The normal reading should be approximately 3.5 psid.) ☐
B. **IF** necessary, **THEN** **DISPATCH** IMs to VERIFY instrument operation. ☐
C. **IF** there are indications of a broken Core Spray header, **THEN** **CONSIDER** the associated Core Spray system INOPERABLE and **TAKE** appropriate action as required by Tech Spec 3.5.1. ☐
D. **IF** there are no indications of a Core Spray header break, **THEN** **REFER TO** Tech Spec table 3.3.5.1 and TRM 3.3.3.3. ☐

References:

1-45E620-2-1	GE 730E930-2 & -8	47W600-59
1-47E610-75-1	Technical Specifications 3.3.5.1	
	TRM 3.3.3	

Brunswick NRC 2010 #9

9. 211000 K1.01 001

Which one of the following identifies the relationship between the SLC system and Core Spray Line Break Detection differential pressure instrument?

The ___(1)___ leg of this DP instrument senses ___(2)___ core plate pressure via the SLC/Core Differential Pressure penetration.

A. (1) variable
(2) below

B. (1) variable
(2) above

C. (1) reference
(2) below

D✓ (1) reference
(2) above

QUESTION 33

Unit 1 has scrammed and is in an ATWS condition.

- During the performance of 1-EOI-Appendix 3A, SLC INJECTION, the Unit Operator ATC placed the Standby Liquid Control (SLC) pump control switch in START-A.

The following plant conditions exist:

- RPV pressure is 1020 psig.
- SLC discharge pressure is 1100 psig.
- ONLY ONE of the SLC System 1 squib firing circuits has actuated.

Based on these conditions, which ONE of the following describes the capability of the SLC system to shutdown the reactor?

A. SLC is injecting normally at full flow, and reactor shutdown will occur as designed

B. SLC is injecting at reduced flow, and reactor shutdown will occur later than designed

C. SLC pump 1A relief valve has lifted, the 1B SLC pump must be started to shutdown the reactor as designed

D. SLC pump 1A failed to start, the 1B SLC pump must be started to shutdown the reactor as designed

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000 K3.01	
	Importance Rating	4.3	

Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on the following: K3.01 ability to shutdown the reactor under certain conditions

Explanation: A **CORRECT-** Each squib valve contains a primer subassembly. Each subassembly contains redundant primers and firing circuits for high reliability. Both circuits fire on pump start. A squib firing circuit failure will not prevent the other squib firing circuit in the subassembly from opening the squib valve. RPV pressure and SLC pressure parameters are normal for injection.

B. Incorrect- Plausible if the candidate thinks that only one valve fired and can handle only partial system flow.

C. Incorrect- Plausible if the candidate does not recall that the squibs are in parallel and thinks that the A squib did not fire.

D. Incorrect – Plausible if the candidate does not recall that the squibs are in parallel and thinks that the A squib did not fire. Additionally the candidate may think that initiating SLC B may handle only partial system flow.

Technical Reference(s): OPL171.036, 1- EOI APPENDIX 3A

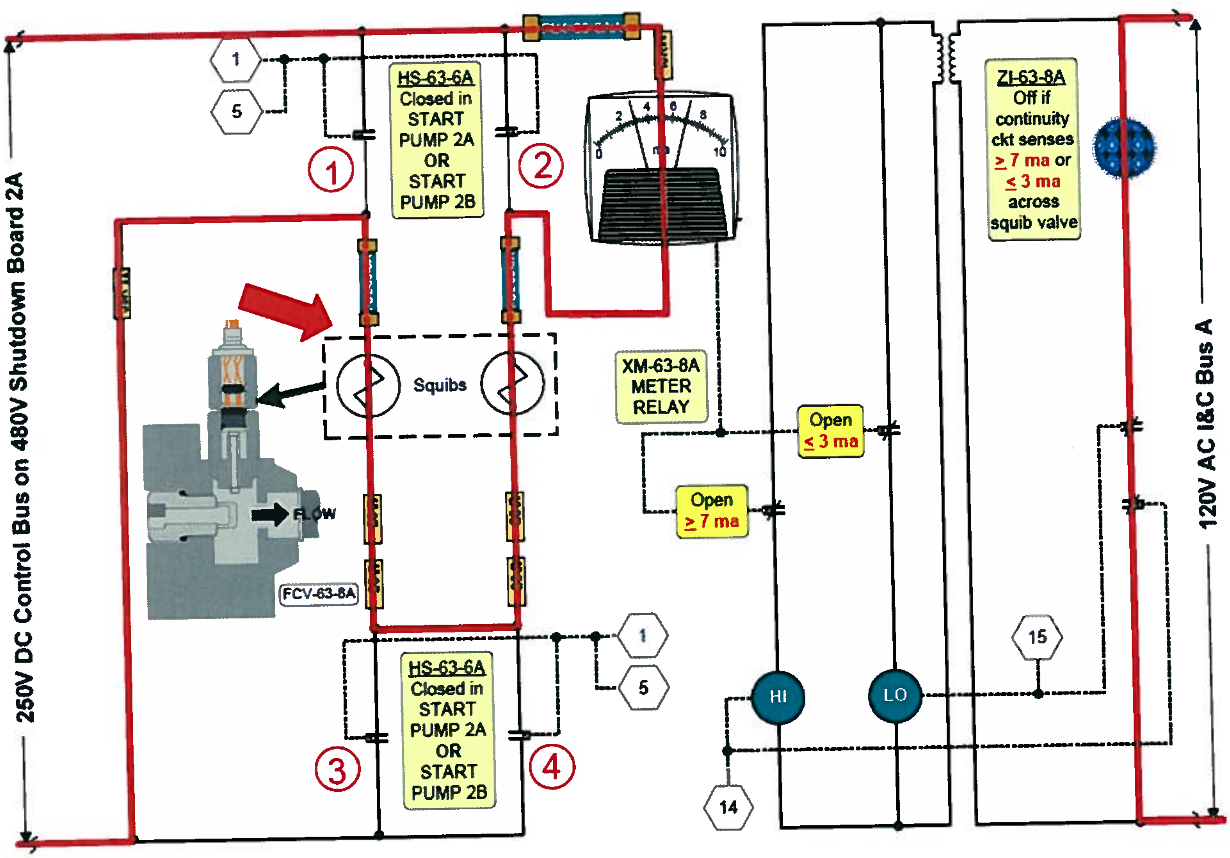
Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Oyster Creek 2010 #23
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	55.41 (6) Design, components, and function of reactivity control mechanisms and instrumentation.

Explosive Squib Valve

Downstream of the pumps, the discharge piping splits into two separate headers; each containing an explosive A(B) SLC SQUIB VALVE (FCV-63-8A and 8B). Each Squib Valve is a 100% capacity, dual-squib (2 primers), shear-plug, designed for zero-leakage. The zero leakage design ensures boron will not leak into the reactor when the pumps are being tested. Two firing squibs are installed in each valve for high reliability in that either squib firing will shear open the valve. Power for each squib firing circuit is via their respective 250V DC Control Power on the 480V Shutdown Boards A and B.



EXAMINATION ANSWER KEY

ILT 10-1 Combined RO & SRO NRC Exam

23 **ID: 10-1 NRO23** **Points: 1.00**

During an ATWS condition, the URO placed the Standby Liquid Control (SLC) System keylock on Panel 4F to FIRE SYS 1. The following plant conditions exist:

- RPV pressure is 1020 psig
- SLC discharge pressure is 1100 psig
- **ONLY ONE** of the SLC System 1 squib firing circuits actuates

Based on these conditions, which statement is correct regarding the capability of the SLC system to inject boron into the reactor?

A. SLC is injecting normally at full flow and reactor shutdown will occur as designed.

B. SLC is injecting at reduced flow and reactor shutdown will occur later than designed.

C. SLC is **NOT** injecting and System 2 must be initiated to shutdown the reactor as designed.

D. SLC is **NOT** injecting and initiating System 2 will **NOT** shutdown the reactor as designed.

Answer: A

QUESTION 34

Unit 3 is operating at 90% and Turbine Stop Valve Closure-RPS Trip Logic System Function Test, 3-SR-3.3.1.1.14 (8I), is being performed.

Main Turbine Stop Valve, MSV-2, is simulated CLOSED.

On panel 3-9-7, the Unit Operator depresses and holds MSV-1 TEST pushbutton 3-HS-47-141 until the MSV-1 position indicator 3-ZI-1-74 indicates 0%.

Which ONE of the following completes the statements?

During this surveillance step, annunciator REACTOR CHANNEL A AUTO SCRAM 3-9-5B (window 1) ___(1)___ alarm

The RPS Trip Logic considers Main Turbine Stop Valve, MSV-1, CLOSED when the valve position is ___(2)___.

A. (1) will NOT
   (2) <90% OPEN

B. (1) will
   (2) < 5% OPEN

C. (1) will
   (2) <90% OPEN

D. (1) will NOT
   (2) <5% OPEN

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000 K4.01	
	Importance Rating	3.4	

212000 Knowledge of the REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 System redundancy and reliability

Explanation: **C CORRECT** –In this step of the surveillance MSV-2 is simulated closed by de-energizing relay RLY-99-5AK10E. When the operator tests MSV-1 and valve position is <90%, relay RLY-99-5AK10A will de-energize and RPS channel A1 will be de-energized. This will cause REACTOR CHANNEL A AUTO SCRAM 3-9-5B (window 1) to annunciate.

A. Incorrect- First Part: Incorrect: When the operator tests MSV-1 and valve position is <90%, relay RLY-99-5AK10A will de-energize and RPS channel A1 will be de-energized. This will cause REACTOR CHANNEL A AUTO SCRAM 3-9-5B (window 1) to annunciate. Plausible if the candidate is unfamiliar with the SR and does not know that MSV-2 is simulated closed. Second Part: Correct.

B. Incorrect- First Part: Correct. Second Part: Incorrect but plausible because MSV-1 slow closes to 5%, then fast closes from <5%.

D. Incorrect- First Part: Incorrect: When the operator tests MSV-1 and valve position is <90%, relay RLY-99-5AK10A will de-energize and RPS channel A1 will be de-energized. This will cause REACTOR CHANNEL A AUTO SCRAM 3-9-5B (window 1) to annunciate. Plausible if the candidate is unfamiliar with the SR and does not know that MSV-2 is simulated closed. Second Part: Incorrect but plausible because MSV-1 slow closes to 5%, then fast closes from <5%.

Technical Reference(s): 3-OI-99, 3-SR-3.3.1.14(8I)

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (6) Design, components, and function of reactivity control mechanisms and instrumentation.

D. Both RPS divisions systems are affected by this procedure.

1. Half Scrams for the A RPS System uses Limit Switches on each Turbine Stop Valve as inputs to both the scram circuits. Both relays in each system must be DE-ENERGIZED to cause a half scrams. (ICS Point SOE035 for Reactor Trip Actuator A1 or A2.)

ICS PT	Bus	Relay	Limit Switch for the following valves (<90% Open)		Fuse
SOE038	A1	RLY-99-5AK10A	SV-1	(ZS-1-74F)	FU1-1-74A
		RLY-99-5AK10E	SV-2	(ZS-1-78F)	FU1-1-78A
SOE040	A2	RLY-99-5AK10C	SV-3	(ZS-1-84F)	FU1-1-84A
		RLY-99-5AK10G	SV-4	(ZS-1-88F)	FU1-1-88A

6.0 ACCEPTANCE CRITERIA

A. Responses which fail to meet the following acceptance criteria constitute unsatisfactory surveillance procedure results and require immediate notification of the Unit Supervisor at the time of failure.

1. A RPS Half Scram signal from Channel A1 is generated by test closing Turbine Stop Valve MSV-1 and simulating closed MSV-2.

2. A RPS Half Scram signal from Channel A2 is generated by test closing Turbine Stop Valve MSV-4 and simulating closed MSV-3.

3. An RPT Trip System A signal is generated by test closing MSV-1 and simulating closed MSV-2 during the performance of this procedure.

4. An RPT Trip System B signal is generated by test closing MSV-4 and simulating closed MSV-3 during the performance of this procedure.

5. A RPS Half Scram signal will extinguish SCRAM SOLENOID GROUP A LOGIC RESET 1, 2, 3 and 4 indicating lights (4) on Panel 3-9-5.

Date ____________

NOTES

1) Steps 7.4.2[1] through 7.4.2[5] should be performed expeditiously to minimize steam system perturbations.
2) TSVs will slow close for the first 95% of travel and fast close for the last 5% of travel.
3) Step 7.4.2[1] thru Step 7.4.2[5] data and sign off(s) may be performed after the completion of Step 7.4.2[5].

7.4.2 Stroke Testing of 3-FCV-1-74

[1] On Panel 3-9-7

SIMULTANEOUSLY PERFORM the following:

- **DEPRESS** and **HOLD** MSV-1 TEST push-button, 3-HS-47-141 until Step 7.4.2[5]

  AND

- **START** the stopwatch. ________

[2] On Panel 3-9-7

WHEN MSV-1 position indicator, 3-ZI-1-74 reads 0%, **THEN**

STOP the stopwatch. ________

[3] On Panel 3-9-17, Bay 1

- [NRC/C] **CHECK** 3-RLY-099-05AK10B, RPS CH B1 TURB STOP VLV SV-1 CLOSURE is DE-ENERGIZED. (LER 50-260/93008). ________
- **CHECK** 3-IL-099-5A-DS16B, SYS A TURBINE STOP VLV indicating light is EXTINGUISHED. ________

[4] On Panel 3-9-15, Bay 1

CHECK 3-RLY-099-05AK10A, RPS CH A1 TURB STOP VLV SV-1 CLOSURE relay DE-ENERGIZED. ________

[5] **RELEASE** MSV-1 TEST push-button, 3-HS-47-141. ________

[6] **CHECK** MSV-1 TEST valve returns to OPEN position as indicated on 3-ZI-1-74. ________

Illustration 3
(Page 6 of 17)

Actions to Place RPS Instruments in Tripped Condition (Tech Spec Table 3.3.1.1-1)

CAUTION

This table was written for the removal of one fuse. If removing two fuses, research the logic carefully for effects on RPS and RPT's. If two fuses are removed, one fuse in an "A" channel and one fuse in a B channel, and a Rx scram and turbine trip occur while these fuses are removed, they must be reinstalled in order to reset the scram and RPT logic.

NOTE

Device Function corresponds to the Tech Spec Table 3.3.1.1 Functions.

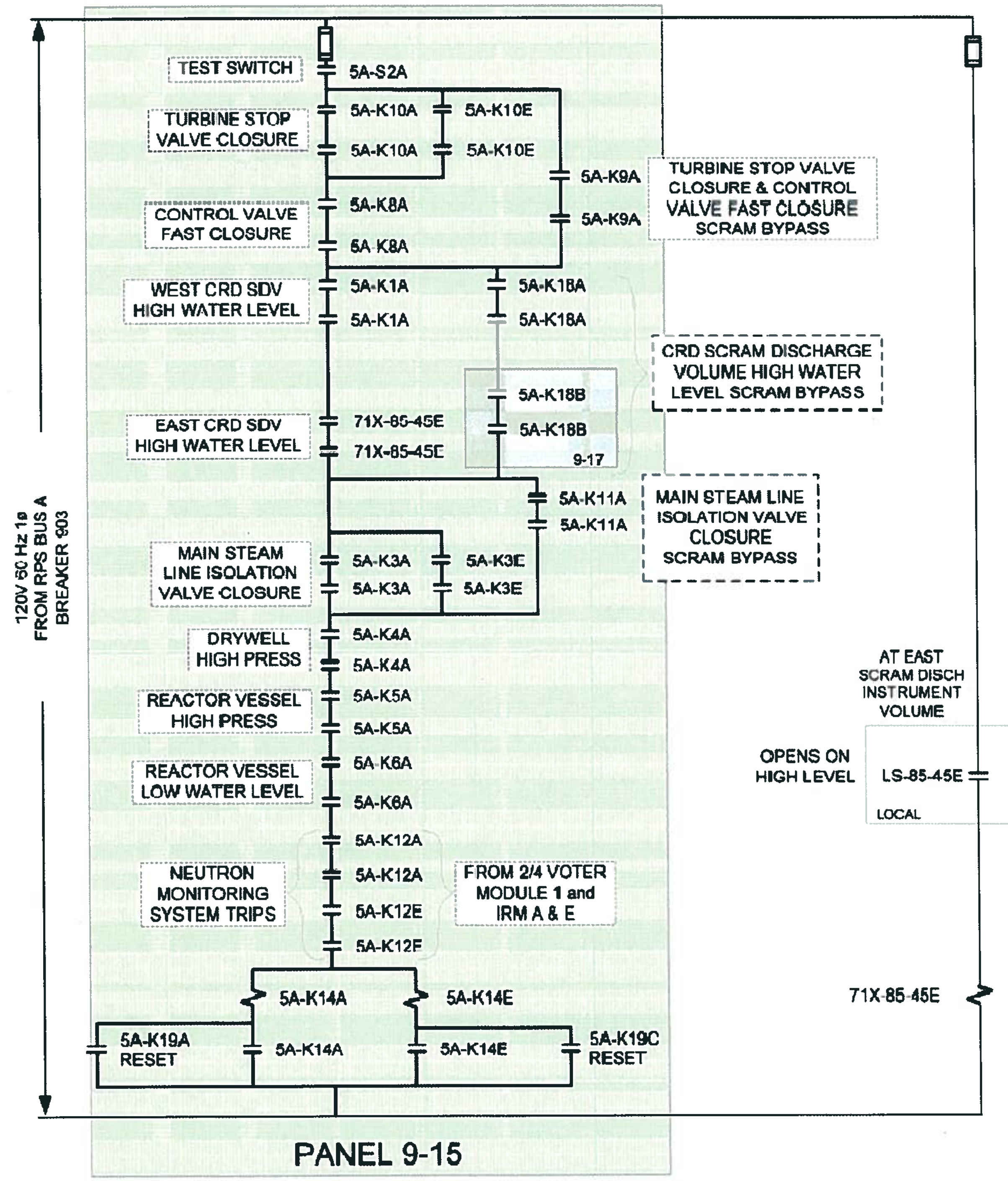
DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
TURB STOP VLV #1 3-FCV-1-74 Function: 8	A1 CHANNEL 3-FU1-1-74A (5A-F10A)	3-RLY-099-05AK10A	9-15	3-730E915-9	NONE	BOTH CIRCUITS FOR ANY SINGLE TURB STOP VLV CAN BE DEENERGIZED AND NO 1/2 SCRAM OR RPT LOGIC ACTUATION OCCURS.
	B1 CHANNEL 3-FU1-1-74B (5A-F10B)	3-RLY-099-05AK10B 3-RLY-099-05AK10K	9-17	3-730E915-10 3-45E763-11(RPT)		
TURB STOP VLV #2 3-FCV-1-78 Function: 8	A1 CHANNEL 3-FUI-1-78A (5A-F10E)	3-RLY-099-05AK10E 3-RLY-099-05AK10J	9-15 9-17	3-730E915-9 3-45E763-11(RPT)	NONE	BOTH CIRCUITS FOR ANY SINGLE TURB STOP VLV CAN BE DEENERGIZED AND NO 1/2 SCRAM OR RPT LOGIC ACTUATION OCCURS.
	B2 CHANNEL 3-FU1-1-78B (5A F10D)	3-RLY-099-05AK10D		3-730E915-10		

Date ____________

7.4.1 RPS Channel A1 Test Alignment (continued)

NOTE

The RPS and RPT logic is functionally tested by partially closing Turbine Stop Valve MSV-1 and simulating closed MSV-2 to satisfy the RPS Channel A1 Logic. The valves themselves will be full stroked closed if stroke timing of the TSVs is required.



TP-2: RPS AUTOMATIC TRIP CHANNEL A1 (TYPICAL FOR A2, B1, B2)

QUESTION 35

A Unit 1 plant startup is in progress with the REACTOR MODE SELECTOR switch in STARTUP.

IRM indications are as follows:
C 57/125 on Range 6
H 27/125 on Range 6

Which ONE of the following describes the conditions that will occur if IRM C and IRM H are ranged down to Range 5?

A. Half scram ONLY

B. Rod block ONLY

C. Rod block AND a Half scram

D. Rod block AND a Full scram

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215003A3.03	
	Importance Rating	3.7	

Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITORING (IRM) SYSTEM including: A3.03 RPS status

Explanation: **C CORRECT** – Ranging down one range will cause IRM C to go upscale and IRM H to read 69.5/125. IRM C will cause a rod block (104.6/125) and a half scram (116.4/125) on RPS A

A. Incorrect- Plausible if the candidate does not recall that a rod block also results from this event or if the new IRM reading is calculated incorrectly.

B. Incorrect- Plausible if the candidate does not recall that a 1/2 scram also results from this event if the new IRM reading is calculated incorrectly.

D. Incorrect - This distractor is plausible if the candidate miscalculates the reading for IRM H.

Technical Reference(s): 1-OI-99

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: Oyster Creek 2010 #17

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

G. The IRMs produce the following trip outputs to the Reactor Manual Control System rod withdrawal block circuitry:

1. High (> 104.6 on 125 scale).
2. Inop (module unplugged, mode switch **NOT** in OPERATE, HV power supply low voltage, loss of 24VDC power supply to IRM drawer).
3. Downscale (< 7.5 on 125 scale), bypassed if range switch set to position 1.
4. Detector wrong position (detector **NOT** full in).

3.0 PRECAUTIONS AND LIMITATIONS (continued)

H. The IRMs produce the following trip outputs to the Reactor Protection System auto-scram circuitry:

1. High-High (> 116.4 on 125 scale).
2. Inop (module unplugged, mode switch **NOT** in OPERATE, HV power supply low voltage, loss of 24VDC power supply to IRM drawer).
3. In addition, by removing the blue shorting links (2 total links), the IRMs are placed in the non-coincident trip logic where any <u>one</u> channel, if tripped, will produce a full reactor scram. The 2/4 Voters are also in this logic such that a trip output from any one Voter yields a full Reactor Scram.

Oyster Creek 2010 #17

17 **ID: 10-1 NRO17** **Points: 1.00**

A plant startup is in progress with the REACTOR MODE SELECTOR switch in STARTUP. An event then occurs and IRM 15 fails INOP.

Which of the following conditions will occur, if any, as a result of this event?

A. A 1/2 scram **ONLY**

B. A rod block **ONLY**

C. A rod block **AND** a 1/2 scram

D. **NEITHER** a rod block **NOR** 1/2 scram

Answer: C

QUESTION 36

SRM 'A' drawer loses power during a Unit 1 plant startup. All IRMs are on range 4. The RPS shorting links are installed.

Which ONE of the following completes the statement?

The power supply to the SRM 'A' drawer is ___(1)___ and the plant response to the SRM A loss of power is ___(2)___.

A. (1) RPS
(2) a Reactor Scram and rod block

B. (1) ±24VDC
(2) a Reactor Scram and rod block

C. (1) ±24VDC
(2) a rod block ONLY. A Reactor Scram will NOT occur

D. (1) RPS
(2) a rod block ONLY. A Reactor Scram will NOT occur

Answer: **C**

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

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

Explanation: **C CORRECT**- the loss of the 24 VDC will cause a loss of SRM logic modules and an inop trip and a withdraw block. No scram signal will occur.

B. Incorrect- +24 VDC is the power supply to the SRM HVPS, however this is plausible if the candidate does not recall the DC power supply. A scram signal will NOT occur, however this is plausible if the candidate confuses this with an IRM HI HI/INOP signal which will generate a scram signal when the MODE switch is NOT in RUN.

A. Incorrect- A scram signal will NOT occur, however this is plausible if the candidate confuses this with an IRM HI HI/INOP signal which will generate a scram signal when the MODE switch is NOT in RUN.

D. Incorrect - +24 VDC is the power supply to the SRM HVPS. This is plausible if the candidate does not recall the DC power supply.

Technical Reference(s): , 1-ARP-9-5A window 13

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Vermont Yankee 2009 #3
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

SRM HIGH/INOP
13

(Page 1 of 1)

Sensor/Trip Point:

Relay K-20

A. 6.8×10^4 cps
B. INOP.
   1. Hi voltage low.
   2. Module unplugged.
   3. Mode switch **NOT** in operate.
   4. Loss of ± 24VDC to monitor.

Sensor Location: Panel 1-9-12, MCR.

Probable Cause:

A. One or more sensor trip conditions exist.
B. SI (or SR) in progress.
C. SRM detectors **NOT** withdrawn.
D. Malfunction of sensor.

Automatic Action: Rod withdrawal block below range 8 on IRM detector **NOT** bypassed, and Rx Mode Sw. **NOT** in Run.

Operator Action:

A. **VALIDATE** initiating condition using multiple indications. ☐
B. **CHECK** ROD withdrawal has been inhibited. ☐
C. **IF** permissible during startup (Mode 2) operation, **THEN** **BYPASS** affected SRM detector. **REFER TO** 1-OI-92. ☐
D. [NRC/C] **IF** one or more SRM recorder readings are downscale, **THEN** **CHECK** for loss of ± 24VDC power. ☐
E. **NOTIFY** Instrument Maintenance that functional test of any monitors indicating an INOP condition, including a downscale reading, are required before the instrument can be considered operable. [NRC IE Item 86-40-03] ☐
F. **REFER TO** Tech. Spec. Sect. 3.3.1.2, Table 3.3.1.2-1, TRM Tables 3.3.4-1 and 3.3.5-1. ☐

References: 1-45E620-6-1 1-730E237-8

Vermont Yankee 2009 #3

Proposed Question: Common 3

Given the following conditions:

- The plant is performing a startup
- All equipment is operable
- The RPS shorting links are installed

Then, the SRM 'A' drawer loses power

Which one of the following describes what supplies the SRM 'A' drawer and what is the plant response?

A. 125 VDC.
A Reactor Scram and rod block will occur

B. +24VDC.
A Reactor Scram and rod block will occur.

C. +24VDC.
A rod block will occur. A Reactor Scram will NOT occur.

D. 125 VDC.
A rod block will occur. A Reactor Scram will NOT occur.

Proposed Answer: C

QUESTION 37

Which ONE of the following completes both statements for the APRM and Voter power supply arrangement?

Each APRM chassis is powered from ___(1)___ Quadruple Low Voltage Power Supply (QLVP).

The four Low Voltage Power Supplies (LVPS) are powered from ___(2)___.

A (1) One
(2) the QLVP(s)

B. (1) One
(2) RPS

C. (1) Two
(2) the QLVP(s)

D. (1) Two
(2) RPS

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005 K2.02	
	Importance Rating	2.6	

Knowledge of electrical power supplies to the following:
K2.02APRM channels

Explanation: B CORRECT – The QLVPS supplies the loads only in its cabinets. The QLVPS is powered from both RPS Busses with LVPS 1 and 2 from RPS A and 3 and 4 from RPS B

A Incorrect – First Part: Correct. Second Part: Incorrect. Plausible that the candidates may confuse QLVP and LVPS.

C Incorrect – First Part: Incorrect. Each APRM gets power from the QLVP in it's cabinet. Second Part: Incorrect. Plausible that the candidates may confuse QLVP and LVPS.

D Incorrect –First Part: Incorrect. Plausible if the candidate confuses the fact that 2 LVPs are powered from each RPS bus. Second Part: Incorrect Plausible because 2 LVPs are powered from each RPS bus

Technical Reference(s): 2-OI-92B

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.148 V.B.8

Question Source:	Bank:	
	Modified Bank:	
	New	X
Question History:	Previous NRC None	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

O. The OPRM channel/cell provides an INOPERATIVE ALARM (Inverse Video) when the quantity of operating OPRM cells is less than 23. When the number of LPRMs in a cell is reduced to less than 2, the OPRM Cell is considered Inoperable. The OPRM function is disabled when the reactor mode switch is in a position other than RUN or the Reactor is operating outside of the OPRM Auto Enable region.

P. The Operators Display Assembly, which normally monitors the LPRM or APRM power, will automatically switch over to OPRM monitoring when the reactor is placed in the region of potential instability. The region of potential instability is bounded by ≥25% power and <60% total recirc drive flow (OPRM Auto Enable region) from any one of the channels.

Q. The message "OPRM TRIP ENABLED" is displayed for each APRM when entering the power/flow region where instability can occur. The message will be replaced with "ANTICIPATED INSTABILITY" whenever a Pre Trip (alarm) setpoint has been reached by any of the OPRM algorithms. If an oscillation trip exists, as defined by the OPRM trip setpoints, the message is replaced with "INSTABILITY DETECTED" and when two of these types of trips occur, an RPS automatic scram is received.

R. The operator has the ability to transfer the display back from OPRM to APRM by depressing the "ETC" softkey.

S. There are a total of four Operators Display Assemblies (ODAs), 2 for the APRM/OPRMs and 2 for the RBM. Each APRM ODA is shared by 2 APRM/OPRMs. All four ODA's are powered by I & C BUS "A".

T. The following are power supplies for the APRM/OPRM:
(Panel 2-9-14 is made up of 5 Chassis, 4 APRMs and 1 RBM)

1. There are five Quadruple Low Voltage Power Supplies (QLVPS), one per bay Panel 2-9-14.

2. Each QLVPS receives power from both RPS busses. LVPS 1 and 2 are fed from RPS A, LVPS 3 and 4 are fed from RPS B.

3. For each QLVPS, LVPS 1 and 4 feed the APRM and RBM A Chassis and LVPS 2 and 3 feed LPRM and RBM B Chassis.

4. Each Voter is powered from the RPS bus it serves, such that those assigned to RPS subchannels B1 and B2 are powered from RPS B and those assigned to RPS subchannels A1 and A2 are powered from RPS A. These power supplies are seen at the bottom of the panels on the QLVPS and are indicated energized by the green illuminated lights.

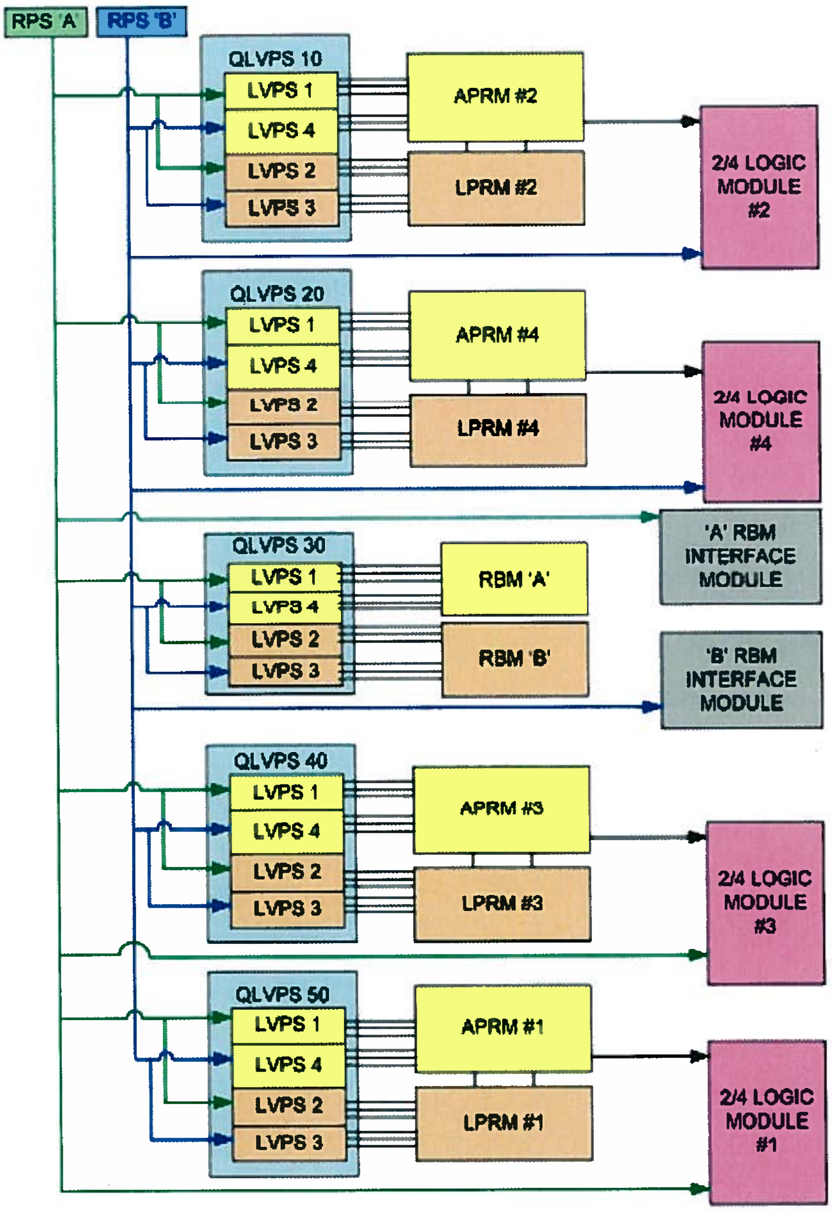
Illustration 1
(Page 1 of 5)

APRM Trip Outputs

APRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
APRM Downscale	≥5%	1. Rod Block if REACTOR MODE SWITCH in RUN.
APRM Inop	1. APRM Chassis Mode not in OPERATE (keylock to INOP). 2. Loss of Input Power to APRM. 3. Self Test detected Critical Fault in the APRM instrument. 4. Firmware Watchdog timer has timed out	1. One Channel detected, no alarm or RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).
APRM Inop Condition	1. < 20 LPRMs in OPERATE, or < 3 per level.	1. <20 LPRMs total or <3 per level results in a Rod Block and a trouble alarm on the display panel. This does not yield an automatic APRM trip, but does, however, make the associated APRM INOP.
APRM High	1. DLO ≤ (0.66W + 59%) SLO ≤ (0.66W(W-10%) + 59%) [W = Total Recirc Drive Flow in % rated]. 2. Neutron Flux Clamp Rod Block ≥ 113% 3. ≤ 10% APRM Flux.	1. Rod Block if REACTOR MODE SWITCH in RUN. 2. Rod Block in all REACTOR MODE SWITCH positions except RUN.
APRM High High	1. DLO ≤ (0.66W + 65%) 2. SLO ≤(0.66(W-10%) + 65%) [W = Total Recirc Drive Flow in % rated]. 2. ≤ 119% APRM Flux. 3. ≤ 14% APRM Flux.	1. Scram. 2. Scram in all REACTOR MODE SWITCH positions except RUN.
APRM Flow Converter	1. ≤ 5% mismatch between APRM Channels. 2. 107% Flow monitor upscale.	1. Flow compare inverse video alarm. 2. Rod Block.
OPRM Inop	< 23 Operable Cells - A cell is inop when it has < 2 operable LPRM's	Annunciation Only
OPRM Pre-Trip Condition	Any one of three algorithms, period, growth, or amplitude exceeds its pre-trip alarm setpoint for an operable OPRM cell.	Rod Block
OPRM Trip	Any one of the three algorithms, period, growth, or amplitude for an operable OPRM cell has exceeded its trip value:	1. One Channel detected, no RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).

All OPRM setpoints are bypassed when the Reactor Mode Switch is not in RUN or the Reactor is not operating in the Power/Flow region where instabilities can occur (≥25% Power & <60% Recirc Drive Flow).



TP-14: POWER DISTRIBUTION BLOCK DIAGRAM

QUESTION 38

All Main Steam Line Isolation Valves (MSIVs) have closed and a reactor scram has occurred on Unit 3.

RPV pressure is being maintained 800 to 1000 psig using the SRVs.
RPV level is being maintained using RCIC in MANUAL due to a controller failure.

Which ONE of the following describes the RCIC System response with no operator action as reactor pressure is lowered from 1000 to 800 psig?

RCIC Pump speed will ___(1)___ and RCIC pump discharge pressure will ___(2)___.

A. (1) lower
(2) lower

B. (1) lower
(2) remain constant

C. (1) remain constant
(2) rise

D. (1) remain constant
(2) lower

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000 A1.02	
	Importance Rating	3.3	

Ability to predict and/or monitor changes in parameters associated with the operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including:
A1.02RCIC pressure

Explanation: D CORRECT – When in MANUAL, a constant signal is sent to the signal converter and is used to control RCIC at a constant <u>speed</u> as reactor pressure drops, speed will remain constant and discharge pressure will drop as reactor pressure against which the pump is working, drops. With a constant speed and lower resistance, flow will increase.

A Incorrect – If the candidate believes that manual control maintains a constant valve position then as steam pressure drops, speed will drop. With the pump working against a lower pressure, the discharge pressure will drop. This is also re response while in AUTO

B Incorrect – See A above for part 1 with 2 the operator may believe that with a constant valve position the pump will discharge at a constant pressure

C Incorrect –Candidate may believe that at a constant speed and lower pressure to work against, discharge pressure may rise as does RCIC flow

Technical Reference(s): 3-OI-71

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Illustration 2
(Page 1 of 2)

RCIC Flow Controller Operation
3-FIC-71-36A

The RCIC SYSTEM FLOW/CONTROL controller, 3-FIC-71-36A, has 2 modes of operation. MANUAL and AUTO.

In Manual it controls the output ma signal with 2 rates of change available. The Fast rate is ≈ 30 seconds to travel full scale. The Fast rates are selected by moving the Manual control lever to the ←← or the →→ position. The Slow rate is ≈ 40 seconds to travel full scale. The Slow rates are selected by moving the manual control lever to the ← or the → position. A momentary movement of the manual control lever to the ← or the → position will result a change of 0.1% or ≈ 0.7 gpm.

In Auto it also controls the ma signal but with a different means of controlling. The signal is changed in Auto by changing the setpoint. The rate for full scale travel of the setpoint is ≈ 45 seconds. A momentary depressing of the setpoint in either the raise or lower keys will result a change of 0.1% or ≈ 0.7 gpm.

Both Auto to Manual and Manual to Auto transfers are "bumpless" transfers.

The "CASCADE" [C] function is **NOT** used.

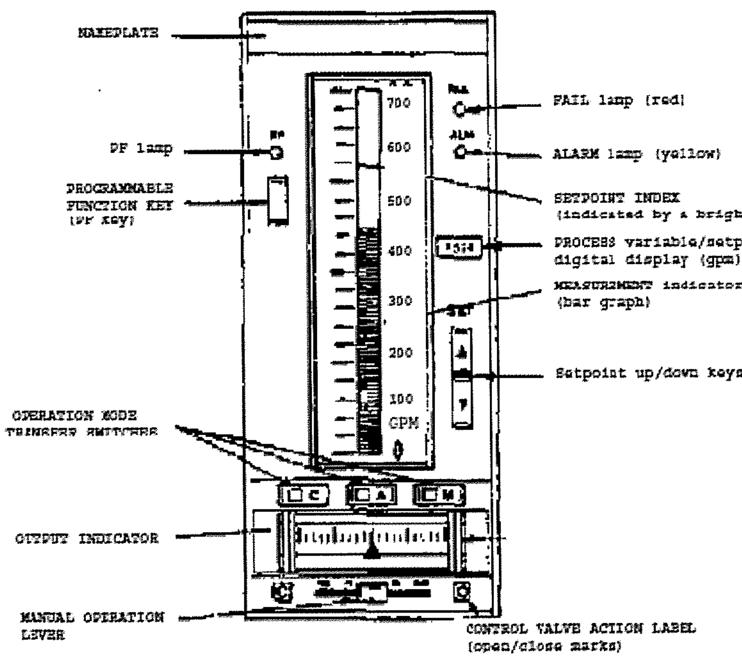


Illustration 2
(Page 2 of 2)

RCIC Flow Controller Operation
3-FIC-71-36A

PROGRAMMABLE FUNCTION KEY (PF key)

While depressed (PF lamp ON) digital display shows "SETPOINT". When **NOT** depressed, PF lamp "OFF" digital display shows system flow. Display will default back to system flow when PF key is released.

STATUS LIGHTS

ALM (alarm) flashes (a flashing alarm lit will override a continuously lit ALM light.)

Data-protect battery needs to be replaced. (RCIC operable)

ALM (alarm) continuously lit

Indicates the controllers high or low limit alarm is actuated or the input/output signal line is open (RCIC inoperable)

FAIL lit

Indicates a serious failure has occurred in the controller. (RCIC inoperable)

QUESTION 39

Unit 2 RCIC was operating following an initiation when a spurious RCIC turbine trip occurs.

The US attempts a RCIC turbine reset by having an UO close and then open the RCIC TURBINE TRIP/THROTTLE VALVE, 2-FCV-71-9.

The UO completes the task and the US has the following indications of RCIC:

- RCIC TURB TRIP/THROTTLE VALVE, 2-ZI-71-9 - green light ON, red light OFF
- Handswitch RCIC TURB TRIP/THROTTLE VALVE, 2-HS-71-9 - red light ON, Green light OFF
- RCIC TURBINE CONTROL VALVE, 2-FCV-71-10 - green light ON, red light ON
- RCIC STEAM LINE INBD, 2-FCV-71-2 - green light OFF, red light ON

Which ONE of the following describes the current condition of RCIC?

A. RCIC is tripped requiring local mechanical linkage to be reset.

B. RCIC is reset and will start as soon as RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, begins to open.

C. RCIC is reset and will start after RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, is full open.

D. RCIC is tripped requiring RCIC AUTO ISOL LOGIC A & B RESET pushbuttons to be depressed.

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000 K5.06	
	Importance Rating	2.7	

Knowledge of the operational implications of the following concepts as they apply to REACTOR CORE ISOLATION COOLING SYSTEM (RCIC):
K5.06 Turbine operation

Explanation: **A** CORRECT – RCIC is tripped and the linkage must be reset locally.

B- Incorrect – RCIC is NOT reset. Plausible if the candidate believes that Handswitch RCIC TURB TRIP/THROTTLE VALVE, 2-HS-71-9 - red light ON, Green light OFF indicates that the RCIC turbine is reset.

C- Incorrect – RCIC is NOT reset. Plausible if the candidate believes that Handswitch RCIC TURB TRIP/THROTTLE VALVE, 2-HS-71-9 - red light ON, Green light OFF indicates that the RCIC turbine is reset.

D- Incorrect –RCI is tripped not isolated. Plausible because an isolation will cause a trip, but a trip does not cause an isolation.

Technical Reference(s): 2-OI-71

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: Fitzpatrick 2008 #37

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

8.4 RCIC Turbine Trip

NOTES

1) The following signals cause a RCIC turbine trip. The RPV High Water Level Trip Signal closes the RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34. All other trip signals close the RCIC TURBINE TRIP/THROTTLE VLV, 2-FCV-71-9 and RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34.
   - High RPV Water Level (+51 inches, Auto Reset at -45")
   - Low Pump Suction Pressure (10 inches Hg vacuum)
   - High Turbine Exhaust Pressure (50 psig)
   - Turbine Overspeed (Mechanical 122.3% of rated signal)
   - Automatic Isolation
   - Manual Push-button
2) All operations are performed at Panel 2-9-3 unless otherwise noted.

[1] **IF** RCIC Turbine did not trip from high RPV water level, **THEN**

VERIFY the following automatic actions:

A. RCIC TURB TRIP/THROTTLE VLV, 2-FCV-71-9, closes. ☐

B. RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, closes. ☐

C. RCIC TURB SPEED, 2-SI-71-42A, indicates zero RPM. ☐

[2] **IF** RCIC Turbine tripped from high RPV water level, **THEN**

VERIFY the following automatic actions:

A. RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, closes. ☐

B. RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, closes. ☐

C. RCIC TURBINE SPEED, 2-SI-71-42A, indicates zero RPM. ☐

8.4 RCIC Turbine Trip (continued)

[3] **IF** RCIC initiation signal is present, RCIC trip was NOT due to high water level **AND** further RCIC operation is desired, **THEN**

RESET RCIC Turbine and **RETURN** to operation as follows:

[3.1] **VERIFY** trip condition has been corrected. ☐

CAUTION

Any time the RCIC controller is placed in "MANUAL" RCIC is to be considered "NOT OPERABLE". However, RCIC is still available.

NOTE

RCIC SYSTEM FLOW/CONTROL Controller, 2-FIC-71-36A, is placed in MANUAL with the demand zeroed to prevent turbine overspeed or hunting when the RCIC TURB TRIP/THROT VALVE, 2-FCV-71-9, is reopened.

[3.2] **PLACE** RCIC SYSTEM FLOW/CONTROL, 2-FIC-71-36A, in MANUAL by DEPRESSING MANUAL operation mode transfer switch and **ADJUST** manual operation lever to minimum. ☐

NOTES

1) A mechanical overspeed trip requires manual reset of the trip linkage at RCIC turbine. To reset, push linkage toward throttle valve until tappet nut collar latches trip lever.
2) To reset RCIC TURB TRIP/THROT VALVE, 2-FCV-71-9, the valve operator must be driven fully closed, then reopened

[3.3] **RESET** RCIC TURB TRIP/THROT VALVE, 2-FCV-71-9. ☐

[3.4] **VERIFY** RCIC TURB TRIP/THROT VALVE, 2-FCV-71-9, and RCIC TURBINE CONTROL VALVE, 2-FCV-71-10, open ☐

QUESTION 40

Unit 2 was operating at 100% Reactor Power with RHR Pump 2B tagged. A Loss of Coolant Accident with a subsequent Loss of Off Site Power has resulted in the following plant conditions:

- Reactor Water Level is (-)125 inches
- Drywell Pressure is 4.1 psig
- B and D 4KV Shutdown Boards are de-energized
- RHR Pump 2A tripped

Which ONE of the following identifies the MINIMUM action, if any, that will prevent the Automatic Depressurization System (ADS) logic from an Auto-Initiation?

A. NO action is required

B. Place ONLY ADS Logic Inhibit Switch 'A' to INHIBIT

C. Place ONLY ADS Logic Inhibit Switch 'B' to INHIBIT

D. Place BOTH ADS Logic Inhibit Switches 'A' AND 'B' to INHIBIT

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	218000 K4.01	
	Importance Rating	3.7	

Knowledge of the AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Prevent inadvertent initiation of ADS logic

Explanation: **A CORRECT**- Pump running permissive is not met. It is the same permissive for System 1 and 2 ADS logic. Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running is required. No RHR Pumps or Core Spray Pumps are running.

B. Incorrect- Plausible in that system 1 channel A logic would be made up with Core Spray pump A running. But because S/D board B doesn't have power to Core Spray pump C, and Core Spray works in pairs, Core Spray pump A is not running. Additionally RHR pump C doesn't have power and RHR pump A tripped.

C. Incorrect- Plausible in that system 2 channel A logic would be made up with Core Spray pump A running. But because S/D board B doesn't have power to Core Spray pump C, and Core Spray works in pairs, Core Spray pump A is not running. Additionally RHR pump C doesn't have power and RHR pump A tripped.

D. Incorrect - Plausible in that if the right combination of Core Spray Pumps were running or any RHR Pump running, this would be correct answer.

Technical Reference(s): OPL171.043, 2-OI-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1006 #40
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis: X

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

INSTRUCTOR NOTES

d. EOI Appendix 8G crossties CAD to DWCA — PROCEDURE USE & ADHERENCE

4. ADS systems controls — TP-2

a. Consists of pressure and water level sensors arranged in the trip systems that control a solenoid-operated pilot air valve

b. The solenoid-operated valve controls the pneumatic pressure applied to a diaphragm actuator which controls the SRV directly — DCN 51106 Cable & Switch configuration / modifications

c. Cables from sensors lead to the Control Room where logic arrangements are formed in cabinets

d. Control channels are separated to limit the effects of electrical failures

e. A two-position control switch is provided in the Control Room for control of the ADS valves

1) Two positions are OPEN and AUTO — HP Use SELF-CHECKING

2) In OPEN, the switch energizes a DC solenoid which allows pneumatic pressure to be applied to the diaphragm actuator of the relief valve — Pressure relief consists of actuation of reactor pressure on internal pilot or by electro-pneumatic operation via pressure switches.

NOTE:
The relief valves can be manually opened to provide a controlled nuclear system cooldown under conditions where the normal heat sink is not available

3) In AUTO, the valves are controlled by the ADS logic and pressure relief logic

f. Four of the six ADS valves may also be controlled from a backup control board which is provided to facilitate plant shutdown and cooldown from outside the Control Room — UNIT DIFFERENCE, DCN 51106 adds new panel "25-658" to Unit 1

5. Automatic Depressurization Initiation Logic

a. The following conditions must be met before automatic depressurization will occur — Obj. V.B.4 Obj. V.C.3 Obj. V.D.3 Obj. V.E.4

1) Two coincident signals of high drywell pressure (+2.45 psig) and low low low reactor vessel

INSTRUCTOR NOTES

water level (-122")
OR
-122" for 265 sec. — LT-3-58A-D

2) A confirmatory low reactor vessel water level signal (+2") (Tech Spec Value 0") — LT-3-184 LT-3-185

3) Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running — Obj. V.C.4 Obj. V.D.4

3.4 Main Steam Relief Valve (MSRV / ADS)

A. Whenever both the acoustic monitor and the temperature indication on a relief valve fail to indicate in the Control Room, the Technical Specifications Section 3.3.3.1 should be consulted to determine what limiting conditions for operation apply.

B. In the event that a relief valve fails to function as designed and the cause of the malfunction is not clearly determined and then corrected, the valve should be considered inoperable and Technical Specifications Section 3.5.1 and 3.4.3 should be consulted to determine what limiting conditions for operation apply.

C. ADS will initiate when ALL of the following conditions are met:

1. A confirmatory Low reactor water level signals (+2.0 inches), REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE, 2-9-3C Window 3

2. Two coincident signals for each of the following parameters:

   a. high drywell pressure (+2.45 psig) in conjunction with low low low reactor water level (-122 inches), ADS BLOWDOWN HIGH DRYWELL PRESS SEAL-IN, 2-XA-55-9-3C Window 33 and RX WTR LVL LOW LOW LOW ECCS/ESF INIT 2-LA-3-58A, 2-XA-55-9-3C Window 28

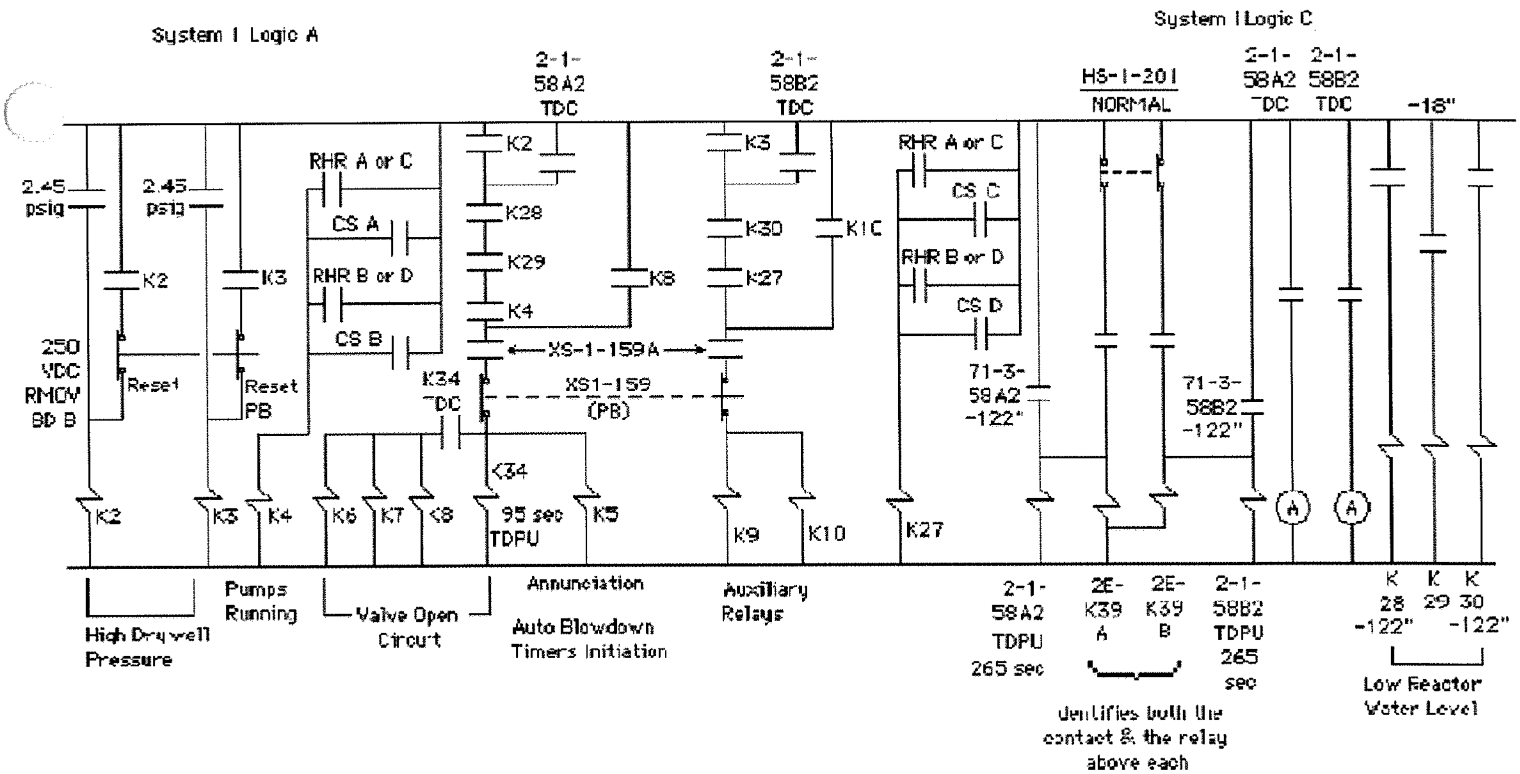
   OR

   b. low low low reactor water level (-122 inches), RX WTR LVL LOW LOW LOW ECCS/ESF INIT 2-LA-3-58A, 2-XA-55-9-3C Window 28, for 265 seconds (High drywell pressure bypass)

3. One RHR pump OR two Core Spray pumps (A or B and C or D) running, RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 2-XA-55-9-3C Window 10.

4. When ALL of the above logic is satisfied, then a 95 second timer starts (ADS BLOWDOWN TIMERS INITIATED, 2-XA-55-9-3C, Window 11) and the timer must be timed out to initiate ADS blowdown.

D. Depressing 2-XS-1-159 and -161 on Panel 2-9-3 will reset the ADS Blowdown Timers. They also reset an ADS initiation, if the timers have timed out. ADS will re-initiate upon subsequent timing out of the timer provided the low level and pump logic signals still exist. The timer setpoint is 95 seconds, however setpoint tolerance allows it to be as low as 77 seconds.


System I Logic A
System I Logic C
2-1-
58A2
TDC
2-1-
58B2
TDC
HS-1-201
NORMAL
-18"
2.45
psig
K2
K3
RHR A or C
CS A
RHR B or D
CS B
K28
K29
K4
K8
K30
K27
K10
RHR A or C
CS C
RHR B or D
CS D
250
VDC
RMOV
BD B
Reset
Reset
PB
K34
TDC
XS-1-159A
XS1-159
(PB)
71-3-
58A2
-122"
71-3-
58B2
-122"
K34
K2
K3
K4
K6
K7
K8
95 sec
TDPU
K5
K9
K10
K27
High Drywell
Pressure
Pumps
Running
Valve Open
Circuit
Annunciation
Auto Blowdown
Timers Initiation
Auxiliary
Relays
2-1-
58A2
TDPU
265 sec
2E-
K39
A
2E-
K39
B
2-1-
58B2
TDPU
265
sec
dentifies both the
contact & the relay
above each
K
28
-122"
K
29
K
30
-122"
Low Reactor
Water Level

Browns Ferry 1006 NRC #40

HLT 0810/1006 Written Exam

40. 218000 K4.01

Unit 2 was operating at 100% Reactor Power with RHR Pump 2B tagged. A Loss of Coolant Accident with a subsequent Loss of Off Site Power has resulted in the following plant conditions:

- Reactor Water Level is (-)125 inches
- Drywell Pressure is 4.1 psig
- B **AND** D 4KV Shutdown Boards are de-energized
- RHR Pump 2A tripped

Which ONE of the following identifies the **MINIMUM** action, if any, that will prevent the Automatic Depressurization System (ADS) from an Auto-Initiation?

A. **NO** action is required

B. Place **ONLY** ADS Logic Inhibit Switch 'A' to INHIBIT

C. Place **ONLY** ADS Logic Inhibit Switch 'B' to INHIBIT

D. Place **BOTH** ADS Logic Inhibit Switches 'A' **AND** 'B' to INHIBIT

QUESTION 41

Given the following plant conditions:

- During performance of 2-SR-3.3.1.1.13, "Reactor Protection and Primary Containment Isolation Systems Low Reactor Water Level Instrument Channel B2 Calibration," 2-LIS-3-203D, RX WATER LEVEL LOW, fails to actuate.

- It is determined that the failure is due to an inoperable switch and a replacement is NOT available for 4 days.

- The Shift Manager has determined that the proper action is to trip the inoperable channel ONLY.

Which ONE of the following completes the statements?

The Channel will be placed in a tripped condition by ____(1)____.

The effect on the Unit will be ___(2)___.

A. (1) Removing the fuse associated with 2-LIS-3-203D
(2) NO Primary Containment Isolation

B. (1) Removing the fuse associated with 2-LIS-3-203D
(2) PCIS Groups 2, 3 and 6 Inboard Isolations

C. (1) Placing a trip into the Analog Trip Unit associated with 2-LIS-3-203D
(2) NO Primary Containment Isolation

D. (1) Placing a trip into the Analog Trip Unit associated with 2-LIS-3-203D
(2) PCIS Groups 2, 3 and 6 Inboard Isolations

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002 A2.06	
	Importance Rating	3.0	

Ability(a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.06 Containment instrumentation failure

Explanation: **A** CORRECT – Pulling the fuse associated with L1S-:-3-203D de-energizes the 'B2' Channel of RPS causing a 1/2 scram in RPS 'B' and a 1/4 isolation in PCIS. No PCIS valves will reposition.

B - Incorrect: PCIS logic causes a "1/4-isolation" signal BUT no PC IV devices actuate. This is plausible because the action to ensure the trip input is correct. In addition, the "1/4-isolation" applies to the PCIS groups identified in the distractor.

C - Incorrect: The method of inputting the trip is incorrect. Tripping an ATU cannot be ensured via a clearance. This is plausible because the RPS and PCIS response is correct.

D - incorrect: The method of inputting the trip is incorrect. Tripping an ATU cannot be ensured via a clearance. This is plausible because the "1/4-isolation" applies to the PCIS groups identified in the distractor.

Amplification: Basically, when a signal received from a sensor reaches the ATU, it is compared to the setpoint value programmed into the ATU. When the setpoint is exceeded, the ATU transmits a TRIP output to the associated logic, whether RPS or PCIS. Essentially converting an ANALOG input into a DIGITAL output. The ATU has a feature which allows it to generate a TRIP output for testing purposes. It could serve the same function as removing power from the input sensor, but is not considered reliable enough to comply with Tech Spec requirements for "placing a channel in the TRIP condition.

Technical Reference(s): 2-OI-64, 2-OI-99

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X	
	Modified Bank:	
	New:	
Question History:	Previous NRC: BFN 2008 #13	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis:	X

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

BFN Unit 2	Primary Containment System	2-OI-64 Rev. 0117 Page 105 of 151

Appendix A
(Page 1 of 10)

Actions to Place PCIS in Tripped Condition

NOTE

Water level designators (1-8)are listed for relationship to the applicable device only.

(T.S. Tables 3.3.6.1-1,3.3.6.2-1, & 3.3.7.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARM	REMARKS
2-LIS-3-203A RX WATER LEVEL LOW (Level 3)	2-FU1-3-203AA (5A-F6A)	5AK6A 5AK25A 16AK5A 16AK6A	9-15	2-730E915-9 2-730E927-7 2-45E671-26	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203B RX WATER LEVEL LOW (Level 3)	2-FU1-3-203BA (5A-F6B)	5AK6B 5AK25B 16AK5B 16AK6B	9-17	2-730E915-10 2-730E927-8 2-45E671-38	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203C RX WATER LEVEL LOW (Level 3)	2-FU1-3-203CA (5A F6C)	5AK6C 5AK25C 16AK5C 16AK6C	9-15	2-730E915-9 2-730E927-7 2-45E671-32	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
2-LIS-3-203D RX WATER LEVEL LOW (Level 3)	2-FU1-3-203DA (5A-F6D)	5AK6D 5AK25D 16AK5D 16AK6D	9-17	2-730E915-10 2-730E927-8 2-45E671-44	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B. CAUSES 1/4 ISOLATION IN PCIS GROUPS 2,3, 6 AND 8. NO PCIS DEVICES ACTUATE.
Table 3.3.6.1-1: Function 2a and 5h Table 3.3.6.2-1: Function 1 Table 3.3.7.1-1: Function 1						

BFN Unit 2	Reactor Protection System	2-OI-99 Rev. 0079 Page 74 of 79

Illustration 3
(Page 6 of 11)

Actions to Place RPS Instruments in Tripped Conditions (TS Table 3.3.1.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARMS	REMARKS
2-LIS-3-203A RX WATER LEVEL LOW (Level 3) A1 CHANNEL Function: 4	2-FU1-3-203AA (5AF6A)	2-RLY-099-05AK06A 2-RLY-099-5A-K25A 2-RLY-064-16AK5A 2-RLY-064-16AK6A	9-15	2-730E915-9 2-730E927-7 2-45E671-26	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203B RX WATER LEVEL LOW (Level 3) B1 CHANNEL Function: 4	2-FU1-3-203BA (5AF6B)	2-RLY-099-05AK06B 2-RLY-099-5A-K25B 2-RLY-064-16AK5B 2-RLY-064-16AK6B	9-17	2-730E915-10 2-730E927-8 2-45E671-38	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203C RX WATER LEVEL LOW (Level 3) A2 CHANNEL Function: 4	2-FU1-3-203CA (5AF6C)	2-RLY-099-05AK06C 2-RLY-099-5A-K25C 2-RLY-064-16AK5C 2-RLY-064-16AK6C	9-15	2-730E915-9 2-730E927-7 2-45E671-32	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B-1 REACTOR CHANNEL A AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL A NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
2-LIS-3-203D RX WATER LEVEL LOW (Level 3) B2 CHANNEL Function: 4	2-FU1-3-203DA (5AF6D)	2-RLY-099-05AK06D 2-RLY-099-5A-K25D 2-RLY-064-16AK5D 2-RLY-064-16AK6D	9-17	2-730E915-10 2-730E927-8 2-45E671-44	2-XA-55-4A-2 RX VESSEL WTR LEVEL LOW HALF SCRAM 2-XA-55-5B2 REACTOR CHANNEL B AUTO SCRAM	ALARMS AND 1/2 SCRAM IN CHANNEL B NO PCIS DEVICES ACTUATE. 1 channel actuated for secondary containment and CREV initiation
NOTE: Device Function corresponds to the TS Table 3.3.1.1 Functions.						

Browns Ferry ILT 0810 #13

13. RO 223002A2.06 001/C/A/T2G1/PCIS//223002A3.01//RO/SRO/BANK

Given the following plant conditions:

- During performance of 2-SR-3.3.1.1.13, "Reactor Protection and Primary Containment Isolation Systems Low Reactor Water Level Instrument Channel B2 Calibration," 2-LIS-3-203D fails to actuate.
- It is determined that the failure is due to an inoperable switch and a replacement is NOT available for 4 days.
- The Shift Manager has determined that the proper action is to trip the inoperable channel ONLY.

Which ONE of the following describes how this is accomplished and the effect on Unit status?

A.✔ Remove the fuse associated with 2-LIS-3-203D. A half scram will result and NO Primary Containment Isolation Valves will realign.

B. Remove the fuse associated with 2-LIS-3-203D. A half scram will result and PCIS Groups 2, 3 and 6 Inboard Isolation Valves will close.

C. Place a trip into the Analog Trip Unit associated with 2-LIS-3-203D. NO half scram will result and NO Primary Containment Isolation Valves will realign.

D. Place a trip into the Analog Trip Unit associated with 2-LIS-3-203D. A half scram will result and PCIS Groups 2, 3 and 6 Outboard Isolation Valves will close.

QUESTION 42

Unit 1 was performing a plant startup per 1-GOI-100-1A, UNIT STARTUP. PRIOR to entering MODE 1, a malfunction in the Turbine EHC system caused reactor pressure to slowly lower and stabilize at 750 psig.

Which ONE of the following completes the statement?

The Main Steam Isolations Valves and Main Steam Line Drains are ___(1)___ because the Group 1 ___(2)___.

A. (1) Closed
(2) isolated on Main Steam Line Pressure $\leq$ 852 psig ONLY

B. (1) Open
(2) isolation is bypassed with the MODE switch NOT IN RUN

C. (1) Closed
(2) isolated on Main Steam Line Pressure $\leq$ 852 psig with the MODE switch NOT IN RUN

D. (1) Open
(2) isolation is bypassed with Main Steam Line $\leq$ 852 psig

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002 K4.04	
	Importance Rating	3.2	

Knowledge of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: K4.04 Automatic bypassing of selected isolations during specified plant conditions

Explanation: **B CORRECT**- MSL Low Pressure ≤852 psig will cause a Group 1 isolation, and is only bypassed with the Mode Switch in RUN.

A- Incorrect- First part: Incorrect, the valves will still be open because the Mode Switch is in STARTUP. Second part: Incorrect, plausible if the candidate confuses when the isolation is bypassed. Additionally, ≤852psig is the correct setpoint for the isolation.

C- Incorrect- First part: Incorrect, they valves will still be open because the Mode Switch is in STARTUP. Second part: exact opposite of the correct logic

D- Incorrect- First Part: Correct. Second part: Incorrect, plausible if the candidate confuses when the isolation is bypassed. Additionally, ≤852psig is the correct setpoint for the isolation.

Technical Reference(s): OPL171.009, OPL171.017, 1-OI-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

OPL171.017

A basic description of each group is as follows:

1. Group 1

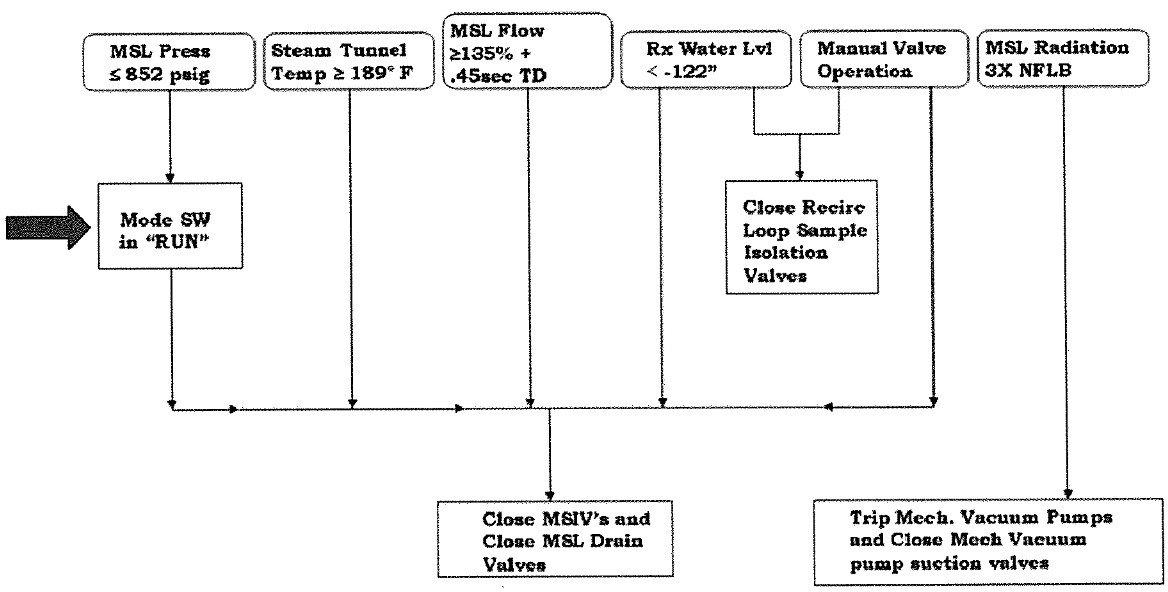
This group includes the Main Steam Isolation Vavles (MSIVs), main steam line drains, and reactor water sample line isolation valves.

The signals which will initiate a Group 1 isolation are as follows:

RPV low low low level (-122" or Level 1)
*MSL High Flow .45 sec TD (135%)
*MSL Area High Temperature (189°F)
*MSL Low Pressure (852 psig Mode Switch in RUN)

*(MSIVs and MSL Drains only)

OPL171.009



1-OI-1

3.2.2 MSIV Isolation

A. Main steam tunnel temperature should not be allowed to exceed 189°F to prevent MSIV isolation.

B. Whenever reactor pressure is reduced to 852 psig and the reactor mode switch is in RUN position, the MSIVs will close.

QUESTION 43

Unit 2 is operating at 100% power when a break in the Drywell Control Air System in the drywell between the drywell (penetration X-50) and the check valve (2-32-2516) causes pressure to decay.

NO operator action has been taken.

Which ONE of the following describes the operation of the MSRVs with loss of the Drywell Control Air System?

The ADS valves ___(1)___ and the non-ADS valves ___(2)___.

A. (1) can be manually operated 5 times and will still open on an ADS signal
(2) can be manually operated once

B. (1) can be manually operated 5 times and will open on an ADS signal
(2) CANNOT be manually operated

C. (1) CANNOT be manually operated but will open on an ADS signal
(2) can be manually operated once

D. (1) CANNOT be manually operated but will open on an ADS signal
(2) CANNOT be manually operated

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	239002 K6.02	
	Importance Rating	3.2	

Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES:
K6.02 Air (nitrogen) supply

Explanation: **B** CORRECT –The ADS valves are equipped with an accumulator that allows 5 operations of the ADS SRVs. The non-ADS SRVs do not have an accumulator and cannot be manually opened but will open in the safety mode.

A Incorrect – First Part: Correct. Second Part: Plausible because the candidate may believe that an SRV will operate once confusing it with the MSIV accumulators.

C Incorrect – First Part: Incorrect. Plausible because the candidate may assume that ADS valve operation is available but only for an actual ADS signal. Second Part: Plausible because the candidate may believe that an SRV will operate once confusing it with the MSIV accumulators.

D Incorrect – First Part: Incorrect. Plausible because the candidate may assume that ADS valve operation is available but only for an actual ADS signal. Candidate may assume that ADS valve operation is available but only for an actual ADS signal. Second Part: Correct.

Technical Reference(s): 2-AOI-32A-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	X
	New	
Question History:	Previous NRC Clinton 2007 #6	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] IF ANY EOI entry condition is met, THEN

ENTER the appropriate EOI(s). ☐

NOTES
1) The MSIV air accumulators are designed to provide for one closing actuation following loss of air supply. Once closed the valve is held closed by the springs.
2) The ADS MSRV air accumulators are provided to assure that the valves can be held open following failure of the air supply to the accumulators, and they are sized to contain sufficient air for a minimum of five valve operations. Operations of the ADS MSRV should be limited to 5 times.
3) Nitrogen Tanks supply pressurized nitrogen to the Drywell Control Air System via the DWCA SUPPLY REGULATORS 2-PREG-32-49A and 2-PREG-32-49A (lead regulator will be set at 100 psig and backup regulator set at 5-8 psig lower)
4) DWCA NITROGEN REG STATION BYPASS VLV, 2 BYV 032 0141 can be used to maintain approximately 98 psig in DWCA Receiver Tanks A & B when required by plant conditions

QUESTION 44

Unit 3 is at 85% power. All three Reactor Feed Pumps are in Automatic Level Control. Reactor Vessel Narrow Range instrument, LT-3-53, has a small leak on the reference leg tap. The Feedwater Level Control System (FWLCS) is in 3-element control. The Narrow Range instruments indicate as follows:

• LT-3-53	44 inches
• LT-3-60	33 inches
• LT-3-206	31 inches
• LT-3-253	33 inches

Which ONE of the following indicates the Reactor Vessel Level that the FWLCS is using as the current RPV level? (Round to nearest whole number)

A. 37 Inches

B. 35 inches

C. 33 inches

D. 32 inches

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	259002 K5.03	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: K5.03 Water level measurement

Explanation: **D CORRECT**- The high level indicated by LT-3-53 is discarded since it is >8 inches above the average. The other 3 signals are averaged for the appropriate control level.

A. Incorrect- Plausible because this is the average of the 3 highest values.

B. Incorrect- Plausible because this is the average of all levels if the highest level is not discarded.

C. Incorrect- Plausible since this is the level that would be maintained if the two highest middle levels were selected.

Technical Reference(s): OPL171.012

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 2004 #30
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

B. Component Description

1. Reactor Water Level — Obj. V.B.1, Obj. V.D.5

a. Four independent narrow range level transmitters (LT-3-53,-60,-206 and 253). They are differential pressure transmitters connected to water reference condensing chambers. Digital readouts are spanned for a reactor level of -10 to + 70 inches but, analog range is still 0" to 60".

b. The control algorithm checks the signal quality. If BAD (failed or out-of-range high or low), the signal is discarded. If GOOD, the signal is further processed.

c. Each level signal is pressure compensated for density differences by the algorithm and the four signals are averaged.

d. The algorithm validates each level signal by comparing them to the average. Level signals that deviate from the average by more than 8 inches are declared invalid, and are discarded from the average.

f. The average level value is used for the single element and three element control logic's.

(1) The individual density compensated levels are output to Control Room indicators. — Obj. V.B.1

(2) The average level is output to one pen of a two-pen recorder.

(3) Reactor Vessel high and low level alarms are generated by comparing the average level to high (>39") and low (<27") setpoints.

(4) The average level is also used in the Recirculation pump runback level interlock logic within the algorithm. — Obj. V.B.7, Obj. V.C.6

g. If one level signal is BAD or invalid, the algorithm will calculate the average of the three remaining level signals and will control on that value. — Obj. V.B.6, Obj. V.C.5

Illustration 8
(Page 1 of 7)

RFWCS Instrumentation

1.0 NARROW RANGE REACTOR WATER LEVEL

1.1 Components

LEVEL A, 1-LI-3-53

LEVEL B, 1-LI-3-60

LEVEL C, 1-LI-3-206

LEVEL D, 1-LI-3-253

1.2 Description

The instruments are located on Panel 1-9-5 along with their corresponding bypass pushbuttons. These instruments provide two types of indication and ranges; analog (0 to 60 inches) and digital (-10 to 70 inches). Each instrument has an amber light which illuminates when the signal has been bypassed automatically by the RFW Control System or manually by the Unit Operator.

1.3 System Operation

The RFW Control System will use a level signal provided the system determines the signal to be good and valid. A **GOOD** level signal is one that has **NOT** failed and is on scale. A **VALID** level signal is one that does **NOT** deviate from the average (or median) level by more than 8 inches.

The RFW Control System validates each narrow range level signal by comparing them to the average. A level signal that deviates from the average by more than 8 inches is declared invalid and is bypassed. A level signal that is declared bad by the RFWCS will also be bypassed automatically.

To avoid individual on-scale but faulty level signals from skewing the average, a secondary validation process is used to compare the average level to the median of the valid signals. If the average value differs from the median value by more than 4 inches, then the RFWCS will validate each level signal to the median value instead of the average. In this case, any level signal that varies by more than 8 inches from the median is declared invalid and bypassed by the system.

BFN 2004 ILT 0301 #30

Unit 3 Reactor Vessel Narrow Range instrument, LT-3-53, has a small leak on the reference leg tap. The Feedwater Level Control System (FWLCS) is in 3-element control. The Narrow Range instruments indicate as follows:

- LT-3-53	44 inches
- LT-3-60	33 inches
- LT-3-206	31 inches
- LT-3-253	33 inches

Which ONE of the following indicates the Reactor Vessel Level that the FWLCS is using as the current RPV level? (Round to nearest whole number)

A. 32 inches

B. 33 inches

C. 35 inches

D. 37 inches

Answer: A

QUESTION 45

Which ONE of the following completes the statements?

In accordance with 0-OI-65 section 8.5.1, “SGT Train A decay heat removal using Fan A and SBT Train A Decay Heat Damper,” the Decay Heat Removal Dampers for the Standby Gas Treatment System __(1)__ at a plenum temperature of __(2)___.

A. (1) automatically open
   (2) 150° F

B. (1) automatically open
   (2) 270° F

C. (1) must be manually opened
   (2) 270° F

D. (1) must be manually opened
   (2) 150° F

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	261000 A1.07	
	Importance Rating	2.8	

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: A1.07 SBGTS train temperature

Explanation: **D CORRECT**- First Part: These dampers do NOT automatically open. The fans are designed for post-LOCA conditions where the iodine adsorption has caused the charcoal bed temperature to rise > 150°F. Second Part: Correct, the dampers should be manually opened by the operator in 0-OI-65 section 8.5.1, "SGT Train A decay heat removal using Fan A and SBT Train A Decay Heat Damper at 150°F.

A. Incorrect- First Part: Incorrect, the dampers do NOT automatically open. Plausible if the candidate believes they automatically open on High Temp. Second Part: Correct.

B. Incorrect- First Part: Incorrect, the dampers do NOT automatically open. Second Part: Incorrect, 270° F is the temperature where Iodine desorption is expected to occur per 0-OI-65.

C. Incorrect- First Part: Correct. Second Part: Incorrect, 270° F is the temperature where Iodine desorption is expected to occur per 0-OI-65.

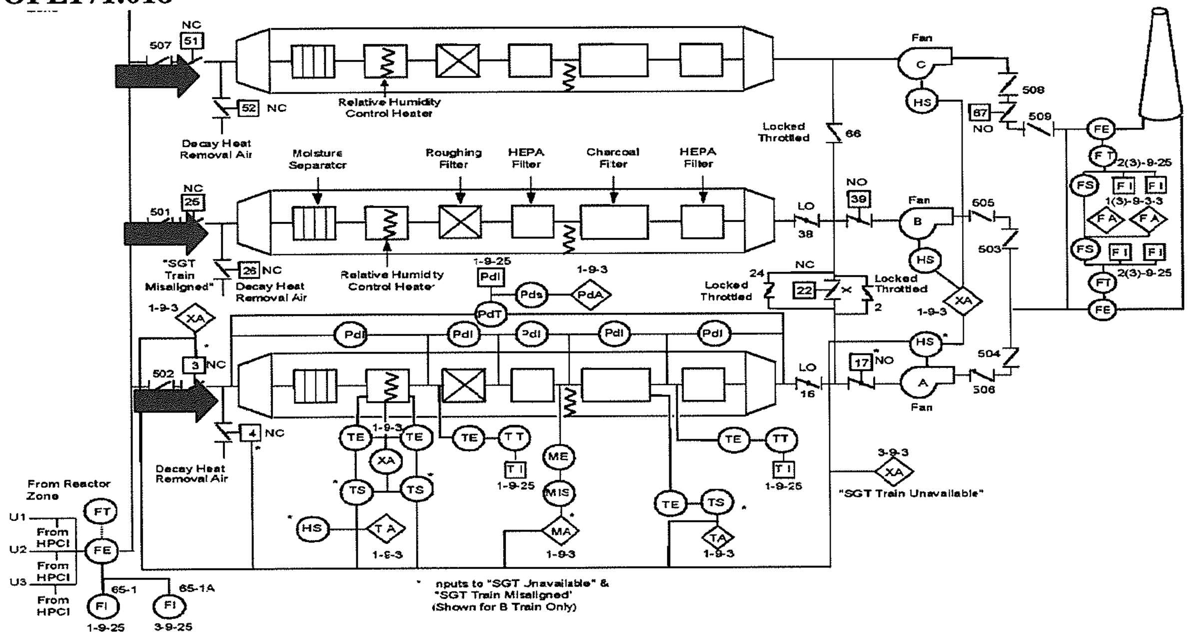
Technical Reference(s): O-OI-65, OPL171.018

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1006 NRC #46
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.



(3) Switch position CLOSE causes annunciation.(HSs 3A, 25A, and 51A)

h. Train outlet valves (65-16, 38, 67)

(1) Normally open

(2) No automatic functions

(3) Switch position CLOSE causes annunciation.(HS-65-67A)

i. Fan inlet valves (65-17, 39)

(1) Normally open

(2) No automatic functions

(3) Switch position CLOSE causes annunciation.(HSs 17A and 39A)

j. Decay heat dampers (65-4, -26 and -52)

(1) Normally closed

(2) No automatic functions

(3) Opening this damper renders the associated train inoperable, for secondary containment purposes

(4) Switch position OPEN causes annunciation.(HS 4A, 26A, 52A)

(5) Decay heat removal should be initiated if charcoal bed temperature rises to 150°F due to iodine adsorption, when the train is no longer in service

(6) Iodine desorption is expected to occur should charcoal bed exit temperature rise to approximately 270°F.

3.0 PRECAUTIONS AND LIMITATIONS

A. Upon a secondary containment isolation, the SGT System is designed to maintain a negative 1/4-inch of H_20 vacuum in Secondary Containment with an inleakage flow of 12,000 cfm.

B. [NRC/C] All three trains will remain in operation during an accident to satisfy single failure criteria and to minimize the potential release of radioactivity from the Reactor Building into the Control Building air supply intake ducts. [NRC NCO 88 0193 004]

C. [NER/C] Steps should be taken to minimize dust loading and to prevent paint vapors, petroleum fumes, welding smoke, and other airborne contaminants from reaching the HEPA filters and charcoal adsorbers. Normal ventilation should be in operation for a minimum of two (2) hours after painting, fire, smoke, or chemical release has terminated prior to operating SGT System. [CAQR SQP890064]

D. If the SGT System is run within 16 hours of the completion of painting in the areas specified in MAI-5.3 or MAI-5.7, Control of Volatile Organic Compounds section, a determination is to be made using those procedures as to whether additional actions are required to verify SGT System operability. Exceeding MAI-5.7 limits requires performing 0-SR-3.6.4.3.2(A)(B)(C) to verify SGT can perform its intended function.

E. When all SGT Trains are secured and any evolution has the potential to discharge radioactive effluents through the main stack, one Unit 2 and one Unit 3 Stack Dilution Fan should remain in operation. This requirement provides clean air flow through the dilution cross-tie to SGT ducts. This prevents the potential back flow of radioactive effluents through the SGT duct work.

F. The alignment of SBGT trains to perform the PURGING function cannot be used when the average reactor coolant temperature is above 212°F since a postulated LOCA could impact the ability for the SBGT trains to perform their safety function. If the primary containment purge system is inoperable and the average reactor coolant temperature is less than or equal to 212°F, the standby gas treatment system venting path will provide the required filtration. The standby gas treatment system is **NOT** the normal means for PURGING operations since the vent path from containment is a much more restrictive flowpath (slower) than the purge system.

G. In the event that the train charcoal filter temperature rises to 150°F due to iodine adsorption following a LOCA, decay heat removal mode of operation should be initiated when the train is no longer in service.

8.5 SGT Train A Decay Heat Removal

NOTES

1) This section provides methods to remove decay heat from the charcoal filters after flow is terminated through SGT Train A.
2) The SGT System will be operated from Panel 1-9-25 throughout this section.
3) Sections 8.5.1 and 8.5.2 utilizes Fan A for decay heat removal; Section 8.5.3 utilizes Fan B. These sections are stand alone.

8.5.1 SGT Train A decay heat removal using Fan A and SGT Train A Decay Heat Damper

[1] **CLOSE** SGTS FILTER BANK A INLET DAMPER, using 0-HS-65-3A. ☐

[2] **VERIFY OPEN** SGTS FAN A INLET DAMPER, using 0-HS-65-17A. ☐

[3] **START** SGTS TRAIN A FAN, using 0-HS-65-18A/1. ☐

[4] **OPEN** SGTS FILTER BANK A DECAY HEAT DAMPER, using 0-HS-65-4A. ☐

INSTRUCTOR NOTES
Obj. V.B.11
Obj. V.C.8
Obj. V.D.6

d. SGT Controls are different for SGT between the units. Operation is recommended from Units 1 and 2 because of SGT shutdown capability and availability of SGT instrumentation.

(1) U-1, Panel 9-25 Train A & B blower & damper controls Train C decay heat damper control Train A & B htr controls, filter bank temp & filter DPs

(2) U-2, Panel 9-25 Train C blower & damper controls Train C filter DPs & temps

(3) U-3, Panel 9-25 Train A, B, C START pushbuttons Train A, B, C valve indications

BFN 1006 NRC #46

Which ONE of the following completes the statements?

In accordance with 0-OI-65, "Standby Gas Treatment System," section 8.5, "SGT Decay Heat Removal," the Decay Heat Removal Dampers for the Standby Gas Treatment System __(1)__ at a plenum temperature of 150°F.

While operating in this mode, SGT flow indication __(2)__ be monitored in the Control Room.

A. (1) automatically open
(2) can

B. (1) automatically open
(2) can **NOT**

C. (1) must be manually opened
(2) can **NOT**

D. (1) must be manually opened
(2) can

QUESTION 46

Which ONE of the following statements describes a qualified Electrical Distribution path for one of the required Technical Specification offsite circuits for Units 1 and 2?

A. From the 500kV switchyard, through USST 1A to 4.16kV Unit Board 2B, to 4.16kV Shutdown Bus 1, to 4.16kV Shutdown Boards C and D

B. From the 500kV switchyard, through USST 1B to 4.16kV Unit Board 1A, to 4.16kV Shutdown Bus 1, to 4.16kV Shutdown Boards A and B

C. From Athens 161kV transmission system, through CSST B to Unit Board 2B, to Start Bus 1A, to Shutdown Bus 2, to 4.16kV Shutdown Boards A and B

D. From Trinity 161kV transmission system, through CSST A to Unit Board 2A, to Start Bus 1A, to 4.16kV Shutdown Bus 2, to 4.16kV Shutdown Boards A and B

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262001.K2.01	
	Importance Rating	3.3	

262001 Knowledge of electrical power supplies to the following: K2.01 Off-site sources of power

Explanation: **B CORRECT**- From the 500kV switchyard, through USST 1B to 4.16kV Unit Board 1A, to 4.16kV Shutdown Bus 1, to 4.16kV Shutdown Boards A and B.

A. Incorrect because USST 2B is the connection to 2B Unit Board. Plausible if the candidate cannot remember the station electrical distribution flowpath from the 500kV switchyard to the 4.16kV Shutdown Boards.

C. Incorrect because the path must be through Shutdown Bus 1. Plausible if the candidate cannot remember the station electrical distribution flowpath from the 500kV switchyard to the 4.16kV Shutdown Boards.

D. Incorrect because the Unit Board must be either 1A or 2B. Plausible if the candidate cannot remember the station electrical distribution flowpath from the 500kV switchyard to the 4.16kV Shutdown Boards.

Technical Reference(s): Technical Specification 3.8.1, OPL171.036

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Unit 1 and 2 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

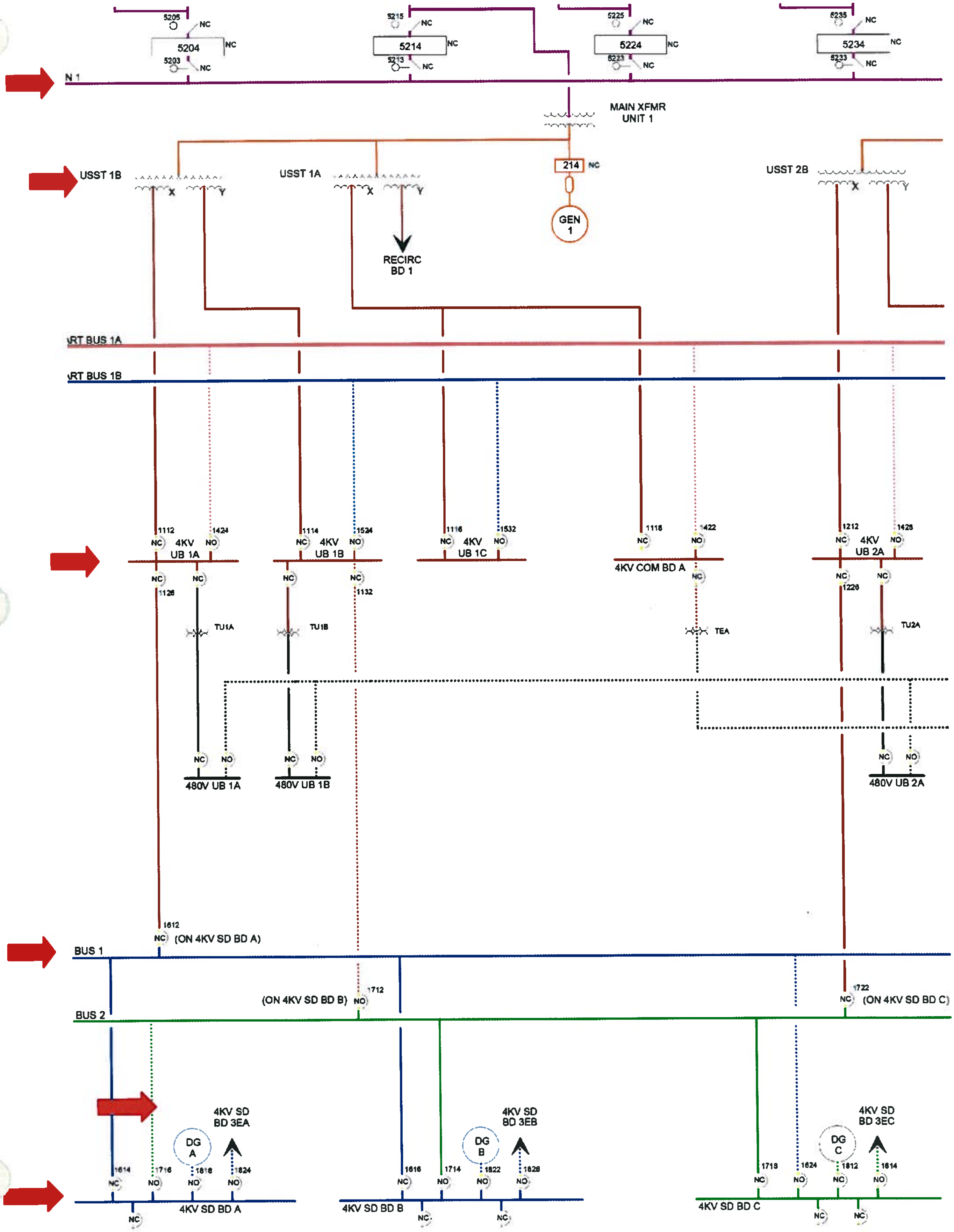
APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------NOTE------------------------------

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME



5205
NC
5204
NC
5203
NC
5215
NC
5214
NC
5213
NC
5225
NC
5224
NC
5223
NC
5235
NC
5234
NC
5233
NC
N 1
MAIN XFMR
UNIT 1
USST 1B
USST 1A
USST 2B
X
Y
214
NC
GEN
1
RECIRC
BD 1
START BUS 1A
START BUS 1B
1112
1424
4KV
UB 1A
1114
1524
4KV
UB 1B
1116
1532
4KV
UB 1C
1118
1422
4KV COM BD A
1212
1428
4KV
UB 2A
1126
1132
1226
TU1A
TU1B
TEA
TU2A
NC
NO
480V UB 1A
480V UB 1B
480V UB 2A
1612
(ON 4KV SD BD A)
BUS 1
1712
(ON 4KV SD BD B)
1722
(ON 4KV SD BD C)
BUS 2
4KV SD
BD 3EA
DG
A
1614
1716
1818
1824
4KV SD BD A
4KV SD
BD 3EB
DG
B
1616
1714
1822
1828
4KV SD BD B
4KV SD
BD 3EC
DG
C
1718
1624
1812
1814
4KV SD BD C

QUESTION 47

At panel 0-9-23-7, the following conditions exist for the "A" 4KV Shutdown Board:

- 0-43-211-A, 4kV SD BD A AUTO/LOCKOUT RESET switch is in the TRIPPED condition
- Alt Supply Breaker is CLOSED
- Norm Supply Breaker is OPEN

Which ONE of the following identifies how the 4KV system will respond if the Unit Operator places the 0-43-211-A switch to the RESET position?

The Shutdown Board will __(1)__ Transfer AND will be supplied from __(2)__.

A. (1) FAST
(2) Shutdown Bus 1

B. (1) FAST
(2) Shutdown Bus 2

C. (1) SLOW
(2) Shutdown Bus 1

D. (1) SLOW
(2) Shutdown Bus 2

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262001.K5.02	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: K5.02 Breaker control

Explanation: **C CORRECT**- First Part: The only automatic transfer of power on a shutdown board is a delayed (slow) transfer. Second Part: The normal power supply to the "A" 4Kv S/D Board is Shutdown Bus 1.

A. Incorrect. First Part: Incorrect. ALL 4kv Shutdown Board AUTO transfers are SLOW (delayed). Plausible in that manual transfers are FAST. Second Part: Correct.

B. Incorrect. First Part: Incorrect. ALL 4kv Shutdown Board AUTO transfers are SLOW (delayed). Plausible in that manual transfers are FAST. Second Part: Incorrect, Shutdown Bus 2 supplies ALTERNATE 1 power to 4kv Shutdown Bus "A".

D. Incorrect. First Part: Correct. Second Part: Incorrect, Shutdown Bus 2 supplies ALTERNATE 1 power to 4kv Shutdown Bus "A".

Technical Reference(s): OPL171.036, 0-OI-57A

Proposed references to be provided to applicants during examination: None

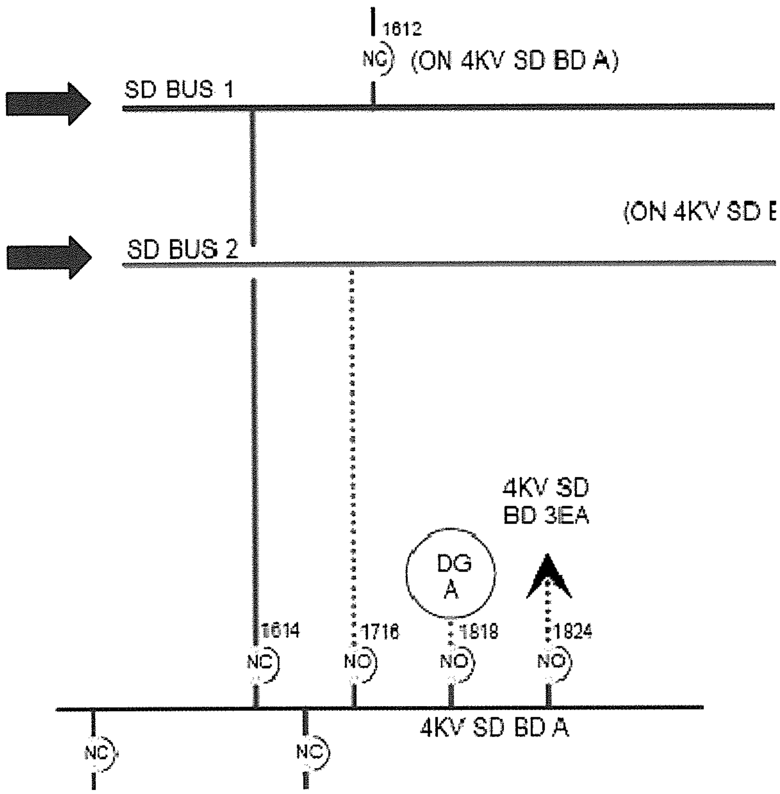
Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 1006 NRC # 47

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.



OPL171.036

(7) CASx (CASA or CASB) accident signal (after 5 second delay via BBRX relay) — -122" RxVL OR 2.45 DWP AND < 450# RPV

I. 4kV Shutdown Boards (Normal Power Seeking)

Refer to prints 15E-500 series Key Diagram of STDBY Aux. Power System
Obj. V.B.6.c
Obj. V.C.1.c
Obj. V.D.6.c

1. Power sources

a. 4kV supplies to each U1/2 Shutdown Board: are as follows:

Board	NORMAL Supply
A	Shutdown Bus 1
B	Shutdown Bus 1
C	Shutdown Bus 2
D	Shutdown Bus 2

The first alternate is from the other Shutdown Bus. The second alternate is from the diesel generator. The third alternate is from the U3 diesel generators via a U3 Shutdown Board.

SBO

3 ½ via bustie board

½ ½ via other SD Bus

7. Shutdown Board Transfer Scheme

a. The only automatic transfer of power on a shutdown board is a delayed (slow) transfer. In order for the transfer to take place, the bus transfer control switch (43Sx) must be in AUTOMATIC.

Obj. V.B.8.c
Obj. V.C.2.c
Obj. V.D.8.c
Procedural Adherence when transferring boards

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Illustration 1
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Auxiliary Power Supplies and Bus Transfer Schemes

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE	REMARKS
9	4kV Common Bd. A	Unit SS TR 1A (BKR 1118)	Start Bus 1A (BKR 1422)	Automatic delayed transfer from the normal to the alternate source is initiated by undervoltage on the normal source, subject to voltage check on the alternate source, and automatic return is initiated by normal voltage on normal source. Manual transfers in either direction are fast type.
10	4kV Common Bd. B	Unit SS TR 2A (BKR 1218)	Start Bus 1B (BKR 1522)	

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE 1	ALTERNATE 2	ALTERNATE 3
11	4-kV Shutdown Bd. A	Shutdown Bus 1 (BKR 1614)	Shutdown Bus 2 (BKR 1716)	Diesel Generator A (BKR 1818)	Shutdown Bd. 3EA (BKR 1824)
12	4-kV Shutdown Bd. B	Shutdown Bus 1 (BKR 1616)	Shutdown Bus 2 (BKR 1714)	Diesel Generator B (BKR 1822)	Shutdown Bd. 3EB (BKR 1828)
13	4kV Shutdown Bd. C	Shutdown Bus 2 (BKR 1718)	Shutdown Bus 1 (BKR 1624)	Diesel Generator C (BKR 1812)	Shutdown Bd. 3EC (BKR 1814)
14	4kV Shutdown Bd. D	Shutdown Bus 2 (BKR 1724)	Shutdown Bus 1 (BKR 1618)	Diesel Generator D (BKR 1816)	Shutdown Bd. 3ED (BKR 1826)

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Illustration 1
(Page 5 of 7)

Auxiliary Power Supplies and Bus Transfer Schemes

REMARKS: Automatic delayed transfer from the normal to alternate 1 source is initiated by undervoltage on the normal source and automatic return is initiated by normal voltage on normal source. These transfers are blocked after time delay in the presence of an accident signal. When an accident signal is present, alternate 1 source breakers are tripped. Also, on 4kV Shutdown Bd A, B, C and D, the common accident signal auto trip from U-3 bus tie breakers (Alternate 3), has been removed. All diesel generators are automatically started by an accident signal, loss of voltage on its shutdown board for 1.5 seconds or degraded voltage for 4 seconds on its shutdown board. After five (5) seconds with no voltage on the shutdown board, all its supply breakers and all its loads except 4160-480V transformers are automatically tripped. Alternate 2 source is then automatically connected. A second level voltage protection is provided for each 4kV shutdown board which will operate an undervoltage relay. If voltage reduces to that board and after 7.43 seconds (from the initial time zero) the feed to the board is tripped, the auto transfer is blocked and motor breakers on the board are tripped. 1.36 seconds later the DG breaker closes in on that shutdown board. Manual return to the normal auxiliary power system is permitted if normal auxiliary power system voltage returns and if a unit is **NOT** in early stage of accident. Units 1 and 2 shutdown boards can be manually tied to their respective 3 unit shutdown board. When doing this, Unit 3's breaker must be closed in on a dead line (interlocked t prevent closing in on an energized line) then Units 1 and 2 respective shutdown breaker can be synchronized to tie the two boards together. Provision is included for backfeeding diesel generator power from the shutdown boards into the 4160V unit boards for hot standby shutdown cooling if all plant power, other than diesel generator power, is lost. For this purpose, means are provided to manually synchronize 4kV shutdown boards.

BFN 1006 NRC #47

47. 262001 K5.02

At panel 0-9-23-7, the following conditions exist for the "A" 4KV Shutdown Board:

- 0-25-211-A/24A, 4kV SD BD A BKR 1716 SYNC switch is ON
- 0-43-211-A, 4kV SD BD A AUTO/LOCKOUT RESET switch is in the TRIPPED condition
- Alt Supply Breaker is CLOSED
- Norm Supply Breaker is OPEN

Which ONE of the following identifies how the 4KV system will respond if the Unit Operator places the 0-43-211-A switch to the RESET position?

The Shutdown Board will __(1)__ Transfer **AND** will be supplied from __(2)__.

A. (1) FAST
(2) Shutdown Bus 1

B. (1) FAST
(2) Shutdown Bus 2

C. (1) SLOW
(2) Shutdown Bus 1

D. (1) SLOW
(2) Shutdown Bus 2

QUESTION 48

Unit 2 is operating at 90% when an electrical fault occurs causing a loss of Unit Preferred Panel 2-9-9 Cabinet 6. The Unit Supervisor has directed increased monitoring of affected plant equipment until Electrical Maintenance can resolve the problem and restore power to Cabinet 6.

In accordance with 2-AOI-57-4, Loss of Unit Preferred, which ONE of the statements below describes the affected equipment that receive increased monitoring and the reason?

A. Monitor Drywell to Suppression Chamber ΔP because the loss of power to PS-64-137B causes the ΔP compressor to start and run continuously.

B. Monitor Panel 2-9-3 indications due to a loss of ECCS and RCIC Trip System Bus Power Monitors.

C. Monitor Control Rod Drive seal temperatures due to CRD SYS FLOW CONTROL VLV 1A/B, 2-FCV-85-11(A/B) failing closed.

D. Monitor Suppression Chamber to Reactor Building ΔP due to the associated vacuum breakers 2-FCV-64-20 and 2-FCV-64-21 failing open.

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002.K3.17	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: K3.17 Process monitoring

Explanation: **C CORRECT** : Per 2-AOI-57-4, Loss of Unit Preferred, CRD SYS FLOW CONTROL VLV 1A/B, 2-FCV-85-11(A/B) CLOSE ON A LOSS OF POWER TO Panel 2-9-9 cabinet 6 Unit Preferred resulting in a loss of cooling water flow to CRD seals. Control Rod Drive seal temperatures should therefore be monitored.

A- Incorrect. Plausible since this is a result of a loss of I&C Bus A, 2-AOI-57-5A.

B- Incorrect. Plausible because this is affected by a loss of annunciators to panel 2-9-3, 2-AOI-57.9.

C- Incorrect. Plausible in that this is a result of a loss of the Division I ECCS ATU Panel 9-81, 2-AOI-57-11.

Technical Reference(s): 2-AOI-57-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: No

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

4.2 Subsequent Actions (continued)

NOTE

CRD SYS FLOW CONTROL VLV 1A/B, 2-FCV-85-11A/B closes on a loss of power to Panel 2-9-9 Cabinet 6 Unit Preferred resulting in a loss of normal cooling water flow to CRD seals. CRD temperatures should be monitored and operation with 2-FCV-85-11A/B closed, limited to less than 1 hour. 2-FCV-85-11A/B can be manually opened, if required. **REFER TO** 2-OI-85.

[4] **PERFORM** the following for the CRD system:

[4.1] **MONITOR** CRD temperatures while 2-FCV-85-11(A/B) are closed. ☐

[4.2] **IF** CRD seal temperatures rise to the alarm setpoint OR the Unit Preferred system cannot be restored within one hour, **THEN**

DISPATCH personnel to MANUALLY OPEN 2-FCV-85-11(A/B). **REFER TO** 2-OI-85 (Otherwise N/A). ☐

QUESTION 49

The following are indicated on Battery Board Room 2 Panel 1:

- BATTERY BOARD 2 250V DC BUS GND INDICATOR 0-GI-280-0002/103 is (-)4V
- Voltage (0-EI-280-0002/102) is 255 VDC
- Amps (0-II-280-0002/101) is 340 Amps Discharge

The following are indicated on Battery Board Room 250V Battery Charger 2A:

- Charger DC Voltage is 255 VDC
- AC ON light is lit
- All other lights are extinguished
- The equalize timer is set for 0 hours

Which ONE of the following describes the expected Battery Board Room No. 2 Panel 1 voltage trend and the reason for that trend with NO operator action?

A. LOWER, because the bus load exceeds the charger's capacity

B. LOWER, because a ground exceeding the allowable normal value is indicated

C. RISE, because the charger's output voltage setting is too high

D. RISE, because an equalizing charge has been completed

Answer: A

QUESTION 49

The following are indicated on Battery Board Room 2 Panel 1:

- BATTERY BOARD 2 250V DC BUS GND INDICATOR 0-GI-280-0002/103 is (-)4V
- ~~Battery Board 2 Panel 1~~ Voltage (0-EI-280-0002/102) is 255 VDC
- ~~Battery Board 2 Panel 1~~ Amps (0-II-280-0002/101) is 340 Amps ~~Discharge~~ ?

The following are indicated on Battery Board Room 250V Battery Charger 2A:

- Charger DC Voltage is 255 VDC
- Charger DC Current is 300 Amps +
- AC ON light is lit
- All other lights are extinguished
- The equalize timer is set for 0 hours

WITH NO OPERATOR ACTION, which ONE of the following describes the expected Battery Board Room No. 2 Panel 1 voltage trend and the reason for that trend?

A. LOWER, because the bus load exceeds the charger's capacity

B. LOWER, because a ground exceeding the allowable normal value is indicated

C. ~~RISE~~, because the charger's output voltage setting is ~~too high~~ the same as the battery board voltage

D. ~~RISE~~, because an equalizing charge has been completed

Answer: **A**

REMAIN THE SAME

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

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

Explanation: **A** CORRECT – Battery Board 2 Panel 1Amps (340 Amps) Discharge is greater than Charger DC Current is 300Amps.

B Incorrect – Plausible because a small ground is indicated but is within the allowable value of ±30 V

C Incorrect – Plausible, because while the charger's output is 255V, the normal value is greater than 250V.

D Incorrect –Plausible because with the amps out greater than amps in, it does not matter that the charger was or may be in an equalizing charge.

Technical Reference(s): 0-OI-57D

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	X
	New	

Question History: Previous NRC : None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

6.1 Normal Operations (continued)

TABLE 2
250 VOLT DC UNIT BATTERY SYSTEM

LOCATION	PARAMETER	NORMAL RANGE
Battery Board Room No. 2(1,3) 250V Charger 2A (1,3,2B) Control Bay 593'	DC Volts	Greater than 250 Volts
	DC Amperes	Greater than zero less than 300 amps
	Power on Light	Illuminated
	Transformer Overtemp Light	Extinguished
	Overvoltage DC Light	Extinguished
	Undervoltage DC Light	Extinguished
	Undervoltage AC Light	Extinguished
Battery Board Room No. 2(1,3) Panel 1 Control Bay 593'	DC Volts	Greater than 250 Volts
	DC Amperes	Zero amps
	Bus Ground Indication(1)	Zero Volts
Battery Board Room No. 4 250V Charger 4 Turbine Bldg. 586'	DC Volts	Greater than 250 Volts
	DC Amperes	Greater than zero less than 300 amps
	Power on Light	Illuminated
	Transformer Overtemp Light	Extinguished
	Overvoltage DC Light	Extinguished
	Undervoltage DC Light	Extinguished
	Undervoltage AC Light	Extinguished

(1) **IF** a ground of an absolute valve greater than or equal to 30 volts is indicated, **THEN**

REFER TO 0-GOI-300-2.

5.1.2 Placing the 250V Battery 2 in Service to Battery Board 2 (continued)

[5] **PERFORM** the following in Battery Board Room 2:

[5.1] **VERIFY** that 250V BATTERY BOARD INSTRUMENTATION, 0-BKR-280-0002/104 is ON. ☐

[5.2] **VERIFY** that 250V DC INCOMING COPPER SLUG, 0-FUDS-280-0002/111 is CLOSED. ☐

[5.3] **CLOSE** 250V BATTERY 2 TIE TO DC BOARD 2, 0-BKR-280-0002/110 by pumping breaker handle four times. ☐

[5.4] **CHECK** the following indications of normal operation on Panel 1:

- BATTERY BOARD 2 250V DC BUS VOLTMETER, 0-EI-280-0002/102 indicates greater than 250 Volts. ☐
- BATTERY 2 AMMETER, 0-II-280-0002/101 indicates zero amps. ☐
- BATTERY BOARD 2 250V DC BUS GND INDICATOR, 0-GI-280-0002/103 indication is less than 30 (plus or minus). ☐

[6] **IF** battery voltage is less than 250 Volts as indicated on BATTERY BOARD 2 250V DC BUS VOLTMETER, 0-EI-280-0002/102, **THEN**

NOTIFY the Unit Supervisor/SRO of low battery voltage. ☐

QUESTION 51

All three Units are operating at 100% power when a series of tornados in the area causes a loss of all offsite power. A "DIESEL GEN 3A OVERLOAD" annunciator is received on panel 3-9-23A when the 3A diesel generator starts. Subsequently, a loss of coolant accident (LOCA) occurs and an accident signal is received on Unit 3.

With NO operator action, which ONE of the following describes the status of the Unit 3 Core Spray pumps <u>one minute after</u> the LOCA?

A. No Core Spray pump is running

B. Core Spray pumps 3B and 3D are running ONLY

C. Core Spray pumps 3B, 3C, and 3D are running ONLY

D. All Core Spray pumps are running

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	264000 K3.03	
	Importance Rating	4.1	

Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: Major loads powered from electrical buses fed by the emergency generator(s)

Explanation: **D CORRECT** : All four unit 3 diesels are running and supplying their respective S/D Boards. Generator overload is not a diesel generator trip.

A. Incorrect. Plausible because the Unit 3 accident signal trips the diesel output breakers. However they would reclose due to the loss of offsite power condition.

B. Incorrect. Plausible because the candidate may consider the 3A D/G tripped on overload. There has to be power on the 3A S/D Board for the companion Core Spray pump 3C to auto start.

C. Incorrect. Plausible because the candidate may consider the 3A D/G tripped on overload and only the 3A Core Spray pump affected.

Technical Reference(s): 3-ARP-9-23A, OPL171.045

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

3-ARP-9-23A

BFN Unit 3	Panel 9-23 3-XA-55-23A	3-ARP-9-23A Rev. 0017 Page 17 of 42

DIESEL GEN 3A OVERLOAD
11

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Sensor/Trip Point:

Aux Relay:	Relay:	
(51X)	(51)	DG "B" phase output
3-51X-082-2547A	3-51-082-2547A	current ≥ 495 amps

Sensor Location: Protective Relay Cab. (25-47A)
Elevation 565
Diesel Generator Bldg.

Probable Cause:

A. Improper loading of generator.
B. Malfunction of voltage regulator.
C. Malfunction of speed controller.
D. Mode switch failure.
E. Sensor malfunction.

Automatic Action: Trips Shutdown Bd 3EA normal (1334), 3-BKR-211-03EA/007, and alternate (1726), 3-BKR-211-03EA/004, feeder breaker if diesel is parallel with system.

OPL171.045 Core Spray

i. CS Pump will not auto start if companion Pump's 4KV SD BD doesn't have power to prevent a single pump auto start and runnout in a loop. Both boards must have either NVA logic energized or DGVA logic deenergized to pick up the start time delay relay.

QUESTION 52

All three Units are operating at 100% power with Control Air Compressors in their normal line up. The following annunciators are received:

- SCRAM PILOT AIR HEADER PRESS LOW, **2**-9-5B, (Window 28)
- AIR COMPRESSOR ABNORMAL, **1**-9-20 (Window 29)

CONTROL AIR PRESSURE, on Panel **1**-9-20, indicates 84 psig and lowering.

Which ONE of the following completes the statements?

A cause of the AIR COMPRESSOR ABNORMAL annunciator is ___(1)___.

The operator required action(s) per 0-AOI-32-1, Loss of Control and Service Air Compressors, is(are) to ___(2)___.

A. (1) a trip of Air Compressor F
(2) verify OPEN **2**-FCV-32-28, 29, and 91, CONTROL AIR ISOLATION VALVE SECONDARY CTMT

B. (1) a trip of Air Compressor F
(2) verify OPEN 0-PCV-33-1, SERVICE AIR TO CONTROL AIR ISOLATION VALVE

C. (1) a trip of Air Compressor G
(2) verify OPEN **2**-FCV-32-28, 29, and 91, CONTROL AIR ISOLATION VALVE SECONDARY CTMT

D. (1) a trip of Air Compressor G
(2) verify OPEN 0-PCV-33-1, SERVICE AIR TO CONTROL AIR ISOLATION VALVE

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000 G2.4.45	
	Importance Rating	3.7	

300000 Instrument Air System. G2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.

Explanation: **D CORRECT** : First Part: A trip of Air Compressor G will cause the AIR COMPRESSOR ABNORMAL, 1-9-20 (Window 29) annunciator. Second Part: 0-AOI-32-1, Loss of Control and Service Air Compressors, directs the Unit 2 operator to verify 0-PCV-33-1, SERVICE AIR TO CONTROL AIR ISOLATION VALVE **IF** CONTROL AIR PRESSURE, is less than 85psig.

A. Incorrect. First part: Incorrect, a trip of Air Compressor **F** will NOT cause the AIR COMPRESSOR ABNORMAL, 1-9-20 (Window 29) annunciator. Plausible, the candidate could easily confuse the three service air compressors E, F, and G. Second Part: Incorrect, Plausible because 0-AOI-32-1, Loss of Control and Service Air Compressors, directs the Unit 2 operator to verify OPEN 2-FCV-32-28, 29, and 91, CONTROL AIR ISOLATION VALVE SECONDARY CTMT **IF** CONTROL AIR PRESSURE, **is within normal operating range (85-110psig) AND** SCRAM PILOT AIR HEADER PRESS LOW, Illuminates.

B. Incorrect. First part: Incorrect, a trip of Air Compressor **F** will NOT cause the AIR COMPRESSOR ABNORMAL, 1-9-20 (Window 29) annunciator. Plausible, the candidate could easily confuse the three service air compressors E, F, and G. Second Part: Correct.

C. First part: a trip of Air Compressor **F** will NOT cause the AIR COMPRESSOR ABNORMAL, 1-9-20 (Window 29) annunciator. Second Part: Incorrect. Plausible because 0-AOI-32-1, Loss of Control and Service Air Compressors, step 4.2[5] directs the Unit Operator to OPEN 0-PCV-33-1, SERVICE AIR TO CONTROL AIR ISOLATION VALVE **IF** Control Air pressure is less than 85 psig.

Technical Reference(s): 0-AOI-32-1, 1-ARP-9-20B, 2-ARP-9-5B, 0-ARP-25-118A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

AIR COMPRESSOR ABNORMAL 0-XA-32-49	
	29

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Sensor/Trip Point:

Actuation of any alarm on local panel 0-LPNL-925-0118A.

Sensor Location: Panel 0-LPNL-925-0118A Elevation 565 Col. T-2 N-LINE

Probable Cause:

A. **G Control Air Compressor**: Function on microcontroller reaches Alert or Shutdown setpoint listed below:

FUNCTION	ALERT	SHUTDOWN
Vibration Stage 1	.80 mil	1.0 mil
Vibration Stage 2	.75 mil	.94 mil
Lube Oil Pressure	18 psig	16 psig
Lube Oil Temperature High	125°F	130°F
Lube Oil Temperature Low	90°F	85°F
Air Temperature Stage 1	120°F	125°F
Discharge Air Temperature	120°F	125°F
Seal Air Pressure Low	---	6 psig

B. A,B,C,D Control Air Compressors alarm from 1-LPNL-925-118A.

Automatic Action:

Possibility of one or more of the following to occur:

- Control Air Compressor G trip.
- Control Air Compressor G surge (auto loading and unloading).
- Control Air Compressor A,B,C,D trip.
- PCV-33-1 opens at Control Air pressure 85 psig lowering.

Continued on Next Page

Annunciator Window Legend

COMPR A ABNORMAL XA-32-6A 1	COMPR A LOW VOLTAGE TRIP EA-32-27A 2	SERVICE AIR SUPPLY TO CONTROL AIR (0-PCV-33-1 OPEN) XA-32-33 3	4
COMPR B ABNORMAL XA-32-6B 5	COMPR B LOW VOLTAGE TRIP EA-32-27B 6	CONT & SER AIR DISCH PRESS LOW FA-32-2 7	8
COMPR C ABNORMAL XA-32-6C 9	COMPR C LOW VOLTAGE TRIP EA-32-27C 10	CONT AIR PRESS LOW PA-32-1 11	12
COMPR D ABNORMAL XA-32-6D 13	COMPR D LOW VOLTAGE TRIP EA-32-27D 14	15	16
COMPR G ABNORMAL XA-32-2901 17	18	19	20

0-XA-32-49, AIR COMPRESSOR ABNORMAL, Window 29
(Page 2 of 2)

Operator Action:

A. **CHECK** for any other Control Air alarms. ☐

B. **CHECK** Control Air Header pressure using:
- CONTROL AIR PRESSURE, 1-PI-32-20, on Panel 1-9-20. ☐
- ICS (Air Compressor G display). ☐

> [1] With a sudden drop in Control Air System demand, G Control Air Compressor can experience some air backflow through the compressor. This is known as a surge condition. When this surge flow reaches 6 psid as sensed by 0-PDS-032-2968, the compressor will alarm (TROUBLE light illuminated) and automatically unload (Inlet Flow Control Valve goes to throttled position and Bypass Control Valve goes fully open). The alarm will reset and the compressor will automatically reload after 6 seconds for the first 3 surges in 10 minutes. If a fourth surge occurs within 10 minutes, the compressor will stay in an alarm condition and will remain unloaded until the RESET key is depressed and held for 10 seconds, on the CMC Controller

C. **IF** more than three compressor surges occur within ten minutes, **THEN**: (Otherwise N/A)
- **CHECK** the compressor remains in an unloaded condition until the RESET key on the CMC Controller is reset. ☐
- **PERFORM** the following on the CMC Controller to reset the compressor
  - **DEPRESS** RESET key and hold for 10 seconds on the CMC Controller. ☐
    - **CHECK** ALARM light on the CMC Controller extinguished. ☐
    - **CHECK** Compressor automatically reloads ☐

D. **IF** Control Air pressure is lowering (normal press approximately 100 psig), **THEN**
OPEN SERVICE AIR SUPPLY TO CONTROL AIR, 0-HS-33-1A/1, on Panel 1-9-20. ☐

E. **DISPATCH** personnel to investigate Control Air compressors for abnormalities or alarms. ☐

F. **COORDINATE** with Unit 2 and Unit 3. ☐

G. **NOTIFY** Unit Supervisor. ☐

H. **REFER TO** 0-OI-32. ☐

References: 1-45E620-12-2 0-45E769-5 0-45E779-4

SCRAM PILOT AIR HEADER PRESS LOW 2-PS-85-38B 28	Sensor/Trip Point: 2-PS-85-38 66 psig

(Page 1 of 2)

Sensor Location: 2-LPNL-925-0018B
El 565'
Rx Bldg
Col R-12 N-LINE

Probable Cause:

A. SI (or SR) in progress.
B. Failure of scram pilot header pressure regulators 2-PCV-85-66 or 2-PCV-85-67.
C. Air supply valve 2-FCV-32-91 failed closed.
D. Control Air System failure.
E. Sensor malfunction.

Automatic Action: None

Operator Action:

A. On Panel 2-9-20, **CHECK** CONTROL AIR PRESSURE, 2-PI-32-88. ☐
B. **IF** low **THEN**
REFER TO 0-AOI-32-1. ☐
C. On Panel 2-9-20, **CHECK OPEN** 2-FCV-32-91. ☐
D. **DISPATCH** personnel to check CRD SCRAM VALVE PILOT AIR HDR PRESS, 2-PI-85-38 on 2-LPNL-925-0018B, Elevation 565', Rx Bldg. ☐
E. Behind 2-LPNL-925-0018A, Rx bldg El 565', **CHECK** CRD CA FILTER INLET, 2-PI-85-66A (-67A) and CRD CA FILTER OUTLET, 2-PI-85-66B (-67B). ☐
F. **IF** DP across CRD CA **FILTER** to 2-PCV-85-67 is high, **THEN** ☐
1. **VERIFY OPEN** 2-85-244, AIR HEADER XTIE SOV. ☐
2. **CLOSE** 2-85-243, AIR HEADER SOV and 2-85-262, HEADER SHUTOFF VLV. ☐
3. BL0W DOWN filter by opening and then releasing petcock on filter. ☐
4. **OPEN** 2-85-243, AIR HEADER SOV and 2-85-262, HEADER SHUTOFF VLV. ☐

Continued on Next Page

4.2 Subsequent Actions (continued)

[5] **IF** CONTROL AIR PRESSURE, as indicated by 1-PI-32-20 on Panel 1-9-20 or 2(3)-PI-32-88 on Panel 2(3)-9-20, is less than 85 psig, **THEN** (Otherwise N/A)

- **PLACE** SERVICE AIR SUPPLY TO CONTROL AIR, 0-HS-33-1A/1, on Panel 1-9-20, in OPEN. ☐
- **PLACE** SERVICE AIR SUPPLY TO CONTROL AIR, 0-HS-33-1A/3 on Panel 3-9-20 in OPEN. ☐

[6] **IF** CONTROL AIR PRESSURE, as indicated by 2(3)-PI-32-88 on Panel 2(3)-9-20, is within normal operating range **AND** SCRAM PILOT AIR HEADER PRESS LOW, (2(3)-XA-55-5B, Window 28) Illuminates, **THEN** (Otherwise N/A)

- **VERIFY OPEN** CONTROL AIR ISOLATION VALVE SECONDARY CTMT, 2(3)-FCV-32-28. (**IF** necessary, **THEN**

  Refer to Attachment 1 for control switch numbers and locations.) ☐

- **VERIFY OPEN** CONTROL AIR ISOLATION VALVE SECONDARY CTMT, 2(3)-FCV-32-29. (**IF** necessary, **THEN**

  Refer to Attachment 1 for control switch numbers and locations.) ☐

- **VERIFY OPEN** CONTROL AIR ISOLATION VALVE SECONDARY CTMT, 2(3)-FCV-32-91. (**IF** necessary, **THEN**

  Refer to Attachment 1 for control switch numbers and locations) ☐

- **VERIFY OPEN** Unit 2(3) bypass valves around Control Air Secondary Containment Isolation Valves 2(3)-BYV-032-0576C, 2(3)-BYV-032-0577C, 2(3)-BYV-032-2385. (**IF** necessary, **THEN**

  Refer to Attachment 1 for valves and locations) ☐

b. Controls

TP-3

(1) Valve controls

(a) Secondary Containment isolation valves

TP-1

i. Control air to reactor building isolation valves (32-28 and 32-29)

ii. Control air to the CRD modules isolation valve (32-91)

a. Each of these valves has two solenoids. There are two handswitches for each solenoid. Therefore, two handswitches in the Control Room and two handswitches local in the Turbine Building for each valve.

b. Normal line-up has all four handswitches in the open position. Placing any one of the four handswitches in the open position will open the valve. However, all four handswitches must be in the closed position for the valve to close.

Manual bypass valves around 32-28, 32-29, 32-91 on all 3 units are open per valve checklist. This change occurred Jan. 2006.
EDC 65754
DCN 51122
PIC 66696
EDC 64201

c. Units 2 and 3 are the only ones with valves 32-28, 32-29, and 32-91 still in place. Unit 1 valves were removed and replaced with pipe per PIC 66696 to DCN 51122. EDC 64201 had already downgraded these valves from Safety Related Secondary Containment Isolation Valves to Non-Quality Related system isolation valves. These valves are not required for operation. Control Room and local handswitches for these valves were removed also. The manual bypass valve around each valve is still in place.

Attention to detail

QUESTION 53

All three units are operating at 100% power in a normal plant lineup when the operating A3 EECW Pump trips. The Unit Operator attempts to start the C3 EECW pump and it fails to start.

Which ONE of the following describes the impact of the A3 EECW Pump trip and failure of the C3 EECW pump to start?

The backup cooling water supply to the ______ has been lost.

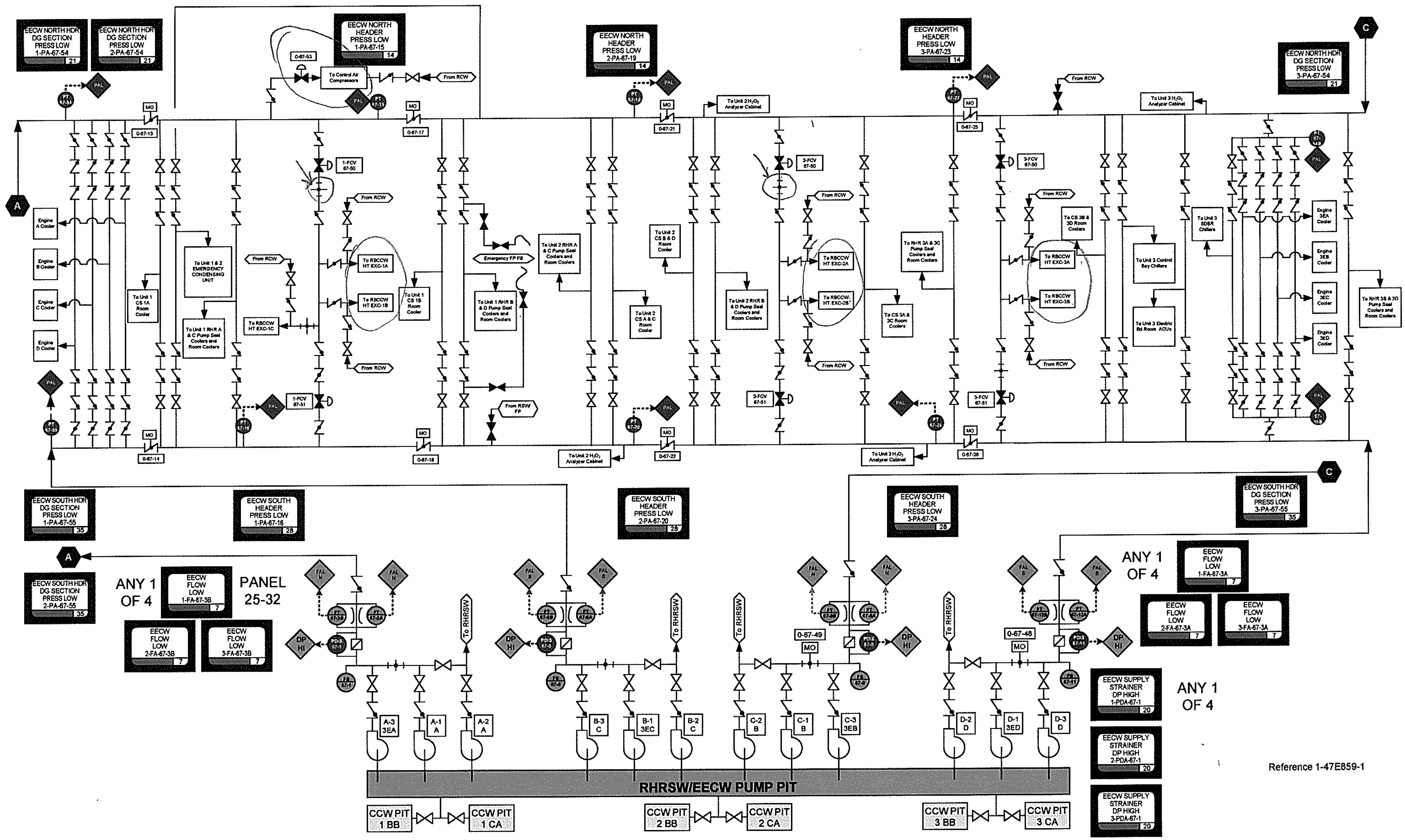
A. control air compressors ONLY

B. Unit 1 and 2 RBCCW heat exchangers ONLY

C. control air compressors and the Unit 3 RBCCW heat exchanger ONLY

D. control air compressors and all RBCCW heat exchangers

Answer: C



Reference 1-47E859-1

3.0 PRECAUTIONS AND LIMITATIONS (continued)

F. RHRSW Pumps B3 and D3 (14 second delay if diesel supplying board, 28 second delay if normal voltage available), and when lined up for EECW operation, A1 and C1, will auto-start when:

1. Any Unit 1 or 2 Core Spray pump starts.

2. Any Unit 1 or 2 Diesel Generator starts.

G. RHRSW Pumps A3 and C3, and when lined up for EECW operation, B1 and D1, will auto-start when (14 second delay if diesel supplying board, 28 second delay if normal voltage available):

1. Any Unit 3 Core Spray pump starts.

2. Any Unit 3 Diesel Generator starts.

H. EECW System backup water supply valve (FCV-67-53) to the control air compressors will auto open at 30 psig lowering RCW pressure, if EECW pressure is ≥ 106 psig. The valve will auto close on EECW pressure dropping to < 106 psig.

I. EECW System backup water supply valves (FCV-67-50 [North header] and 51 [South header]) to the RBCCW heat exchangers will open at 15 psig lowering RCW pressure if EECW pressure is equal to or greater than the setpoint. These valves will close on EECW pressure dropping below the setpoint. Once closed, the closure seals in until manually reset in accordance with Section 8.7. The north header supply to Unit 1 RBCCW, the north header supply to Unit 2 RBCCW and the South header supply to Unit 3 RBCCW are normally isolated with a manual valve; therefore no flow will occur when either 1-FCV-67-50, 2-FCV-67-50 or 3-FCV-67-51 opens. The EECW pressure setpoints for these valves are listed below in psig:

	Unit 1	Unit 2	Unit 3
FCV-67-50	90	91	92
FCV-67-51	107	109	113

J. The EECW discharge strainer automatically starts its cleaning cycle on pump discharge flow, and the flush valve opens automatically

QUESTION 53

All three units are operating at 100% ~~power in a normal plant lineup~~ when the ~~operating~~ A3 EECW Pump trips. The Unit Operator attempts to start the C3 EECW pump and it fails to start.

Which ONE of the following describes the impact of the A3 EECW Pump trip and failure of the C3 EECW pump to start?

The AUTOMATIC backup cooling water supply to the ______ has been lost.

A. control air compressors ONLY

B. Unit 1 and 2 RBCCW heat exchangers ONLY

C. control air compressors and the Unit 3 RBCCW heat exchanger ONLY

D. control air compressors and the Unit 1, Unit 2, and Unit 3 RBCCW heat exchangers

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000 K1.01	
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause-effect relationships between CCWS and the following:
K1.01 Service water system

Explanation: **C** CORRECT – The A3 and C3RHRSW Pumps are on the North EECW header which will automatically align to the control air compressors and all 3 RBCCW heat exchangers, but the U1 andU2 RBCCW heat exchangers are manually isolated from the North header and will only automatically align to the South header.

A Incorrect – Plausible if the candidate forgets that the Unit 3 RBCCW heat exchanger supply is isolated from the South header.

B Incorrect – Plausible if the candidate confuses the North and South headers. The answer is correct for loss of the South header.

D Incorrect –Plausible if the candidate forgets the South header is available for RBCCW heat exchangers.

Technical Reference(s): 0-OI-67

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New	
Question History:	Previous NRC: Nine Mile 1 2007 #36	
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Nine Mile Point #1 2007 NRC exam #36

EXAMINATION ANSWER KEY

Reactor Operator Written Examination Nine Mile Point Unit 1 2007

36	USERID: NRC 2006 RO 36	SYSID: 21261	Points: 1.00

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

- H1-4-2, R BUILDING SW PRESS/SERV W PUMP HDR PRESS LOW is in alarm
- N1-SOP-18.1 SW Failure / Low INTAKE LEVEL has been entered
- Neither Service Water Pump can be started

Which one of the following is the required action and the effect on plant operations?

A. Start an ESW Pump. Power operations can continue because all cooling water is restored.

B. Start an ESW Pump. A plant shutdown or scram is required due to partial loss of cooling water.

C. Lineup fire water to RBCLC heat exchanger. Power operations can continue because all cooling water is restored.

D. Lineup fire water to RBCLC heat exchanger. A plant shutdown or scram is required due to partial loss of cooling water.

Answer: B

Answer Explanation: B. is correct. Per N1-SOP-18.1 an ESW pump must be started. Since ESW can only supply RBCLC HXs and not the TBCLC HXs, a plant shutdown or scram will be required due to loss of cooling to TBCLC loads.
A is incorrect. Since ESW will not supply TBCLC loads, all cooling water is NOT restored. Power operations cannot be continued, since TBCLC loads no longer have a heat sink.
C. & D. are incorrect. Fire water is only lined up if ESW pumps cannot be started.

QUESTION 54

Unit 3 is operating at 100% power with the following indications on Panel 9-5:

Which ONE of the following completes the statement below?

Using ONLY the CRD CONTROL SWITCH (3-HS-85-48) the operator may withdraw control rods ___(1)___, and insert control rods ___(2)___.

A. (1) by single notch ONLY
(2) by single notch ONLY

B. (1) by single notch ONLY
(2) by single notch OR continuously

C. (1) by single notch OR continuously
(2) by single notch ONLY

D. (1) by single notch OR continuously
(2) by single notch OR continuously

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201002 K4.04	
	Importance Rating	3.3	

Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following:
K4.04 "Single notch " rod withdrawal and insertion

Explanation: **B** CORRECT – The ROD WITHDRAW PERMISSIVE light indicates that control rods can be withdrawn but without the use of the RONOR switch the rods can only be notched out.

A- Incorrect – Plausible if the candidate believes that the control rods are notched in and out using the CRD Control Switch and the RONOR switch must be used for continuous in or out

C- Incorrect – Plausible if the candidate believes that the control rods are notched in, and can be notched out or continuously withdrawn using the CRD Control Switch and the RONOR switch.

D- Incorrect –Plausible if the candidate believes that the switch operates the same for insertion and withdrawal.

Technical Reference(s): OPL171.029

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

4. Manual Control Switches (Panel 9-5) TP-3

a. CRD Control Switch (HS-85-48) Obj. V.B.2.a

(1) HS-85-48 has three positions (listed counter clockwise).

(a) ROD OUT NOTCH

(b) OFF

(c) ROD IN

(d) Spring return to OFF

(2) Initiates notch in and notch out cycles (notch movement means moving control rod from one even position indication to the next, i.e., one notch on the CRD index tube)

(3) If held in NOTCH OUT, will complete one notch out cycle and stop

INSTRUCTOR NOTES

(4) If held to ROD IN, will move control rod continuously in until released

b. CRD Notch Override Switch (85-47) — Obj. V.B.2.b

(1) HS-85-47 has three positions (listed counter clockwise).

(a) NOTCH OVERRIDE

(b) OFF

(c) EMERG ROD IN

(d) Spring return to OFF

(2) NOTCH OVERRIDE position — Consider consequences of Improper Performance!! SOER 84-2 <u>STAR</u>

(a) Allows for continuous withdrawal of control rod

(b) Used in conjunction with CRD Control Switch - ROD OUT NOTCH position.

(c) Amber light above switch is energized during notch override action.

(d) Continuous control rod withdrawal may be used when a control rod is to be withdrawn greater than 3 notches, except for high notch worth areas as identified by the Reactor Engineer — Obj. V.B.2.e

(3) EMERG ROD IN position — Obj. V.D, E.9

(a) Acts directly on directional control valves to bypass timer

(b) No settle function, forces water past seals in CRD while settling into notch

5. Indicating Lights (Panel 9-5, apron section)

TP-3, TP-13
Obj. V.D,E.6

a. Movement control lights (above CRD Control Switch)

(1) ROD OUT PERMIT

Obj. V.B.4

(a) Blue light

INSTRUCTOR NOTES

(b) Indicates no rod withdraw blocks present

(c) Light indicates that rod can be withdrawn (RMCS or RWM rod withdraw block present when light is not energized).

(d) Has no effect on insert motion, i.e., rod can be inserted with light deenergized.

QUESTION 55

A Unit 3 startup is in progress and power is 5%. The Rod Worth Minimizer (RWM) is in service with Sequence Control ON.

Which ONE of the following completes the statements?

If a control rod within the currently selected Rod Group, but NOT the next rod in the programmed sequence, is selected and continuous withdraw is attempted, then control rod withdrawal ___(1)___ is permitted by the RWM.

Per NPG-SPP 10.4, Reactivity Management Program , the control rod is ___(2)___.

A. (1) one notch
(2) mispositioned. The operators immediately enter 3-AOI-85-7, Mispositioned Control Rod

B. (1) one notch
(2) NOT mispositioned. The operators use 3-OI-85, Control Rod Drive System, Sections 6.7, Control Rod Insertion, and 6.8, Operations With Rod Worth Minimizer Insert/Withdraw Errors to move the control rod

C. (1) continuously until either switch is released
(2) mispositioned. The operators immediately enter 3-AOI-85-7, Mispositioned Control Rod

D. (1) continuously until either switch is released
(2) NOT mispositioned. The operators use 3-OI-85, Control Rod Drive System, Sections 6.7, Control Rod Insertion, and 6.8, Operations With Rod Worth Minimizer Insert/Withdraw Errors to move the control rod

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201006 A2.05	
	Importance Rating	3.1	

Ability to (a) predict the impact of the following on the ROD WORTH MINIMIZER SYSTEM (RWM); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.05 Out of sequence rod movement

Explanation: **A** CORRECT – The RWM would see the rod as an improper selection and generate a select error but not a select block. The RWM allows an error to be made and then imposes blocks to force the operator to fix the error. Only a single notch withdrawal would be allowed. This is a mispositioning because the wrong rod was moved.

B Incorrect – First Part: Correct. Second Part: Incorrect. Plausible because SPP-10.4, Reactivity Management Program, allows a control to be out of position by one notch and NOT considered mispositioned.

C Incorrect – First Part: Incorrect. Plausible in the one rod test mode, one rod can be continuously withdrawn until the withdrawal is stopped then it must be inserted to 00. Second Part: Correct.

D Incorrect –First Part: Incorrect. Plausible in the one rod test mode, one rod can be continuously withdrawn until the withdrawal is stopped then it must be inserted to 00. Second Part: Incorrect. Plausible because SPP-10.4, Reactivity Management Program, allows a control to be out of position by one notch and NOT considered mispositioned.

Technical Reference(s): 3-OI-85, SPP-10.4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New : X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

3.8 Rod Worth Minimizer (RWM) (continued)

I. A withdraw error occurs if:

1. A rod in the currently latched group is withdrawn past the withdraw limit for the group, OR

2. A rod in a group lower than the one currently latched is withdrawn past the withdraw limit for its group, OR

3. A rod in a group higher than the one currently latched is withdrawn past the insert limit for its group.

J. A select error occurs if:

1. With the reactor operating below the LPAP, a rod other than one contained in the currently latched group is selected, unless conditions for latching up or down are met, OR

2. With a rod block applied, any rod other than an error rod is selected, OR

3. When operating in the Sequence Control Mode, a rod is skipped.

K. An insert block occurs if:

1. With two insert errors existing, a rod is moved to cause a third insert error, OR

2. A withdraw error has been made, a withdraw block applied, and a rod other than the withdraw error rod is selected.

L. A withdraw block occurs if:

1. A withdraw error is made, OR

2. With three insert errors existing and an insert block present, a rod other than one of the insert errors is selected.

M. A select block occurs if:

1. The RWM Bypass Switch is in normal and the RWM program is <u>not</u> running; i.e., following return to normal from bypass and the program has <u>not</u> been initialized, OR

2. The RWM Bypass Switch is in normal and the program stops due to software error.

6.0 DEFINITIONS (continued)

Mispositioned Control Rod - A control rod meeting one of the following criteria. This definition does not distinguish between mispositionings caused by equipment failure and those caused by human error since the Reactivity impact is identical

- Control rod left in a position other than the intended position and not corrected before completion of the confirmation step of the associated Control Rod Movement Instructions.
- Control rod moved more than one notch beyond the intended position. [BWR]
- Rods misaligned by greater than 12 steps. [PWR]
- Control rod moved in the wrong direction.
- Wrong control rod moved.
- Drifting control rod.
- Unexpected single control rod scram.
- Control rod moved to a position inconsistent with the Reactivity Plan or associated predictions

3.8 Rod Worth Minimizer (RWM) (continued)

I. A withdraw error occurs if:

1. A rod in the currently latched group is withdrawn past the withdraw limit for the group, OR

2. A rod in a group lower than the one currently latched is withdrawn past the withdraw limit for its group, OR

3. A rod in a group higher than the one currently latched is withdrawn past the insert limit for its group.

J. A select error occurs if:

1. With the reactor operating below the LPAP, a rod other than one contained in the currently latched group is selected, unless conditions for latching up or down are met, OR

2. With a rod block applied, any rod other than an error rod is selected, OR

3. When operating in the Sequence Control Mode, a rod is skipped.

K. An insert block occurs if:

1. With two insert errors existing, a rod is moved to cause a third insert error, OR

2. A withdraw error has been made, a withdraw block applied, and a rod other than the withdraw error rod is selected.

L. A withdraw block occurs if:

1. A withdraw error is made, OR

2. With three insert errors existing and an insert block present, a rod other than one of the insert errors is selected.

M. A select block occurs if:

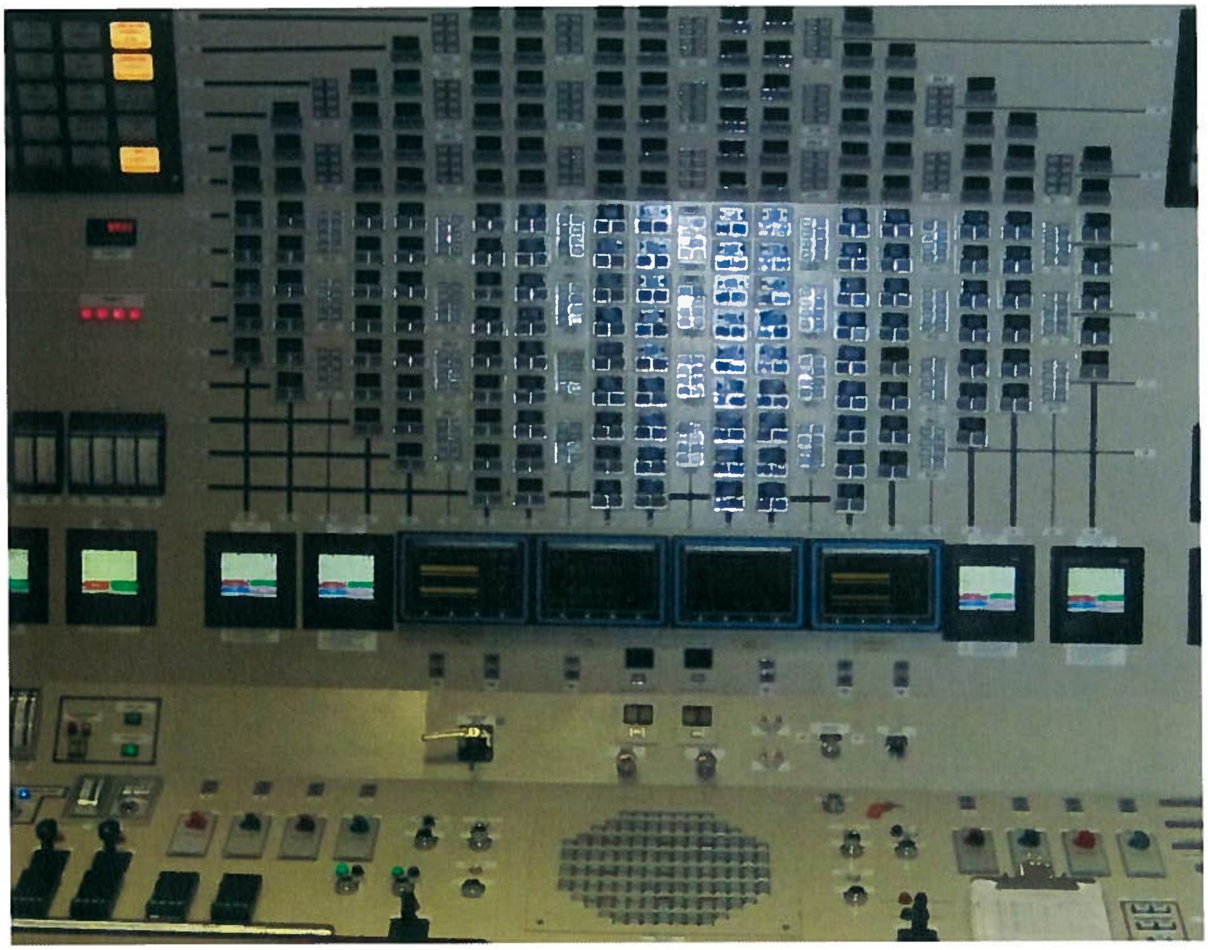
1 The RWM Bypass Switch is in normal and the RWM program is not running; i.e., following return to normal from bypass and the program has not been initialized, OR

2. The RWM Bypass Switch is in normal and the program stops due to software error.

QUESTION 56

Unit 2 is operating at 100% power when a loss of power occurred.

Current indications:



Which ONE of the following has been lost?

A. ICS power

B. Unit Preferred

C. Plant Preferred

D. Unit Non-Preferred

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	214000 K6.01	
	Importance Rating	2.5	

Knowledge of the effect that a loss or malfunction of the following will have on the ROD POSITION INFORMATION SYSTEM:
K6.01 A. C. electrical power

Explanation: B CORRECT – The loss of Unit Preferred will cause a loss of RPIS

A Incorrect – Plausible since the loss of 120VAC ICS power results in a loss of power to the Rod Worth Minimizer (RWM).

C Incorrect – Plausible since the name is similar and significant equipment is powered from Plant Preferred.

D Incorrect –Plausible since the loss of Unit Non- Preferred will cause a loss of power to RMCS/CRD components and prevent control rod motion.

Technical Reference(s): 2-AOI-85-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge:	X
	Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for Loss of Unit Preferred (Battery Board 2 Panel 11, Control Room Pane l2-9-9 Cabinet 5 Non-preferred and Unit Preferred to Panel 2-9-9 Cabinet #6).

NOTE

A loss of the Unit Preferred power source results in the following boards being de-energized:

- Battery Board 2 Panel 11
- Panel 2-9-9 Cabinet 5 Non-preferred
- Panel 2-9-9 Cabinet 6 Unit Preferred (if panel does **NOT** auto transfer to alternate)

2.0 SYMPTOMS

A. Loss of indication to CRD rod control indicator lights, Panel 2-9-5 (i.e. rod out permissive, timer switch malfunction - select block, rod withdraw, insert and settle, notch override and stabilizing valve indicator lights.)

B. The following recorders fail downscale (Panels 2-9-6, 2-9-7, 2-9-8):

1. Turbine Generator Vibration (2-XR-47-15),
2. TURB GEN ECC/SPEED/VALVE POSN (2-XR-47-16),
3. TURB EXP AND TEMP MN XFMR Winding Temp (2-XR-47-20/2-TR-47-20)
4. RFPT/RFP VIB and ECC (2-XR-3-177).

C. PANEL 2-9-9 CABINET 1, 2,3 or 6 CONTROL POWER TRANSFER (2-XA-55-7A, Window 15), on Panel 2-9-7, is in alarm on transfer of Unit Preferred Cabinet 6.

D. Unit 2 Panel 2-9-9, Cabinet 6 UNIT PFD 120V AC NORMAL and ALTERNATE SUPPLY lights 2-XI-57-600A and 2-XI-57-600B extinguish.

E. UNIT PFD SUPPLY ABNORMAL (2-XA-55-8B, Window 35) on Panel 2-9-8 is in alarm.

2.0 SYMPTOMS (continued)

F. Loss of RPIS. REFER TO 2-AOI-85-4.

G. RFW Control System Panel Display Stations on Panel 2-9-5 disabled. PDS Controls are inoperative and displays become blank. The RFW Control System continues to control system parameters according to water level setpoint.

H. The following RFW Control System annunciators in alarm on Panel 2-9-6:

1. RFPT GOVERNOR POWER FAILURE OR GOVERNOR ABNORMAL (2-XA-55-6C, Window 12).

2. RFWCS TROUBLE (2-XA-55-6C, Window 28).

I. The following EHC Control System annunciators in alarm on Panel 2-9-6:

1. EHC POWER ABNORMAL (2-XA-55-7B Window 5)

2. EHC/TSI SYSTEM TROUBLE (2-XA-55-7B Window 6)

J. EHC Control System PLU 1 (power load unbalance) can bypass with a sustained loss of power to Panel 9-9 Cabinet 5. An uninterruptible power supply will keep the PLU energized for approximately 15 minutes after normal power is lost.

K. EHC Control System HMI on Panel 2-9-31 may become blank if power is lost to Panel 9-9 Cabinet 6. An uninterruptible power supply will keep this component energized for approximately 15 minutes after normal power is lost.

L. RECIRC FLOW SYSTEM TROUBLE ALARM (2-XA-55-4A, Window 23).

M. Loss of power to CRD Select Modules.

N. ANN: PNL 2-9-21 SYS LEAK DETECTION POWER FAILURE (2-XA-55-3D, Window 31) on loss of power to Panel 2-9-21 Steam Leak Detection Panel.

O. TIP isolation signal when Cabinet 5 (Breaker 503) is de-energized.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **STOP** all control rod movement. ☐

4.2 Subsequent Actions

NOTE
Reference TRM 3.3.5, RPIS Indicated Channel Operability, for applicable 7 or 30 day LCO relating to an inoperable RPIS indication.

[1] **IF** control rod movement is required with a Total loss of RPIS, **THEN:**

MANUALLY SCRAM reactor. ☐

[2] **NOTIFY** the Operations Superintendent and Reactor Engineer for actions to be taken in a timely manner. ☐

[3] **NOTIFY** Technical Support to help determine the extent of loss of RPIS. ☐

[4] **IF** control rod was in motion when RPIS failed **AND** position of that control rod can not be determined, **THEN:**

CONSIDER that Control Rod Inoperable. **REFER TO** Tech. Spec. 3.1.3. ☐

[5] **VERIFY ON** Breaker 612, Panel 2-9-27 ROD POSITION INFO SYS **FEED** FROM UNIT PREFERRED 120VAC at Panel 2 9 9 Cabinet 6. ☐

QUESTION 57

Unit 2 is operating at 100% power with the following RPV Pressure Indications on Panel 9-5

- 2-PI-3-54 1045 psig
- 2-PI-3-61 966 psig
- 2-PI-3-207 1020 psig

Which ONE of the following is the reactor pressure used by the Feedwater Level Control System (FWLCS)?

A 1010.3 psig

B. 1020.0 psig

C. 1032.5 psig

D. 1035.0 psig

Answer: **A**

Nuc Boiler Instr.

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	216000 A4.02	
	Importance Rating	3.3	

Ability to manually operate and/or monitor in the control room:
A4.02 Channel select controls

Explanation: **A CORRECT** – Because no pressure indicator is ≥ 75 psi from the average, no pressure is considered invalid. Therefore reactor pressure used by the Feedwater Level Control System (FWLCS) is the average of the 3 instruments.

B Incorrect – Plausible since this is the median value.

C Incorrect – Plausible, this is the average of the 2 highest values.

D Incorrect –Plausible, since this is the default value if pressure is not available.

Technical Reference(s): 2-OI-3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New X
Question History:	Previous NRC None
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

4.3 System Operation

The RFW Control System will use a pressure signal provided the system determines the signal to be good and valid. A GOOD pressure signal is one that has not failed and is on scale. A VALID signal is one that does not deviate from the average pressure by more than 75 psig.

The RFWCS validates each Wide Range Pressure by comparing them to the average. Any pressure signal that deviates from the average by more than 75 psig is declared invalid and is bypassed. A WR Pressure signal that is declared bad by the system will also be automatically bypassed.

To avoid individual on-scale but faulty pressure signals from skewing the average, the average is compared to the median of the WR Pressure signals. If the average value differs from the median value by more than 75 psig, the RFWCS will validate each pressure signal to median value, instead of average. In this case, any pressure signal that deviates from median by more than 75 psig is declared invalid and is automatically bypassed by the system.

QUESTION 58

A small LOCA has occurred on Unit 2. The following conditions now exist:

- All control rods are at position 00
- RPV Pressure is 950 psig and steady
- RPV Water Level dropped to (-)40 inches and is now 20 inches and slowly rising
- Drywell Pressure reached 12.5 psig and is now 1.82 psig and slowly lowering
- Suppression Pool Temperature peaked 93° F and steady
- ~~RHR Loop~~ I is in Drywell Sprays

Which ONE of the following completes the statements?

The RHR Loop I Suppression Chamber Spray Valves (2-FCV-74-57, RHR SYS I SUPPR CHBR/POOL ISOL VLV and 1-FCV-74-58, RHR SYS I SUPPR CHBR SPRAY VALVE) ___(1)___ automatically shut.

In accordance with 2-EOI-2, PRIMARY CONTAINMENT CONTROL, RHR Loop I ___(2)___.

A. (1) have
(2) may be placed in standby readiness

B (1) have
(2) MUST be placed in Suppression Pool Cooling

C (1) have NOT
(2) may be placed in standby readiness

D. (1) have NOT
(2) MUST be placed in Suppression Pool Cooling

Answer: **C**

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

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

Explanation: **C** CORRECT – With no accident signal present (-122 inches OR RPV pressure ≤ 450 psig with drywell pressure ≥ 2.45 psig) then there is no auto closure of the suppression Chamber spray valves on low drywell pressure. There is no longer an entry condition to EOI-2

A Incorrect – First Part: Incorrect. Plausible since the valves would auto close if a LPCI initiation signal was present. Plausible since the valves would auto close if a LPCI initiation signal was present. Second Part: Correct.

B Incorrect – First Part: Incorrect. Plausible since the valves would auto close if a LPCI initiation signal was present. Second Part: Incorrect. Plausible, if Suppression Pool Temperature was 95° F or greater then 2-EOI-2, step SP/T-3 requires all available SP Cooling except for pumps requiring continuous operation to maintain adequate core cooling.

D Incorrect – First Part: Incorrect. Second Part: Incorrect. Plausible, if Suppression Pool Temperature was 95° F or greater then 2-EOI-2, step SP/T-3 requires all available SP Cooling except for pumps requiring continuous operation to maintain adequate core cooling.

Technical Reference(s): 2-EOI-2, 2-OI-74

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New X

Question History: Previous NRC None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

3.6 Interlocks (continued)

7. If Unit 2 reactor pressure exceeds 100 psig or a Group II isolation occurs on Unit 2 while Shutdown Cooling is in operation, the following will occur for the given condition:

   (100 psig) RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.

   (Group II) RHR SYS I and II LPCI INBD INJECT VALVEs, 2-FCV-74-53 and 2-FCV-74-67, close and Unit 2 RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.

8. To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton is required to be depressed after either of following occur:

   a. Isolation signal has been reset **OR**

   b. 2-FCV-74-47 **OR** 2-FCV-74-48 is fully closed.

9. If after a GROUP II Isolation, RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67) is given an OPEN signal prior to depressing the RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET 2-XS-74-126(132), then the valve will travel full open and full close unless given a close signal prior to traveling full open.

10. The RHR spray/cooling valves, 2-FCV-74-57(71), receive an auto closure signal in the presence of a LPCI initiation signal and they are interlocked to prevent opening if the in-line torus spray valve, 2-FCV-74-58(72), is not fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

    a. Reactor level is greater than 2/3 core height **AND**

    b. LPCI initiation signal is present **AND**

    c. The select reset switch is in the SELECT position.

    The requirements for greater than 2/3 core height and a LPCI initiation signal may be BYPASSED using the keylock bypass switch, 2-XS-74-122/30.

3.6 Interlocks (continued)

11. If primary containment cooling is desired with reactor level at less than2/3 core height, the keylock bypass switch is required to be placed in BYPASS before the select reset switch is placed in SELECT to ensure relay logic is made up.\

12. The RHR torus spray valves, 2-FCV-74-58(72), have the same in-line valve interlocks as those outlined in 3.6A 10 for the torus spray/cooling valves. Additionally these valves have an interlock preventing opening unless drywell pressure is ≥1.96 psig which cannot be bypassed.

13. The RHR torus cooling/test valves, 2-FCV-74-59(73), receive an auto closure signal in the presence of a LPCI initiation signal. Auto closure may be bypassed by the same conditions/actions outlined in Step 3.6A.10

14. The RHR containment spray valves, 2-FCV-74-60(74) and 61(75), have in-line valve interlocks similar to these described in Step 3.6A.10 through 3.6A.12 for the RHR torus spray valves 2-FCV-74-57(58) and 71(72).

15. If 2-FCV-74-59(73) LOCA CLOSURE TIME light (2-IL-74-59Y Loop I; 2-IL-74-73Y Loop II) on Panel 2-9-3 is extinguished due to its associated valve being opened, that Loop is inoperable for LPCI.

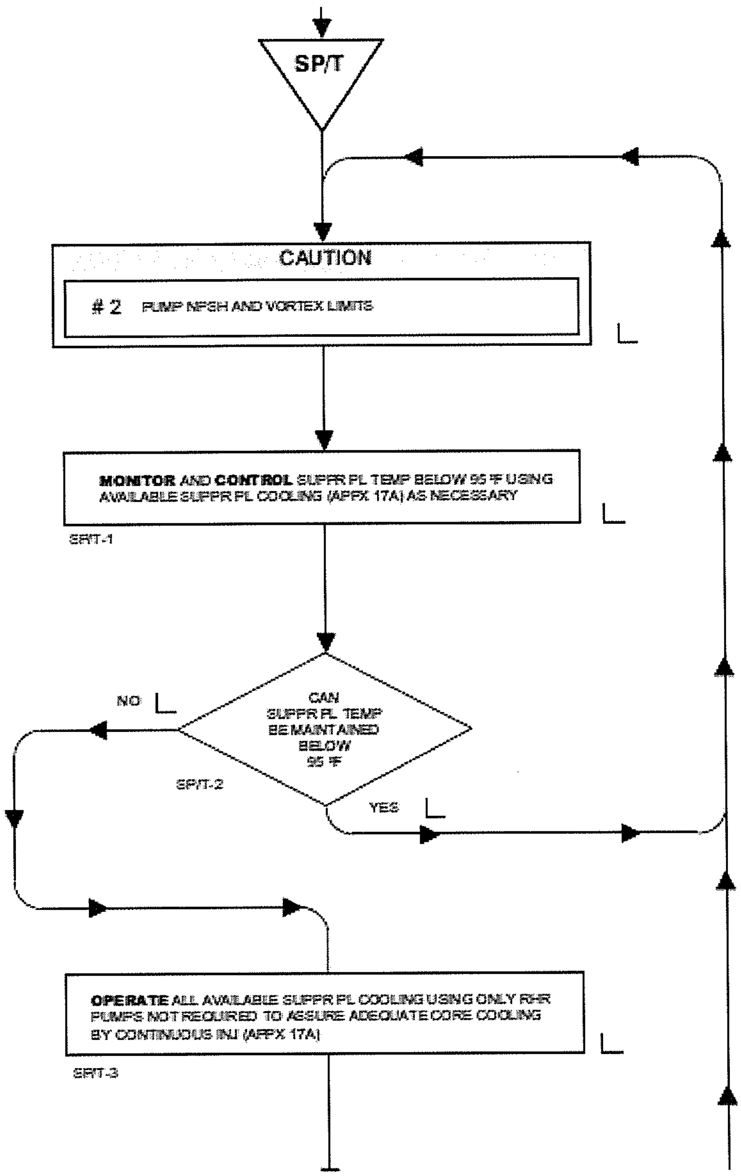
16. If 2-HS-74-148(149) RHR SYSTEM I (II) MIN FLOW INHIBIT switch is in the INHIBIT position, the pumps on that loop do not have automatic minimum flow protection.

17. If RHR pumps 2A(B) and 2C(D) SD COOLING SUCT VLV, 2-FCV-74-2(25) AND 13(36) are not fully closed, the RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57(71) cannot be opened.

    a. 2-HS-074-0057D placed in BYPASS position, defeats the interlocks on 2-FCV-74-57 and the SD COOLING SUCT VLV 2-FCV-74-2 AND 13.

    b. 2-HS-074-0071C placed in BYPASS position, defeats the interlocks on 2-FCV-74-71 and the SD COOLING SUCT VLV 2-FCV-74-25 AND 36.

18. If RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57(71) is not fully closed, the RHR pumps 2A(B) and 2C(D) SD COOLING SUCT VLV, 2-FCV-74-2(25) AND 13(36) cannot be opened.



SP/T
CAUTION
2 PUMP NPSH AND VORTEX LIMITS
MONITOR AND CONTROL SUPPR PL TEMP BELOW 95 °F USING AVAILABLE SUPPR PL COOLING (APPX 17A) AS NECESSARY
SP/T-1
CAN SUPPR PL TEMP BE MAINTAINED BELOW 95 °F
SP/T-2
NO
YES
OPERATE ALL AVAILABLE SUPPR PL COOLING USING ONLY RHR PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING BY CONTINUOUS INJ (APPX 17A)
SP/T-3

QUESTION 59

On Unit 3, the Mode Switch is in REFUEL, AND ALL control rods are inserted. The Refueling Bridge operator grappled a fuel bundle, raised the grapple, AND commenced moving the bundle towards the core.

Which ONE of the following completes the statement?

As the Refueling Bridge moves towards the core, it will _________.

A. continue over the core AND will initiate a control rod block

B. continue over the core AND will cause NO protective actions

C. stop before it reaches the core AND will initiate a control rod block

D. stop before it reaches the core AND will cause NO protective actions

Answer: **A**

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

Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT : Fuel handling equipment interlocks

Explanation: A CORRECT – With the fuel grapple loaded and the platform over the core in REFUEL a rod block is produced but motion is not blocked.

B Incorrect – Plausible if the candidate believes that a rod must be withdrawn or mode switch changed to produce a rod block

C Incorrect – Plausible if the candidate does not recognize the role of rod position in the logic

D Incorrect –Plausible if the candidate confuses the logic for movement and rod block

Technical Reference(s):0-GOI-100-3A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

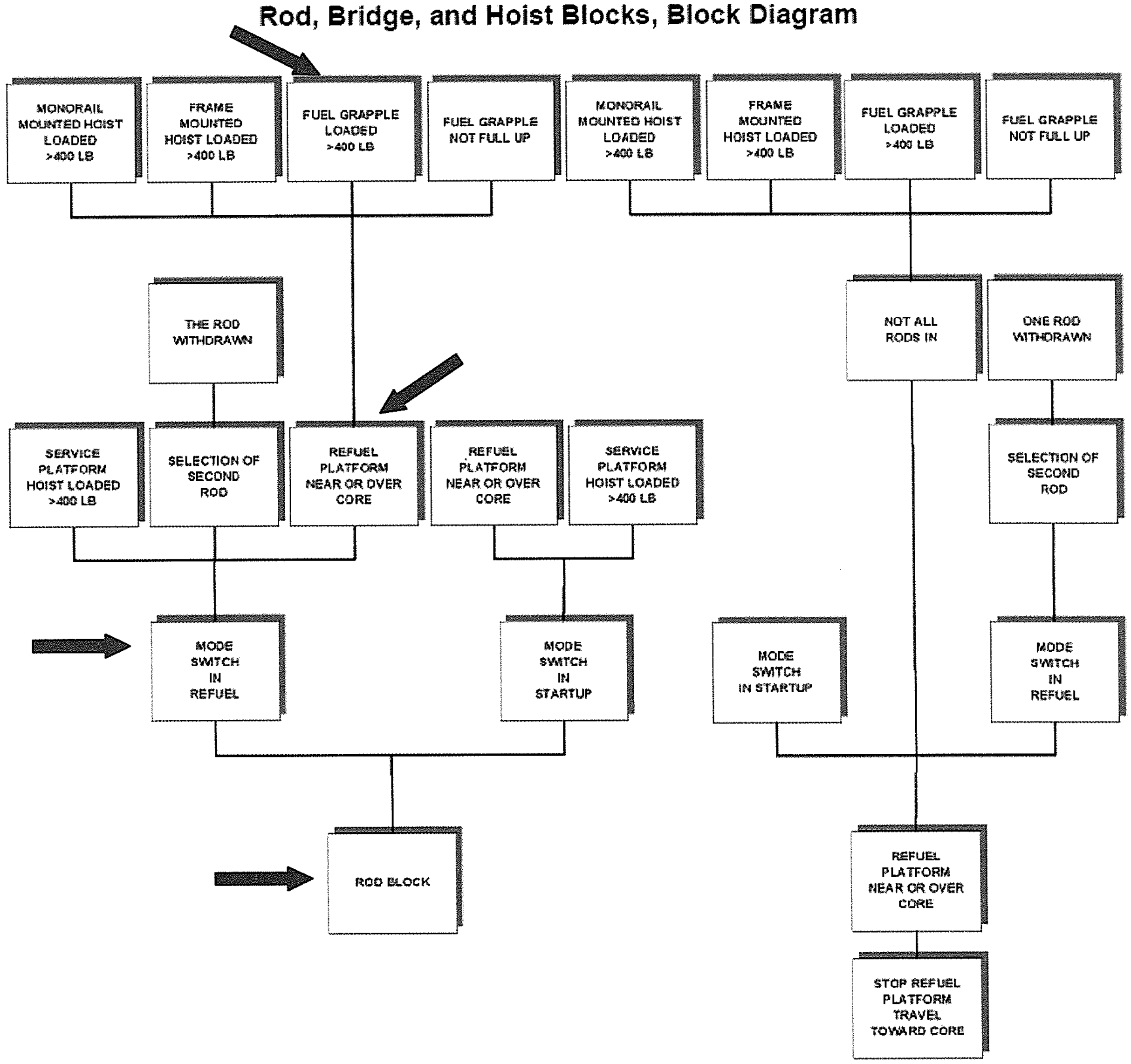
Question History: Previous NRC None

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

10 CFR Part 55 Content: 55.41 (7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Illustration 2
(Page 2 of 2)

Rod, Bridge, and Hoist Blocks, Block Diagram



QUESTION 60

Unit 1 has automatically scrammed and all control rods did NOT fully insert.

The following conditions exist:

- Reactor power is 17% RTP
- RPV pressure is 970 psig being maintained by the Main Turbine Bypass Valves
- RPV level is being maintained at 25 inches by the feedwater system
- Suppression Pool temperature is 91° F and rising slowly

Which ONE of the following describes the actions necessary to control reactor water level in accordance with the EOIs?

A. Maintain RPV level (+)2 to (+)51 using the feedwater system.

B. Slowly lower RPV level using the feedwater system until Reactor Power is less than 5%, then maintain level in a 50 inch band.

C. Terminate and prevent injection to the RPV until RPV level is (-)50 inches OR Reactor Power is less than 5% Reactor Power, whichever level is HIGHER, then maintain RPV level in a 50 inch band.

D. Terminate and prevent injection to the RPV until RPV level is -50 inches, then maintain RPV level (-)50 to (-)100 inches.

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	259001 G2.4.6	
	Importance Rating	3.7	

Reactor Feedwater System
G2.4.6Knowledge of EOP mitigation strategies

Explanation: **D** CORRECT – Power is greater than 5% so C-5 must be entered. Step C5-10 requires terminating and preventing and C5-12 stops lowering level and C5-13 sets the level band as in this answer.

A Incorrect – Plausible this is the correct action if power is less than 5% in EOI-1

B Incorrect – Plausible natural tendency is for candidates to slowly lower level in a controlled manner.

C Incorrect –Plausible as this is the correct initial action but should stop at lower value.

Technical Reference(s): 1-EOI-1, C-5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 ((10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

IS RX POWER ABOVE 5% OR UNKNOWN

C5-9

NO

YES

C5 INJ SYSTEM STATUS

SYSTEM	APPX	INJ STOPPED AND PREVENTED
CNDS AND FW	4	
HPCI	4	
LPCI	4	
CS	4	

STOP AND **PREVENT** ALL INJ INTO RPV EXCEPT FROM RCIC, CRD, AND SLC (APPX 4)

C5-10

WHEN RPV WATER LEVEL DROPS BELOW -50 IN.

THEN **CONTINUE**

C5-11

D

C5-2

STOP LOWERING RPV WATER LVL

C5-12

RECORD WATER LEVEL FOR STEP C5-12:

_______________ IN.

_______________ IN.

CAUTION

#5 RAPID RPV INJ MAY CAUSE CORE DAMAGE

#2 PUMP NPSH AND VORTEX LIMITS

#3 ELEVATED SUPPR CHMBR PRESS MAY TRIP RCIC

#6 HPCI OR RCIC SUCTION TEMP ABOVE 140 °F

MAINTAIN RPV WATER LVL BETWEEN -180 IN. AND THE LEVEL TO WHICH IT WAS LOWERED WITH THE FOLLOWING INJ SOURCES:

INJ SOURCE	APPX	INJ PRESS
CNDS AND FW	5A	1250 PSIG
CRD	5B	1640 PSIG
RCIC, WITH CST SUCTION IF POSSIBLE	5C	1200 PSIG
HPCI, WITH CST SUCTION IF POSSIBLE	5D	1200 PSIG
CNDS	6A	480 PSIG

QUESTION 61

Regarding the Offgas charcoal adsorbers, which ONE of the following completes the statement?

High relative humidity is an operational concern in the offgas system because ____(1)_____ results in ___(2)_____.

A. (1) freezing
   (2) plugging of the adsorbers

B. (1) charcoal acidity
   (2) increased corrosion

C. (1) charcoal fines
   (2) a reduction of the available charcoal volume

D. (1) wet charcoal
   (2) heat in the adsorbers

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	271000 K5.11	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to the OFFGAS SYSTEM
K5.11 Explain the necessity of reducing relative humidity for carbon bed filters

Explanation: **D** CORRECT –Dewpoint temperature in the offgas system is kept low to prevent wetting of the charcoal which can cause heat in the adsorber beds.

A Incorrect – Plausible because the charcoal adsorbers operate below freezing and any moisture can freeze and block passage through the bed.

B Incorrect- Plausible because the addition of moisture frequently increases corrosion rates but is not the concern in this case.

C Incorrect – Plausible. Wetting charcoal does tend to produce fines but this will not impact the volume of charcoal.

Technical Reference(s): OPL171.030, OI-66

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (13) Procedures and equipment available for handling and disposal of radioactive materials and effluents.

INSTRUCTOR NOTE:

c. Drains to off-gas condensate drain sump in Radwaste.

(1) FSV-66-23 open for normal operation

Auto close at 3.5# off gas press

(2) A loop seal in the drain line prevents escape of gas to Radwaste.

d. Off-gas pretreatment and linear radiation monitoring system normally takes a sample from the inlet to the holdup volume.

Sample taken at holdup volume inlet and returned to SJAE inlet or inlet to 6 hour hold up volume

e. Actual holdup time will be approximately 6 hours due to removal of radiolytic H_2 and O_2. This lowers the flow rate and increases holdup time.

14. Cooler condensers (two required - one operating, one standby)

CAN BE OPERATED IN PARALLEL

a. Location: Off-Gas Treatment Building

b. Purpose: Further cools gas-steam mixture for moisture control. Cools to 45°F. Excess moisture in the charcoal adsorber lowers the electro-static attraction of iodine.

Obj V.B.1
See Handout 2
Moisture also creates heat in the beds.

5.11 Aligning Charcoal Filters for Parallel Flow

NOTE

The charcoal beds can be aligned for either parallel or series flow, but normally parallel flow is preferred. Performing the following steps at Panel 9-53 aligns the charcoal beds for parallel flow. If series alignment is preferred, Section 8.10 is required to be performed in lieu of the following steps.

CAUTION

The charcoal adsorbers are required to be aligned in the treatment mode prior to reaching 25% power.

[1] **PLACE** OFFGAS TREATMENT SELECT handswitch, 2-XS-66-113, in TREAT. ☐

[2] **OPEN** CHARCOAL ADSORBER TRAIN 2 INLET VALVE, using 2-HS-66-117. ☐

[3] **OPEN** CHARCOAL ADSORBER TRAIN 1 DISCH VALVE, using 2-HS-66-118. ☐

[4] **CLOSE** CHARCOAL ADSORBER TRAINS SERIES VLV, using 2-HS-66-116. ☐

[5] **CHECK** dewpoint temperature on OFFGAS REHEATER TEMPERATURE recorder, 2-TRS-66-108, indicates 45°F or less (Blue Pen). ☐

CAUTION

A Reheater Inlet Dewpoint Temperature above 48°F may cause wetting of the charcoal beds.

QUESTION 62

Which ONE of the following describes the power supplies to the Stack Gas Radiation Monitors, 0-RM-90-147 and 0-RM-90-148, scintillation detectors?

A. ±48VDC Annunciator Power Distribution System

B. ±24VDC from the Neutron Monitoring System

C. 120VAC from the Reactor Protection system (RPS)

D. 120VAC from the Instrumentation & Controls (I&C) Power Distribution System

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	272000 K2.03	
	Importance Rating	2.5	

Knowledge of electrical power supplies to the following:
K2.03 Stack gas radiation monitoring system

Explanation: **B** CORRECT – 0-RM-90-147 and 0-RM-90-148 scintillation detectors are powered from +24VDC from the Neutron Monitoring System .

A Incorrect. Plausible if the candidate confuses the two low voltage DC systems (+24VDC from the Neutron Monitoring System and +48VDC Annunciator Power Distribution System).

C Incorrect – Plausible because other radiation monitors are powered from the 120VAC Reactor Protection system (RPS).

D Incorrect – Plausible because other radiation monitors are powered from the 120VAC Instrumentation & Controls (I&C) Power Distribution System.

Technical Reference(s): OPL171.033

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC None

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

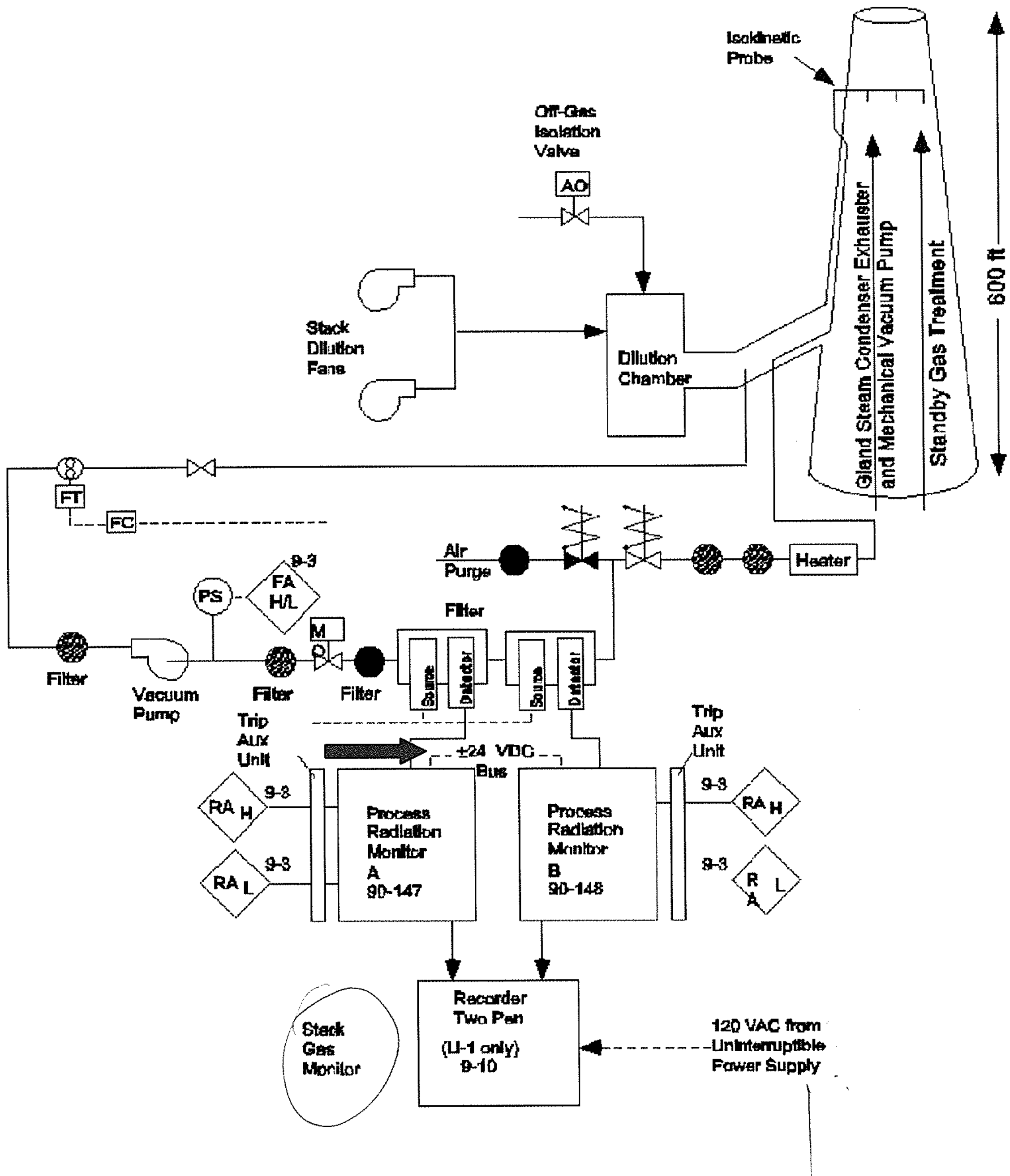
10 CFR Part 55 Content: 55.41 (13) Procedures and equipment available for handling and disposal of radioactive materials and effluents.

e. The monitors for the main steam line, Off-Gas radiation, stack gas radiation, process liquid, and reactor/refuel ventilation monitors are located on Panel 9-10 in each MCR. — Obj. V.B.2; Obj. V.C.2

(1) Panel 9-10 has power supplies: ±24 V DC from the Neutron Monitoring System, I&C, and RPS. — Obj. V.B.3.f, 3.k; Obj. V.C.3.f, 3.j

(2) In general, the scintillation detectors have ± 24 v DC supply. — Obj. V.B.3.f, 3.k; Obj. V.C.3.f, 3.j

(3) The MSL monitors and Reactor/Refuel Ventilation monitors are fed from RPS — Obj. V.B.3.f; Obj. V.C.3.f


Isokinetic Probe
Off-Gas Isolation Valve
AO
Stack Dilution Fans
Dilution Chamber
Gland Steam Condenser Exhauster and Mechanical Vacuum Pump
Standby Gas Treatment
600 ft
FT
FC
Air Purge
Heater
PS
9-3
FA H/L
M
Filter
Filter
Vacuum Pump
Filter
Filter
Source
Detector
Source
Detector
Trip Aux Unit
±24 VDC Bus
Trip Aux Unit
9-3
RA H
9-3
RA L
Process Radiation Monitor A 90-147
Process Radiation Monitor B 90-148
9-3
RA H
9-3
RA L
Recorder Two Pen
(U-1 only)
9-10
Stack Gas Monitor
120 VAC from Uninterruptible Power Supply

Appendix 2 Monitor Summary

Monitor	ID No	Tech Spec	Monitor Power	Type	Indications		Function
Main Steam Lines	136 137	None	RPS A RPS A	Ion	CR NUMAC digital display (2) Recorder (RR-90-135) 1 selector switche	Alarms	- DNSCL - HIGH (1.5 x NFLB) - HI/HI/INOP (3 x NFLB) Vac Pmp Vlvs, Vac Pmps,
Off Gas Pre-Treatment	157	None	I&C 'A'	Ion	CR Meter Recorder RR-90-157	Alarms	- OG AVG ANNUAL RELEASE LIMIT EXCEEDED - OG PRETREATMENT HIGH - OG PRETREATMENT DNSC - OG SAMPLE FLOW ABNML
Flux Tilt	160	None	NMS	Ion	CR Meter Recorder, One pen RR-90-160		Indication only
Stack Gas	147 148	ODCM 1.1.2	NMS	Scintillation	Indicator(U1) Recorder(U1)	Alarms	- HIGH - HIGH/HIGH - DNSCL/INOP - FLOW ABNORMAL

INSTRUCTOR NOTES

c. The sample passes through two shielded chambers where the radiation level of the stack gas is measured by two scintillation detectors, one located in each shielded chamber

Obj. V.B.2
Obj. V.C.2

PRINT:
0-729E814-1A

QUESTION 63

Unit 2 Reactor Building Ventilation Supply fan "B" is tagged for maintenance.

Reactor Building Ventilation Supply and Exhaust fans "A" are in service in FAST when the Supply Fan trips.

Which ONE of the following completes the statement?

The Unit 2 Reactor Building Train "A" Exhaust Fan ___(1)___ IMMEDIATELY trip and the Unit 2 Reactor Building pressure will ___(2)___.

A. (1) will NOT
(2) lower

B. (1) will NOT
(2) rise

C. (1) will
(2) lower

D. (1) will
(2) rise

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	288000 K3.05	
	Importance Rating	3.1	

Knowledge of the effect that a loss or malfunction of the PLANT VENTILATION SYSTEMS will have on the following: K3.05 Reactor building pressure

Explanation: **A** CORRECT – The Reactor Building is maintained at a slightly negative pressure by having slightly larger exhaust fans. Without the Supply fan, the pressure in the reactor building will lower and the DP will rise (get more negative). The Supply Fan will NOT IMMEDIATELY trip; however, as it continues to run it will draw pressure in the reactor building down to where it will cause a static lockout and trip.

B. Incorrect – First Part: Correct. Second Part: Incorrect, Plausible if the candidate confuses the relationship between the sizing of the supply and exhaust fans, and or doesn't understand the DP maintained on the Reactor building.

C. Incorrect – First Part: Incorrect, Plausible if the candidate believes that a trip of the supply fans cause a direct trip of the exhaust fan. Second Part: Correct.

D. Incorrect – First Part: Incorrect, Plausible if the candidate believes that a trip of the supply fans cause a direct trip of the exhaust fan. Second Part: Incorrect, Plausible if the candidate confuses the relationship between the sizing of the supply and exhaust fans, and or doesn't understand the DP maintained on the Reactor building.

Technical Reference(s): 2-ARP-9-3D

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.060 V.B.2

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

REACTOR ZONE DIFFERENTIAL PRESSURE LOW 2-PDA-64-27	32

(Page 1 of 1)

Sensor/Trip Point:

2-PDIS-064-27 -0.17 in. of water

Sensor Location: Panel 2-25-213, Rx Bldg El. 639'

Probable Cause:

A. Securing/Alternating Refuel Zone Fans.
B. Trip of any Rx Bldg Zone Exh. Fan.
C. PCIS Group 6 Isolation.
D. Differential Pressure switches fail closed.
E. Rapidly changing Barometric pressure or high winds.
F. Normal ventilation in service with Standby Gas Treatment System running at the same time.
G. High energy line break in Secondary Containment.

Automatic Action: Annunciation only.

Operator Action:

A. **IF** the alarm is intermittent, **THEN** **CHECK** for high wind conditions (ex., >20 mph) on ICS. ☐
B. **IF** high wind conditions CANNOT be confirmed, **THEN** **REQUEST** personnel to check local Reactor Building differential pressure. ☐
C. **IF** alarm is due to high wind conditions, **THEN** EOI-3 entry is NOT required. ☐
D. **IF** alarm is valid, **THEN** **INFORM** Unit Supervisor of 2-EOI-3 entry condition. ☐
E. **REQUEST** personnel to check fans locally for any apparent problems. ☐
F. **REFER TO** 2-OI-30B and **PLACE** standby fan in service to restore normal differential pressure. ☐

References: 2-45E620-2 2-47E610-64-1

a. All isolations must be manually reset and SGT stopped after initiation. Isolation will not clear until ventilation switch for affected zone is moved to OFF. Fans may then be restored.

b. Static Pressure Isolation — Obj. V.B.2, 4

Obj. V.C.2, 4

c. The signals are +0.5 in. of H2O / -1.0 in. of H2O. This only isolates affected zone, trips fans. — Obj. V.D. 2

a. Controls, operation, and electrical feeds are basically the same as reactor building fans.

b. Reactor zone and refueling zone static pressure is maintained at .25 inch water negative by use of larger exhaust fans and static pressure regulation. Dampers mounted in each fan inlet are designed to gradually close or open in response to static pressure regulators to maintain building pressure and ensure an elevated, monitored release. — Obj. V.B.2

Obj. V.C.2

c. PdIC 64-1, located on the refuel floor, regulates suction dampers to Refueling Zone Supply Fans. PdIC 64-2, located on elevation 639 reactor building regulates suction dampers to Reactor Zone Supply.

QUESTION 64

Given the following plant conditions:

- High radiation has been detected in the air inlet to the Unit 3 Control Room.
- Radiation Monitor RE-90-259B is reading 275 cpm above background.

Which ONE of the following describes the Control Room Emergency Ventilation (CREV) System response?

A. NEITHER CREV unit will automatically start under these conditions.

B. BOTH CREV units will automatically start and continue to run with suction from the normal outside air path to Elevation 3C.

C. The Selected CREV unit will automatically start and will continue to run until Control Bay Ventilation is restarted; then, it will automatically stop.

D. The Selected CREV unit will automatically start. The Standby CREV unit will begin to auto-start; but, will ONLY run if the selected CREV unit fails to develop sufficient flow.

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290003 A3.01	
	Importance Rating	3.1	

290003 Ability to monitor automatic operations of the CONTROL ROOM HVAC including:
A3.01 Initiation/reconfiguration

Explanation: **D** CORRECT – The Control Room Emergency Ventilation (CREV) system automatically starts on high radiation sensed by Radiation Monitor RE-90-259B. The actual setpoint is 221 cpm. The radiation level that is given in the stem was chosen because the Tech Spec required setpoint if 270 cpm. This requires the candidate to determine the difference for operational validity. There are two CREV units. One is selected as "Primary" and the other "Standby". On initiation, both CREV units receive a start signal, however the "Standby" unit's signal is delayed 30 seconds to allow the "Primary" unit enough time to start. If the start is successful, the flow from the "Primary" unit will remove a start permissive to the "Standby" unit and it will abort it's startup. CREV units must be manually secured once the initiating conditions have cleared.

A. Incorrect . This is plausible because the Tech Spec initiation setpoint is 270 cpm, which is less than the given radiation level. However, the actual CREV initiation setpoint is 221 cpm.

B. Incorrect. This is plausible since both CREV units receive a start signal on a valid initiation. However, the CREV unit NOT selected will experience a 30 second time delay on initiation and will only complete its start sequence if the selected CREV unit fails to start.

C. Incorrect. This is plausible because the start sequence is correct. However, once initiated, CREV must be manually secured. There is no automatic shutdown capability, only trips.

Technical Reference(s): 0-OI-31, Tech Spec 3.3.7.1, OPL171.067

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: Browns Ferry 0610 NRC Exam #34

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

10 CFR Part 55 Content: 55.41 7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

7. Control Room Emergency Ventilation (CREV) is designed to supply and process the outdoor air needed for pressurization during isolated conditions. There are 2 CREV units rated at 3000 cfm each. A CREV unit consists of Motor-driven fan, (power supply is from 480V RMOV Bd 1A for CREV Fan A; RMOV Bd 3B for CREV Fan B), HEPA filter (common), charcoal filter assemblies located in the CREVS Equipment Room, charcoal heater, and inlet isolation damper and a backflow check outlet damper. They are designed to maintain a positive pressurization to 1/8" w.g. minimum to the control room.

   Tech. Spec. 3.7.3
   Obj.V.B.2/ V.C.6 /V.C.7
   (Old CREV Units abandoned in place as Auxiliary Pressurization Systems)
   TP-4
   2-47E2865-4

   a. A CREV may be started manually from control room Panel 2-9-22 if local control switch is in AUTO position via a 3 position, spring-return to center switch. (STOP-AUTO-START). Actuates only the CREVS unit & associated damper, not the isolation dampers.

      Red indicating lights on panel 3-9-21 to provide indication of CREV Fan A and/or B running on Unit 3. Annunciators are on panel 9-6 for all units.

   b. There is also a 2 position maintained contact, one per train, AUTO-INITIATE/ TEST switch which is used to perform system level actions for that train (primarily testing). It provides the same response as auto start.

   c. Local start at local control station in Relay Room is done using a 2 position maintained, one per train, AUTO-TEST switch. Isolation dampers do not operate automatically if started from local panel.

   d. Automatic start signals are:

      Obj. V.B.1/V.B.2
      Obj. V.C.1
      Obj. V.C.17

      (1) High radiation of 221 cpm above background + 2 Min TD (270 cpm Tech Specs) in air inlet ducts to Control Room from (Radiation monitor RM 90-259A Units 1 & 2, Radiation monitor RM 90-259B Unit 3). Either monitor starts selected CREV unit.

      T. S. 3.3.7.1

(2) Reactor zone ventilation systems radiation high ≥72 MR/hr

e. On receipt of a start signal, normal outside air paths (see below) to elevation 3C are isolated. The selected CREV unit starts once the inlet damper is full open. This supplies pressurizing air to the Unit 1, 2 and 3 Control Rooms. One CREV unit can supply all three control rooms, so the STBY CREV unit will not normally start. Once started, the CREV unit will continue to run until manually secured by first clearing the high radiation signals and the PCIS signals (otherwise equipment cycling will occur)

The inlet damper is normally closed & fails closed. Damper opening takes ~70 seconds. While in the intermediate position both red & green lights will be lit on 2-9-22. The unit heater will energize 10 sec. after the damper is full open to allow the fan to come up to speed. High Rad or PCIS signal will energize relays in Div I (CR1-A) and Div II (CR1-B). Contacts from the CR1 relays are used to energize solenoids to isolate the M.C.R. normal intake dampers (150B,D,E,F, and G)

5.29 Manual Initiation of Control Room Emergency Ventilation (CREV) System

NOTES

1) The CREV System automatically initiates from:
   - PCIS Group 6

     Reactor vessel water level at "LEVEL 3"

     Drywell pressure at 2.45 psig

     Reactor zone exhaust radiation at 72 mr/hr

     Refuel zone exhaust radiation at 72 mr/hr
   - Control Room High Radiation

     221 CPM above background on U1 & 2 (3) Control Room

     Radiation-Gas Radiation Recorder, 0-RR-90-259A(B)
2) The CREV System is normally in Standby Readiness.
3) Performance of this instruction requires the use of two (2) Briggs and Stratton keys.
4) Dampers 0-FCO-31-150B, D, E, F, and G close automatically on auto initiation or a start from the control room using CREV TRAIN A (B) INIT/CB ISOL 0-HS-31-150A (0-HS-31-150B) on Panel 2-9-22. The dampers will NOT operate in response to the local fan control switches (0-HS-31-7214B or 0-HS-31-7213B) or to the control room fan control switches (0-HS-31-7214A or 0-HS-31-7213A).

[1] **IF** CREV failed to initiate on a valid initiation signal, or it is desired to place CREV in service with all associated functions and isolations, **THEN**

PERFORM the following at Panel 2-9-22 unless otherwise specified:

[1.1] **VERIFY** CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, (0-LPNL-25-0628, El. 617, CREV room) is in the desired position for the train to be started. ☐

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 3	1,2,3,(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 528 inches above vessel zero
2. Drywell Pressure - High	1,2,3	2	B	SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 2.5 psig
3. Reactor Zone Exhaust Radiation - High	1,2,3 (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
4. Refueling Floor Exhaust Radiation - High	1,2,3, (a)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 100 mR/hr
5. Control Room Air Supply Duct Radiation - High	1,2,3, (a)	1	D	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 270 cpm above background

(a) During operations with a potential for draining the reactor vessel.

3.5 **CREV and CREV instrumentation operability issues (continued)**

G. CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 may be placed in either the "A" or "B" position, depending on the operability status of the CREV trains. When a CREV train is inoperable, it will NOT be selected as lead. When both CREV trains are operable, the preferred position for CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, is in TRAIN "A" which makes the "A" CREV the lead train. In the event that "A" CREV is INOP, CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 is required to be placed in the TRAIN "B" position so the "B" CREV will initiate, without a time delay, as the lead train.

H. When one of the CREV trains is inoperable for testing, the CREV UNIT PRIMARY SELECTOR SWITCH, 0-XSW-031-7214 is required to be aligned to the train which is NOT under testing conditions to ensure the non-test train will initiate under an actual initiation signal.

BFN ILT 0610 NRC Exam #34

34. Given the following plant conditions:

High radiation has been detected in the air inlet to the Unit 3 Control Room.

- Radiation Monitor RE-90-259B is reading 275 cpm above background.

Which ONE of the following describes the Control Room Emergency Ventilation (CREV) System response?

A. NEITHER CREV unit will automatically start at the current radiation level.

B. BOTH CREV units will automatically start with suction from the normal outside air path to Elevation 3C.

C. The Selected CREV unit will automatically start and will continue to run until Control Bay Ventilation is restarted; then, it will automatically stop.

D. The Selected CREV unit will automatically start. The Standby CREV unit will begin to auto-start; but, will ONLY run if the selected CREV unit fails to develop sufficient flow.

QUESTION 65

Which ONE of the following describes the piping configuration for the RWCU system return flow back to the reactor?

A. Units 2 and 3 can return via "A" or "B" Feedwater spargers.

B. Units 1 and 2 can return via "A" or "B" Feedwater spargers.

C. ONLY Unit 3 can return via "A" or "B" Feedwater spargers.

D. ONLY Unit 1 can return via the "A" or "B" Feedwater spargers.

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	290002 K1.14	
	Importance Rating	2.9	

Knowledge of the physical connections and/or cause effect relationships between REACTOR VESSEL INTERNALS and the following: RWCU

Explanation: **C** CORRECT: 3-OI-69 directs return through manual valve ISV-069-0625 to Feedwater line A **OR** ISV-069-580 to feedwater line B: Procedurally you can return to only one feedwater line at a time.

A-Incorrect- Unit 2 does not return through feedwater line A. Unit 3 can return through either A or B. This is plausible if the candidate knows that two units return through feedwater line B but thinks it is units 2 and 3 not 1 and 2.

B- Incorrect - Units 1 and 2 return through feedwater line B. This is plausible because it is just the opposite of the correct answer.

D- Incorrect- Unit 1 returns through feedwater line B ONLY. Plausible if the candidate knows one unit can return through both lines but believes that to be Unit 1.

Technical Reference(s): OPL171.013

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: No

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

10 CFR Part 55 Content: 55.41 41 (3) Mechanical components and design features of reactor primary system.

e. Water is returned to the reactor vessel via the following path:

Obj. V.B.3
Obj. V.C.3

(1) Regenerative heat exchanger (shell side)

(2) Return isolation valve (69-12)

(3) Thermal sleeve

(4) Feedwater Line B (Units 1 & 2), Feedwater Line A & B (Unit 3)*

*The valve line-up in 3-OI-69 has been revised to show ISV-069-0625 RWCU SYSTEM RETURN MANUAL ISOL. Valve open or closed. *Either 3-ISV-069-0625 <u>or</u> 3-ISV-069-0580, BUT NOT BOTH, may be OPEN during power operation .

UNIT DIFFERENCE
PCR 07004710 added new section, Unit 3 can return to either "A" or "B" feedwater line.

QUESTION 66

Which ONE of the following describes how the Offgas System functions to create a substantial reduction in the release of radioactive materials to the environment?

A. The system reduces the volume of Offgas flow; AND delays the release of hydrogen and oxygen to the environment.

B. The system reduces the volume of Offgas flow; AND delays the release of xenon and krypton to the environment.

C. The system recombines short-lived radioactive gases; AND delays the release of hydrogen and oxygen to the environment.

D. The system recombines short-lived radioactive gases; AND delays the release of xenon and krypton to the environment.

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.1.27	
	Importance Rating	3.9	

G 2.1.27 Knowledge of system purpose and/or function.

Explanation: **B** CORRECT: The offgas recombiner reduces the total volume of the offgas flow by recombining radiolytically disassociated hydrogen and oxygen to produce water vapor. After recombination, the offgas flow consist of small volume amounts of fission product and activation gases carried in the airflow arising out of in-leakage to the condenser. This offgas stream is delayed for decay of short lived radioactive isotopes, and then conditioned to a low moisture content and the proper temperature for maximum delay in the charcoal adsorber system. The long holdup time produced by the charcoal adsorbers permits the xenon and krypton gases to decay and be removed by the charcoal adsorber and a HEPA filter. This vapor stream is condensed in the offgas condenser and the moisture is drained to the main condenser.

A-Incorrect. First part correct, Second part wrong. This is plausible because the operator may incorrectly believe that it is the Hydrogen and Oxygen that undergoes a delayed release.

C- Incorrect. First part wrong, second part wrong. This is plausible because the operator may incorrectly believe that it is the short lived activation gases (i.e. N-16, O-19, and N-13, which are all gases expected to be within the system) that are recombined, and that it is the Hydrogen and Oxygen that undergoes a delayed release.

D- Incorrect. First part wrong, Second part correct. This is plausible because the operator may incorrectly believe that it is the short lived activation gases (i.e. N-16, O-19, and N-13, which are all gases expected to be within the system) that are recombined.

Technical Reference(s): OPL171.030

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:	
Question History:	Previous NRC: No	
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis :	

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

INSTRUCTOR NOTES

X. Lesson Body

A. General Description

Review the System Health report prior to teaching.

1. Function

a. The purpose of the Off-Gas System is to minimize and control the release of radioactive krypton, xenon, N^{13}, N^{17}, and 0^{19} isotopes by allowing optimum decay time before discharge to the atmosphere. Additionally, the system minimizes the explosion potential within the Off-Gas System through dilution and controlled recombination of gaseous radiolytic hydrogen and oxygen. The recombination process provides the added benefit of volume reduction in the quantity of gas that is to be processed and exhausted.

Obj. V.B.1

b. The Off-Gas System also reduces the amount of radioactive particulate material released to the atmosphere through filtration of the off-gas before release to the off-gas vent pipe (stack).

c. The Off-Gas system draws and maintains main condenser vacuum. Lowering the vacuum in the condenser lowers cycle efficiency and raises the cost of electrical production. If condenser vacuum lowers below 21.8" Hg Vacuum, the main turbine will trip. If condenser vacuum lowers below 7" Hg Vacuum, the reactor feedpumps will trip and the main steam bypass valves will be lost.

Obj. V.B.1

QUESTION 67

Unit 3 was at 100% Reactor Power when a leak resulted in Suppression Pool Level of 11.4 feet and slowly lowering. The required actions of the EOIs have been performed.

Which ONE of the following completes the statement?

Two minutes after initiating required EOI actions, Wide Range Reactor Pressure Indication(s) available on Control Room Panel(s) __(1)__ will be __(2)__.

A. (1) 3-9-5 ONLY
(2) stable

B. (1) 3-9-5 ONLY
(2) lowering

C. (1) 3-9-3 AND 3-9-5
(2) stable

D. (1) 3-9-3 AND 3-9-5
(2) lowering

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.1.31	
	Importance Rating	4.6	

G2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Explanation: **D CORRECT**: Wide Range Pressure indication is available on both 3-9-3 and 3-9-5. Part 2 correct – Per 3-EOI-2, if Suppression Pool Level cannot be maintained > 11.5 feet, Reactor Scram and Emergency Depressurization are required.

A- Incorrect: First Part: Incorrect – Plausible in that this would be the correct answer if the question asked where Narrow Range Pressure indication is available. Second Part: Incorrect – Plausible in that in accordance with 3-EOI-2, reactor scram is required if Suppression Pool cannot be maintained >11.5 feet. Two minutes after the scram, reactor pressure would be stable. However, this is incorrect since 3-EOI-2 also required ED for this condition.

B- Incorrect: First Part: Incorrect – Plausible in that this would be the correct answer if the question asked where Narrow Range Pressure indication is available. Second Part: Correct.

C- INCORRECT: First Part: Correct – See Explanation D. Second Part: Incorrect – Plausible in that in accordance with 3-EOI-2, reactor scram is required if Suppression Pool cannot be maintained >11.5 feet. Two minutes after the scram, reactor pressure would be stable. However, this is incorrect since 3-EOI-2 also required ED for this condition.

Technical Reference(s): 3-EOI-1, 3-EOI-2, OPL171.003

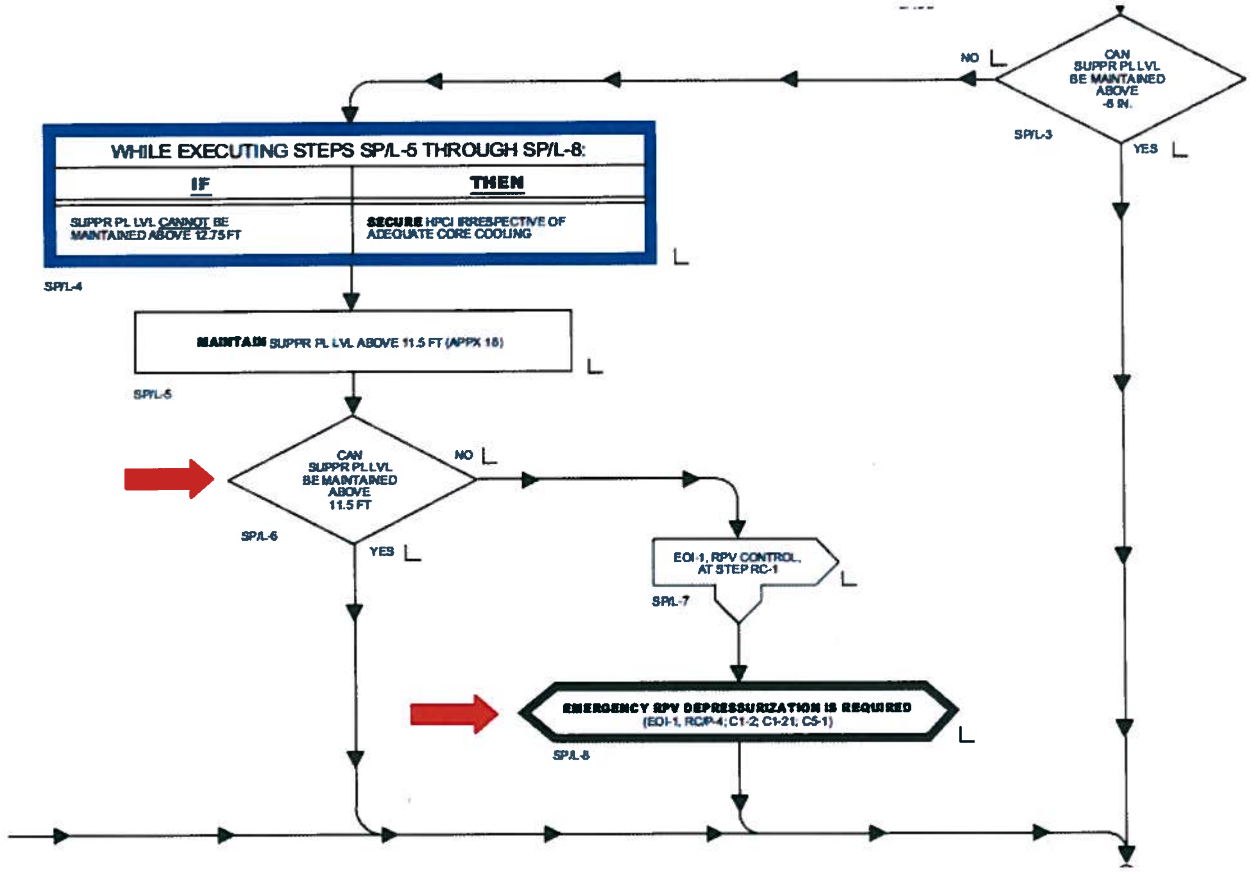
Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.203 V.B.13

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Browns Ferry 1102 #15
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

3-EOI-2, Primary Containment Control



Browns Ferry 1102 #15

ILT 1102 Written Exam

15. 295030 G2.1.31

Unit 3 was at 100% Reactor Power when a leak from the Torus resulted in Suppression Pool Level of 11.4 feet. Required actions of the EOIs have been performed.

Which ONE of the following completes the statement below?

Two minutes after initiating required EOI actions, Wide Range Reactor Pressure Indication(s) available on Control Room Panel(s) __(1)__ will be __(2)__.

A. **(1)** 3-9-5 **ONLY**
(2) stable

B. **(1)** 3-9-5 **ONLY**
(2) lowering

C. **(1)** 3-9-3 **AND** 3-9-5
(2) stable

D. **(1)** 3-9-3 **AND** 3-9-5
(2) lowering

QUESTION 68

Which ONE of the following describes a condition that is in compliance with 0-GOI-100-3C, Fuel Movement Operations During Refueling?

A. Spent Fuel Storage Pool Temperature is 67°F.

B. 2 way directional bridge movements may be performed in the Spent Fuel Pool with the grapple loaded.

C. Fuel may be stored in the defective storage rack adjacent to the sipping rack while fuel sipping is in progress.

D. The refuel platform air system should be in operation when the mast is submerged.

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.1.36	
	Importance Rating	3.0	

G 2.1.36 Knowledge of procedures and limitations involved in core alterations.

Explanation: **D** CORRECT: IAW 0-GOI-100-3C . The refuel platform air system should be in operation when the mast is submerged.

A-Incorrect. **68**°F is the minimum temperature, per criticality analysis, therefore **67**°F is NOT in compliance. Plausible if the candidate does not know the minimum temperature requirements for the Spent Fuel Storage Pool.

B- Incorrect. Plausible because P&L R in 0-GOI-100-3C discusses both 3 way movements and 2 way movements in the Spent Fuel Storage Pool area.

C- Incorrect. Plausible because P&L G discusses ALL bundles must be placed in racks however it lists this exception- No fuel is to be stored in the defective fuel storage rack adjacent to the sipping rack when sipping is in progress.

Technical Reference(s): 0-GOI-100-3C

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: No
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis :

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

0-GOI-100-3C, Fuel Movement Operations During Refueling

3.5 Fuel or Fuel Related Component (FRC) Handling (continued)

J. The refuel platform mast air system should be in service and not isolated when the mast is in the water. Isolating the air while the mast is submerged, allows water to bleed into the air system. It is recommended that condensate be drained from from the air compressor tank approximately once every 4 hours when the bridge is in use to prevent loss of air to grapple.[PER 221107]

G. No fuel is to be stored in the defective fuel storage rack adjacent to the sipping rack when sipping is in progress. Bundles shall be placed in racks.
REFER TO NPG-SPP-05.8.

N. The Fuel Pool Cooling System is only required to be in service to maintain fuel pool water within specifications listed in NPG-SPP-05.3 and water temperature less than 125°F but greater than 72°F.

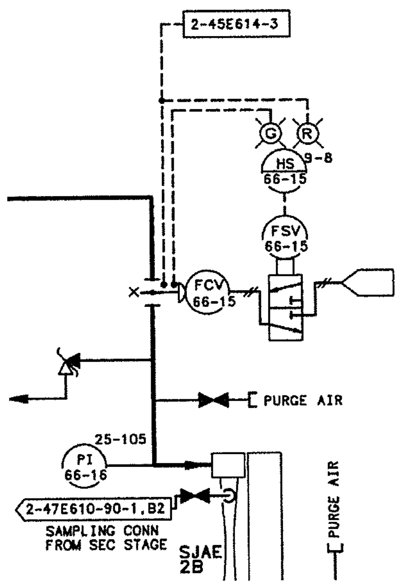
1. [NER/C] Temperature in the SFSP below 68°F exceeds the temperature reactivity assumed in the criticality analysis. [INPO SER 90-017]

2. [NER/C] Maintaining water temperature below 100°F minimizes the release of soluble activity. [GE SIL 541]

R. Three-direction movements are <u>NOT</u> allowed in the Spent Fuel Pool. Two-direction horizontal movements (X and Y directions only) are allowed with the Main Fuel Handling Grapple NOT loaded; however, extreme caution shall be used when utilizing two-direction horizontal movements in the Spent Fuel Pool or three-direction movements in the vessel area.

QUESTION 69

The operating crew is implementing a clearance for the **Unit 2** Steam Jet Air Ejector 2A, Handswitch 2-HS-66-15, at panel 2-9-8 has been placed in the "CLOSED" position.



Which ONE of the following predicts how the valve and actuator will respond if the solenoid valve power is removed?

2-FCV-66-15 will ___(1)____, and air will be ___ (2)___ the 2-FCV-66-15 actuator.

A. (1) stroke open
   (2) vented off of

B. (1) stroke open
   (2) supplied to

C. (1) remain closed
   (2) supplied to

D. (1) remain closed
   (2) vented off of

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.2.15	
	Importance Rating	3.9	

G2.2.15 Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.

Explanation: **D** CORRECT: With the handswitch in CLOSED 2-FCV-66-15 will be closed. When power is removed from the solenoid, the air will remain vented off of the actuator and the valve will remain closed. The candidate must use the drawing provided to determine the impact of the a loss of electrical power on the operation of 2-FCV-66-15 (i.e. is the solenoid energized when the valve is open or closed, does the solenoid vent air off the actuator when the solenoid is energized or de-energized)

A-Incorrect. Plausible if the candidate is confused which state (energized or de-energized) the solenoid is shown in the drawing. In addition the candidate needs to know which way the valve fails on a loss of power.

B- Incorrect. Plausible if the candidate is confused which state (energized or de-energized) the solenoid is shown in the drawing. In addition the candidate needs to know which way the valve fails on a loss of power.

C- Incorrect. Plausible if the candidate is confused which state (energized or de-energized) the solenoid is shown in the drawing. In addition the candidate needs to know which way the valve fails on a loss of power.

Technical Reference(s): Drawing 2-47E610-66-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

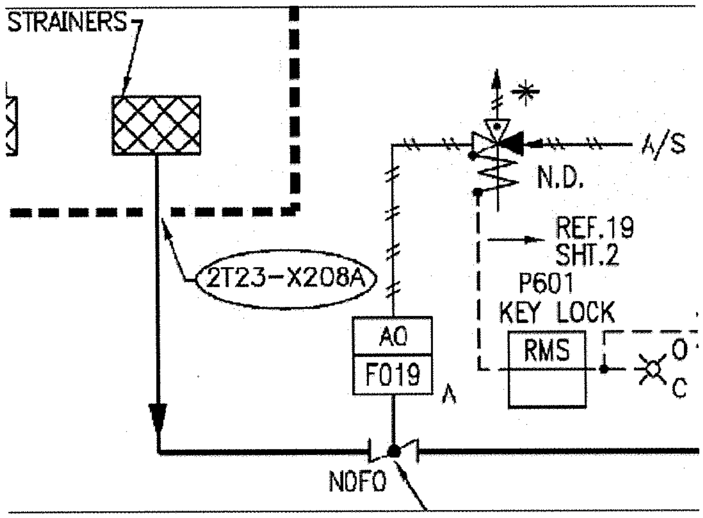
Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: Hatch 2009 #68
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis :

10 CFR Part 55 Content: 55.41 (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Hatch 2009 #69

69. G2.2.15 001

The crew is implementing a tagout for the **Unit 2** Core Spray system and the 2E21-F019A, "Torus Suction Valve" keylock control switch at panel P601 has been placed in the "CLOSED" position.



Which ONE of the following predicts how the valve and actuator will respond if the solenoid valve power is removed?

2E21-F019A will __(1)__.

Air will be __(2)__ the 2E21-F019A actuator.

A.✓ (1) stroke open
(2) vented off of

B. (1) stroke open
(2) supplied to

C. (1) remain closed
(2) vented off of

D. (1) remain closed
(2) supplied to

QUESTION 70

Which ONE of the following completes the statement?

The MCPR fuel cladding integrity Safety Limit ensures that _______.

A. during normal operation and transients at least 99.9% of the fuel rods do NOT experience Transition Boiling.

B. the peak cladding temperature during the design basis LOCA does not exceed the limits in 10CFR46.

C. the calculated total oxidation of the cladding shall not exceed 0.17 times the total cladding thickness before oxidation.

D. fuel thermal-mechanical design limits are not exceeded anywhere in the core during normal operation.

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.2.25	
	Importance Rating	3.2	

G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Explanation: **A** CORRECT: This question tests the bases for the MCPR safety limit by use of distracters composed from the bases/failure mechanism from other thermal limits. Candidates often confuse the mechanisms associated with safety/thermal limits.

B-Incorrect. This is the failure mechanism due to APLHGR following a LOCA. Nothing stated in the stem indicates that APLHGR is being violated. Plausible since it is one of the safety limit failure mechanisms.

C- Incorrect. This one of the effects from elevated clad temperatures (APLHGR). Zirc Water reactor occurs when clad temperatures reach 2200°F, which is the temperature associated with APLHGR limit violation. Nothing stated in the stem indicates that APLHGR is being violated. Plausible since it is a condition associated with one of the safety limit violations.

D- Incorrect. This is the failure mechanism for LHGR violation and nothing stated in the stem indicates that LHGR is being violated.

Technical Reference(s): TS 2.0 Safety Limits, TS Bases for safety limits B2.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Hatch 2009 #70
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis :

10 CFR Part 55 Content: 55.41 (1) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

TS 2.0 Safety Limits
TS Bases for safety limits B2.1.1

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormal operational transients. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

Lesson Plan OPL171.087 Technical Specifications

Iterative computer models are run by the various fuel vendors calculating a margin to MCPR for the various power excursion transients terminated by plant protective actions (SCRAM / EOC-RPT). Uncertainties in exact core operating state and uncertainties in the calculational methodologies result in the safety limit being established sufficiently above a MCPR of 1.0 to ensure that 99.9% of the fuel rods will never exceed boiling transition.

QUESTION RO 71

Which ONE of the following completes the statement below in accordance with Technical Specification 3.1.3 Control Rod OPERABILITY?

If one withdrawn control rod is stuck, stuck control rod separation criteria must be verified ________.

A. immediately

B. within 15 minutes

C. within 30 minutes

D. within 1 hour

Correct Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.2.39	
	Importance Rating	3.9	

G2.2.39 K&A: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

A - **CORRECT: IMMEDIATELY**

B – INCORRECT: 15 minutes is a time for reactor pressure TS 3.4.10

C - INCORRECT: 30 minutes for TS 3.4.9 RCS heatup/cooldown

D– INCORRECT: numerous times in TS have one hour action statements

Technical Reference(s): Tech Specs 3.1.3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.006 V.B.22

Question Source:	Bank:	
	Modified Bank:	
	New	X
Question History:	Previous NRC:	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each control roc.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	-----NOTE----- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ---------- A.1 Verify stuck control rod separation criteria are met.	Immediately
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

QUESTION 72

A Unit 2 startup is in progress. The Mode Switch is in RUN, and reactor power is 16%.

There are indications of a leak in the drywell, and a drywell entry is being planned.

Which ONE of the following completes the statement?

Drywell entry AT POWER with the Mode Switch in RUN must be authorized by the__________.

A. Plant Manager

B. Site Vice President

C. Radiation Protection Manager

D. Shift Manager

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.3.13	
	Importance Rating	3.4	

G2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Explanation: **A** CORRECT – 2-GOI-200-2 requires Plant Manager permission for drywell entries at NOT/NOP (Precaution 3.1.H) and for entries with the Mode Switch in RUN (3.2.E)

B. Incorrect Plausible if the candidate believes that only the highest on site manager must approve the entry.

C Incorrect – Plausible if the candidate believes that the senior radiation manger is sufficient.

D Incorrect – Plausible if the candidate believes that NO entry at power is permitted.

Technical Reference(s): 2-GOI-200-2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC None	
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	(12) Radiological safety principles and procedures.

2-GOI-200-2 , Primary Containment Initial Entry and Closeout

H. Permitting access to the Drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the Reactor and operating for extended periods with significant leaks in the Primary System, leak inspections are scheduled during startup periods, when the Primary System is at or near rated operating temperature and pressure. These entries require Plant Manager permission.

Entry to the Drywell to perform a leak inspection, when the Primary system is at or near rated operating temperature and pressure, requires ________.

QUESTION RO 73

The Unit 3 is operating at 100%. An Assistant Unit Operator reports that the Hydrogen Water Chemistry System has just been placed into service per 3-OI-4, Hydrogen Water Chemistry System, section 5.4.

Which ONE of the following completes the statement?

If the H2 injection rate were adjusted from the Normal Unit 3 H2 Injection rate at 100% power to___(1)____, the Main Steam Line Radiation levels will be ___(2)___.

A. (1) 16 scfm
(2) HIGHER, due to a HIGHER than normal Hydrogen injection rate

B. (1) 12 scfm
(2) HIGHER, due to a HIGHER than normal Hydrogen injection rate

C. (1) 16 scfm
(2) LOWER, due to a LOWER than normal Hydrogen injection rate

D. (1) 12 scfm
(2) LOWER, due to a LOWER than normal Hydrogen injection rate

Correct Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.3.14	
	Importance Rating	3.4	

G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Explanation: **A CORRECT** : Unit 3 normal rate is 12 scfm. An increased hydrogen flowrate (16 scfm) will cause an increase in MSL rad levels, due to the excess carryover of N16 / ammonia.

B - INCORRECT: First Part: Incorrect, the normal rate is 12 scfm, so there would be NO change in the MSL rad levels. Plausible because 16 scfm is the normal injection rate for Unit 2.

C – INCORRECT: Raising the hydrogen injection rate will NOT lower MSL rad levels. Plausible because 16 scfm is the normal injection rate for Unit 2.

D – INCORRECT: First Part: Incorrect, the normal rate is 12 scfm, so there would be NO change in the MSL rad levels. Plausible if the candidate does not know the normal Unit 3 injection rate.

Technical Reference(s): 3-OI-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: No

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

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

C. The hydrogen flow controller directs hydrogen flow from the hydrogen supply system to the suction of each condensate booster pump. This controller has two modes of operation, Power Determined Setpoint mode and Operator Determined Setpoint mode.

1. Automatic/Power Determined Setpoint mode automatically changes hydrogen injection flow in response to changes in reactor power. This is accomplished by the PLC computing the desired hydrogen flow rate based on reactor feedwater flow. The Automatic/Power Determined Setpoint mode is used for normal operation of the HWC System and also used when reducing hydrogen injection related dose rates to support Maintenance, Chemistry, or Radiation Protection activities while the plant is operating.

3.3 The following precautions pertain to Hydrogen Water Chemistry System operation: (continued)

D. Important hydrogen injection flow rate values are as follows:

1. Minimum H_2 Injection Rate allowed to be entered on the OIU: 3 scfm. This is the injection rate normally used when lowering HWC for ALARA considerations or maintenance purposes per Section 6.0, Normal Operations. When 5 scfm is entered in the OIU for Automatic/Power Determined Mode, H_2 Injection Rate will lower automatically to a new scfm depending on the new power level, i.e., 5 scfm for 100% power; when power is lowered to 90%, the injection rate will automatically roll back to 4.5 scfm and so on.

2. Normal H_2 Injection Rate (100% Reactor Power): 12 scfm (This value is set by Chemistry with the performance of CI-13.1, Chemistry Program. Chemistry will notify Operations should this value change). "Off Normal" operating conditions may require other injection rates which must be coordinated with the System Engineer, Chemistry, Radiation Protection, and approved by the Unit Supervisor/SRO.

3. Maximum H_2 Injection Rate allowed: 25 scfm.

5.4 Placing the Hydrogen Water Chemistry System in Service (continued)

- Oxygen Controller is in Automatic (SF10) / Hydrogen Determined Setpoint mode (F10).
- Hydrogen Controller is in Automatic (SF1)/ Operator Determined Setpoint mode (SF1).

[12.9] **SET** the hydrogen flow DESIRED (F3) setpoint to 3 SCFM, and then **PRESS** ENTER.

[12.10] **CHECK** the following indications on the OIU:

- Hydrogen flow ramps up to 3 SCFM.
- Oxygen flow ramps up to 1.5 SCFM after a time delay (delay up to 15 min for low power operation).
- Offgas oxygen concentration stabilizes at 21% ± 5%.
- Power Determined Setpoint (F6) is set at 12 SCFM. **REFER TO** Section 6.1.

[12.11] **WHEN** Reactor Power is >20% and Oxygen concentration has stabilized at 21% ± 5%, **THEN**

TRANSFER the hydrogen controller from Automatic / Operator Determined Setpoint mode to Automatic / Power Determined Setpoint mode by pressing F1.

[12.12] **CHECK** that hydrogen flow ramps up to a pre-programmed value based on plant power and that oxygen flow follows hydrogen flow after the programmed time delay (delay up to 15 min for low power operation).

[12.13] **WHEN** steady state oxygen flow and indication is achieved, **CHECK** Offgas oxygen concentration stabilizes at 21% ± 5%.

QUESTION RO 74

The Unit 1 reactor has just automatically scrammed. The following plant conditions exist:

- Reactor power 3%
- Reactor water level +10 inches (slowly rising, lowest level observed was (-)25 inches)
- Reactor pressure 1047 psig
- Suppression Pool Level 16 feet
- Suppression Pool Temperature 94° F and slowly rising
- Drywell temperature is 140° F and slowly rising

Which ONE of the following completes the statements?

The Unit-1 Unit Supervisor will enter EOI- 1 ___(1)___.

The Operator at the Controls will immediately initiate ARI ___(2)___.

A. (1) and C-5 ONLY
(2) ONLY

B. (1) EOI-2, and C-5
(2) ONLY

C. (1) and C-5 ONLY
(2) and trip the Reactor Recirculation Pumps

D. (1) EOI-2, and C-5
(2) and trip the Reactor Recirculation Pumps

Correct Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.4.1	
	Importance Rating	4.6	

G2.4.1 Knowledge of EOP entry conditions and immediate action steps.

Explanation: **B CORRECT** : C-5 entry is required since EOI-1 RC/L-1 applies, "It has NOT been determined that the reactor will remain subcritical under all conditions without boron…Exit RC/L and Enter C-5. EOI-2 is required because Suppression Pool level is above -1 inch. Which corresponds to approximately 15 feet.

A - Incorrect: First Part: Incorrect. EOI-2 is required because Suppression Pool level is above -1 inch. Which corresponds to approximately 15 feet. Plausible if the candidate does not recognize the 16 foot suppression pool level requires entry to EOI-2. Second Part: Correct.

C – INCORRECT: First Part: Incorrect. EOI-2 is required because Suppression Pool level is above -1 inch. Which corresponds to approximately 15 feet. Plausible if the candidate does not recognize the 16 foot suppression pool level requires entry to EOI-2. Second Part: Incorrect. Tripping the Recirc Pumps is not required since power is below 5%. Plausible because EOI-1 step RC/Q-7 directs tripping the recirc pumps if power is >5% or unknown.

D – INCORRECT: First Part: Correct. Second Part: Incorrect. Tripping the Recirc Pumps is not required since power is below 5%. Plausible because EOI-1 step RC/Q-7 directs tripping the recirc pumps if power is >5% or unknown.

Technical Reference(s): 1-EOI-1 and 2 and C-5

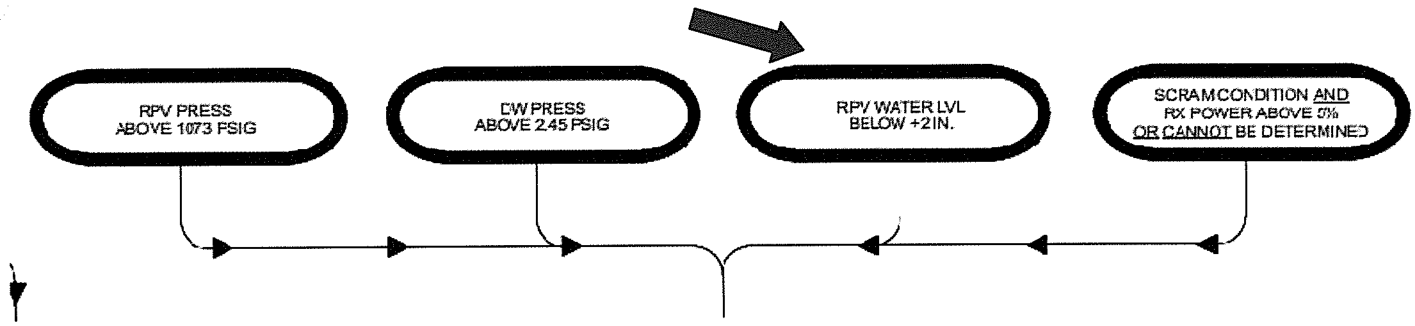
Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

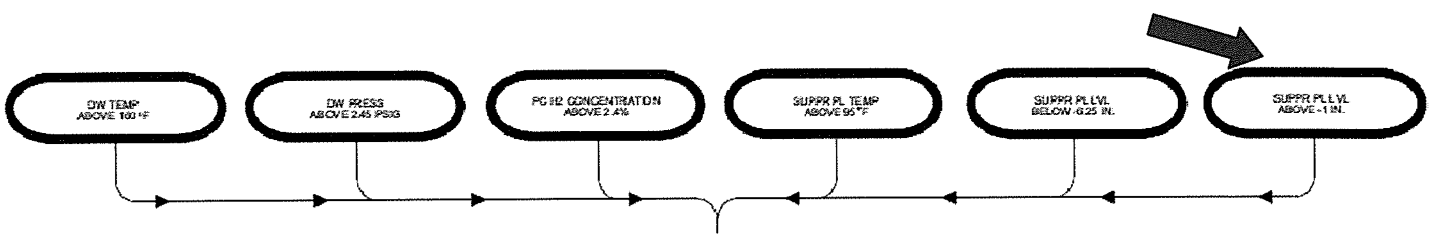
Question Source:	Bank: Modified Bank: X New	
Question History:	Previous NRC: River Bend 2008 #74	
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis	

10 CFR Part 55 Content: 55.41 (5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

1EOI-1 Entry Conditions

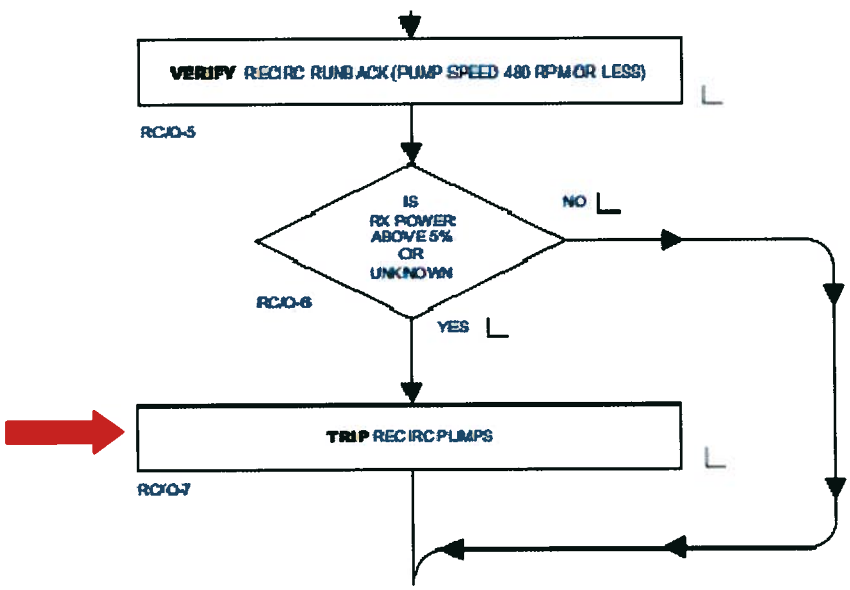


1-EOI-2 Entry Conditions



WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
IT HAS **NOT** BEEN DETERMINED THAT THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS (SEE NOTE)	**EXIT** RC/L AND **ENTER** C5, LEVEL/POWER CONTROL
RPV WATER LVL CANNOT BE DETERMINED	**EXIT** RC/L AND **ENTER** C4, RPV FLOODING
PC WATER LVL CANNOT BE MAINTAINED BELOW 105 FT OR SUPPR CHMBR PRESS CANNOT BE MAINTAINED BELOW 55 PSIG	**STOP** INJ INTO THE RPV FROM SOURCES EXTERNAL TO THE PC NOT REQUIRED FOR ADEQUATE CORE COOLING.

RC/L-3



QUESTION 74 Rev 1

Examination Outline Cross-Reference:	Level	RO☒ SRO[
	Tier #	3
	Group #	Emergency F
	K/A #	G. 2.4.1
	Importance Rating	4.6

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question:

The reactor has just scrammed. The following plant conditions exist:

• Reactor power	3%
• Reactor water level	17 inches (lowest level observed was 15 in
• Reactor pressure	1047 psig
• Suppression Pool Level	20 feet 2 inches
• Drywell H2	0.4%
• Drywell pressure	0.2 psid

Which of the following represents the required EOP(s) to enter?

A. EOP-1 and EOP-2

B. EOP-2 only

C. EOP-1A and EOP-2

D. EOP-1 only

Proposed Answer: B

QUESTION 75

A Unit 1 startup is in progress. Reactor power has reached Range 4 on all IRMs and is continuing to increase. Control rod 26-27 is at position 12 and being withdrawn to position 16. While the control rod is driving the collet fingers fail.

Which ONE of the following describes the impact of this failure on the core?

When the operator stops driving the Control Rod, it will move ___(1)___ the core, and, by procedure the operator at the controls will select the Control Rod and ___(2)___.

A. (1) INTO
(2) INSERT it to the FULL IN position

B. (1) INTO
(2) CONTINUOUSLY INSERT it to position 00 using EMERG ROD IN

C. (1) OUT OF
(2) INSERT it to the FULL IN position

D. (1) OUT OF
(2) CONTINUOUSLY INSERT it to position 00 using EMERG ROD IN

Correct Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.4.9	
	Importance Rating	3.8	

G2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Explanation: **C** CORRECT: Either foreign material from the reactor entering and lodging in the CRD collet area, wedging the collet open OR collet retainer tube cracking failure can cause a control rod to drift OUT. Per 1-AOI-85-6, Rod Drift OUT: IF a Control Rod is moving from its intended position without operator actions, THEN SELECT the drifting control rod and INSERT to the FULL IN (00) position.

A - INCORRECT: First Part: Incorrect, the control rod will drift out. Plausible if the candidate confuses the failure of collet fingers with the failure of scram outlet valve which causes a drift IN. Second Part: Correct.

B- INCORRECT: First Part: Incorrect, the control rod will drift out. Plausible if the candidate confuses the failure of collet fingers with the failure of scram outlet valve which causes a drift IN. Second Part: Incorrect. Plausible because this is the action for a Control Rod Drop.

D – INCORRECT: First Part: Correct. Second Part: Incorrect. Plausible because this is the action for a Control Rod Drop.

Technical Reference(s): 1-AOI-85-5, 1-AOI-85-6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

4.0 OPERATOR ACTIONS

CAUTION

[NRC/C] Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. [NCO 940245010]

4.1 Immediate Actions

[1] **IF** multiple Control Rod drifts are identified, **THEN**

MANUALLY SCRAM the Reactor and enter 1-AOI-100-1. ☐

4.2 Subsequent Actions

[1] **IF** a Control Rod is moving from its intended position without operator actions, **THEN**

SELECT the drifting control rod and **INSERT** to the FULL IN (00) position. ☐

[2] **IF** Control Rod Drive does NOT respond to INSERT signal, **THEN**

PERFORM the following: (Otherwise N/A)

[2.1] **REDUCE** Total Core Flow, as indicated on TOTAL CORE FLOW/CORE PRESS DROP, 1-XR-68-50 on Panel 1-9-5, by approximately 10% to control possible power increase. ☐

[2.2] [NER/C] **IF** drifting control rod is causing Reactor power to rapidly rise at a rate which can NOT be controlled by reducing recirculation flow, **THEN**

MANUALLY SCRAM the Reactor. (Otherwise N/A) [INPO SER 90-015] ☐

[3] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. ☐

[4] **IF** another Control Rod Drift occurs before Reactor Engineering completes the evaluation, **THEN**

MANUALLY SCRAM Reactor and enter 1-AOI-100-1. ☐

4.2 Subsequent Actions (continued)

NOTES

1) A control rod that will NOT latch could be indicative of either the following:
   - A failed open withdraw directional control valve (1-FCV-085-40C).
   - Foreign material from the Reactor entering and lodging in the CRD collet area, wedging the collet open.
   - Collet retainer tube cracking failure.
2) If control rod does not latch at position 00, Step 4.2[6] may be skipped to avoid control rod drift to full out position when removed from service.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **VERIFY** automatic actions which have occurred. ☐

[2] **IF** NO scram occurs, **THEN**

REDUCE Reactor power by 10% from the power prior to event. ☐

[3] **OBTAIN** concurrence of SM and **CONTINUOUSLY INSERT** dropped control rod to position 00 using EMERG ROD IN switch. ☐

[4] **WHEN** control rod is fully inserted, **THEN**

REMOVE associated HCU from service. **REFER TO** 1-OI-85. ☐

[5] **ADJUST** reactor power and flow as directed by Reactor Engineer/Unit Supervisor to stay within required thermal and feedwater temperature limits. **REFER TO** 1-GOI-100-12 or 1-GOI-100-12A for the power reduction. ☐

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** multiple rods are drifting into core, **THEN**

MANUALLY SCRAM Reactor. **REFER TO** 1-AOI-100-1. ☐

4.2 Subsequent Actions

[1] **IF** the Control Rod is moving from its intended position without operator actions, **THEN**

INSERT the Control Rod to position 00 using CONTINUOUS IN. (Otherwise N/A) ☐

[2] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. ☐

[3] **IF** another Control Rod Drift occurs before Reactor Engineering provides a verbal or written evaluation, **THEN**

MANUALLY SCRAM Reactor and enter 1-AOI-100-1. ☐

[4] **CHECK** Thermal Limits on ICS by running OFFICIAL 3D. ☐

[5] **ADJUST** control rod pattern as directed by Reactor Engineer and **CHECK** Thermal Limits on ICS (RUN OFFICIAL 3D).. ☐

[6] **IF** CRD Cooling Water Header DP is excessive and causing the control rod drift, **THEN**

ADJUST CRD SYSTEM FLOW CONTROL, 1-FIC-85-11, and CRD DRIVE WATER PRESS CONTROL VLV, 1-HS-85-23A, as required to establish the following: (Otherwise N/A)

- CRD DRIVE WTR HDR DP, 1-PDI-85-17A, between 250 and 270 psid ☐
- CRD SYSTEM FLOW CONTROL, 1-FIC-85-11, between 40 and 65 gpm ☐
- CRD CLG WTR HDR DP, 1-PDI-85-18A, at about 20 psid or as close as possible while maintaining flow and Header pressure. ☐