QUESTION 1 Rev 2

U2 is in mode 5 with Refueling in progress, with the following system alignments:

- 2B RHR pump is in shutdown cooling.
- Both 2A and 2B Reactor Recirc pumps are tagged out of service.
- 2A RPS is powered from its alternate source.

Subsequently:

- Reactor Water Level drops to 0 inches and then recovers to + 75 inches.
- The only action taken by the crew was to reset any actuation(s) that may have occurred.

Note:
RHR Sys II LPCI Inboard Injection Valve, 2-FCV-74-67
RHR Sys II LPCI Outboard Injection Valve, 2-FCV-74-66
RHR Shutdown Cooling Suction Outboard Isolation Valve, 2-FCV-74-47
RHR Shutdown Cooling Suction Inboard Isolation Valve, 2-FCV-74-48

Which one of the following describes the **minimum** actions required, in accordance with 2-AOI-74-1, Loss of Shutdown Cooling, prior to restarting the 2B RHR pump to restore Shutdown cooling?

A. CLOSE the 2-FCV-74-67, OPEN the 2-FCV-74-66, then OPEN 2-FCV-74-47 and 2-FCV-74-48.

B. CLOSE the 2-FCV-74-66, OPEN the 2-FCV-74-67, then OPEN 2-FCV-74-47 and 2-FCV-74-48.

C. CLOSE the 2-FCV-74-67 OPEN 2-FCV-74-66, then OPEN 2-FCV-74-48 only.

D. CLOSE the 2-FCV-74-66 OPEN 2-FCV-74-67, then OPEN 2-FCV-74-47 only.

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295001 G2.1.23	
	Importance Rating	4.3	4.4

Partial or Complete Loss of Forced Core Flow Circulation; Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Justification for K/A match: 295001 Partial or Complete Loss of Forced Core Flow Circulation. This includes both Reactor Recirc and RHR Shutdown Cooling flow loss. With a conduct of operations generic K/A of 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. We are saying that since the first part of the K/A is an A/EAP that we can use the integrated plant procedure (AOI-74-1), which is an Abnormal Procedure, to make it fit the overall K/A.

Explanation: CORRECT B When Reactor water level drops to 0 inches, a Group 2 isolation will occur, closing the RHR valves for the loop that is in Shutdown Cooling. In accordance with 2-AOI-74-1, Loss of Shutdown Cooling, once the Group 2 isolation has been reset, as stated in the stem, the actions that are necessary are covered in step 4.2[14] 2-FCV-74-66 must be closed, 2-FCV-74-67, 2-FCV-74-47, and 2-FCV-74-48 must be opened.

A. **Incorrect because –** 2-FCV-74-67 will be closed due to the PCIS isolation the operator is directed to close 2-FCV-74-66 then open 2-FCV-74-67 so that the Operator can control flow with the 2-FCV-74-66 (throttle valve) when the pump is started.
Plausible if the candidate knows that one of the injection valves closes due to the isolation but does not remember which one.

C. **Incorrect because –** 2-FCV-74-67 will be closed due to the PCIS isolation the operator is directed to close 2-FCV-74-66 then open 2-FCV-74-67 so that the Operator can control flow with 2-FCV-74-66 throttle valve when the pump is started and because 2-FCV-74-47 must also be opened.
Plausible if the candidate knows that one of the injection valves closes due to the isolation but does not remember which one and if the candidate believes only the RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-48 closed on the group 2 isolation. This would be correct for a loss of RPS A only, for instance, with RHR loop II in SDC.

D. **Incorrect because –** 2-FCV-74-48 must also be opened.
Plausible in that the operator is directed to close 2-FCV-74-66 then open 2-FCV-74-67 so that the Operator can control flow with 2-FCV-74-66 (throttle valve) when the pump is started.
2-FCV-74-47 would close and 2-FCV-74-48 would remain open due to a loss of RPS B only, with RHR loop II in SDC.

Technical Reference(s): 2-AOI-74-1 Rev39 and 2-OI-99 Rev81

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.044 R19 OBJ 4.i OPL171.017 R16 OBJ 2.g and 4

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: BFN 1205 Q30
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41 (b)(10)

4.2 Subsequent Actions (continued)

[14] **IF** the Group 2 PCIS Isolation has been reset, **THEN** (Otherwise **N/A**)

RETURN the affected loop of RHR to Shutdown Cooling as follows.

[14.1] **CLOSE** RHR SYS I(II) LPCI OUTBD INJECT VALVE, 2-FCV-74-52(66).

[14.2] **OPEN** RHR SYS I(II) LPCI INBD INJECT VALVE, 2-FCV-74-53(67)

[14.3] **VERIFY** RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 2-HS-74-148(149) in **INHIBIT**

[14.4] **VERIFY CLOSED** RHR SYSTEM I(II) MIN FLOW VALVE, 2-FCV-74-7(30).

[14.5] **VERIFY CLOSED** RHR PUMP 2A(2B) and 2C(2D) SUPPR POOL SUCT VLVs, 2-FCV-74-1(24) and 2-FCV-74-12(35).

[14.6] **VERIFY OPEN** RHR PUMP 2A(2B) and 2C(2D) SD COOLING SUCT VLVs, 2-FCV-74-2(25) and 2-FCV-74-13(36).

[14.7] **OPEN** RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48

[14.8] **IF** the tripped pump has been determined to be in operating condition and with Unit Supervisor permission**, THEN:**

RESTART tripped RHR pump(s) RHR PUMP 2A(2C)(2B)(2D) using 2-HS-74-5A(16A)(28A)(39A)

Illustration 1
(Page 3 of 4)

RPS Bus A or B Power Transfer

B. Loss of power to RPS Bus A only will result in the following events in addition to those listed for RPS Bus A or B power loss:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-74-48	RHR shutdown cooling inboard suction	CLOSES
FCV-74-53	RHR System I inboard injection	CLOSES

Illustration 1
(Page 4 of 4)

RPS Bus A or B Power Transfer

C. Loss of power to RPS Bus B only will result in the following events in addition to those listed for RPS Bus A or B power loss:

VALVE	FUNCTION/SYSTEM	ACTION
FCV-74-47	RHR shutdown cooling outboard suction	CLOSES
FCV-74-67	RHR System II inboard injection	CLOSES

BFN 1205

QUESTION 30

Unit 2 is aligned with RHR Loop I in shutdown cooling and Loop II in standby readiness.

A leak occurs which results in the following conditions:

- Reactor level is at zero inches and slowly lowering
- Drywell Pressure is at 3.0 psig and slowly rising
- RHR pumps A and C have tripped

Which ONE of the following completes the statement below for the **MINIMUM** action(s) required to establish injection with RHR Loop II in accordance with 2-OI-74, RHR System?

After 2-FCV-74-47 or 2-FCV-74-48 closed ____.

NOTE: 2-FCV-74-47, RHR Shutdown Cooling SUCT OUTBD ISOL VLV
2-FCV-74-48, RHR Shutdown Cooling SUCT INBD ISOL VLV

A. place 2-HS-74-67A, RHR SYS II LPCI INBD INJECT VALVE to open

B. depress the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132

C. depress the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132 and then place 2-HS-74-67A, RHR SYS II LPCI INBD INJECT VALVE to open

D. depress the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132, place 2-HS-74-67A, RHR SYS II LPCI INBD INJECT VALVE to open, then place 2-HS-74-66A, RHR SYS II LPCI OUTBD INJECT VALVE to open

QUESTION 2 Rev 0

Units 1, 2, and 3 are operating at 100% power.

Subsequently:

A loss of all off site power occurs.

The following conditions exist:

- The C Diesel Generator is supplying the C 4KV shutdown board.
- The 3EB Diesel Generator is supplying the 3EB 4KV shutdown board.
- All other Diesel Generators **failed to start**.

Assume No Operator Actions Have Been Taken

Which one of the following completes the statement below?

Unit (s) _____ is (are) in a station black out.

A. 1 only

B. 2 only

C. 1 and 3 only

D. 2 and 3 only

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295003AK1.06	
	Importance Rating	3.8	4.0

Partial or Complete Loss of A.C. Power; Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: (CFR: 41.8 to 41.10) Station blackout:

Justification for K/A match: To meet the first part of the K/A the question sets up a loss of A.C. power, and then asks the operators to determine which of the Units are technically a Station Blackout. This meets the second part asking the operational implication of that loss of AC Power. The implication is that some of the Units are going to be in a Station Blackout and have to perform the actions in accordance with that condition.

Explanation: CORRECT D: With only the C and 3EB D/G supplying power to their 4KV SD boards and no operator actions taken Unit 2 and Unit 3 480V SD boards are de-energized placing them in a station black out. The 1B 480V SD board would have power therefore unit 1 is not in a station black out.

A. **Incorrect because –** 1B 480V SD board is energized by the C D/G.
Plausible if the candidate thinks that 4KV Shutdown boards A and B supply 480V Shutdown boards1A and 1B, 4KV Shutdown boards 3A and 3B supply 480V Shutdown boards 3A and 3B, and that 4KV Shutdown boards C and D supply 480V Shutdown boards 2A and 2B.

B. **Incorrect because –** 3A and 3B 480V SD boards are also de-energized.
Plausible if that candidate knows the U1 and U2 distribution but thinks that 4KV Shutdown boards 3A and 3B supply 480V Shutdown boards 3A and 3B.

C. **Incorrect because –** 1B 480V SD board is energized by the C D/G.
Plausible if the candidate thinks that 4KV Shutdown boards A and B supply 480V Shutdown boards1A and 1B, but knows that 4KV Shutdown boards 3A and 3C supply 480V Shutdown boards 3A and 3B.

Note: B 4KV Shutdown board feeds the 1C and 2C RHR and CS pumps while the C 4KV Shutdown board feeds the 1B and 2B RHR and CS pumps. This divisional arrangement adds to the above plausibility.

Technical Reference(s): 0-AOI-57-1A Rev 97

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.036 ILT objective 6.f, 8f, 13

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: 2010 Brunswick Q38
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(10)

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, and operator actions for a loss of all offsite power, and for a station blackout (SBO). Station Blackout (SBO) is defined as a loss of 161 and 500kV systems and a failure of the two diesel generators which supply normal power to the two 480V Shutdown Boards on a unit. Exiting the SBO can occur through Cross-connect capabilities as long as it does not place the Non-SBO unit in jeopardy. Analysis takes credit for only one unit being in an SBO Event. The actions in this instruction are also applicable to a loss of all AC power. [NER/C] This instruction provides procedural guidance for restoring offsite power. [SOER 90-1]

Brunswick 2010

38. 295003 A2.05 001

Both Units were operating at rated power when ALL switchyard PCB position indications turn green.

Diesel Generator status:

DG1	Running loaded
DG2	Under clearance
DG3	Running loaded
DG4	Tripped on low lube oil pressure

Which one of the following identifies the AOP(s) that Unit One and Unit Two are required to perform?

Unit One is required to perform __(1)__.
Unit Two is required to perform __(2)__.

A. (1) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses
(2) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

B. (1) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses
(2) 0AOP-36.2, Station Blackout

C. (1) 0AOP-36.2, Station Blackout
(2) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

D. (1) 0AOP-36.2, Station Blackout
(2) 0AOP-36.2, Station Blackout

QUESTION 3 Rev 0

Which one of the following completes the statements below concerning the 250 VDC Unit batteries and battery chargers?

The Class 1E Unit Batteries have the capacity to compensate for a __ (1) __ Station Blackout event during multi-unit operations **Without Operator action.**

In accordance with 1/2-AOI-57-1D, 480V Load Shed, if the load shed logic can **NOT** be reset the 2A 250V Battery charger may be returned to service by placing the charger select switch in __ (2) __.

A. (1) 4 hour
   (2) OFF then back to ON

B. (1) 12 hour
   (2) OFF then back to ON

C. (1) 4 hour
   (2) EMERG

D. (1) 12 hour
   (2) EMERG

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295004 AK2.01	
	Importance Rating	3.1	3.1

Partial or Complete Loss of D.C. Power; Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: (CFR: 41.7 / 45.8) Battery charger

Justification for K/A match: To match the first condition of being in a partial or complete loss of DC power, a load shed is initiated during which the battery charger will lose power. The procedure 1/2-AOI-57-1D, 480V Load Shed, provides the interrelationship aspect concerning which charger goes with which battery and when the A/C source must be reconnected to the charger.

Explanation: CORRECT C: Part 1 – According to OPL171.037 Batteries 5 and 6 were added to increase the capacity for the Class 1E Unit Batteries to compensate for a 4 hour Station Blackout (SBO) event during multi-unit operations. Part 2 – In accordance with 1/2-AOI-57-1D if load shed logic cannot be reset **PLACE** Battery Charger EMER/OFF/ON Select Switch in EMERG.

A. **Incorrect because –** The switch must be placed in EMERG to energize the charger. **Plausible** in that many of the breaker schemes at BFN require taking the control switch to off to reset the anti pump coil prior to restarting a component.

B. **Incorrect because –** BFN has a 4 hour coping time without operator action and the switch must be placed in EMERG to energize the charger.
Plausible because extending the battery capacity to 12 hours is a strategy during an extended loss of AC power however, this requires manual operator actions. Part 2 is plausible see A above.

D. **Incorrect because –** Part 1 incorrect see B above. Part 2 correct.
Plausible because – Part 1 see B above and Part 2 is correct.

Technical Reference(s): OPL 171.037 Rev 13, 0-OI-57D Rev 152, 1/2-AOI-57-1D Rev 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.037 objective 1 OPL171.072 objective 4 & 5

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: NA
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(10)

B. Systems

1. The 250V Unit and Plant DC System

a. Description
This DC system consists of six batteries, their chargers, and the associated circuitry, switches, distribution panels, indicators, and alarms. There is one battery and one charger per Unit for Class 1E safety related systems. One Plant/Station battery and charger per unit for Class Non-1E systems and components and one common spare charger (2B). Batteries 5 and 6 were added to increase the capacity for the Class 1E Unit Batteries to compensate for a 4-hour Station Blackout (SBO) event during multi-unit operations. Most Class Non-1E loads have been removed from the unit batteries and loaded on the plant/station batteries 4, 5, and 6. The DC System provides motive power for DC motor operated valves, pumps, and control/logic power for ECCS.

Note: class 1E is safety related electrical

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0152 Page 16 of 303

3.0 **PRECAUTIONS AND LIMITATIONS** (continued)

F. 250V Unit Battery Charger 1, 2A, 2B and 3 Emergency ON select switch bypasses battery charger emergency load shed contacts. Placing the select switch in Emergency ON reestablishes charger operations with an accident signal present and Diesel Generator voltage available.

4.2 Subsequent Actions (continued)

[14] **PERFORM** the following for the 250V Unit Batteries:

- **MONITOR** Batt Bd amps and VOLTS on Panel 1(2)-9-8.
- **PLACE** the battery charger back in service within 30 minutes after loss of the charger to the battery.

[15] **IF** load shed logic can **NOT** be reset and operation of the battery charger is required, **THEN** (Otherwise N/A)

[15.1] **PLACE** Battery Charger EMER/OFF/ON Select Switch in EMERG.

[15.2] **WHEN** load shed logic is reset, **THEN PLACE** Battery Charger EMERG/OFF/ON Select Switch in ON.

480V LOAD SHEDDING LOGIC SYSTEM OPL171.072R12 page 15

D. Loads which are locked out until the accident signal is clear and reset or "DGVA" resets

f. Battery Chargers 1, 2a, 2b, 3 - Each charger has a emergency on switch which bypasses load shed contacts

QUESTION 4 Rev 0

Which one of the following completes the statement below?

In the event of a Main Turbine trip without bypass valves from full power, a Reactor Scram is initiated to anticipate the __ 1 __ and to prevent exceeding the __ 2 __ safety limit.

A. (1) rapid reduction in Reactor water level
   (2) Reactor water level

B. (1) rapid reduction in Reactor water level
   (2) MCPR

C. (1) rapid increase in Reactor pressure and Neutron flux
   (2) Reactor water level

D. (1) rapid increase in Reactor pressure and Neutron flux
   (2) MCPR

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295005 AK3.01	
	Importance Rating	3.8	3.8

Main Turbine Generator Trip; Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.5 / 45.6) Reactor SCRAM

Justification for K/A match: This is one of the analyzed transients identified for BWRs in the UFSR. So that meets the requirement for being in Tier 1, Knowledge of the reasons for the Reactor SCRAM, this is met when asking which Safety Limit it is there to prevent.

Explanation: CORRECT D: In accordance with OPL171.028, Reactor Protection System, the Turbine Stop Valves 10 percent closure anticipates the pressure and neutron flux rise caused by the rapid closure of the Turbine Stop Valves (Turbine Trip). In accordance with OPL171.222, Transient Analysis, for Event #4-Main Turbine Trip without Bypass Valves the Most limiting failure mechanism is Over-pressurization of the RPV and fuel damage caused by a change in Critical Power Ratio resulting from a rapid reduction in Void Fraction.

A. **Incorrect because –** The scram is initiated to anticipate the rapid increase in Reactor pressure and Neutron flux and prevents exceeding the MCPR safety limit.
Plausible in that indicated Reactor water level does initially drop rapidly on a turbine trip see OPL171.222, Transient Analysis event 4 curve showing post accident range indicated level lowering to -120 inches. Part 2 Plausible in that there is a Reactor water level safety limit at top of active fuel which is -162 inches.

B. **Incorrect because –** The scram is initiated to anticipate the rapid increase in Reactor pressure and Neutron flux.
Plausible because – Part 1 see A above and Part 2 correct.

C. **Incorrect because –** The scram prevents exceeding the MCPR safety limit.
Plausible because – Part 1 is correct. See A above for Part 2.

Technical Reference(s): OPL 171.222 Rev 4 , OPL 171.028 Rev 19

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.222 OBJ 4 OPL 171.028 OBJ 13.c and 16

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)(5)

2. Turbine Stop Valves, 10 percent closure anticipates the pressure and neutron flux rise caused by the rapid closure of the Turbine Stop Valves.

Event #4-Main Turbine Trip without Bypass Valves

Transient description: This event is initiated by a Turbine Trip with a failure of Bypass Valves to operate.
FSAR Assumption differences: The FSAR transient was initiated from 100% power and 105% Core 'flow. NOTE: This is the most limiting abnormal operating transient. The assumptions are very close to actual plant conditions.
Most limiting failure mechanism: Overpressurization of the RPV and fuel damage caused by a change in Critical Power Ratio resulting from a rapid reduction in Void Fraction.
Event Diagnosis Key Parameter Changes:

1. **Power Level-**Initially spikes very high until the scram on TCV Closure terminates the rapid increase.

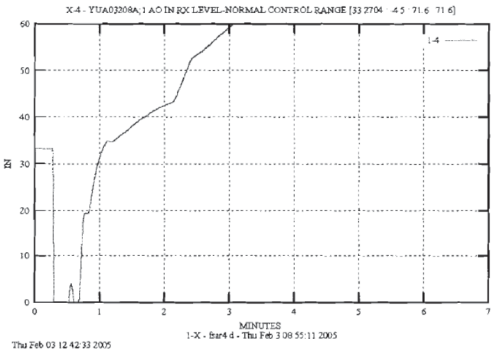
2. **Total Steam Flow-**Drops to almost zero as the TCVs and TSVs close. Only auxiliary steam loads remain.

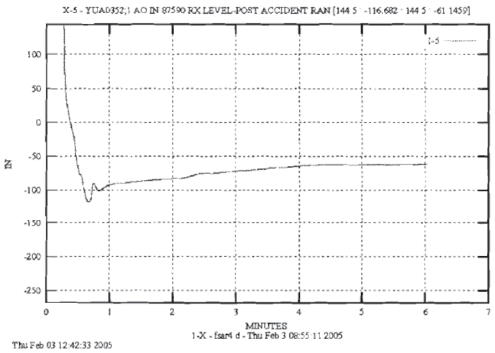
3. **Main Generator Megawatts-**Immediately drops to zero due to generator trip.

4. **Reactor Pressure-**Rapidly rises to MSRV setpoint and remains high due to decay heat. Pressure oscillates around the MSRV lift setpoint.

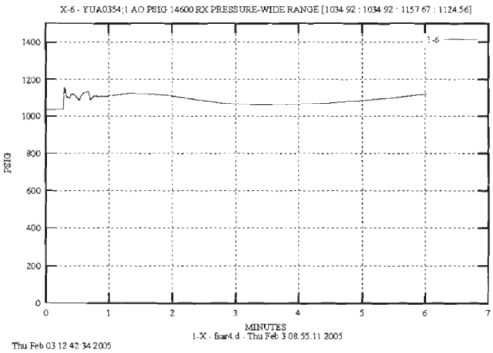
5. **Reactor Water Level-**Drops initially due to void collapse as water in the downcomer is directed into the active fuel region. Reactor Feed Pumps restore level. Small spikes in the level trend are due to lifting MSRVs.

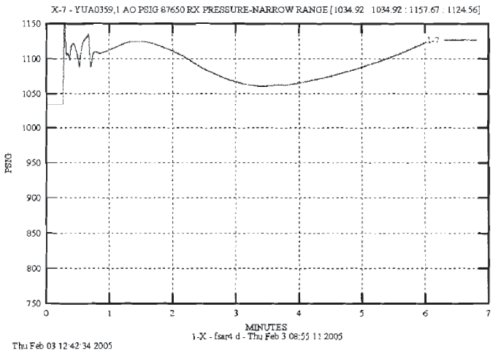
Main Turbine Trip Without Bypass Valves





Event - 4





QUESTION 5 Rev 0

Unit 1 is operating at 100% power when the B RPS MG set output breaker trips open causing a half scram.

Which one of the following describes required actions to place 1B RPS on alternate power in accordance with 1-AOI-99-1?

A. Verify Circuit Protector 1B1 and 1B2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 1

B. Verify Circuit Protector 1B1 and 1B2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 2

C. Verify Circuit Protector 1C1 and 1C2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 1

D. Verify Circuit Protector 1C1 and 1C2 are Reset,
   Place the RPS bus 1B normal/alt transfer switch to ALT in Battery Board Rm 2

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295006 AA1.01	
	Importance Rating	4.2	4.2

SCRAM; Ability to operate and/or monitor the following as they apply to SCRAM: (CFR: 41.7 / 45.6) RPS

Justification for K/A match: Loss of an RPS Bus will cause a ½ Scram and entry into 1-AOI-99-1. To meet the part of the K/A that talks about the ability to operate or monitor RPS, the question relies on the students knowledge of the RPS system and the mitigating steps in the AOI to restore power to the bus, meeting the operate portion of the K/A.

Explanation: CORRECT C: In accordance with 1-AOI-99-1 Loss of Power to one RPS bus, the following actions are taken prior to transferring RPS B to alternate power:
The memory lights inside1C1 and 1C2 Circuit Protectors are verified reset, the 1C1 and 1C2 Circuit Protectors are verified reset, then RPS B is transferred to alternate using the 1B NORM/ALT switch in Battery Board room 1.

A. **Incorrect because –** The alternate power supply uses the 1C1 and 1C2 circuit protectors.
Plausible that the candidate would think that the 1B1 and 1B2 Circuit Protectors remain in the B RPS circuit while on alternate power.

B. **Incorrect because –** The alternate power supply uses the 1C1 and 1C2 circuit protectors and the transfer is done in battery board room 1.
Plausible see A above and because the Unit 1 120VAC Unit Preferred system is transferred to alternate power in Battery Board room 2 (see 1-AOI-57-4)

D. **Incorrect because –** The transfer is done in battery board room 1.
Plausible because - Part 1 is correct and the Unit 1 120VAC Unit Preferred system is transferred to alternate power in Battery Board room 2 (see 1-AOI-57-4).

Technical Reference(s): 1-AOI-99-1 Rev 22 and 1-AOI-57-4 Rev 31

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.028 R 19 ILT OBJ 10

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC: N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(7)	

4.2 Subsequent Actions (continued)

[8] **RESTORE** Alternate Power to RPS Bus B as follows:

[8.1] **VERIFY** memory lights inside the RPS Circuit Protector cabinet are RESET:
- RPS CKT PROTECTOR RESET PB CAB 1C1, 1-HS-099-0001C1
- RPS CKT PROTECTOR RESET PB CAB 1C2, 1-HS-099-0001C2

[8.2] **VERIFY** Circuit Protectors 1C1 and 1C2 are RESET:
- RPS CIRCUIT PROTECTOR 1C1 RESET, 1-HS-099-0001C1/1
- RPS CIRCUIT PROTECTOR 1C2 RESET, 1-HS-099-0001C2/1

[8.3] **VERIFY** ALTERNATE PWR AVAILABLE TO RPS BUS 1B, 1-IL-099-0001BD, light is ILLUMINATED at RPS Bus 1B MG Control Panel.

[8.4] **PLACE** RPS BUS 1B NORM/ALT TRANSFER SWITCH, 1-XS-099-0001B, to ALT, in Battery Board Rm 1.

[8.5] **NOTIFY** Unit 1 Operator that RPS 1B is on alternate power supply.

4.2 Subsequent Actions (continued)

[7] **DISPATCH** personnel to determine status of UNIT PFD SYSTEM INVERTER AND RECTIFIER, 1-INV-252-0001. ☐

NOTE
Upon reenergization of the Unit Preferred Bus (Battery Board 1 Panel 11) Panel 1-9-9 Cabinet 6 Unit Preferred will auto-transfer back to the normal source.

[8] **IF** ALTERNATE SOURCE AVAILABLE (P13), 1-IL-252-0001B, is illuminated, **THEN SWAP** to Alternate Source supply to Unit Preferred (Battery Board 1 Panel 11) as follows:

[8.1] **CLOSE** ALT SOURCE AC OUTPUT (B5), 1-BKR-252-0001B.

[8.2] **DISPATCH** personnel to Battery Board 2 Sync and Speed Control Panel (Battery Board 2 Panel 11) and **PERFORM** the following:

[8.2.1] **PLACE** UNIT 1 PFD SYSTEM TRANSFORMER SOURCE SYNC SW SS-2, 1-HS-252-01/SS-2 in ON.

[8.2.2] **OPEN** UNIT 1 BKR 1001, 0-HS-280-001/1001.

[8.2.3] **CLOSE** UNIT 1 BKR 1002, 0-HS-280-001/1002.

[8.2.4] **PLACE** UNIT 1 PFD SYSTEM TRANSFORMER SOURCE SYNC SW SS-2, 1-HS-252-01/SS-2 in OFF.

QUESTION 6 Rev 0

The Shift Manager has directed entering 3-AOI-100-2, Control Room Abandonment, due to heavy smoke in the U3 MCR.

Which one of the following completes the statements below concerning 3-AOI-100-2, Control Room Abandonment?

The immediate actions are taken to SCRAM the reactor and to place the unit in the most stable configuration possible to __ (1) __.

If the Reactor fails to scram when the immediate actions are performed, you are to continue in 3-AOI-100-2, because the subsequent actions will direct __ (2) __.

A. (1) allow time to prepare for plant cooldown
   (2) initiating ARI

B. (1) allow time to prepare for plant cooldown
   (2) pulling RPS Scram Solenoid Fuses

C. (1) limit the heat load on CCW
   (2) initiating ARI

D. (1) limit the heat load on CCW
   (2) pulling RPS Scram Solenoid Fuses

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295016AK3.01	
	Importance Rating	4.1	4.2

Control Room Abandonment; Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: (CFR: 41.5 / 45.6) Reactor SCRAM

Justification for K/A match: 3-AOI-100-2 is the control room abandonment abnormal, so the first part is met. The first part of the question meets the second part of the K/A, Knowledge of the reason for scramming, A single part question could not be written on just the reason for scramming the reactor when you leave the control room, so a second part was added to allow construction of a two by two question. It is related, but does not match the K/A, but is related to the control room abandonment procedure.

Explanation: Correct B: Part 1- OPL 171.208, CONTROL ROOM ABANDONMENT states:
The immediate actions are taken to scram the reactor and to place the unit in the most stable configuration possible to allow time to prepare for plant cooldown.
Part 2 - NOTE on 3-AOI-100-2 page 6 states:
If rods fail to insert or scram solenoids fail to deenergize in Steps 4.1[4] and 4.1[5], then Step 4.2[1] will pull RPS Scram Solenoid Fuses.

A. **Incorrect because –** 3-AOI-100-2 does not direct initiating ARI.
Plausible because Part 1 is correct and initiating ARI is the expected action for a scram failure when not in 3-AOI-100-2

C. **Incorrect because –** The heat load will be directed to the torus and removed by RHRSW not CCW and 3-AOI-100-2 does not direct initiating ARI.
Plausible because the CCW system normally rejects heat to the heat sink and will remain in service. Part 2 is plausible see A above.

D. **Incorrect because –** The heat load will be directed to the torus and removed by RHRSW not CCW.
Plausible because the CCW system normally rejects heat to the heat sink and will remain in service and because Part 2 is correct.

Technical Reference(s): 3-AOI-100-2 Rev 22

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.208 NLO OBJ 2 ILT OBJ 5

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(10)	

Lesson Plan Content

II. Presentation

A. Control Room Abandonment AOI-100-2 Procedure Overview

2. The reactor is shut down by manual scram from the Control Room. Water level is maintained using the Reactor Core Isolation Cooling (RCIC) System, controlled from Panel 25-32. After the MSIVs are closed, a plant cooldown at normal rates is conducted using four ADS valves located at 25-32. One loop of RHR is placed in Suppression Pool Cooling.

a.When the reactor has been depressurized, RCIC is shut down and the RHR subsystem is placed in Shutdown Cooling.

b. The immediate actions are taken to scram the reactor and to place the unit in the most stable configuration possible to allow time to prepare for plant cooldown. Before leaving the Control Room, the following actions must be taken:
1) If core flow is >60% then reduce core flow to between 50-60%. Obj.ILT/LOR-1
2) Manually scram the reactor at Panel 9-5.
3) Place the reactor Mode Switch in SHUTDOWN.

4.0 OPERATOR ACTIONS

4.1 Immediate Action

NOTES

1) The immediate action to "DEPRESS REACTOR SCRAM A and B pushbuttons" is required to be completed prior to evacuating the control room.
2) Steps should be performed in order, however, Steps 4.1[7], 4.1[10], 4.1[11], and 4.1[12] may be performed at anytime while performing the immediate actions.

[1] **IF** core flow is above 60%, **THEN:** (Otherwise N/A)

LOWER core flow to between 50-60%. ☐

[2] **DEPRESS** REACTOR SCRAM A and B pushbuttons. ☐

[3] **PLACE** REACTOR MODE SWITCH in SHUTDOWN. ☐

NOTE

If rods fail to insert or scram solenoids fail to deenergize in Steps 4.1[4] and 4.1[5], then Step 4.2[1] will pull RPS Scram Solenoid Fuses.

4.2 Unit 3 Subsequent Actions

[1] **IF** ALL control rods were **NOT** fully inserted **AND** RPS failed to deenergize, **THEN:**(Otherwise N/A)

DIRECT an operator to Unit 3 Auxiliary Instrument Room to perform Attachment 9. ☐

Attachment 9
(Page 1 of 3)

Removal and Replacement of RPS Scram Solenoid Fuses

1.0 CONTROL BAY - UNIT 3 AUXILIARY INSTRUMENT ROOM - EL 593'

[1] **OBTAIN** fuse pullers. ☐

[2] **LOCATE** Terminal Strip CC inside Panel 3-9-15, Bay 2 (Rear). ☐

[3] **REMOVE** the following fuses (located at bottom of terminal strip CC):

RPS BUS A (Panel 3-9-15, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	REMOVED
CC	FOUR (4)	3-FU1-085-0037AA	☐
CC	FIVE (5)	3-FU1-085-0039A/2	☐
CC	SIX (6)	3-FU1-085-0039A/3	☐
CC	SEVEN (7)	3-FU1-085-0039A/4	☐

[4] **LOCATE** Terminal Strip CC inside Panel 3-9-17, Bay 2 (Rear). ☐

[5] **REMOVE** the following fuses (located at bottom of terminal strip CC): ☐

RPS BUS B (Panel 3-9-17, Bay 2 - Rear)

BLOCK	NUMBER	FUSE ID	REMOVED
CC	FOUR (4)	3-FU1-085-0037BA	☐
CC	FIVE (5)	3-FU1-085-0039B/2	☐
CC	SIX (6)	3-FU1-085-0039B/3	☐
CC	SEVEN (7)	3-FU1-085-0039B/4	☐

[6] **NOTIFY** UO at Panel 3-25-32 upon completion of fuse removal. ☐

QUESTION 7 Rev 1

Unit 1 is operating 100% power, when an RBCCW leak develops causing the 1-FCV-70-48 RBCCW Sectionalizing valve to close, isolating the leak.

Which one of the following components has **NOT** lost cooling water?

A. Drywell equipment drain sump

B. Fuel pool cooling heat exchangers

C. Reactor water cleanup pump seal coolers

D. RWCU Non-regenerative heat exchangers

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295018 AK2.01	
	Importance Rating	3.3	3.4

Partial or Complete Loss of Component Cooling Water; Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: (CFR: 41.7 / 45.8) System loads

Justification for K/A match: Partial RBCCW loss meets the first part of the K/A. The second part asks the interrelation of that loss on systems loads. When RBCCW Sectionalizing valve is closes, due to low pressure, the non-critical loads are shed from RBCCW cooling. That meets the second part of the K/A.

Explanation: CORRECT A: When 1-FCV-70-48, RBCCW Sectionalizing valve, is closed it isolates the Non-essential loop loads. The loads listed in answer B, C, and D are Non-essential loop loads and therefore have lost cooling. The Drywell equipment drain sump has not lost cooling.

B. **Incorrect because –** this is a Non-essential loop load.
Plausible because of the attention FPC has gotten since the Fukushima events that the candidate may think FPC is on the essential loop.

C. **Incorrect because –** this is a Non-essential loop load.
Plausible because the Recirc pumps seal coolers have not lost cooling.

D. **Incorrect because –** this is a Non-essential loop load.
Plausible because the loss of cooling to the RWCU Non-regenerative heat exchangers will cause RWCU to isolate.

Technical Reference(s): OPL 171.047 Rev 12,

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.047 R12 OBJ 2 and 3

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:	41(b)(4)

2. RBCCW Heat Loads

a. Essential loop loads
- 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
- 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 Rev 0

A rupture in the control air header has occurred.

- Control air pressure indicates 25 psig and lowering in the U3 Control Room.
- 3-AOI-32-2, Loss of Control Air has been entered.
- Several U3 Control Rods failed to insert during the transient.
- The US directs inserting Control Rods in accordance with 3-EOI Appendix-1D

Which one of the following completes the statement below?

In order to insert Control Rods the Unit Operator is required to dispatch personnel to manually _____.

A. **open** the 3-FCV-85-11A, CRD Flow Control Valve, and **open** the 3-PCV-85-23, CRD Drive Water Pressure Control Valve

B. **close** the 3-FCV-85-11A, CRD Flow Control Valve, and **open** the 3-PCV-85-23, CRD Drive Water Pressure Control Valve

C. **open** the 3-FCV-85-11A, CRD Flow Control Valve, and **close** the 3-HCV-85-586, Charging Water SOV

D. **close** the 3-FCV-85-11A, CRD Flow Control Valve, and **close** the 3-HCV-85-586, Charging Water SOV

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295019 AA2.02	
	Importance Rating	3.6	3.7

Partial or Complete Loss of Instrument Air; Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: (CFR: 41.10) Status of safety-related instrument air system loads

Justification for K/A match: The questions sets up a partial loss of instrument air (control air) to match the first part of the K/A. The second part is covered when asking about what has to be done to compensate for how some Control Rod Drive valves fail on a loss of instrument air. This is a safety related load for the instrument air system.

Explanation: CORRECT C: On loss of control air 3-FCV-85-11A(B), CRD Flow Control Valve will fail closed and CRD system flow will be through the charging water header. IAW 3-AOI-32-2, Loss of Control Air step 4.2[7] if Drive water pressure is require to insert control rods manually open 3-FCV-85-11A(B) and (since the scram cannot be reset) throttle/close 3-HCV-85-586 charging water SOV.

A. **Incorrect because –** 3-PCV-85-23 is downstream of the drive water header so opening the valve would lower D/P and adjusting the valve would not help unless 3-HCV-85-586 is closed.
Plausible in that opening 3-FCV-85-11A(B) is correct and 3-PCV-85-23 is normally adjusted IAW 3-OI-85 to control drive water D/P.

B. **Incorrect because –** 3-AOI-32-2 directs opening 3-FCV-85-11A(B)
Plausible in that 3-FCV-85-11A is an air operated valve that could fail open and does have and installed hand wheel/dogging device. 3-PCV-85-23 is adjusted by 3-OI-85 to control drive water D/P.

D. **Incorrect because –** 3-AOI-32-2 directs opening 3-FCV-85-11A(B)
Plausible because - 3-FCV-85-11A is an air operated valve that could fail open and does have and installed hand wheel/dogging device and closing 3-HCV-85-586 is correct.

Technical Reference(s): 3-AOI-32-2 Rev 23; 3-EOI Appendix 1D Rev 3;
3-OI-85 Rev 82.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.005 R20 OBJ 28G, 171.054 R16 OBJ B.7

Question Source:	Bank:	
	Modified Bank:	X
	New:	
Question History:	Previous NRC: 1102 Q33	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(7)	

OPL171.054, Control and Service Air (Includes DWCA System), Rev# 16

9. COMPONENT FAILURE MODES

a. Feedwater startup bypass valve fails as is at 65 psig

b. Temperature control valves for RBCCW The ten-inch valves fails closed,
The six-inch valve opens to a predetermined position

c. Temperature control valves for Main Turbine Lube Oil Coolers The 4 inch valve fails closed
The 8 inch valve fails open

d. Inboard MSIVs fail closed and the operating medium for the MSRVs to be lost

e. Reactor scrams because scram valves fail open

f. Outboard MSIVs fail closed

g. CRD flow control valve fails closed (manually open)

BFN Unit 3	Loss Of Control Air	3-AOI-32-2 Rev. 0023 Page 8 of 24

4.2 Subsequent Actions (continued)

[7] **IF** CRD drive water pressure is required to insert control rods which failed to scram, **THEN PERFORM** the following: (Otherwise N/A)

[7.1] **MANUALLY OPEN** CRD Flow Control Valve, 3-FCV-85-11A or B

[7.2] **THROTTLE/CLOSE** charging water with 3-HCV-85-586 to establish drive water pressure

BFN Unit 3	Insert Control Rods Using Reactor Manual Control System	3-EOI Appendix-1D Rev. 0003 Page 3 of 6

1.0 INSTRUCTIONS

[2] **IF** Reactor Scram or ARI CANNOT be reset, **THEN DISPATCH** personnel to close 3-SHV-085-0586, CHARGING WATER SOV (RB NE, El 565 ft).

Modified from BFN 11-02 NRC Exam

Proposed Question: **# 33**

Unit 2 was operating at 100% Reactor Power, when the plant experienced a complete loss of the Control Air system. The following plant conditions exist:

- **ALL** eight Scram Solenoid Group A/B Logic Reset Lights are **NOT** lit
- Recirc Pumps are Tripped
- Reactor Power is 20%

You are the OATC and have been directed to perform 2-EOI Appendix 1D, "Insert Control Rods Using Reactor Manual Control System" (RMCS).

Based on the above conditions which ONE of the following responses contains the correct steps to manually insert **AND** determine when the control rods are inserted?

Verify CRD Pump operating, **____(1)____** , direct manually opening CRD Flow Control Valve

(2-FCV-85-11A or B), verify Mode Switch in SHUTDOWN, bypass the Rod Worth Minimizer, CRD Power Switch ON, select control rod, **AND** place CRD **____(2)____**.

A. **(1)** reset ARI

(2) Control Switch in ROD IN, until green 00 is lit, on the four rod display

B. **(1)** reset ARI

(2) Notch Override Switch in EMERG IN, until the control rod stops moving inward

C. **(1)** direct closure of CHARGING WATER SHUTOFF, 2-SHV-85-586

(2) Control Switch in ROD IN, until the green 00 is lit, on the four rod display

D. **(1)** direct closure of CHARGING WATER SHUTOFF, 2-SHV-85-586

(2) Notch Override Switch in EMERG IN, until the control rod stops moving inward

Proposed Answer: **D**

QUESTION 9 Rev 2

Unit 2 is in day 2 of a forced outage with the following conditions:

- Currently in Mode 4
- Moderator Temperature band is 150° F to 180° F
- Both Reactor Recirc pumps are OFF with suction valves open and discharge valves closed
- RHR pump 2A is in Shutdown Cooling

Subsequently:

The 2B Recirc Pump discharge valve is inadvertently opened and **fails to close**.

Which one of the following completes the statements below 4 hours after the 2B Recirc Pump discharge valve opened?

ASSUME no other operator actions are taken.

The RHR outlet temperature from the 2A RHR Heat Exchanger will __ (1) __ and actual moderator temperature will __ (2) __.

A. (1) Remain the same
   (2) Remain the same

B. (1) Lower
   (2) Lower

C. (1) Rise
   (2) Rise

D. (1) Lower
   (2) Rise

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295021 AK2.07	
	Importance Rating	3.1	3.2

Loss of Shutdown Cooling; Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: (CFR: 41.7 / 45.8) Reactor recirculation

Justification for K/A match: To match the first part of the K/A the conditions in the question set up the unit in cold shutdown with RHR in Shutdown Cooling. To match the second part of the K/A, the interrelationship with Reactor Recirculation(RR), the RR pump's discharge valve is opened and will not reclose, diverting RHR flow back through the RR pump, bypassing the core, therefore a loss of shutdown cooling.

Explanation: Correct D: When the 2B Recirc Pump discharge valve opens and fails to close, a Reactor core bypass is established. IAW 2-OI-74 8.8.1[30.7] this would result in Shutdown cooling being out of service. With SD Cooling flow bypassing the Reactor Core, RHR HX 2A RHR outlet temperature will lower due to a reduced heat load and the Core moderator temperature will rise due to reduced cooling flow through the core.

A. **Incorrect because** – Neither temperature will remain the same.
Plausible if the candidate does not understand that this will allow shutdown cooling flow to bypass the Reactor core which has happened in the industry.

B. **Incorrect because** – Moderator temperature will rise.
Plausible if the candidate assumes that the volume of water from the un-isolated Recirc loop and associated ambient heat losses will cool both the Moderator temperature and the SD Cooling loop.

C. **Incorrect because** – RHR temperatures will lower due to reduced heat load.
Plausible if the candidate assumes that the volume of water from the un-isolated Recirc loop will increase the heat load on RHR and the Moderator temperature will rise.

Technical Reference(s): 2-OI-74 Rev 171, 2-SR-3.4.9.5-7 Rev 8

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.0044 R19 OBJ 11d,12.g

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Hatch 2007 Q51
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(3)

8.8.1 Initiation / Operation of RHR Loop I in Shutdown Cooling (continued)

[30.7] **PLACE** CAUTION ORDER on Recirc Suction and Discharge valves of both loops stating "When Recirc Loop is **NOT** in service, EITHER the suction OR the discharge valve SHALL remain closed unless taking exception on SDC LCO for removal of SDC for up to 2 hours in an 8 hour period." (REFER TO Technical Specification 3.4.8 Note 1.) [OE 7481, SER 95-025]

8.8.1 Initiation / Operation of RHR Loop I in Shutdown Cooling (continued)

CAUTIONS

1) When the reactor is in Mode 4 or Mode 5, failure to maintain ONE of the following could result in thermal stratification:

- Shutdown Cooling mode with reactor water level greater than 70 inches on 2-LI-3-55, **OR**:
- A Reactor Recirc Pump running, **OR**:
- Shutdown Cooling flow greater than 7,000 gpm (With the cavity flooded, maintain a shutdown cooling flow of at least 6000 gpm)

NOTES

4) Reactor water level should be maintained greater than 70" any time SDC is in operation, when possible. Exceptions to this level band are allowed to perform approved procedures (i.e., flushing, testing, floodup, etc.), provided Shutdown Cooling flow is maintained greater than 7,000 gpm, when level is less than 70", to prevent thermal stratification.

3.4 Shutdown Cooling (continued)

E. [NER/C] When the reactor is in cold shutdown (MODE 4 or Mode 5), adequate flow up through the core and into the downcomer region is required to ensure thermal stratification does not occur. With inadequate flow, stratification can occur and cause erroneous temperature indications. To provide adequate mixing:

1. Shutdown cooling flow is required to be maintained with reactor water level between 70 inches and 90 inches on 2-LI-3-55, This level band should be maintained any time SDC is in operation, when possible. Exceptions to this level band are allowed to perform approved procedures (i.e. flushing, testing, floodup, etc.), provided Shutdown Cooling flow is maintained greater than 7,000 gpm, when level is less than 70 inches, to prevent thermal stratification.

OR

2. Maintain a Recirculation pump running. [SER 95-025]

G. [NER/C] To ensure thermal stratification does not occur, the following flow requirements apply during Shutdown Cooling operation with no Recirculation pumps running: [SER 95-025]], (Q22462A)

1. With the reactor vessel head installed, maintain shutdown cooling flow of at least 7,000 - 10,000 gpm.

2. With the reactor cavity flooded, maintain shutdown cooling flow of at least 6,000 gpm.

Modified from Hatch 2007 NRC exam

51. 295021AK2.07 001

Unit 1 is in Mode 4 preparing for startup after a forced mid-cycle shutdown with the following conditions:

- RHR Loop "B" is in Shutdown Cooling w/ 7900 gpm
- Both Recirc Pumps OFF w/ discharge valves closed and suction valves open
- 1E41-R605, RHR Water Temp on Panel 1H11-P614 is 185°F
- RWCU Inlet temperature is 187°F
- RPV level is 37"

As the operator was performing a surveillance on the "1B" Recirc Pump discharge valve, the valve opened as required but would not re-close. The auxiliary operator is currently investigating the valve motor breaker.

Given these plant conditions, which ONE of the following describes how this valve being open will affect RWCU and RHR temperatures?

Actual core coolant temperature will _____.
RHR heat exchanger inlet water temperature will _____.

A. rise / lower

B. remain the same / lower

C. rise / rise

D. remain the same / rise

Note: On Unit 1, RHR shutdown cooling suction comes from the "B" Recirc loop. (different on Unit 2).

A. Correct.

B. Incorrect because core is not receiving any forced circulation; given this decay heat load, the actual core coolant temperature will rise. Plausible because applicant may not recognize core bypass conditions because shutdown cooling remains in service.

C. Incorrect because RHR inlet water temperature will lower because this water is no longer circulating around hot fuel. Plausible if applicant knows core temperature is rising.

D. Incorrect because core is not receiving any forced circulation; given this decay heat load, the actual core coolant temperature will rise. Also incorrect because RHR inlet water temperature will lower because this water is no longer circulating around hot fuel. Plausible if applicant thinks that the RHR pump is dead headed.

QUESTION 10 Rev 0

A Refueling Accident has occurred on U2 resulting in the following indications:

(See next page for picture)
What is the Refueling Zone Detector B reading and is it above or below the trip setpoints?

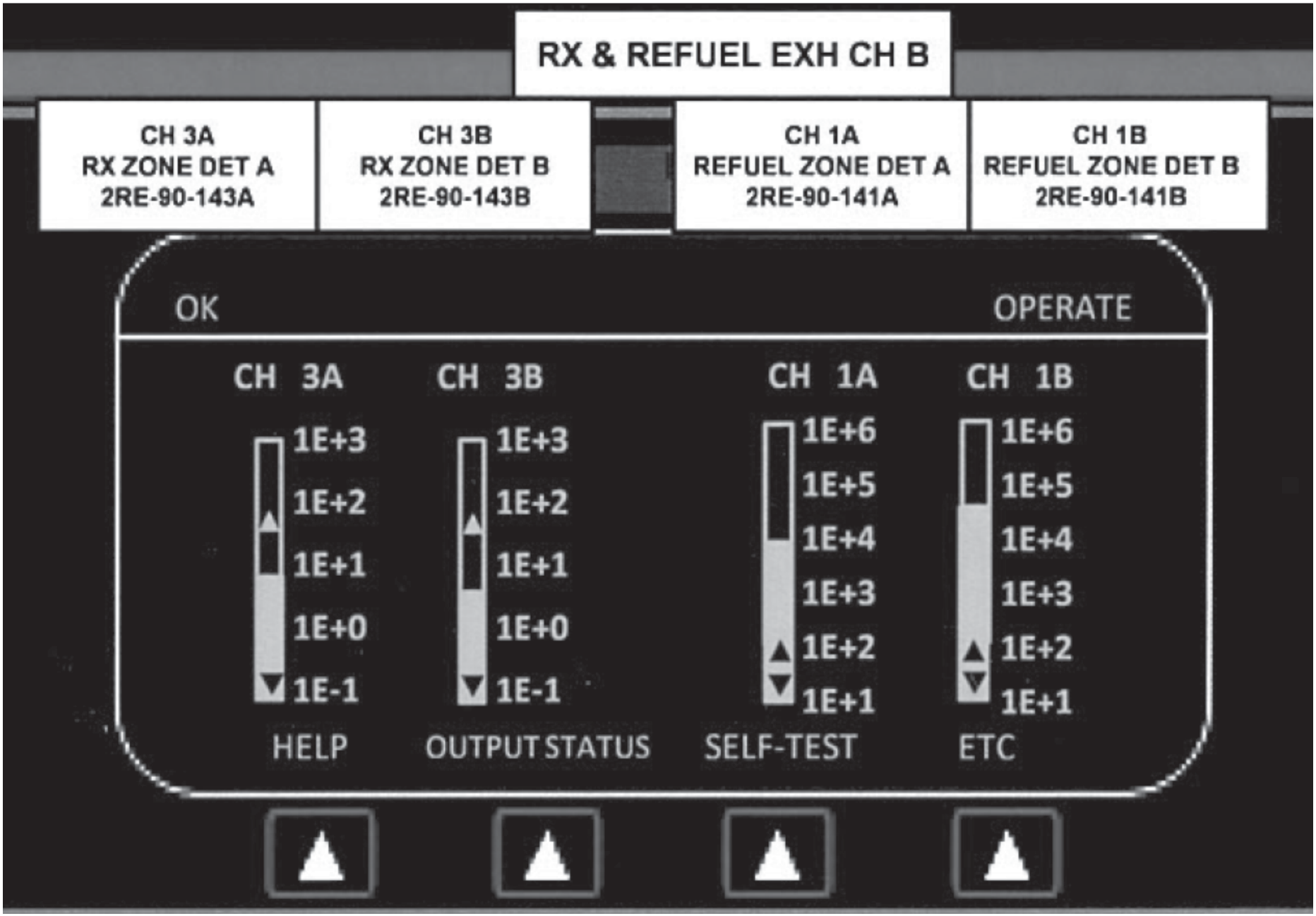
A. 10,000 mR/hr; below

B. 10,000 mR/hr; above

C. 40,000 mR/hr; below

D. 40,000 mR/hr; above

Answer: D


RX & REFUEL EXH CH B
CH 3A
RX ZONE DET A
2RE-90-143A
CH 3B
RX ZONE DET B
2RE-90-143B
CH 1A
REFUEL ZONE DET A
2RE-90-141A
CH 1B
REFUEL ZONE DET B
2RE-90-141B
OK
OPERATE
CH 3A
1E+3
1E+2
1E+1
1E+0
1E-1
CH 3B
1E+3
1E+2
1E+1
1E+0
1E-1
CH 1A
1E+6
1E+5
1E+4
1E+3
1E+2
1E+1
CH 1B
1E+6
1E+5
1E+4
1E+3
1E+2
1E+1
HELP
OUTPUT STATUS
SELF-TEST
ETC

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295023 AA1.04	
	Importance Rating	3.4	3.7

Refueling Accidents; Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7) Radiation monitoring equipment

Justification for K/A match: The question matches the K/A, because it is asking about monitoring the refuel radiation monitors during a refueling accident. The monitors are reading much higher than normal, which is indicative of a refueling accident.

Explanation: Correct D: During the refueling accident, the refuel zone detectors could be reading higher than 10 R/hr and as indicated on the picture provided, the 2RE-90-141B which is the B channel Refuel Zone detector is reading between 10,000 and 100,000 mR/hr somewhat higher than half way between the two, or 40,000 mR/hr.

A. **Incorrect because –** This is the Refuel Zone A reading, not the Refuel Zone B reading.
Plausible since they are displayed on the same screen right next to each other.

B. **Incorrect because –** This is the Refuel Zone A reading, not the Refuel Zone B reading.
Plausible since they are displayed on the same screen right next to each other.

C. **Incorrect because –** This reading is above the trip setpoint
Plausible since the refuel rad monitors have a different range than the Rx zone detectors, but they have the same setpoint.

Technical Reference(s): OPL171.033, Rev 14

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.033 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:	41(b)(11)	

5. Reactor Building (RM-90-142, 143) /Refuel Zone (RM-90-140, 141) Ventilation Radiation Monitoring System,

   b. Four gamma sensitive GM instrumentation channels monitor the radiation from the reactor zone exhaust and four identical channels monitor the radiation from the refueling zone

      (1) These are physically located on the side of the ventilation ducts on the refuel floor

      (2) Refuel Zone monitors RM-90-140(A & B) and Reactor Zone monitors 90-142(A & B) are powered from RPS 'A'

      (3) Refuel Zone monitors RM-90-141(A & B) and Reactor Zone monitors 90-143(A & B) are powered from RPS 'B'

      (4) The instrumentation channels are similar to an area radiation monitoring system channel.

   c. Alarms

      (1) REACTOR ZONE EXHAUST RADIATION HIGH (55-3A-21) Alarm setpoint is 72 mr/hr

         (a) Reactor zone and refueling zone monitors work independently of each other for trip actuation

         (b) High radiation trip setpoint is 72 mr/hr for the refueling and reactor zones Rad-monitor auto resets when alarm is clear

         (c) Trip logic for the refueling and the reactor zones is identical, and the following combinations will generate a trip:

      Two high level trips in the same channel, (division) –OR Two-out-of-two, once

      One downscale trip in each channel (division) –OR One-out-of-two, twice

      One monitor INOP in each channel (division) –OR Loss of RPS power to either channel

      (2) Automatic actions (a) Refuel Zone Trip

         1) Isolate Refuel Zone ALL 3 UNITS

         2) Starts Standby Gas Treatment System Opens SBGT suction to Refuel Zone

         3) PCIS Group 6 isolation

4) Starts CREVs

(b) Reactor Zone Trip

1) Isolate Control Room,
Reactor Zone, and Refueling Zone ventilation

2) Starts Standby Gas
Treatment System Opens SBGT suction to affected unit's Reactor Zone

3) Start CREVs

4) PCIS Group 6 isolation

(3) REACTOR ZONE EXHAUST RADIATION MONITOR DNSCL (55-3A-35) Alarms when reading <0.20 mr/hr

(4) REFUELING ZONE EXHAUST RADIATION HIGH (55-3A-34) Alarm setpoint is 72 mr/hr Same setpoint as Reactor zone Radiation High

(5) REFUELING ZONE EXHAUST RADIATION MONITOR DNSCL
(55-3A-28) Alarms when reading <18.0 mr/hr

d. Instrumentation

(1) NUMAC Digital Display is provided on Panel 9-10
(2) Units 2 and 3 have 2 Dual-pen recorders on Panel 9-2 RR-90-140/141 and RR-90-142/143 Unit 1 has 4 pen recorder RR-90-144
(3) ICS Display "MISCRAD"
140 Refuel Zone
141 Refuel Zone
142 Reactor Zone
143 Reactor Zone

QUESTION 11 Rev 2

All three Units are operating at 100% power when a small steam leak develops in Unit 2 Drywell.

- The Unit Supervisor directs the Unit Operator to begin venting in accordance with the AOI.

(1) How many SGT train(s) are required to be verified running prior to venting?

(2) When venting is complete, in which Unit's SR-2 should the SGT run time be recorded?

A. (1) 1
(2) 1

B. (1) 1
(2) 2

C. (1) 2
(2) 1

D. (1) 2
(2) 2

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295024EA1.20	
	Importance Rating	3.1	3.1

High Drywell Pressure; Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE:(CFR: 41.7 / 45.6) Standby gas treatment/FRVS

Justification for K/A match: This is a Tier 1, Abnormal /Emergency K/A concerning Standby Gas Trains and High Drywell Pressures. To match the K/A the question places the unit in a rising DW pressure condition which requires the Drywell to be vented. To match the other part of the K/A, ability to operate and /or monitor the SGTs as they apply to high drywell pressure.

Explanation: CORRECT A: With Drywell pressure rising, the2- AOI-64-1 has the operator verify one SGT running before aligning to vent the drywell. Also contained in the AOI, are steps to record the run time in the Unit 1 Control Room 1-SR-2.

B. **Incorrect because –**SGT run times are recorded in 1-SR-2
Plausible because –Part 1 is correct and since Unit 2 vented it is plausible that the run time would be recorded in 2-SR-2.

C. **Incorrect because –** 2-AOI-64-1 directs starting one train prior to venting.
Plausible because – IAW OPL 171.018, Standby Gas Treatment System, 2 of the 3 trains can provide design flow conditions.

D. **Incorrect because –**Part 1 see C above and Part 2 see B above.
Plausiblebecause –Part 1 see C above and Part 2 see B above.

Technical Reference(s): 0-OI-65 Rev 55, 2-AOI-64-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.018 rev 18 HLT V.B 9 and 12

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)(7)

4.2 Subsequent Actions (continued)

[2.5.4] **VERIFY RUNNING** a Standby Gas Treatment Fan STGTS TRAIN C(A)(B) (Panel 2-9-25).

4.2 Subsequent Actions (continued)

[2.5.13] **RECORD** SGTS Train(s) run time in appropriate Control Room Reactor narrative log for transfer to 1-SR-2.

b. All 3 SGT trains auto-start on initiation and run until manually stopped.
c. 2 of the 3 trains can provide design flow conditions.

QUESTION 12 Rev 0

Which ONE of the following describes a basis for the initiation of Alternate Rod Insertion (ARI) on high Reactor pressure?

ARI is initiated to…

A. provide an independent and redundant means of depressurizing the reactor scram air header to affect a reactor scram.

B. protect against exceeding the Heat Capacity Temperature Limit.

C. protect against exceeding the Reactor Pressure safety limit.

D. provide an independent and redundant means of de-energizing the scram pilot solenoid valves.

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295025EK3.06	
	Importance Rating	4.2	4.4

High Reactor Pressure; Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: (CFR: 41.5 / 45.6) Alternate rod insertion

Justification for K/A match: To match the K/A, the question asks straight forward, what is the reason for the initiation of Alternate Rod Insertion (ARI) on high Reactor pressure.

Explanation: CORRECT A: IAW EOIPM 0-V-C discussion of step RCQ-4 Initiation of ARI provides an independent and redundant means of depressurizing the reactor scram air header and operating the scram discharge volume vent and drain valves as required to affect a reactor scram.

B. Incorrect because – ARI provides an independent and redundant means of depressurizing the reactor scram air header.
Plausible in that inserting control rods will help to prevent exceeding HCTL however, that is not the basis for ARI. Initiation of SLC (another reactivity control method) does protect against exceeding HCTL.

C. Incorrect because – ARI provides an independent and redundant means of depressurizing the reactor scram air header.
Plausible in that inserting control rods will help to limit the Reactor Pressure rise it is not the basis for ARI. The Recirc pump RPT breakers are tripped by Reactor high pressure adding negative reactivity to the core. This reduces generated core heat sufficiently such that the MSRVs can protect the reactor from over-pressurization.

D. Incorrect because – ARI does not de-energizing the scram pilot solenoid valves.
Plausible in that ARI does provide a redundant means of inserting control rods however it is independent of the scram solenoid valves.

Technical Reference(s): 1-OI-85 rev 40, OPL171.005 rev 18, OPL171.007 rev 25, EOIPM 0-V-C rev 02

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.005 rev18 OBJ 23

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Clinton 2007 NRC Exam Q#24

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

10 CFR Part 55 Content: 41(b)(5)

3.2 ATWS/ARI/RPT

A. The ARI system auto initiation can be reset after a 30 sec time delay and all initiation signals are reset.

B. The ATWS/ARI/RPT is activated by either two low levels (≤ -45 in) or two high pressures 1148 psig, or manual initiation pushbutton.

1. An automatic signal from either A or B trip channel causes two actions:

a. It opens one of the two RPT breakers on each of the two recirculation pumps,

AND

b. It energizes one of the two identical sets of four ATWS/ARI/RPT valves.

2. Manual initiation from either A or B trip channel only initiates the ARI portion of the system. The RPT will not trip from manual initiation.

OPL171.005, Control Rod Drive (CRD) Hydraulics , Rev. 18

o. ATWS/ARI (Alternate Rod Insertion) System

1) Purpose - to provide an alternate method of shutting down the reactor should control rods fail to insert on a scram.

OPL171.007 ,Reactor Recirculation System, Revision 25 page 39 and 40

b. Anticipated Transient Without Scram (ATWS) Recirculation Pump Trip (RPT)

1) For most anticipated transients, a reactor scram to shutdown the reactor and rapidly reduce the core heat generation is an important step to ensure that no plant damage occurs.
a) If such a transient should occur and a scram did not result, an ATWS event would have occurred.

2) The transients having the greatest potential for significant core damage are those leading to a pressure rise in the reactor vessel.
a) Consequently, Brown's Ferry has installed the ATWS Recirculation Pump Trip (ATWS- RPT) in conjunction with the Alternate Rod Insertion (ARI) System.

3) The ATWS trip of the Recirculation Pumps is initiated by meeting logic coincidence requirements for either of the following two signals:
a) RPV High Pressure at 1148 psig.
b) RPV Low Reactor Water Level at -45 inches.

4) The ATWS-RPT trip opens the RPT breakers and secures both Recirculation Pumps.

5) When logic requirements are met in either channel, the RPT breakers are tripped, adding negative reactivity to the core.
a) This reduces generated core heat sufficiently such that the MSRVs can protect the reactor from overpressurization.

6) The ATWS trip of the Recirculation Pumps shares the same auto-initiation logic as the ATWS Alternate Rod Insertion (ARI) circuit.

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION : RC/Q-4

Initiation of ARI provides an independent and redundant means of depressurizing the reactor scram air header and operating the scram discharge volume vent and drain valves as required to affect a reactor scram. The instruction regarding ARI initiation is applicable at this point in the sequence of actions to control reactor power because ARI does not also cause a trip of the reactor recirculation pumps. (Running back recirculation flow to minimum and then tripping the pumps is addressed separately in subsequent steps.)

Clinton 2007 NRC Exam Q #24

Clinton
Question #: 024 Exam Date: 2007/08/20 Facility: 461 Reactor Type: BWR-GE6 Exam Level B K/A 295025 EK3.06

QUESTION:

Which ONE of the following describes a basis for Alternate Rod Insertion (ARI) due to high reactor pressure?

a. ARI limits fuel damage due to pellet expansion to less than 1%.

b. ARI reduces the challenge to the integrity of the Reactor Coolant Pressure Boundary.

c. ARI reduces unnecessary safety relief valve operation that challenges SRV and SRV piping integrity.

d. ARI reduces unnecessary safety relief valve operation that results in undesired heatup of the Suppression Pool.

ANSWER: b.

REFERENCE:
SER ATWS Rule - 10CFR50.62
FUNDAMENTAL
BANK
EXPLANATION:
a is incorrect Limiting operating power level limits fuel damage during accident conditions.
b is correct ARI reduces the challenge to the integrity of the Reactor Coolant Pressure
Boundary.
c is incorrect SRV operation contributes to RPV pressure control and void formation in the core contributes to core cooling and reactivity control by insertion of negative reactivity. SRVs will continue to cycle on high pressure.
d is incorrect SRV operation contributes for RPV pressure control. While ARI shuts down the reactor and thereby reduces SRV cycling, it is not the reason for the ARI trip on a high RPV pressure signal.

K/A Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : Alternate rod insertion: Plant-Specific.

QUESTION 13 Rev 3

U3 was operating at 100% power.
Current conditions are as follows:

- MN STM LINE A RELIEF VALVE 1-179 has opened and attempts to reclose the valve have failed.
- TE-64-161C BAY 5 on 3-TR-64-161,SUPPRESSION POOL TEMPERATURE, Indicates 97°F
- 3-TI-64-161, SUPPR POOL BULK TEMP, indicates is 90°F
- The US has directed the UO to initiate Suppression Pool Cooling in accordance with 3-AOI-1-1, Relief Valve Stuck Open

Evaluate the illustrations attached on the next page.

Which one of the following completes the statements below?

Based on indications provided above all available Suppression Pool Cooling __ (1) __ required to be operated by 3-EOI-2, Primary Containment Control.

Based on the illustrations provided, RHR Loop I __ (2) __ properly aligned for Suppression Pool Cooling.

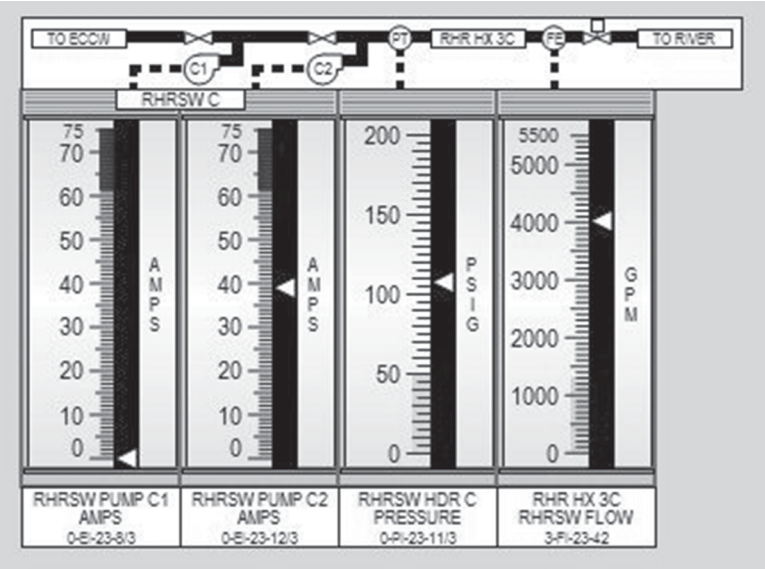
A. (1) is
(2) is

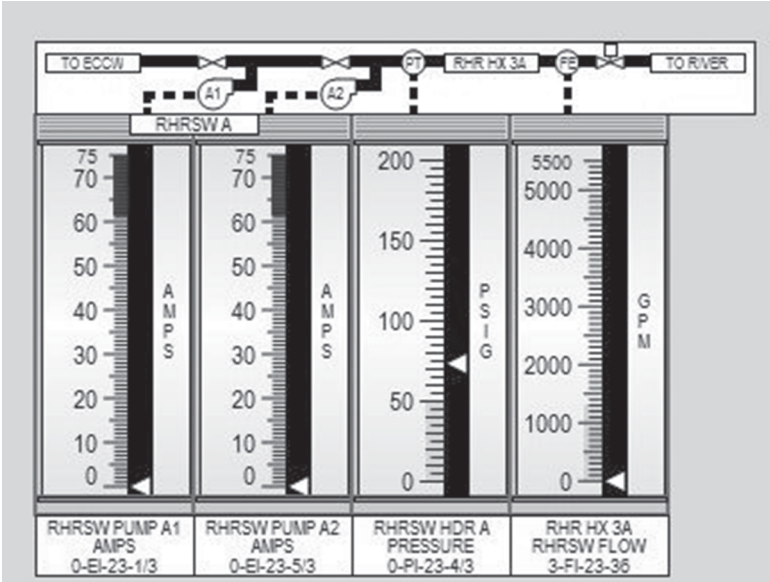
B. (1) is
(2) is NOT

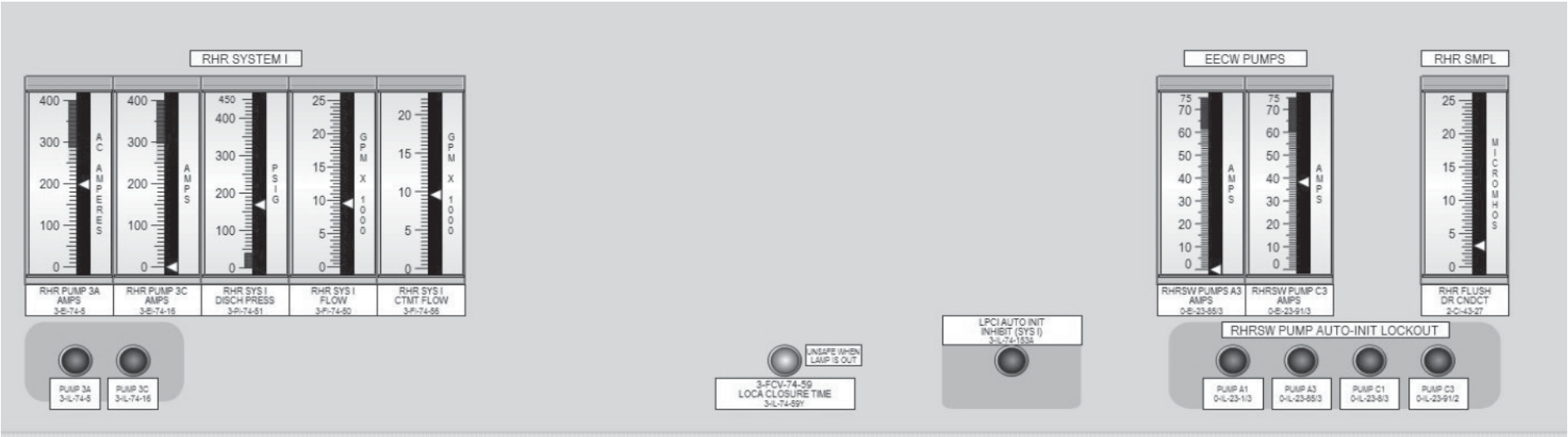
C. (1) is NOT
(2) is

D. (1) is NOT
(2) is NOT

Answer: D


TO ECCW
C1
C2
PT
RHR HX 3C
FE
TO RIVER
RHRSW C
AMPS
AMPS
PSIG
GPM
RHRSW PUMP C1 AMPS
0-EI-23-8/3
RHRSW PUMP C2 AMPS
0-EI-23-12/3
RHRSW HDR C PRESSURE
0-PI-23-11/3
RHR HX 3C RHRSW FLOW
3-FI-23-42


TO ECCW
A1
A2
PT
RHR HX 3A
FE
TO RIVER
RHRSW A
AMPS
AMPS
PSIG
GPM
RHRSW PUMP A1 AMPS
0-EI-23-1/3
RHRSW PUMP A2 AMPS
0-EI-23-5/3
RHRSW HDR A PRESSURE
0-PI-23-4/3
RHR HX 3A RHRSW FLOW
3-FI-23-36


RHR SYSTEM I
AC AMPERES
AMPS
PSIG
GPM X 1000
GPM X 1000
RHR PUMP 3A AMPS
3-EI-74-5
RHR PUMP 3C AMPS
3-EI-74-16
RHR SYS I DISCH PRESS
3-PI-74-51
RHR SYS I FLOW
3-FI-74-50
RHR SYS I CTMT FLOW
3-FI-74-56
PUMP 3A
3-IL-74-5
PUMP 3C
3-IL-74-16
UNSAFE WHEN LAMP IS OUT
3-FCV-74-59
LOCA CLOSURE TIME
3-IL-74-59Y
LPCI AUTO INIT INHIBIT (SYS I)
3-IL-74-159A
EECW PUMPS
AMPS
AMPS
RHRSW PUMPS A3 AMPS
0-EI-23-85/3
RHRSW PUMP C3 AMPS
0-EI-23-91/3
RHR SMPL
MICROMHOS
RHR FLUSH DR CNDCT
2-CI-43-27
RHRSW PUMP AUTO-INIT LOCKOUT
PUMP A1
0-IL-23-1/3
PUMP A3
0-IL-23-85/3
PUMP C1
0-IL-23-8/3
PUMP C3
0-IL-23-91/2

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295026 G2.1.31	
	Importance Rating	4.6	4.3

Suppression Pool High Water Temperature; Ability to locate control room switches, controls, and indications and to determine that they correctly reflect the desired Plant lineup.

Justification for K/A match: K/A 295026 is under the Emergency Plant Evolutions section of the catalog. Part 1 of the question asks if an action in EOI-2 is required based on Suppression Pool temperature indications, satisfying the first part of the K/A. The second part of the K/A is a Generic conduct of operations K/A. The question asks the candidate to evaluate control board indications (attached illustrations) and determine if Suppression Pool Cooling is properly aligned satisfying the Generic part of the K/A.

Explanation: CORRECT D: The EOI-2 entry condition for SP temperature is based on the value of the Tech Spec LCO which is an average temperature. The average SP temperature is indicated as the bulk temperature on 3-TI-64-161 which is given as 90°F. This is below the EOI-2 entry condition therefore; EOI-2 does not require all available SPC. The illustrations provided show that 3A RHR is aligned for SCP however RHRSW is aligned to the 3C RHR heat exchange and not to the 3A RHR heat exchanger and therefore, is not properly aligned.

A. **Incorrect because –** The average (bulk) SP temperature is below 95°F and because RHRSW is aligned to the C RHR heat exchanger instead of the A RHR heat exchanger.
Plausible because – TE-64-161C for BAY 5 of the SP is above 95°F and because the RHR lineup for SPC, including the LOCA closure time light, is correct except for the RHRSW lineup.

B. **Incorrect because** – The average (bulk) SP temperature is below 95°F.
Plausible because – TE-64-161C for BAY 5 of the SP is above 95°F and Part 2 is correct.

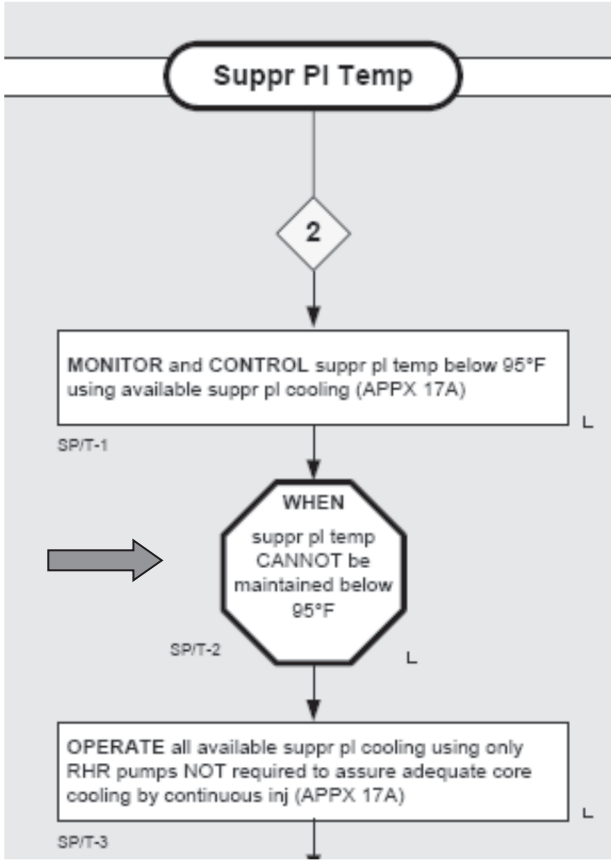
C. **Incorrect because** – RHRSW is aligned to the C RHR heat exchanger instead of the A RHR heat exchanger.
Plausible because – Part 1 is correct and the RHR lineup for SPC, including the LOCA closure time light, is correct except for the RHRSW lineup.

Technical Reference(s): EOIPM Section 0-III-D Rev 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.044 R19 OBJ 2.d

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(8)	



1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: ENTRY CONDITIONS (cont'd)

Suppression pool temperature above *A.31*(95°F), Technical Specification Suppression Pool Temperature LCO

Suppression Pool Average Temperature
3.6.2.1

3.6 CONTAINMENT SYSTEMS
3.6.2.1 Suppression Pool Average Temperature
LCO 3.6.2.1 Suppression pool average temperature shall be:

a. ≤ 95°F when any OPERABLE intermediate range monitor (IRM) channel is > 70/125 divisions of full scale on Range 7 and no testing that adds heat to the suppression pool is being performed;

4.2.4 Other Actions and Documentation

[4] **INITIATE** suppression pool cooling as necessary to maintain suppression pool temperature less than 95°F.

3-OI-74 Rev. 0119 Page 98 of 449

8.5 Initiation of Loop I(II) Suppression Pool Cooling (continued)

[7] **PLACE** RHR Pump and Heat Exchanger A(C) in service as follows:
[7.1] **START** an RHRSW Pump to supply RHR Heat Exchanger A(C).
[7.2] **THROTTLE** RHR HX 3A(3C) RHRSW OUTLET VLV, 3-FCV-23-34(40), as required for cooling. (Refer to caution 2 above.)
[7.3] **IF** required to maintain Total RHRSW Flow for RHRSW Pump A(C) between 4000 and 4500 gpm, **THEN REQUEST** another unit to establish flow for the associated RHRSW Pump A(C) and maintain the Total RHRSW Flow between 4000 and 4500 gpm per 0-OI-23. (Otherwise N/A)
[7.4] **VERIFY CLOSED** RHR SYS I LPCI INBD INJECT VALVE, 3-FCV-74-53.
[7.5] **VERIFY CLOSED** RHR SYS I SUPPR POOL CLG/TEST VLV, 3-FCV-74-59. (N/A if starting the second Loop I RHR Pump C(A))
[7.6] **VERIFY CLOSED** RHR SYS I SUPPR CHBR SPRAY VALVE, 3-FCV-74-58.
[7.7] **VERIFY CLOSED** RHR SYS I DW SPRAY OUTBD VLV, 3-FCV-74-60.
[7.8] **VERIFY OPEN** RHR SYS I SUPPR CHBR/POOL ISOL VLV, 3-FCV-74-57.
[7.9] **START** RHR PUMP 3A(3C) using 3-HS-74-5A(16A).

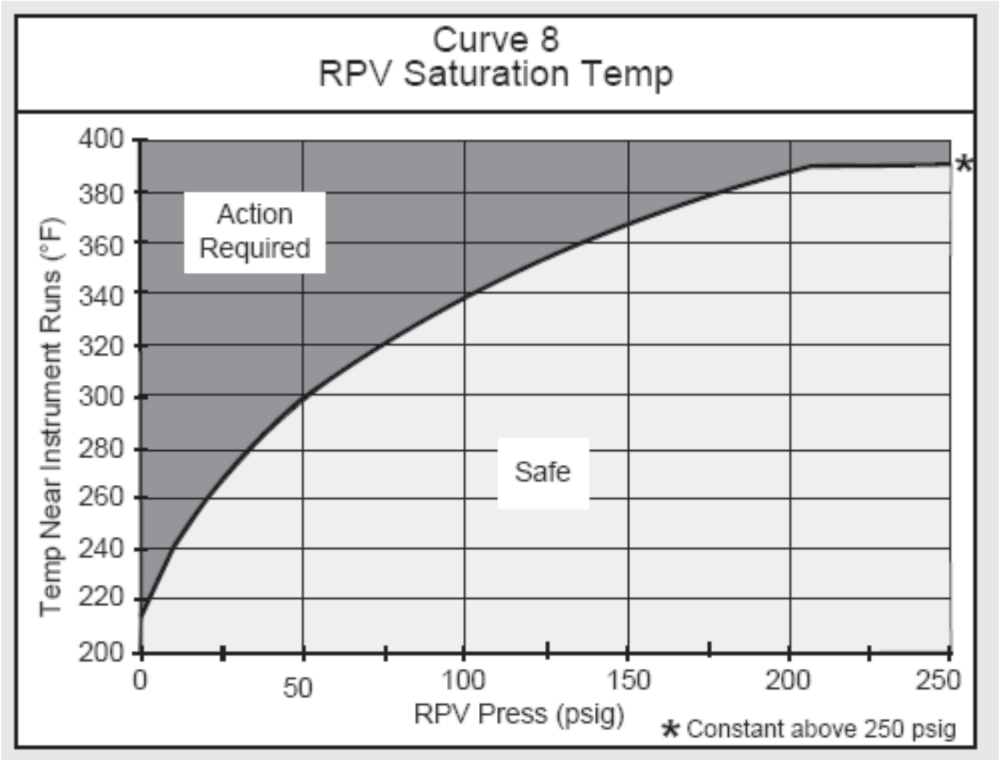
QUESTION 14 Rev 0

The following conditions exist following a LOCA:

- Drywell temperature is 320°F
- Reactor pressure is 50 psig
- LI-3-55, Reactor water level Flood up Range indicates above the minimum level for Drywell Temperature up to 350°F.

In accordance with EOI-5 Caution 1 and curve 8 (see attached) which one of the following completes the statements below?

LI-3-55 __ (1) __ due to __ (2) __.



A. (1) may **NOT** be used
(2) boiling in the instrument run

B. (1) may be unreliable
(2) boiling in the instrument run

C. (1) may **NOT** be used
(2) being calibrated for cold conditions

D. (1) may be unreliable
(2) being calibrated for cold conditions

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295028 EA2.03	
	Importance Rating	3.7	3.9

High Drywell Temperature; Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) Reactor water level

Justification for K/A match: To match the K/A, the conditions are setup where elevated Drywell temperature has affected the vertical instrument run of the selected instrument and therefore the candidate must determine that the instrument is unreliable.

Explanation: CORRECT B: EOI-5 caution 1 states: If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run.

A. **Incorrect because –** The instrument may still be usable if there are no indications of boiling in the instrument runs.
Plausible based on being in the action required range of Curve 8, RPV Saturation Temp and Part 2 is correct.

C. **Incorrect because –** Part 1 see A above and Part 2 the decision is not based on calibration conditions.
Plausible because – Part 1 see A above and Part 2 is plausible because 2-LI-3-55 is calibrated for cold conditions and if above saturation Temp/Press the instrument accuracy may be affected.

D. **Incorrect because –** Part 1 is correct. Part 2 is incorrect see C above.
Plausible because – Part 1 is correct and Part 2 see C above.

Technical Reference(s): EOI-5 rev 0, OPL171.003 rev 20, OPL 171.201 rev 07

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.201 R7 OBJ 11 and 12

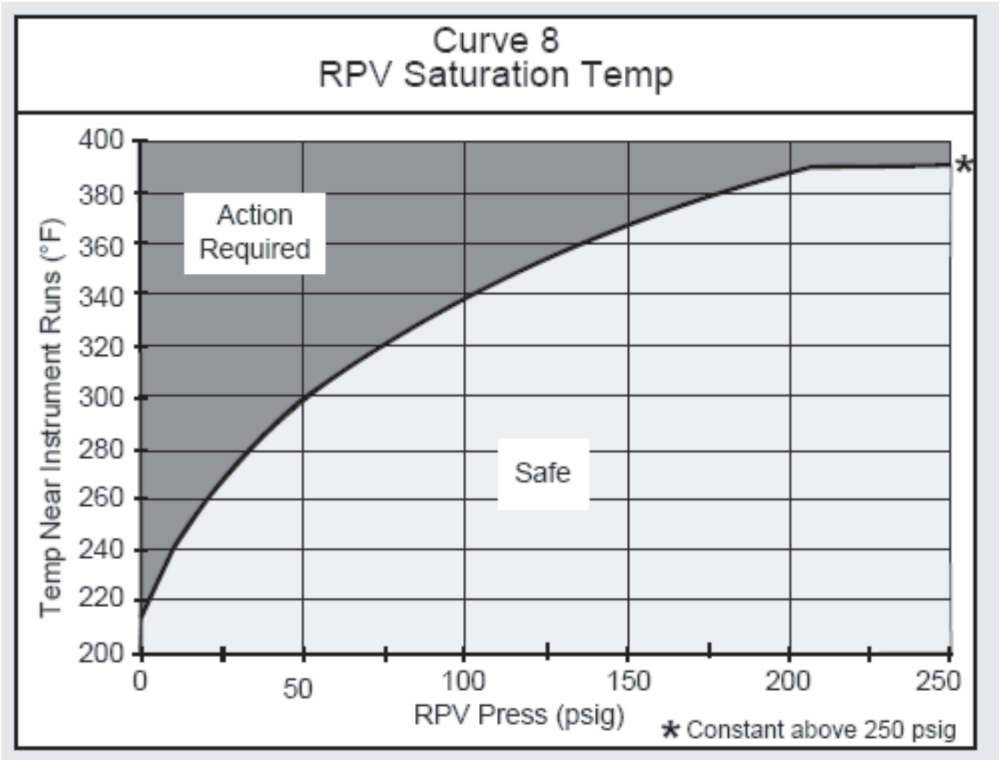
Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(10)	

2-EOI-5 CURVES AND CAUTIONS

An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp.

If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run.

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 100
		-150	N/A	101 to 150
		-145	N/A	151 to 200
		-140	N/A	201 to 250
		-130	N/A	251 to 300
		-120	N/A	301 to 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	Normal 0 to +60	on scale	N/A	below 150
		+5	N/A	151 to 200
		+15	N/A	201 to 250
		+20	N/A	251 to 300
		+30	N/A	301 to 350
LI-3-52 LI-3-62A	Post Accident -268 to +32	on scale	N/A	N/A
LI-3-55	Shutdown Floodup 0 to +400	+10	Below 100	N/A
		+15	100 to 150	N/A
		+20	151 to 200	N/A
		+30	201 to 250	N/A
		+40	251 to 300	N/A
		+50	301 to 350	N/A
		+65	351 to 400	N/A



OPL171.003, REACTOR VESSEL PROCESS INSTRUMENTATION, Rev. 20

3) Shutdown Vessel Flood Range (Flood-up Range)
   a) 0 to +400 inches range covering upper portion of reactor vessel
   b) Referenced to instrument zero
   c) Calibration Conditions:
   - cold conditions
   - Rx Temperature (<212°F, 0 psig)
   - DW temp ~75°F

   d) Provides level indication during vessel flooding or cool down.

OPL 171.201 Revision 7 page 34

Caution 1 part B says instruments “may be unreliable” if Curve 8 is exceeded. This means instruments may continue to be used until and unless erratic indication is observed since momentary excursions (expected in some post LOCA situations) into curve 8 unsafe region will not result in boiling. If, however, indications of boiling are observed then that instrument is unusable until the instrument lines can be cooled and refilled.

QUESTION 15 Rev 0

Unit 3 is operating at 100% power, when the following events occur:

- 3-9-3B window 15 SUPPR CHAMBER WATER LEVEL ABNORMAL alarms.
- EOI-2, Primary Containment Control, is entered.
- EOI appendix 18, Suppression Water Inventory Removal and Makeup, is in progress.

Which one of the following, if valid, would indicate a lowering suppression pool level?

A. Drywell -to-Suppression Chamber Differential pressure is lowering.

B. 3-9-3C window 9 RCIC PUMP SUCTION PRESS LOW 3-PA-71-21A alarms.

C. 3-9-3E window 11 SUPPR POOL DISCH HDR PRESS LOW 3-PA-74-94 alarms.

D. HPCI Pump Suction valves automatically realign.

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295030 G2.4.46	
	Importance Rating	4.2	4.2

Low Suppression Pool Water Level; Ability to verify that the alarms are consistent with the plant conditions

Justification for K/A match: To match Tier 1 and the generic K/A the question sets up conditions of abnormal suppression pool level and asks the candidates to determine if the given indications and alarms are consistent with a Low Suppression Pool Water Level.

Explanation: Correct C: 3-PS-74-94 is on the piping coming from the torus ring header to the RHR pump suctions. As Suppression pool level lowers the static pressure at 3-PS-74-94 will lower and will alarm at 3.6 psi.

A. **Incorrect because –** This would indicate a rising suppression Pool level.
Plausible because - Changing suppression Pool level will change Drywell to Suppression Chamber D/P.

B. **Incorrect because –** RCIC suction is normally aligned to the CST.
Plausible because **-** While the RCIC suction is normally aligned to the CST it can be aligned to the Suppression Pool which could cause this alarm.

D. **Incorrect because –** This would occur for a high level in the Suppression Pool.
Plausible because **-** HPCI Pump Suction valves do automatically realign on Suppression Pool level.

Technical Reference(s): 3-ARP-9-3B Rev 21, 3C Rev 28, 3E Rev 28, 3F Rev 29

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.016 R19 OBJ 8 and 12

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(7)

SUPPR CHAMBER WATER LEVEL ABNORMAL 3-LA-64-54A	15

(Page 1 of 1)

Sensor/Trip Point:

LT-64-54	≤ -5.5" H_2O
	≥ -1.75" H_2O

Sensor Location: RX Bldg, El 519'
NW corner room just inside door

Probable Cause:
A. Suppression Chamber water level abnormal.
B. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** level using multiple indications. ☐
B. **IF** level is low, **THEN**
DISPATCH personnel to check for leaks. ☐
C. **IF** level is high, **THEN**
CHECK for RCIC, HPCI, Core Spray, or RHR draining to Suppression Pool, and **CHECK** 3-TR-64-161 and -162. ☐
D. **REFER TO** 3-OI-74, Sections 8.2, 8.3, and 8.4. ☐
E. **REFER TO** Tech Spec Section 3.6.2.2. ☐
F. **IF** level is above -1" or below -6.25" **AND NOT** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
ENTER 3-EOI-2 Flowchart. ☐
G. **IF** level is above -1" or below -6.25" **AND** in Mode 4 or Mode 5 **THEN** (otherwise N/A)
1. **EVALUATE** plant conditions to **DETERMINE** if 3-EOI-2 entry is appropriate. ☐
2. **RECORD** actions in NOMS log. ☐

References: 3-45E620-3 3-47E610-64-1 GE 730E933-1
Technical Specifications 3.6.2.2

DRYWELL TO SUPPR CHAMBER DIFF PRESS ABNORMAL 3-PDA-64-137	26

(Page 1 of 1)

Sensor/Trip Point:

3-PDS-64-137A or 64-138A	1.13 psid low
3-PDS-64-137C or 64-138C	1.34 psid high

Sensor Location:
Panel 9-19
Elevation 593'
Auxiliary Instrument Room

Probable Cause:
A. Drywell DP compressor malfunction.
B. Excessive N2 makeup to the drywell/torus.
C. Excessive venting from drywell/torus.
D. Sensor malfunction.

Automatic Action:
A. 3-PDS-64-137B and/or 138B.
1. Starts compressor on lowering DP.
2. Stops compressor on rising DP.

Operator Action:
A. **VERIFY** alarm by checking Drywell to Suppression Chamber DP. ☐
B. **REFER TO** 3-OI-64. ☐
C. **REFER TO** Tech Spec Section 3.6.2.6. ☐

References:
3-45E620-8 3-45E777-21 3-47E610-64-2
Technical Specifications 3.6.2.6

RCIC PUMP
SUCTION PRESS
LOW
3-PA-71-21A

9

(Page 1 of 1)

Sensor/Trip Point:

3-PS-071-0021A 10 Inches Hg Vacuum

Sensor Location: Panel 25-31
Rx. Bldg, El 621', R-20 Q-LINE

Probable Cause:
A. Misalignment of suction valves.
B. Sensor malfunction.

Automatic Action:
A. RCIC Turbine trip.
1. RCIC TURB THROT VALVE, 3-FCV-71-9, closes.
2. RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, closes.

Operator Action:
A. **VERIFY** all automatic actions have occurred. ☐
B. **CHECK** suction valve lineup. ☐
C. **IF** RCIC Turbine has tripped, **THEN** **REFER TO** 3-OI-71. ☐
D. **REFER TO** Tech Spec 3.5.3. ☐

References: 3-45E620-2 GE 730E921 Series 0-45E626-2 3-47E610-71-1
Technical Specifications 3.5.3

SUPPR POOL DISCH HDR PRESS LOW 3-PA-74-94	11

(Page 1 of 1)

Sensor/Trip Point:

PS-74-94 3.6 psi

Sensor Location: Panel 25-160
Rx Bldg, El 519', R-18 U-LINE

Probable Cause:
A. Sensor malfunction.
B. Spurious pump start.
C. Low level in suppression pool.

Automatic Action: None

Operator Action:
A. **VERIFY** suppression pool disch. hdr pressure, using 3-PI-74-94, on Panel 3-9-3. ☐
B. **CHECK** Suppression Chamber water level is between -2 inches and -5.5 inches. **IF NOT**, **THEN** **REFER TO** 3-OI-74. ☐
C. **CHECK** for spurious RHR, Core Spray, HPCI or RCIC pump start. **REFER TO** 3-OI-74, 3-OI-75, 3-OI-73, or 3-OI-71, respectively. ☐
D. **VERIFY** ECCS systems are aligned properly. **REFER TO** 3-OI-74, 3-OI-75, 3-OI-73, or 3-OI-71, respectively. ☐
E. **DISPATCH** personnel to Rx Bldg, El 519' to check for leaks and leakage paths from the suppression pool. ☐
F. **REFER TO** Tech Spec 3.6.2.2 ☐

References: 3-45N620-1 47W600-112 3-47E610-74-1 GE 730E920-12
Technical Specifications 3.6.2.2, 5.4 and 5.5
FSAR Section 13.0

SUPPR POOL LEVEL HIGH 3-LA-73-57	12

(Page 1 of 1)

Sensor/Trip Point:

LS-73-57A OR 57B	5.25 inches above instrument zero after 7-second time delay

Sensor Location: Rx Bldg, El 537', N-R15, NW wall, next to Suppr Chbr on a platform (Bay 7)

Probable Cause:
A. Rising water level in Suppression Pool.
B. Sensor malfunction.

Automatic Action: Automatic transfer of HPCI suction from CST to Suppression Pool.

Operator Action:
A. **CHECK** CST 3 and Suppression Pool level using multiple indications. ☐
B. **VERIFY** HPCI Suction automatically transfers to the Suppression Pool. ☐
C. **IF** automatic transfer fails, **THEN** **REFER TO** 3-OI-73. ☐
D. **REFER TO** Tech Spec 3.5.1, 3.5.2 and 3.6.2.2. ☐

References: 3-45E620-1 3-47E610-64-1 3-730E928-3
Technical Specifications 3.5.1, 3.5.2 and 3.6.2.2

QUESTION 16 Rev 0

An ATWS has occurred on Unit 2.

- ATWS actions are complete
- Reactor water level currently indicates +40 inches
- Reactor Power is 46%
- SLC is injecting

Which one of the following completes the statements below?

EOI-1A requires operators to stop and prevent all injection except __ (1) __ to mitigate the consequences of the failure-to-scram.

This action mitigates the failure to scram by __ (2) __ which adds negative reactivity.

A. (1) CRD, and SLC only
(2) reducing natural circulation and increasing the void fraction in the core

B. (1) CRD, and SLC only
(2) increasing natural circulation to mix the injected boron

C. (1) RCIC, CRD, and SLC
(2) reducing natural circulation and increasing the void fraction in the core

D. (1) RCIC, CRD, and SLC
(2) increasing natural circulation to mix the injected boron

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295031EK1.02	
	Importance Rating	3.8	4.1

Reactor Low Water Level; Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.8 to 41.10) Natural circulation:

Justification for K/A match: To match Tier 1, an ATWAS is presented as the initial conditions of the question. To match the K/A, the question asks about terminating and preventing injection, which lowers level to the point where natural circulation is retarded causing more voids to be generated, which is an operation implication of the action of lowering reactor water level.

Explanation: CORRECT C: In accordance with EOIPM 0-V-M Lowering RPV water level is accomplished by terminating and preventing all injection into the RPV, except from SLC, RCIC and CRD. The EOIPM also states: The process by which reactor power is reduced by lowering RPV water level occurs as follows: … As the natural circulation driving head is reduced, the natural circulation flow through the core is reduced. The reduced core flow results in a reduced rate of steam removal from the core. The reduced rate of steam removal results in an increased void fraction inside the shroud. The increased void fraction adds negative reactivity to the reactor

A. **Incorrect because –** RCIC injection is also required to be stopped and prevented.
   Plausible because SLC and CRD may be needed to establish and maintain reactor shutdown conditions. RCIC operation is allowed to continue and (along with SLC and CRD) may prevent RPV water level from dropping to the level that requires emergency RPV depressurization. Part 2 is correct

B. **Incorrect because –** Part 1 see A above. Part 2 Increasing natural circulation before SLC tank level lowers to 67% would increase power.
   Plausible because raising Reactor water level will increase natural circulation and promote mixing the injected boron. This is the bases for raising Reactor water level when SLC tank level lowers to 67% which represents the hot shutdown boron weight.

D. **Incorrect because –** Part 2 see B above.
   Plausible because – Part 1 is correct. Part 2 see B above.

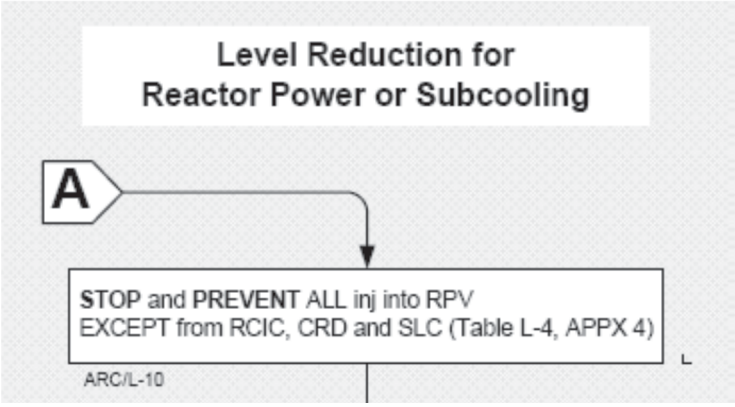
Technical Reference(s): 2-EOI-1A Rev 0, EOIPM 0-V-M Rev 0

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.205 R11 OBJ 2

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(1)	

2-EOI-1A ATWS RPV CONTROL



1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/L-10

Flowpath A is entered from ARC/L-4 when RPV water level must be deliberately lowered to reduce reactor power or core inlet subcooling.

Lowering RPV water level is accomplished by terminating and preventing all injection into the RPV, except from SLC, RCIC and CRD. SLC, RCIC and CRD are relatively low flow systems. SLC and CRD may be needed to establish and maintain reactor shutdown conditions. When restoration of injection is subsequently required but other outside shroud injection systems are incapable of injection, continued RCIC operation (along with SLC and CRD) may prevent RPV water level from dropping to the level that requires emergency RPV depressurization. The marginal increase in integrated power resulting from continued RCIC operation while RPV water level is deliberately lowered has a negligible impact on suppression pool temperature. Table 4 lists the systems that must be terminated and prevented in this step. EOI Appendix 4 provides instructions for accomplishing this task.

With RPV injection terminated and prevented, RPV water level and reactor power decrease at the maximum possible rate allowed by boil off. Failure to completely stop RPV injection flow (with the exception of CRD, RCIC and SLC) prolongs the elevated reactor power condition, thus depositing more energy than necessary into the suppression pool. For RPV water level reductions which uncover the feedwater spargers, failure to completely stop RPV injection flow also delays reduction in core inlet subcooling, thus increasing the possibility of neutronic/thermal-hydraulic instabilities.

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/L-13, ARC/L-14

If all Table Q-1 conditions exist, it is appropriate to continue lowering RPV water level in Step ARC/L-13 in order to effect the desired power reduction until, in Step ARC/L-14, one or more of the Level/Power conditions no longer exist.

The operator is or has been directed to reject as much heat as possible from the RPV to the main condenser (retainment override in ARC/P-3), to place all available suppression pool cooling into operation (EOI-2 SP/T), to trip the recirculation pumps (ARC/Q-5), and to concurrently inject boron and manually insert control rods (ARC/Q-10 and ARC/Q-14). One additional action remains available to mitigate the consequences of a failure to scram condition: deliberately lowering RPV water level to effect a reduction in reactor power. Lowering RPV water level reduces natural circulation driving head and core flow, thereby reducing reactor power and the heat addition rate to the suppression pool. The reactor power reduction achieved by lowering RPV water level may be sufficient to ensure that boron injection can be completed before the Heat Capacity Temperature Limit is reached.

The process by which reactor power is reduced by lowering RPV water level occurs as follows:

1. The reactor is in a natural circulation mode following recirculation pump trip (accomplished in Step ARC/Q-5). The natural circulation driving head is a function of the fluid density difference between the regions inside and outside of the shroud (void fraction directly affects the fluid density inside the shroud) and the height of the fluid columns (RPV water level).

2. As RPV water level is lowered, the height of the fluid columns is reduced, thereby reducing the natural circulation driving head.

3. As the natural circulation driving head is reduced, the natural circulation flow through the core is reduced.

4. The reduced core flow results in a reduced rate of steam removal from the core.

5. The reduced rate of steam removal results in an increased void fraction inside the shroud.

6. The increased void fraction adds negative reactivity to the reactor.

7. The negative reactivity drives the reactor slightly subcritical and power begins to decrease.

1.0 EOI-1A, ATWS RPV CONTROL BASES (continued)

DISCUSSION: ARC/L-7

With boron injected into the lower plenum, little natural circulation and boron mixing occur if RPV water level is lowered to and maintained near the Minimum Steam Cooling RPV Water Level. Three-dimensional scale model tests indicate that the injected boron concentrates in the lower plenum and does not contribute to reactor shutdown until in-core distribution (mixing) is achieved. When an amount of boron sufficient to shut down the reactor has been injected into the RPV, mixing is accomplished by raising RPV water level in ARC/L-9, thereby increasing natural circulation flow through the vessel.

Question 17 Rev 1

Unit 2 is operating at 100% power when an ATWS occurs. The following conditions are noted:

- The OATC reports ATWS actions complete
- All Main Turbine bypass valves failed close
- The BOP operator has opened 2 MSRVs
- Reactor pressure is 1050 psig and rising

In accordance with step ARC-1 and NOTE 1 of EOI-1A which one of the following conditions would allow exiting EOI-1A and entering EOI-1?

A. 19 Control Rods are at notch 02 with all other Control Rods fully inserted.

B. All Control Rods are inserted to 00 except 2 at notch 08.

C. Reactor subcritical with SLC injected into the RPV to a tank level of 67%.

D. SLC has been injected into the RPV to a tank level of 43%.

Answer A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295037 EK1.07	
	Importance Rating	3.4	3.8

SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown; Knowledge of the operational implications of the following concepts as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: (CFR: 41.8 to 41.10) Shutdown margin

Justification for K/A match: To match the Tier and K/A an ATWAS is the setup of the question. To match the knowledge of the operational implications of Shutdown margin part of the K/A, the question asks whether or not shutdown margin is met or not so that the ATWS flowchart could be exited, given the set of conditions in the question.

Explanation: CORRECT A: IAW EOI-1A Note 1 19 Control Rods are at notch 02 with all other Control Rods fully inserted provides adequate SDM to ensure that the Reactor will remain subcritical without boron under all conditions.

B. **Incorrect because –** These conditions do not met EOI note 1.
Plausible because – The total number of notches out for the 2 Rods would be 16 and IAW EOI Note 1 up to 19 notches out is acceptable however, only with 19 rods at notch 02

C. **Incorrect because –** These conditions do not met EOI note 1.
Plausible because – The Reactor is subcritical and Hot Shutdown boron weight has been injected. With SLC injected to a tank level of 67% Reactor water level can be returned to the normal level band.

D. **Incorrect because –** These conditions do not met EOI note 1.
Plausible because – This represents Cold Shutdown boron weight and allows plant cooldown to commence.

Technical Reference(s): Tech Spec amendment No 253, TVA-COLR-BF2C19 Rev 2, EOI-1A Rev 0, EOIPM 0-V-C Rev 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): N/A

Question Source:	Bank:	
	Modified Bank:	X
	New:	
Question History:	Previous NRC:	Peach Bottom 2007 Q18
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(10)	

TECH SPEC
1.0 USE AND APPLICATION
1.1 Definitions

SHUTDOWN MARGIN (SDM)
SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:
a. The reactor is xenon free;
b. The moderator temperature is 68□F; and
c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

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.

TECH SPEC B 3.1.1

ACTIONS A.1
With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted.

B.1
If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods).

COLR

EDMS: L32 140820 800

TVA NPG

Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga TN 37402

Date: August 20, 2014

8 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

NOTE

1 The reactor will remain subcritical without boron under all conditions when:

- Any 19 control rods are at notch **02** with all other control rods fully inserted

OR

- All control rods except one are inserted to or beyond position **00**

OR

- Determined by Reactor Engineering (0-TI-394)

9 **1A**

	IF	THEN
1	The reactor will remain subcritical without boron under all conditions	**STOP** boron inj unless required by other procedures AND 1 EOI-1 RPV Control

Peach Bottom

EXAMINATION ANSWER KEY

2007 NRC RO Rev 1

18 **ID: N-ILT-2101-6-002** **Points: 1.00**

Unit 2 conditions are as follows:

* The Unit has scrammed.
* Seven (7) control rods located randomly throughout the core are stuck between positions 06 and 34.
* None of the seven control rods moved after ARI initiation.
* Reactor pressure is 920 psig.
* Reactor water level is +20 inches (stable on the narrow range).
* Drywell pressure is 1.0 psig.
* Drywell temperature is 130°F.
* Torus temperature is 85°F.
* T-101, "RPV Control", Leg RC/Q Rods was entered from T-100, "Scram", due to ATWS condition.

In accordance with T-101, "RPV Control", which one of the following describes the condition allowing exit from T-101, Leg RC/Q?

A. Cold shutdown boron weight has been injected into the reactor core.

B. ALL control rods, except one, are fully inserted into the reactor core.

C. Reactor power will remain below 4% under ALL conditions without boron.

D. Hot shutdown boron weight has been injected into the reactor core.

Answer: B

QUESTION 18 Rev 1

All three units are operating at 100% power.

A transient occurs on Unit 2 and the following alarms are received:

- 2-9-4C window 27 OG AVG ANNUAL RELEASE LIMIT EXCEEDED
- 2-9-3A window 13 STACK GAS RADIATION HIGH
- 2-9-7A window 3 STACK GAS DILUTION AIR FLOW LOW

The Unit 2 UO reports the following:

- Stack dilution fan 2A tripped and 2B failed to start
- 0-FI-90-271 Stack Gas Flow on Panel 1-9-53 indicates 14,000 scfm

Based on the information provided which one of the following identifies the Stack Gas Radiation Monitor(s) with valid indications.

A. None of the Stack Gas Radiation Monitor indications are valid.

B. Only 0-RM-90-306 WRGERMS indication is valid.

C. Only 0-RM-90-147/148 Stack Gas Monitor indications are valid.

D. 0-RM-90-306 WRGERMS and 0-RM-90-147/148 Stack Gas Monitor indications are valid.

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295038 EA1.01	
	Importance Rating	3.9	4.2

High Off-Site Release Rate; Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.7 / 45.6) EA1.01 Stack-gas monitoring system

Justification for K/A match: This question matches the Tier and K/A, by setting up conditions that give a high off-site release rate and asks the candidate to monitor the Stack-gas monitoring system, and based on knowledge of the precautions and limitation concerning the system to make a judgment about the status of the system.

Explanation: CORRECT C: IAW 2-OI-66 and 2-OI-90 WRGERMS requires a minimum Stack flow of 16,366 scfm in order for accurate Isokinetic Radioactive Release Rate sampling and monitoring however, there is no requirement for Stack Dilution Fans or Filter Cubicle Exhaust Fans to be in service for general stack gas monitoring as provided by 0-RM-90-147/148.

A. **Incorrect because –** 0-RM-90-147/148 are valid even with low flow.
Plausible if the candidate remembers that Stack flow affects WRGERMS but does not remember that 0-RM-90-147/148 is not affected by lowering Stack flow.

B. **Incorrect because –** If stack flow is abnormal 0-AOI-90-2 directs declaring WRGERMS INOP.
Plausible if the candidate remembers that Stack flow affects one of the Stack Gas Rad Monitors but does not remember which one.

D. **Incorrect because –** If stack flow is abnormal 0-AOI-90-2 directs declaring WRGERMS INOP.
Plausible if the candidate remembers that Stack flow does not affect 0-RM-90-147/148 and does not remember that it does affect 0-RM-90-306 WRGERMS

Technical Reference(s): 2-OI-66 Rev 112, 2-OI-90 Rev 85, 2-AOI-90-2 Rev 13, 2-ARP-9-3A Rev 47, 2-ARP-9-7A Rev 32, 2-ARP-9-4C Rev 32

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.033 R14 OBJ V.B.3.b

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC: N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(11)	

OG AVG ANNUAL RELEASE LIMIT EXCEEDED 2-RA-90-157C	27

(Page 1 of 2)

Sensor/Trip Point:

2-RE-90-157 133 R/hr (Alarm from recorder)
(1.33×10^5 mR/hr)

Sensor Location:
Elevation 565'
Turbine Building
Column T-7 B-LINE
Recorder is on Panel 2-9-2.

Probable Cause:
A. Abnormal flow in the off gas system.
B. Resin trap failure (RWCU or Condensate Demins).
C. Fuel damage.

Automatic Action:
None

Operator Action:

A. To determine if the Off Gas Annual Release Rate Limit is exceeded, **PERFORM** the following: ☐
1. **VERIFY** alarm condition on the following: ☐
   - OFFGAS RADIATION recorder, 2-RR-90-266, Panel 2-9-2. ☐
   - OG PRETREATMENT RAD MON RTMR monitor, 2-RM-90-157, Panel 2-9-10. ☐

NOTE

High off-gas flow can sweep settled particulates into flow stream causing momentary rise in monitor parameters. Low off-gas flow can result in improper dilution causing rise in monitor parameters.

2. **VERIFY** off-gas flow and monitor sample flow normal. ☐
3. **NOTIFY** Radiation Protection. ☐
4. **REQUEST** Chemistry perform radiochemical analysis to determine source. ☐
5. With OPS MGT and Shift Manager's permission, **PLACE** charcoal beds in parallel with another unit. **REFER TO** 2-OI-66. ☐
6. **IF** fuel damage is suspected, **THEN** **REFER TO** 2-SR-3.4.6.1 for dose equivalent iodine - 131 determination. ☐

Continued on Next Page

OG AVG ANNUAL RELEASE LIMIT EXCEEDED 2-RA-90-157C, Window 27
(Page 2 of 2)

Operator
Action: (Continued)

7. **REFER TO** 0-SI-4.8.B.1.a.1 and 2-SR-3.4.6.1-a for ODCM compliance and to determine if power level reduction is required. ☐
8. **IF** directed by Shift Manager or Unit Supervisor/SRO, **THEN REDUCE** reactor power to maintain off-gas radiation within ODCM limits. ☐
9. **REFER TO** EPIP-1. ☐

References:

2-45E620-4	2-47E610-90-1	GE 2-729E814 series
2-47E610-55-14	FSAR Sections 1.6.4.4.6, 7.12.2.2, and 13.6.2	Technical Specifications 3.4.6.1-a.

Technical Requirements Manual Sections 3.3.5.1, 3.3.9.1, & 3.7.2.1

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

STACK GAS DILUTION AIR FLOW LOW 2-FA-66-33	3

(Page 1 of 1)

Sensor/Trip Point:

2-FIS-066-0033A 0.1" H_2O after 5 second time delay.
2-FIS-066-0033B

(Both 2-FIS-066-0033A and 2-FIS-066-0033B must pickup to receive this alarm.)

Sensor Location: Panel 25-211
Stack El 599.5'

Probable Cause:
A. Failure/degradation of inservice dilution fan(s).
B. Outlet dampers failed closed.
C. Breaker tripped.
D. Sensor failure.
E. Off-Gas Stack Isolation Dampers **NOT** full open
F. Standby Gas Treatment Train in service.

Automatic Action: Auto start of alternate fan if transfer switch, 2-XS-66-29, is in alternate fan position and alternate fan handswitch is in AUTO.

Operator Action:
A. **VERIFY** alternate fan ON and damper open, (red light illuminated) on Panel 2-9-7. ☐
B. **DISPATCH** personnel to stack to check and report status of the following for both fans:
1. Fan motor. ☐
2. Fan belts. ☐
3. Damper stuck closed. ☐

C. **CHECK** breaker 5C on 480V Diesel Aux Bd A and B. ☐

References: 2-45E620-10 2-47E610-66-1 2-45E771-4

3.0 PRECAUTIONS AND LIMITATIONS (continued)

N. To ensure radioactive effluents being released from the main stack are being properly monitored, the following stack dilution fan operational requirements should be observed:

1. 16,366 SCFM stack flow rate is required to support Wide Range Gaseous Effluent Radiation Monitoring System. However, the optimum flow rate should be maintained at greater than 18,500. Three to four Stack dilution Fans may be required to obtain this flow. This requirement provides minimum main stack flow for accurate Isokinetic Radioactive Release Rate sampling and monitoring.

2. There is no requirement for Stack Dilution Fans or Filter Cubicle Exhaust Fans to be in service for general stack gas monitoring as provided by 0-RM-90-147/148. Plant design has installed the sample probe greater than ten diameters above all flow paths that discharge to the stack. This design provides a well mixed main stack flow for accurate general stack gas radioactive release rate monitoring.

3.0 PRECAUTIONS AND LIMITATIONS

F. The following stack dilution fan operational requirements should be observed:

1. One Unit 2 Stack Dilution Fan is required to be in operation to provide dilution air flow for any potential hydrogen concentration in Offgas flow when Unit 2 Offgas System is required for unit operation.

2. The required flow for stack gas, 0-FI-90-271, is 16,366 scfm. To preclude receiving erroneous alarms, optimum flow is 18,500. Either one or both Stack Gas Dilution Fans can be placed in service to satisfy these requirements. This may require 4 Stack Dilution fans (total for the plant) to be placed in service. This requirement provides minimum main stack flow for correct and accurate isokinetic radioactive release rate sampling and monitoring. Any two Stack Dilution Fans from separate Units and one Filter Cubicle Exhaust Fan, as a minimum, may meet this flow rate.

Date _____

4.2 Subsequent Actions (continued)

[3] **WHEN** cause of alarm has been determined, **THEN**

GO TO appropriate section of this AOI as listed below: ☐

Step 4.2[4]	HIGH or HI HI NOBLE RADIATION
Step 4.2[5]	HOST COMMUNICATION ERROR
Step 4.2[6]	HIGH FILTER DIFF PRESS
Step 4.2[7]	ABNORMAL STACK FLOW
Step 4.2[8]	ABNORMAL NR or HR FLOW
Step 4.2[9]	ABN WRGERM BLDG TEMP

Date _____

4.2 Subsequent Actions (continued)

[7] **IF** ABNORMAL STACK FLOW, **THEN PERFORM** the following:

[7.1] **CHECK** ABNORMAL STACK FLOW in alarm on 0-RM-90-306.

[7.2] **NOTIFY** Unit Supervisor/SRO and request that 0-RM-90-306 be declared inoperable.

[7.3] **CONTACT** Chem Lab as required by LCO actions.

[7.4] **INITIATE** a SR.

QUESTION 19 Rev 1

A fire has occurred in the Unit 3 Reactor Building.

Which ONE of the following completes the statements below?

In accordance with EPIP-17, Fire Emergency Procedure, the __ (1) __ is responsible for sounding the fire alarm bell.

In accordance with 0-AOI-26-1, Fire Response, the reason AUOs are assembled in the Control Rooms is to ensure __ (2) __ is/are completed within the required time.

A. (1) Unit 1 Control Room Unit Operator
   (2) SSI manual actions

B. (1) Unit 1 Control Room Unit Operator
   (2) personnel accountability

C. (1) Unit 3 Control Room Unit Operator
   (2) SSI manual actions

D. (1) Unit 3 Control Room Unit Operator
   (2) personnel accountability

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	600000 AK3.04	
	Importance Rating	2.8	3.4

Plant Fire On Site; Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site.

Justification for K/A match: Tier 1 Abnormal/Emergency K/A for a fire on site. To match this K/A the question gives an initial setup of a fire in Unit 3 and asks the reason for the actions contained in the Fire Abnormal Procedure.

Explanation: Correct A: The Fire alarm bell is sounded from the Unit 1 Control Room only. In accordance with 0-AOI-26-1 Note on page 6, AUOs are assembled in the Control Rooms to ensure SSI manual actions can be completed within the required time.

B. **Incorrect because** – As stated in the AOI, AUOs are assembled in the Control Rooms to ensure SSI manual actions can be completed within the required time.
Plausible because Part 1 is correct and because the AUOs assigned to a Unit are expected to report to their MCR for assembly and accountability within 20 minutes. A fire in the Reactor Building could result in an Alert classification and assembly and accountability may be initiated at the discretion of the SED.

C. **Incorrect because** – even though the fire is in Unit 3 there are no provisions given to sound the fire alarm from that unit. That activity is performed in Unit 1 only.
Plausible because the fire is in the U3 Reactor Building and both U1 and U3 have Fire Protection Display Panels.

D. **Incorrect because** – even though the fire is in Unit 3 there are no provisions given to sound the fire alarm from that unit. That activity is performed in Unit 1 only.
Plausible because the fire is in the U3 Reactor Building and because both U1 and U3 have Fire Protection Display Panels. Part 2 is plausible because if accountability is initiated the AUOs assigned to a Unit are expected to report to their MCR for assembly and accountability within 20 minutes. A fire in the Reactor Building could result in an Alert classification and assembly and accountability may be initiated at the discretion of the SED.

Technical Reference(s): EPIP-17 Rev 33, 0-AOI-26-1 Rev 18.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.049 R15 OBJ V.B.6

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 0801 Q19
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)(10)

BFN Unit 0	FIRE EMERGENCY PROCEDURE	EPIP-17 Rev. 0033 Page 5 of 10

3.2 Initial Notification by Control Room Unit Operator (continued)

B. Initiate the “Fire Alarm Bell”.

BFN Unit 0	High Pressure Fire Protection System	0-OI-26 Rev. 0095 Page 10 of 66

3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. Attachment 7, Fire Protection Display Panels, 0-XI-026-0151 and 3-XI-026-0151, lists the Alarm Messages and Actions from local Fire Panels that input to Unit 1, Panel 1-9-22 and Unit 3, Panel 3-9-20.

BFN Unit 0	Fire Response	0-AOI-26-1 Rev. 0018 Page 6 of 32

4.2 Subsequent Actions (continued)

NOTES

1) The Shift Manager will remain in communication with the Incident Commander and reference 0-SSI-001 for applicability based on the severity of the fire.
2) Each Safe Shutdown Instruction contains illustrations which depict the credited plant/unit equipment and instrumentation for that specific Fire Area.
3) The AUOs are assembled in the Control Rooms to ensure SSI manual actions can be completed within the required time.
4) To ensure that in the event of a APP R fire, containment pressure is not vented below that which is needed to maintain RHR pump NPSH, maintain 1(2,3)-FIC-84-19 in normal position of Manual and “0” scfm.

[4] **IF** directed by the Unit Supervisor, **THEN PERFORM** the following:

[4.1] **NOTIFY** AUOs to report to their assigned Control Room(s), all other AUOs will report to Unit 2 MCR.

FIRE / EXPLOSION

Description	Description	
6.4-U1 TABLE	6.4-U2	UNUSUAL EVENT
Confirmed fire in ANY plant area listed in Table 6.4-U1 **AND** NOT extinguished within 15 minutes. OPERATING CONDITION: ALL	Unanticipated explosion within the protected area resulting in visible damage to ANY permanent structure or equipment. OPERATING CONDITION: ALL	
6.4-A TABLE		ALERT
Fire or explosion in ANY plant area listed in Table 6.4-A affecting safety system performance **OR** Fire or explosion causing visible damage to permanent structure of safety systems in ANY plant area listed in Table 6.4-A. OPERATING CONDITION: ALL		

3.3 Assembly / Accountability

[1] **IF** emergency circumstances warrant Assembly /Accountability, **THEN CONTINUE** in this procedure.

3.4 Site Assembly and Accountability (continued)

F. Special Conditions Concerning Assembly and Accountability

1. If an individual cannot reach their designated assembly area within 20 minutes, they should go to the nearest designated area and card their badge into the accountability reader. They should remain in that assembly area. Appendix A and Appendix B list assembly area locations.

Appendix B
(Page 1 of 1)

Emergency Responders Emergency Facility / Assembly Areas

Designated Assembly Area	Reporting Organizations
Unit 1 and 2 Control Rooms	1. All Operations personnel in Control Bays, Unit 1/2
Unit 3 Control Room	1. All Operations personnel in Control Bays, Unit 3

19. 600000 AK3.04

A fire has occurred in Unit 3 **AND** the Assistant Unit Operators (AUOs) assigned to Unit 3 have been notified to report to the Unit 3 Control Room.

Which ONE of the following identifies who is responsible for sounding the site fire alarm bell in accordance with EPIP-17, "Fire Emergency Procedure" **AND** the reason why extra AUOs (i.e. those not assigned to Unit 3) are directed to assemble in the Unit 2 Control Room in accordance with 0-AOI-26-1, "Fire Response?"

A. The Unit 3 Control Room Unit Operator; personnel accountability.

B. The Unit 1 Control Room Unit Operator; personnel accountability.

C. The Unit 3 Control Room Unit Operator; in the event SSI actions are required.

D. The Unit 1 Control Room Unit Operator; in the event SSI actions are required.

QUESTION 20 Rev 2

In accordance with 0-AOI-57-1E, Grid Instability, what is the **maximum** MVAR outgoing limit to maintain the offsite qualification of both 500-Kv and 161-Kv offsite power sources?

A. + 50

B. + 100

C. + 150

D. + 300

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	700000 AA2.05	
	Importance Rating	3.2	3.8

Generator Voltage and Electric Grid Disturbances; Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8) Operational status of offsite circuit

Justification for K/A match: To match Tier 1 and the K/A, the question sets up conditions where the plant is in the Grid Instability Abnormal Operating Instruction, 0-AOI-57-1E, then asks the straight forward question about the MVAR **maximum** outgoing limit, to match the part of the K/A about the operational status of offsite circuits.

Explanation: CORRECT D: In accordance with 0-AOI-57-1E [6] **IF** grid instability is characterized by system voltage being maintained outside the normal limits of 520 ±10 KV, **THEN PERFORM** the following steps: [6.2.1] **RAISE** reactive power to system voltage returns to 510 KV **OR UNTIL** Generator Reactive Power reaches +300 MVAR.

A. **Incorrect because –**This is the top of the outgoing MVAR band IAW OI-47 for manual voltage regulation.
Plausible because this is a NOTE in the Generator Synchronization and loading section of OI-47, Turbine-Generator System.

B. **Incorrect because –**This is the value for the upper limit if ODMI-944201 is in place and is not addressed in 0-AOI-57-1E.
Plausible in that Operating outside this range raises the possibility of rotor coil distance blocks loosening, which could cause Turbine damage according to ODMI-944201.

C. **Incorrect because –** This is the value for the Automatic Generator Voltage Regulator electronic limit for incoming MVAR (-150MVAR).
Plausible in that this is an operating limit for the incoming MVAR. This limit is intended to prevent the possibility of the Generator slipping a pole.

Technical Reference(s): 0-AOI-57-1E Rev 11, 1-OI-47 Rev 48

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.134, MAIN GENERATOR AND EXCITER Obj. 9

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:	41(b)(10)	

1.0 PURPOSE

This abnormal operating instruction provides guidance for responding to a 500KV Grid Instability condition. This can consist of either grid system frequency deviating from the normal standard of 60 Hertz or an inability of the system to maintain the normal 500KV voltage schedule of 520 ±10kv at the BFN site.

NOTES

3) A 300 Mvar maximum outgoing limit applies to all three units for both 500-Kv and 161-Kv offsite power source qualification.

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:

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

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

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

Appendix A
(Page 2 of 4)
Grid Stability Background and Basis

2.0 STABILITY

A. Grid instability can be caused by a failure in breaker protection schemes that result in loss of transmission lines and an inability of the transmission system to maintain either the required voltage or required system frequency due to current system loading and generating/distributing capacities. During off-normal grid system conditions a reduction in system frequency will occur concurrent with a reduction in system voltage. Conversely a rise in system frequency will result in a coincident rise in grid system voltage.

B. Unit stability is very sensitive to the gross MVAR output level of the generators. The less reactive power they generate the more the stability margin is reduced.

C. Following a system perturbation, if the BFN units are stable, the power swings will be sufficiently damped and the BFN turbine generators return to synchronous speed. If the BFN units are unstable, the power swings are **NOT** sufficiently damped to prevent the uncontrolled acceleration of the BFN turbine generators and their pulling out of step with the system. When this occurs the BFN units will be automatically tripped by protective relaying. During lightly loaded conditions, the stability margins of the turbine generators are reduced because there is less load for damping power swings due to disturbances. Also, the BFN units may be generating at low gross MVAR levels and very likely may be absorbing MVARs in order to control system voltages. This causes the BFN units to be less stable due to low magnetic flux levels in the field windings for damping.

D. To assist in maintaining grid stability, the Load Coordinator may request BFN to raise the incoming reactive power (-MVAR/leading) for each operating unit. Incoming MVARs inherently reduce the individual stability of the turbine generator system whereas outgoing reactive power (+MVAR/lagging) raises it. Normal turbine generator system incoming reactive power is limited to -150 MVARs in automatic voltage regulator operation. Manual voltage regulator operation beyond this limit is allowed only as directed by the Load Coordinator during transmission system disruption contingencies.

5.5 Generator Synchronization and Loading (continued)

NOTES

1) Unless otherwise directed, the normal band for Generator MVARS should be 0 + or - 50 MVARs while regulation is in manual.
2) When operated in Automatic, the Generator Voltage Regulator has an electronic limit which limits incoming reactive, at full load, to approximately 150 MVAR. When operated in Manual, there is an administrative limit of 150 MVAR. These limits are intended to prevent the possibility of the Generator slipping a pole.
3) See Section 8.11.1 for operation with the voltage regulator in Manual.

CAUTION

IF ODMI-944201 is in place, it is desired to maintain generator reactive loading between -100 and +100 MVARs. Operating outside this range raises the possibility of rotor coil distance blocks loosening, which could cause Turbine damage.

QUESTION 21 Rev 0

Unit 3 is operating at 100% power when all three Reactor Feed Pump Turbines trip.

The Reactor is manually scrammed and 3-EOI-1 is entered.

As Reactor water level lowers, which one of the following alarms, if valid would be the **first one to require** the OATC to trip the Reactor Recirc pumps?

ASSUME NO AUTOMATIC actions have occurred.

A. 3-9-5A window 8, REACTOR WATER LEVEL ABNORMAL

B. 3-9-3C window 3, REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE

C. 3-9-3F window 29, RX WTR LVL LOW LOW HPCI/RCIC INIT

D. 3-9-3C window 28, RX WTR LVL LOW LOW LOW ECCS/ESF INIT

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295009 G2.4.45	
	Importance Rating	4.1	4.3

Low Reactor Water Level; Ability to prioritize and interpret the significance of each annunciator or alarm

Justification for K/A match: To match the Tier and K/A the question sets up conditions of low reactor water level (Abnormal Water Level Condition), and asks the candidate to evaluate the Annunciators to interpret when the OATC is to trip the Reactor Recirculation pumps

Explanation: CORRECT C: 3-9-F window 29 alarms at (-) 45 inches Reactor water level. In accordance with 3-OI-68 step 3.8.B.2 the RPT breakers automatically open at (-) 45 inches Reactor water level tripping both Recirc pumps. In accordance with OPDP-1 section 3.3.5.A If an automatic control or an automatic action is confirmed to have malfunctioned, take prompt actions to place that control in manual or to accomplish the desired function.

A. **Incorrect because –** The OATC should manually runback the Recirc pumps however, tripping the pumps is not required at this Reactor water level.
Plausible because when this alarm comes in with a RFPT tripped a Recirc pump automatic runback of both Recirc pumps should have initiated.

B. **Incorrect because –** Tripping the pumps is not required at this Reactor water level.
Plausible in that the setpoint for this alarm is (+) 2 inches Reactor Water level which is an EOI-1 entry condition and would require a manual scram. Note EOI-1A would require tripping the Recirc pumps if Reactor power is > 5%.

D. **Incorrect because –** This would not be the first alarm to indicate that the Recirc pumps should be tripped.
Plausible that the candidate may recognize that the pumps do not have adequate NPSH at this level however, the pumps should be tripped at (-) 45 inches Reactor Water level when they failed to automatically trip.

Technical Reference(s): 3-OI-68 Rev 88, 3-ARP-9-5A Rev 45, 3-ARP-9-3C Rev 29, 3-ARP-9-3F Rev 29, OPDP-1 Rev 34.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.003 R20 OBJ 11H

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:	41(b)(7)

REACTOR WATER LEVEL ABNORMAL 3-LA-3-53	8

(Page 1 of 1)

Sensor/Trip Point:

RFW Control System

3-RLY-003-0053LL	≤ 27 inches
3-RLY-003-0053LH	≥ 39 inches

Average of valid Narrow Range level signals from RFW Control System. Signals originate from 3-LT-3-53, 3-60, 3-206, 3-253.

Sensor Location:
3-RLY-003-0053LL in Panel 3-9-97
3-RLY-003-0053LH in Panel 3-9-97
3-LT-3-53 on Panel 25-5B (Reactor Bldg. El 593', R17-T)
3-LT-3-60 on Panel 25-6B (Reactor Bldg El 593', R19-Q)
3-LT-3-206 on Panel 3-925-0426 (Reactor Bldg 593',R17-T)
3-LT-3-253 on Panel 25-6D (Reactor Bldg 593", R19-Q))

Probable Cause:
A. Reactor water level high or low.
B. Malfunction of sensor.

Automatic Action:
None
(Reactor scram on low level at +2.0 in)
(Main Turbine/RFPT trip and subsequent scram if ≥ 30% Reactor Power on high level at +55 in.)

Operator Action:
A. **CHECK** Reactor water level hi/low using multiple indications including Average Narrow Range Level on 3-XR-3-53 recorder, 3-LI-3-53, 3-LI-3-60, 3-LI-3-206, and 3-LI-3-253 on Panel 3-9-5. ☐
B. **IF** alarm is valid, **THEN** **REFER TO** 3-AOI-3-1 or 3-OI-3. ☐
C. **IF** 3-LI-3-53, 3-LI-3-60, 3-LI-3-206, and 3-LI-3-253 has failed or is invalid, **THEN** with SRO permission, **BYPASS** the affected level instrument. **REFER TO** 3-OI-3, Section 8.2. ☐

References:

3-45E620-6	3-729E895-10	3-47E610-46-1
3-729E895-1	3-AOI-3-1	3-OI-3

REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE	3

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-184	RPV level
LIS-3-185	≤ +2.0 inches

Sensor Location:	LIS-3-184 Panel 9-81 Aux Inst. Rm	LIS-3-185 Panel 9-82 Aux Inst. Rm

Probable Cause:

A. SI/SR is in progress.
B. Low reactor water level (Level 3).
C. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** Reactor water level by multiple indications. ☐
B. **DISPATCH** personnel to Aux Instrument Rm EI 593, to check relays energized:
   1. Panel 9-30, Relay 2E-K29 ☐
   2. Panel 9-33, Relay 2E-K24 ☐

C. **REFER TO** Tech Spec Section 3.3.5.1 and 3.5.1. ☐

References:

3-47E610-3-1 3-45E620-2 47W600-57 and 58 3-730E929 -1 and 2
Technical Specifications 3.5.1 and 3.3.5.1
Technical Specifications Bases 3.3.5.1

RX WTR LVL LOW LOW HPCI/RCIC INIT 3-LA-3-58B	29

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-58A	-45”
LIS-3-58B	-45”
LIS-3-58C	-45”
LIS-3-58D	-45”

Sensor Location:	LIS-3-58A, 58B Aux Inst Rm Panel 9-81	LIS-3-58C, 58D Aux Inst Rm Panel 9-82

Probable Cause:

A. Low RPV water level (Level 2).
B. SI/SR in progress.
C. Sensor malfunction.

Automatic Action: Automatic initiation of HPCI and RCIC.

Operator Action:

A. **CHECK** RPV water level using multiple indications. ☐
B. **REFER TO** the EOIs. ☐

References:

3-45E670-14, -19	3-45E620-1	3-47E610-3 -1
3-730E928-2, -3, and -4	Technical Specification 3.3.5.1	

RX WTR LVL LOW LOW LOW ECCS/ESF INIT 3-LA-3-58A	28

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-58A, B, C and D ≤ -122 inches (RPV low-low-low level)(Level 1)

Sensor Location: Panel 9-81, 82 Aux Instrument Room

Probable Cause:
A. Reactor Water Low Level
B. SI/SR in progress.

Automatic Action: (One out-of-two taken twice logic)

A. The following receive auto start signals:
- Core Spray System
- RHR System LPCI Mode
- Diesel Generators
- RHRSW (EECW) pump

B. ADS blowdown logic input.

Operator Action:
A. **VERIFY** RPV water level using multiple indications. ☐
B. **REFER TO** the EOIs. ☐

References:

3-45E620-2	3-47E610-3-1	47W600-57 and 58
GE 730E930-13, -14 and 19	45E670-13 and 19	

3.8 Electrical

B. A complete list of Recirc System trip functions is provided in Illustration 4. The RPT breakers between the recirc drives and pump motors will open on any of the following:

1. Reactor dome Pressure of 1148 psig (ATWS/RPT). (Both pressure switches in Logic A or both pressure switches in Logic B will cause RPT breakers to trip both pumps.) (2 out of 2 taken once logic)

2. Reactor Water Level -45" (ATWS/RPT). (Both level switches in Logic A or both level switches in Level B will cause RPT breakers to trip both pumps.) (2 out of 2 taken once logic)

3.3.5 Manual Control of Automatic Systems

A. If an automatic control or an automatic action is confirmed to have malfunctioned, take prompt actions to place that control in manual or to accomplish the desired function.

8.4 Resetting Recirc Pump Runback

Recirc Pump runback (both pumps) will occur on any of the following signals:

Total feedwater flow ≤ 19 percent (15 second time delay). (Indicated by annunciators RECIRC LOOP A FLOW LIMITER ENFORCING and RECIRC LOOP B FLOW LIMITER ENFORCING)

Any individual RFP flow is < 19 percent and Reactor water level ≤ 27 inches. (Indicated by annunciators and amber light above 3-HS-68-32 and 3-HS-68-41) or a Reactor scram occurs.

QUESTION 22 Rev 0

U2 is operating at 100% Power when a transient results in fuel cladding failure. The following sequence of event occurs:

07:50 2-9-4C window 34 OG POST TRTMT RADIATION HIGH-HIGH alarms
08:00 2-9-4C window 35 OG POST TRTMT RAD MONITOR Hi-Hi-Hi/INOP alarms and will **NOT** reset.
08:05 The Automatic and Immediate actions of 2-AOI-66-2 were completed.

At 12:00 (NOON) Which one of the following describes the expected system response?

Unit 2 Offgas flow to the 6 Hour Holdup volume will be __ (1) __ the 0800 value.

The indication on 0-RM-90-147/148 Stack Gas RAD Monitors will be __ (2) __ than the values at 0800.

A. (1) lower than
(2) lower

B. (1) lower than
(2) higher

C. (1) the same as
(2) lower

D. (1) the same as
(2) higher

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295017 AA1.06	
	Importance Rating	3.2	3.2

High Off-Site Release Rate; Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE : (CFR: 41.7 / 45.6) Condenser air removal system:

Justification for K/A match: To match the Tier and K/A, the question gives that A transient results in fuel cladding failure and asks the operator to determine the status of the off-gas system.

Explanation: CORRECT A: When 2-9-4C window 35 alarms 2-FCV-66-28 Offgas isolation valve will close. With the Offgas flow isolated indicated off-gas flow to the 6 hour holdup volume will lower with the steam jets in service due to increasing back pressure. When the Unit 2 Offgas flow to the stack is isolated the indication on the Stack gas Rad Monitors will lower but will continue to indicate based on flow from sources other than U2 Offgas.

B. **Incorrect because –** Stack Gas Rad monitor readings will lower due to closure of 2-FCV-66-28.
Plausible because Part 1 is correct and because there is a rising trend in Rad levels given in the stem of the question.

C. **Incorrect because –** Flow to the 6 hour holdup volume will lower.
Plausible if the candidate does not remember that Offgas Post Treatment Rad Hi-Hi-Hi isolates 2-FCV-66-28. Part 2 is correct even if 2-FCV-66-28 remains open due to Operators inserting a scram on Unit 2 IAW 2-AOI-66-2.

D. **Incorrect because –** See A above and because the Stack Gas Rad monitor readings will lower due to closure of 2-FCV-66-28.
Plausible see A above. Part 2 is plausible if the candidate does not remember the 2-FCV-66-28 will isolate because there is a rising trend in Rad levels given in the stem of the question.

Technical Reference(s): 2-ARP-9-4C Rev 32; 2-AOI-66-2 Rev 21

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): N/A

Question Source:	Bank:	
	Modified Bank:	X
	New:	
Question History:	Previous NRC:	Perry 2007-2 Q25
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(7)	

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. ☐

Continued on Next Page

OG POST TRTMT RAD MONITOR HI-HI-HI/INOP 2-RA-90-265C	35

(Page 1 of 1)

Sensor/Trip Point:

2-RM-90-265A	6.2×10^5 cps
2-RM-90-266A	6.2×10^5 cps

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

Probable Cause:
A. Resin trap failure (RWCU or Condensate demins).
B. Fuel damage.

Automatic Action: OFFGAS SYSTEM ISOLATION VALVE 2-FCV-66-28 closes after a 5 second time delay

Operator Action:

A. **VERIFY** alarm condition 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** OFF-GAS SYSTEM ISOLATION VALVE, 2-FCV-66-28 has the Mechanical Restraint **DISENGAGED** and 2-FCV-66-28 is CLOSED. ☐

C. **REFER TO** 2-AOI-66-2. ☐

References:
2-45E620-4 2-45E614-2 2-47E610-90-2 2-115D6410RE-3
GE 2-729E814-6
FSAR Sections 1.6.4.4.6, 7.12.2.2, 7.12.2.3, 7.12.3.3, 9.5.4, and 13.6.2

3.0 AUTOMATIC ACTIONS

B. OFFGAS SYSTEM ISOLATION VALVE, 2-FCV-66-28, automatically closes on any combination of Off Gas Post Treatment Hi Hi Hi, downscale, or inop simultaneously in both channels of the O.G. post treatment radiation monitoring system after 5 seconds. 2-FCV-066-0028 will not perform it's design function to automatically close, when it is mechanically restrained open due to plant conditions.

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** scram has not occurred, **THEN PERFORM** the following:

[1.1] **IF** core flow is above 60%, **THEN REDUCE** core flow to between 50-60%.

[1.2] **MANUALLY SCRAM** the Reactor. (Reference 2-AOI-100-1).

Perry 2007-2 KA-295017AA1.06

QUESTION RO 25

Fuel element failure is indicated by increasing plant radiation levels.

MAIN STEAM LINE RADIATION HIGH alarm is received for all Main Steam Line Radiation Monitors.

MAIN STEAM LINE RADIATION HI HI/INOP alarm is received for Main Steam Line Radiation Monitors A and B.

Which one of the following receives a close signal?

A. Off-Gas Discharge Isolation Valve, N64-F632

B. Reactor Water Sample Isolation Valve, B33-F019

C. Main Steam Line Isolation Valves, B21-F022A-D and B21-F028A-D

D. Mechanical Vacuum Pump Suction Valves, N62-F130A and N62-F130B

BFN 1501

Q 18

Which of the following alarms, if valid, requires an immediate operator action to manually scram the Reactor, to limit Off-Site Release Rate?

A. OG POST TREATMENT RADIATION HI-HI-HI/INOP, 2-9-4C Window 35

B. OG PRE-TREATMENT RADIATION HIGH, 2-9-3A Window 5

C. STACK GAS RAD HIGH, 2-9-3A Window 13

D. OG AVG ANNUAL RELEASE LIMIT EXCEEDED, 2-9-4C Window 27

Answer: A

QUESTION 23 Rev 1

Due to an error while performing surveillance testing on Unit 2 a Group 6 isolation is initiated and all three trains of SGT started.

In accordance with 2-AOI-64-2D, Group 6 Ventilation System Isolation, which one of the following completes the statements below?

Operators are directed to MONITOR _____ closely while Reactor Zone fans are out of service to help determine if the Steam Vault Exhaust Booster Fan is in Service.

A. Reactor Zone exhaust flow on Panel 2-9-25

B. Leak detection temperatures on Panel 2-9-21

C. SGT Total flow on Panel 2-9-20

D. Steam tunnel temperature on Panel 2-9-3

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295020AK2.11	
	Importance Rating	3.2	3.4

Inadvertent Containment Isolation; Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following: Standby gas treatment system/FRVS

Justification for K/A match: To match the Tier and K/A the question establishes conditions that have resulted in an inadvertent isolation and then asks the candidate to determine which observations are required to avoid a PCIS group 1 isolation. Note this question was written to avoid overlap with question #11.

Explanation: CORRECT D: 2-AOI-64-2D step 4.2[3] directs: **MONITOR** Steam tunnel temperature closely while Reactor Zone fans are out of service. Step 4.2[3.1] states: **IF** Steam tunnel temperature rises, **THEN VERIFIES** STEAM VAULT EXH BOOSTER FAN is in service. The caution on the same page states: MSIV's may isolate on a Group I isolation if the time the Reactor Zone fans are removed from service is **NOT** minimized during Reactor power operation and the Steam Vault Exhaust Booster Fan is **NOT** in service. When auto initiated SGT takes suction on the Reactor Zone via the Reactor Zone crosstie damper however SGT flow by its self is not sufficient to prevent the steam tunnel temperature from rising to the group 1 isolation setpoint.

A. **Incorrect because –** 2-AOI-64-2D step 4.2[3] directs: **MONITOR** Steam tunnel temperature closely while Reactor Zone fans are out of service.
Plausible because – SGT takes suction on the Reactor Zone via the Reactor Zone crosstie damper and Reactor Zone flow is what normally cools the Steam Tunnel.

B. **Incorrect because** – 2-AOI-64-2D step 4.2[3] directs: **MONITOR** Steam tunnel temperature closely while Reactor Zone fans are out of service.
Plausible because – This panel is monitored in step 4.2[6] to aid in determining the location of the problem.

C. **Incorrect because** – 2-AOI-64-2D step 4.2[3] directs: **MONITOR** Steam tunnel temperature closely while Reactor Zone fans are out of service.
Plausible because – SGT Total flow is indicated on Panel 2-9-20 and because when auto initiated SGT takes suction on the Reactor Zone via the Reactor Zone crosstie damper . Reactor Zone flow is what normally cools the Steam Tunnel.

Technical Reference(s): 2-AOI-64-2d Rev 33, 0-OI-65 Rev 55

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): N/A

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:	41(b)(7)

CAUTION

MSIV's may isolate on a Group I isolation if the time the Reactor Zone fans are removed from service is **NOT** minimized during Reactor power operation and the Steam Vault Exhaust Booster Fan is **NOT** in service. Steam tunnel temperature should be closely monitored while Reactor Zone fans are out of service.

[3] **MONITOR** Steam tunnel temperature closely while Reactor Zone fans are out of service.

[3.1] **IF** Steam tunnel temperature rises, **THEN VERIFY** STEAM VAULT EXH BOOSTER FAN is in service. REFER TO 2-OI-30B.

[6] **MONITOR** the following to aid in determining the location of the problem:

- AREA RADIATION, 2-RR-90-1 (Panel 2-9-2).
- AIR PARTICULATE MONITOR CONSOLE, 2-MON-90-50 (Panel 2-9-2).
- LEAK DETECTION SYSTEM, TI-69-29 (Panel 2-9-21).

6.0 SYSTEM OPERATIONS

[1] **MONITOR** system operating parameters as follows:

C. SGT Total Flow- sum of 0-FI-65-50B/1(2)(3) and 0-FI-65-71B/1(2)(3). 🕒 **PANEL 1-9-20 PANEL 2-9-20 PANEL 3-9-20**

QUESTION 24 Rev 0

Unit 1 Suppression Pool Level is + 5.5 inches and CST level is 558 feet.

Which ONE of the following completes the statements below?

HPCI Suction __ (1) __ automatically transfer to the Suppression Pool.

RCIC Suction __ (2) __ automatically transfer to the Suppression Pool.

A. (1) will
(2) will

B. (1) will
(2) will **NOT**

C. (1) will **NOT**
(2) will

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

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295029 EK2.02	
	Importance Rating	3.4	3.6

High Suppression Pool Water Level; Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: (CFR: 41.7 / 45.8) HPCI:

Justification for K/A match: To match the Tier 1 aspect and K/A the conditions that are set up in the question deal with abnormal suppression pool water level and its effect on the HPCI System.

Explanation: CORRECT B: Part 1 correct – HPCI Suction automatically swaps to suppression pool on high suppression pool level +5.25" or low CST level Elev <552'6". Part 2 correct - RCIC has no automatic transfer from CST to torus.

A. **Incorrect because -** RCIC does not have an auto swap of its suction path.
Plausible because – 1-OI-71 directs manually swapping the RCIC suction if HPCI suction auto swaps.

C. **Incorrect because -** HPCI does have a suction transfer on high Torus level at 5.25 inches.
Plausible because – The candidate may remember that one of these systems has an auto swap of its suction and the other does not and may chose the incorrect one.

D. **Incorrect because –** HPCI does have a suction transfer on high Torus level at 5.25 inches.
Plausible because - The value given is very close to the high torus level setpoint for HPCI suction transfer and the candidate may not recall the exact setpoint and because RCIC does not have an auto swap of its suction.

Technical Reference(s): OPL171.040 Rev 23; OPL171.042 Rev 20; 1OI-73 Rev25.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.040 obj 6, OPL 171.042 obj 4

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:	41(b)(7)

3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. When any of the following signals are received, HPCI SUPPR POOL OUTBD SUCT VLV, 1-FCV-073-0027, and HPCI SUPPR POOL INBD SUCT VALVE, 1-FCV-073-0026 automatically open, unless a HPCI isolation signal is present.

1. Suppression Pool Level High at +5.25 in.

2. HPCI Pump Suction Condensate Header Level Low at approximately 7000 gallons (El. 552'6" on 1-LS-073-0056A and 1-LS-073-0056B)

6.1 Normal Operation (continued)

[6] **MAINTAIN** Suppression Pool level between -5.5 and -2 in. **REFER TO** 1-OI-74.
[7] **MONITOR** CST 1 LEVEL, 1-LI-2-169A, on Panel 1-9-6.
[8] **IF** ANY of the following conditions occur while RCIC is injecting into the RPV:

- Suppression Pool level reaches +7 in., **OR**
- HPCI suction auto transfers to suppression pool, **OR**
- RCIC Turbine trips on pump low suction pressure, **THEN**

PERFORM the following steps to transfer RCIC Suction to the Suppression Pool:

[8.1] **OPEN** 1-FCV-071-0017 using RCIC SUPPR POOL INBD SUCT VALVE, 1-HS-71-17A.
[8.2] **OPEN** 1-FCV-071-0018 using RCIC SUPPR POOL OUTBD SUCT VALVE, 1-HS-71-18A.
[8.3] **WHEN** 1-FCV-071-0017 AND 1-FCV-071-0018, are fully open, **THEN VERIFY CLOSED** 1-FCV-071-0019 using RCIC CST SUCTION VALVE, 1-HS-71-19A.

Modified Question BFN ILT 1102 Written Exam

24. 295029 EK2.02

Unit 1 Suppression Pool Level is (+) 5 inches and CST level is 550'.
Which ONE of the following completes the statements below?

HPCI Suction __**(1)**__ automatically transfer to the Suppression Pool.

RCIC Suction __**(2)**__ automatically transfer to the Suppression Pool.

A. **(1)** will
(2) will

B. (1) will
(2) will **NOT**

C. **(1)** will **NOT**
(2) will

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

QUESTION 25 Rev 0

U2 is operating at 100% Power when 2-9-3D window 24 MAIN STEAM LINE LEAK DETECTION TEMP HIGH alarms.

The BOP Operator reports that 2-TIS-1-60A; MN STEAM TUNNEL TEMP indicates 162 °F and rising.

Which one of the following completes the statements below?

An EOI-3 entry condition __ (1) __ been met.

The MSIV closure setpoint for the Steam Tunnel temperature is __ (2) __ °F.

A. (1) has
   (2) 189

B. (1) has
   (2) 315

C. (1) has **NOT**
   (2) 189

D. (1) has **NOT**
   (2) 315

Answer: A

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

High Secondary Containment Area Temperature; Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13) Area temperature

Justification for K/A match: To match the Tier and K/A this is an abnormal (High) temperature in the Secondary Containment (Main Steam Tunnel). To match the K/A, the question sets up a high temperature in SC and asks if this is an EOI entry condition (RO knowledge) and then asks the setpoint for the Group one isolation.

Explanation: CORRECT A: EOI-3 is entered when any Secondary Containment area temp above max normal value of table SC-1. SC-1 max normal value is alarmed therefore with 2-9-3D window 24 in alarm due to 2-TIS-1-60A EOI entry is required. The ARP for 2-9-3D window 24 indicates that an impending MSIV isolation will occur at 189°F.

B. **Incorrect because** – this is the wrong temperature, this is the MAX Safe Temperature for that area, not the Group one setpoint.
Plausible because it is a temperature number that a candidate may remember for action to be performed in that area, but it is the wrong one.

C. **Incorrect because** – there is an entry condition for EOI-3 Secondary Containment. It is “Any Secondary Cntmt area temp above Max Normal value of Table SC-1, looking at Table SC-1, Steam tunnel (RB) 9-3D Alarm Window 24 if in alarm is the entry condition.
Plausible because other similar alarms are **not** entry conditions into the EOIs.

D. **Incorrect because** – there is an entry condition for EOI-3 Secondary Containment. It is “Any Secondary Cntmt area temp above Max Normal value of Table SC-1, looking at Table SC-1, Steam tunnel (RB) 9-3D Alarm Window 24 if in alarm is the entry condition. Also this is the wrong temperature, This is the MAX Safe Temperature for that area, not the Group one setpoint.
Plausible because other similar alarms are **not** entry conditions into the EOIs. Also because it is a temperature number that a candidate may remember for action to be performed in that area, but it is the wrong one.

Technical Reference(s): 2-ARP-9-3D Rev 29, 2-EOI-3 Rev 16.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.204 OBJ V.B.2, OPL 171.016 OBJ 19.a

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:	41(b)(10)	

MAIN STEAM LINE LEAK DETECTION TEMP HIGH 2-TA-1-60	24

(Page 1 of 2)

Sensor/Trip Point:

2-TIS-1-60A 160°F

2-TS-1-60B through D

Sensor Location: Panel 2-9-3, Main Control Room, El 617', Panel 2-9-21, Main Control Rm, El 617'

Probable Cause:

A. Main Steam, RWCU, Feedwater, RCIC, or HPCI Disch. (Only with HPCI in service and elevated Suppression Pool water Temp.) Line Break.
B. Turb or Rx Bldg cooling/ventilation out of service.
C. Sensor malfunction.
D. Steam Vault Exhaust Booster Fan out of service.

Automatic Action: Impending MSIV Isolation at 189°F area temp.

Operator Action:

NOTE

The following steps may be performed in any order or concurrently as necessary.

A. **CHECK** the following temperature indications: ☐
- MN STEAM TUNNEL TEMP temperature indicator, 2-TIS-1-60A on Panel 2-9-3. ☐
- Temperature Switches 2-TS-1-60B, -60C, or -60D window(s) on Panel 2-9-21. ☐
- RWCU Piping in the Main Steam Tunnel temperature indicators, 2-TIS-69-834A(B)(C)(D), Auxiliary Instrument Room Panels 2-9-83(84)(85)(86) OR ICS'HPTURB' mimic. ☐

B. **CHECK** the following flow indications: ☐
- MAIN STEAM LINE FLOW A(B)(C)(D), 2-FI-46-1(2)(3)(4) on Panel 2-9-5 ☐
- RFW FLOW LINE A(B), 2-FI-3-78A(78B) on Panel 2-9-5. ☐
- RFP 2A(2B)(2C) flow indicators, 2-FI-3-20(13)(6) on Panel 2-9-6. ☐

C. **IF** RCIC is **NOT** in service **AND** 2-FI-71-1A(B), RCIC STEAM FLOW indicates flow, **THEN**
ISOLATE RCIC and **VERIFY** temperatures lowering. ☐

Continued on Next Page

MAIN STEAM LINE LEAK DETECTION TEMP HIGH 2-TA-1-60, Window 24
(Page 2 of 2)

Operator Action: (Continued)

D. **CHECK** for elevated RAD Levels on the following instruments:
- 2-RM-90-20, CRD-HCU West. ☐
- 2-RM-90-29, Suppression Pool. ☐

E. **IF** HPCI is injecting with elevated Suppression Pool Temperature, **THEN**
CONSIDER securing HPCI to determine if it is the source of the leak. ☐

F. **IF** Rx Bldg main steam tunnel temperature is above 160°F on 2-TIS-1-60A on Panel 2-9-3, **THEN**
PERFORM the following: ☐
1. **ENTER** 2-EOI-3 Flowchart. ☐
2. **VERIFY** Rx Zone fans, 2-HS-64-11A at Panel 2-9-25, in fast speed. ☐
3. **VERIFY** Steam Vault Exhaust Booster Fan in service and **REFER TO** 3-OI-30B. ☐

G. **IF** Turb Bldg main steam tunnel temperature is above 160°F on 2-TS-1-60B, -60C, or -60D on Panel 2-9-21, **THEN**
DISPATCH personnel to 480V AC Turb Bldg Vent Bd 2A (Turb Bldg, El 617' T-9.5 M-LINE) to verify TB fans and the Mechanical Spaces Exhaust Fan running. ☐

H. **REFER TO** Tech Spec 3.3.1.1, 3.3.6.1. ☐

References: 2-45E620-2 2-47E610-1-1 GE 2-920D351
Tech Spec 3.3.1.1
Tech Spec 3.3.6.1

2-EOI-3 SECONDARY CONTAINMENT CONTROL

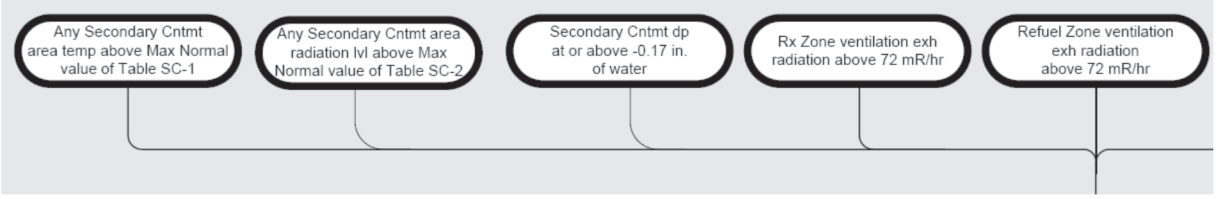


Table SC-1
Secondary Cntmt Area Temp

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	150	FCV-74-47, 48
RHR sys II pumps	XA-55-3E-4	74-95B	Alarmed	210	FCV-74-47, 48
HPCI room	XA-55-3F-10	73-55A	Alarmed	270	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 XA-55-3F-10 XA-55-3E-4	71-41B, C, D 73-55B, C, D 74-95G, H	Alarmed Alarmed Alarmed	200 240 240	FCV-71, 2, 3 FCV-73-2, 3, 81 FCV-74-47, 48
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	170	FCV-74-47, 48
RB el 565 W (RWCU pipe trench)	XA-55-5B-32 (Panel 9-5) XA-55-5B-33 (Panel 9-5)	69-835A, B, C, D (Aux Inst room)	Alarmed	170	FCV-69-1, 2, 12
RWCU hx room	XA-55-3D-17	69-29F, G, H	Alarmed	220	FCV-69-1, 2, 12
RWCU pump A	XA-55-3D-17	69-29D	Alarmed	215	FCV-69-1, 2, 12
RWCU pump B	XA-55-3D-17	69-29E	Alarmed	215	FCV-69-1, 2, 12
RB el 593	XA-55-3E-4	74-95C, D	Alarmed	195	FCV-74-47, 48
RB el 621	XA-55-3E-4	74-95F	Alarmed	155	FCV-43-13, 14

QUESTION 26 Rev 0

Unit 1 is in Mode 5, Units 2 and 3 are in Mode 1.

A Refueling accident occurs on Unit 1 resulting in U1 Reactor and Refuel Zone Ventilation Radiation Monitors reading:

- Reactor Zone 1-RM-90-142A indicates 65mr/hr
- Reactor Zone 1-RM-90-142B indicates 67mr/hr
- Refuel Zone 1-RM-90-140A indicates 75mr/hr
- Refuel Zone 1-RM-90-140B indicates 78mr/hr

- Reactor Zone 1-RM-90-143A indicates 68mr/hr
- Reactor Zone 1-RM-90-143B indicates down scale
- Refuel Zone 1-RM-90-141A indicates 70mr/hr
- Refuel Zone 1-RM-90-141B indicates 69mr/hr

Which one of the following completes the statement below?

Based on these conditions the Control Room Operators are protected from the Release of Radioactive material by _____.

A. Refuel Zone isolation only

B. Reactor and Refuel Zone isolation

C. Reactor Zone isolation and CREV auto initiation

D. Refuel Zone isolation and CREV auto initiation

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295034EK1.01	
	Importance Rating	3.8	4.1

Secondary Containment Ventilation High Radiation; Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: (CFR: 41.8 to 41.10) Personnel protection

Justification for K/A match: This question matches the Tier and K/A in that it involves secondary containment ventilation high radiation conditions indicative of an emergency. To match the second part of the K/A about personnel protection, the question asks how the Control Room Operators are protected from the Release of Radioactive material.

Explanation: CORRECT D: With the Refuel Zone ventilation Radiation monitors reading above 72mr/hr a group 6 PCIS isolation of the Refuel Zone occurs. In addition to the Refuel Zone isolation the CREV system will initiate to pressurize the control rooms and to filter the control room's air supply.

A. **Incorrect because –** Refuel Zone will isolate and CREV will initiate.
Plausible that the candidate may forget that the Refuel Zone Radiation Monitors also initiate CREV. The Control Room Radiation monitors (CREV CAMs) 0-RM-90-259A and B initiate CREV and a Reactor Zone isolation initiates CREV and isolates the Refuel Zone however, a Refuel Zone isolation does not isolate the Reactor Zone.

B. **Incorrect because –** A refuel Zone isolation does not isolate the Reactor Zone.
Plausible if the candidate thinks a Refuel Zone isolation will also isolate the Reactor Zone ventilation but forgets that CREV will initiate.

C. **Incorrect because –** A refuel Zone isolation does not isolate the Reactor Zone.
Plausible in that CREV will auto initiate however, the Reactor Zone will not isolate with one detector down scale or from a Refuel Zone isolation.

Technical Reference(s): 1-OI-90 rev 68, 0-OI-31 Rev 145, OPL 171.067 Rev 18.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.067 OBJ B.2.j

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: BFN 1205 Q #26
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(9)

3.0 PRECAUTIONS AND LIMITATIONS

A. The following Radiation Monitoring subsystems initiate the listed automatic actions and isolations on high radiation trip signals:

3. Refueling Zone Ventilation (72 mr/hr high radiation signal from 2 out of 2 taken once logic or downscale/inop signal from 1 out of 2 taken twice logic.)
   a. Standby Gas Treatment System auto start.
   b. Refueling Zone Vent System isolation.
   c. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)

4. Reactor Zone Ventilation (72 mr/hr high radiation signal from 2 out of 2 taken once logic or downscale/inop signal from 1 out of 2 taken twice logic.
   a. Group 6 Isolation.
   b. Standby Gas Treatment System auto start.
   c. Refueling Zone Ventilation isolation.
   d. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)

6.7 NUMAC Radiation Monitor Operation

NOTES

3) There are two detectors for each channel of the Reactor Zone/Refuel Zone monitors and are indicated on each monitor as follows:

1-RM-90-140/142

Display	Description
CH 2A	CH 2A RX ZONE DET A, 1-RE-90-142A
CH 2B	CH 2B RX ZONE DET B, 1-RE-90-142B
CH 0A	CH 0A REFUEL ZONE DET A, 1-RE-90-140A
CH 0B	CH 0B REFUEL ZONE DET B, 1-RE-90-140B

1-RM-90-141/143

Display	Description
CH 3A	CH 3A RX ZONE DET A, 1-RE-90-143A
CH 3B	CH 3B RX ZONE DET B, 1-RE-90-143B
CH 1A	CH 1A REFUEL ZONE DET A, 1-RE-90-141A
CH 1B	CH 1B REFUEL ZONE DET B, 1-RE-90-141B

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.

BFN **Unit 0**	**Control Bay and Off-Gas Treatment Building Air Conditioning System**	**0-OI-31** **Rev. 0145** **Page 99 of 258**

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)

BFN 1205 QUESTION **26 KA** 295034 EK1.02

With all 3 Units at 100% power the following occurs on Unit 3:

• RWCU Backwash Receiving Tank overflows during a backwash of RWCU Filter Demin.
• 3-RI-90-13A, NORTH CLEAN-UP SYSTEM RB El 593 reads 500 mr/hr
• 3-RI-90-14A, SOUTH CLEAN-UP SYSTEM RB El 593 reads 600 mr/hr
• 3-RI-90-9, CLEAN-UP SYSTEM RB El 621 reads 250 mr/hr
• RX BLDG AREA RADIATION HIGH is in alarm (3-9-3A, window 22)
• 3-RM-90-142, RX ZONE EXH CH A RAD RTMR is reading 85 mrem/hr on BOTH channels
• 3-RM-90-143, RX ZONE EXH CH B RAD RTMR is reading 85 mrem/hr on BOTH channels

Which ONE of the following indentifies the plant impact to these conditions?

A. Control Room pressure remains unchanged and Plant Stack flow rate lowers.

B. Control Room pressure remains unchanged and Plant Stack flow rate rises.

C. Control Room pressure becomes more positive and Plant Stack flow rate lowers.

D. Control Room pressure becomes more positive and Plant Stack flow rate rises.

ANSWER D

QUESTION 27 Rev 3

The Radwaste Operator reports that Unit 1 Reactor Building Floor Drain Sump B level is 50 inches and rising with the B Sump pump running.

Which one of the following completes the statements below?

When Reactor Building Floor Drain Sump B level rises **an additional** __ (1) __ inches Entry into EOI-3, Secondary Containment Control is required.

In accordance with the EOI Program Manual Section 0-V-E, EOI-3 Secondary Containment Control Bases, the reason for isolating a system that is discharging into Secondary Containment is to __ (2) __.

A. (1) 16
   (2) terminate the radioactivity release

B. (1) 16
   (2) prevent the spread of contamination

C. (1) 25
   (2) terminate the radioactivity release

D. (1) 25
   (2) prevent the spread of contamination

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295036 EK3.03	
	Importance Rating	3.5	3.6

Sec Cont High Sump/Area Wtr Lvl; Knowledge of the reasons for the following responses as they apply to SEC CONT HIGH SUMP/AREA WTR LVL: 41.5 Isolating affected systems

Justification for K/A match: This question matches the Tier and K/A of an emergency condition (i.e. entry into Emergency Operating Instructions) for secondary containment water level. To future match the K/A concerning the reason for isolating affected system, the question asks specifically what the program manual states as the reason for the isolation.

Explanation: Correct A: Table SC-3 indicates that the max normal floor drain sump level is 66 inches which corresponds to the sump abnormal alarm and the EOI entry condition. IAW the EOIPM all systems found to be discharging into the area, with the exceptions noted, are isolated to terminate the radioactivity release.

B. **Incorrect because** – IAW the EOIPM the leaking system is isolated to terminate the radioactivity release.
Plausible because – Part 1 is correct and RCI-26 states that: Minimizing the radioactive contamination of areas is one of 4 objectives used to assure that Browns Ferry fully complies with the Nuclear Regulatory Commission's regulatory exposure requirements.

C. **Incorrect because** – The EOI entry condition is a Floor Drain level of 66 inches (50+16=66)
Plausible because – 50+25=75 inches which corresponds with the Rx BLDG Equipment Drain Sump level abnormal alarm.

D. **Incorrect because** – The EOI entry condition is a Floor Drain level of 66 inches (50+16=66) and IAW the EOIPM the leaking system is isolated to terminate the radioactivity release.
Plausible because – 50+25=75 inches which corresponds with the Rx BLDG Equipment Drain Sump level abnormal alarm.

Technical Reference(s): EOIPM Section 0-V-E Rev 3.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.204 R7 OBJ V.B.1

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(10)

LA-77-8B RX BLDG FLR DR SUMP B ABNORMAL 33	Sensor/Trip Point:
	1-LT-077-0008B — Elevation 516' 6"
	2-LT-077-0008B — 66" from bottom of sump
	3-LT-077-0008B
	0-PLC-077-0008B — Excessive leakage

(Page 1 of 2)

Sensor Location: Rx Bldg Floor Drain Sump
Elevation 519 near RHR pump Rm B & D
Reactor Bldg

Probable Cause:
A. Excessive in-leakage.
B. Pump(s) and/or valve malfunction.
C. Valve misalignment or discharge check valve on idle pump stuck open.
D. Level transmitter Malfunction.

Automatic Action: Sump pump starts at Elevation 515.

Operator Action:

A. **IDENTIFY** which unit's sump is in alarm and which condition is responsible for alarm (Excessive leakage or High-High level) by use of PLC HMI Work Station. ☐

B. **IF** high inleakage is the cause of the alarm, **THEN NOTIFY** appropriate Unit Supervisor and request assistance to **LOCATE/STOP** source of excess water. (**REFER TO** print 1-47E852-1, 2-47E852-1, 3-47E852-1). ☐

C. **IF** high level in the sump is the cause of the alarm, **THEN PERFORM** the following:
1. **CHECK** RUNNING or **START** Pump B. ☐
2. **NOTIFY** affected unit Unit Supervisor of an EOI-3 entry condition alarm being sealed in on Radwaste Control Panel (High-High level is an EOI-3 entry condition). ☐
3. **REQUEST** operator assistance from the appropriate Unit Operator to determine cause of alarm by checking the following: (steps a through c can be performed in any logical order and may be suspended when the cause of the High Level is discovered):
   a. Sump local level indicator LI-077-0008BB and pump coupling. ☐
   b. Sump pump B breaker (480V RMOV BD B breaker 1B). ☐
   c. Valve alignment. **REFER TO** OI-77. ☐

Continued on Next Page

LA-77-17 RX BLDG EQUIP DR SUMP ABNORMAL	4

(Page 1 of 1)

Sensor/Trip Point:

1-LT-077-0017	Elevation 514'
2-LT-077-0017	75" from bottom of sump'
3-LT-077-0017	
0-PLC-077-0017	Excessive leakage

Sensor Location: Rx Bldg Equip Sump, Elevation 519
Reactor Bldg

Probable Cause:
A. Excessive in-leakage.
B. Pump(s) and/or valve malfunction.
C. Valve misalignment or discharge check valve of idle pump stuck open.
D. Level Transmitter malfunction.

Automatic Action: Second sump pump starts if due to HI LEVEL, if due to EXCESSIVE INLEAKAGE no auto actions.

Operator Action:

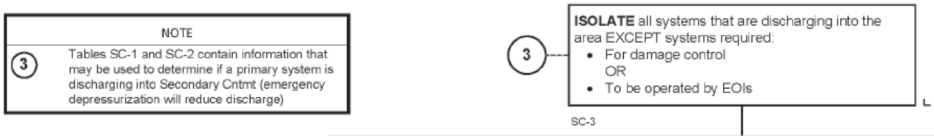
A. **IDENTIFY** which unit's sump is in alarm by use of PLC HMI Work Station. ☐
B. **IDENTIFY** which condition is causing alarm (EXCESSIVE INLEAKAGE) or (HI LEVEL). ☐
C. **IF** EXCESSIVE INLEAKAGE, **THEN**
INVESTIGATE the change in leak rate to sump. ☐
D. **IF** HI LEVEL, **THEN**
CHECK OPEN valve 1(2)(3)-FCV-077-0017B. ☐
1. **CHECK** RUNNING or **START** both A and B sump pumps. ☐
2. **CHECK** for any associated alarm(s) or abnormal indications:
   - Sump high temp alarm REACTOR BLDG EQUIP SUMP TEMP HIGH (XA-55-17A, Window 3). ☐
   - Loss of power indication(s) (pump/valve red & green light extinguished). ☐
3. **REQUEST** operator assistance from the appropriate UO to determine cause of alarm by checking:
   - Pump coupling and sump level on 1(2)(3)-LI-77-17B. ☐
   - Sump pump A 480V RMOV BD C breaker 4B position. ☐
   - Sump pump B 480V RMOV BD B breaker 18B position. ☐
4. **NOTIFY** supervisor and **LOCATE/STOP** source of excess water (see print 1-47E852-2, 2-47E852-2, 3-47E852-2). ☐
5. **LOG** valid events and actions taken in NOMS narrative log. ☐

References:

0-45E779-16	0-47E610-77-1	791E492-10	1-47E852-2
2-47E852-2	3-47E852-2	730E934-20,21,22	

1.0 EOI-3, SECONDARY CONTAINMENT CONTROL BASES (continued)

SC-3



1.0 EOI-3, SECONDARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SC-3

A secondary parameter in Table SC-1, SC-2 or SC-3 above its maximum normal operating value is an indication of a primary system leak, either directly into the area or indirectly through secondary systems. All systems found to be discharging into the area, with the exceptions noted, are isolated to terminate the radioactivity release.

Continued operation of damage control systems required to protect personnel, plant structures, or essential equipment is permitted since these objectives generally take precedence over control of secondary containment area Max Normal levels.

Examples may include fire suppression systems, alternative spent fuel pool makeup and cooling methods, and portable sprays being used to scrub unisolable radioactivity releases.

The objectives of other EOI flowcharts are given higher priority than the objectives of Secondary Containment Control. Systems that must be operated to support the performance of other steps of the EOIs therefore need not be isolated here.

Systems “required” for damage control or other EOI functions include not only those explicitly identified in other EOI steps but also auxiliaries that may be needed to permit continued use of those systems, support the accomplishment of overall EOI objectives, or avoid further degradation of plant conditions. The determination of whether continued operation of a system is “required” necessitates a judgment based on the nature of the event and current plant conditions. The benefits of continued operation must be balanced against potential loss of essential equipment and area access that could further complicate efforts to stabilize the plant. For example, if loss of condenser vacuum could result in loss of preferred injection systems or increase the offsite radioactivity release rate, systems needed to maintain condenser vacuum may remain in operation.

1.0 PURPOSE

C. The fundamental purpose of the Radiation Protection program is to assure that Browns Ferry fully complies with the Nuclear Regulatory Commission's regulatory exposure requirements, which are designed to limit exposure to ionizing radiation to station employees and the general public to levels determined to pose no health or safety risks. Browns Ferry's radiation protection program consists of four objectives that are designed to address this purpose as well as to provide additional margins for radiological safety. Those areas are:

1. Maintaining radiation exposure as low as reasonably achievable
2. Maintaining positive control of radioactive material
3. Minimizing the radioactive contamination of areas, equipment and personnel
4. Minimizing the generation of radioactive waste.

QUESTION 28 Rev 0

A loss of coolant accident occurred on Unit 3
The following conditions exist:

- 3A RHR pump is running in LPCI mode
- Reactor water level has been stabilized at +15 inches
- Drywell spray was initiated using 3B RHR pump
- Drywell Temperature is 230 °F and slowly lowering
- Drywell Pressure is 6 psig and slowly lowering

Subsequently:

- 3-9-3D window 29 RHR/CS DIV I TEMP HIGH alarms
- The Reactor Building AUO reports that the 3A RHR Room Cooler is **NOT** running and would **NOT** start using the local pushbutton

Which one of the following describes the required actions and the reason for these actions?

Place RHR pump __ (1) __ in service in LPCI mode and secure 3A RHR pump in order to prevent 3A RHR pump __ (2) __ damage.

A. (1) 3C
(2) motor

B. (1) 3D
(2) motor

C. (1) 3C
(2) seal

D. (1) 3D
(2) seal

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	203000 A2.08	
	Importance Rating	2.9	3.0

Residual Heat Removal /Low Pressure Coolant Injection: Injection Mode; Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Inadequate room cooling

Justification for K/A match: Tier 2 is the systems type questions, however since it is paired with an A2 K/A (abnormal condition) and that the operation of LPCI is not a normal (non-emergency mode of RHR) this question sets up RHR in LPCI injection, then gives a high temperature alarm, with the report that the room cooler is not functioning properly (matching the K/A about inadequate room cooling) and asks what procedure steps are necessary to mitigate the impact of this condition on LPCI. Since this is a systems Tier, the second part of the question asks what damage to the RHR Pumps might occur based on this loss.

Explanation: CORRECT A: The ARP for 3-9-3D window 29 states: start a spare pump and secure the pump in alarm. Since each RHR pump has a room cooler and RHR loop 1 is aligned in LPCI mode the 3C RHR pump would be placed in service for LPCI. Securing the 3A RHR pump will prevent damage to the motor windings and bearings.

B. **Incorrect because –** This would require securing Drywell spray causing Containment parameters to degrade unnecessarily.
Plausible because – Window 29 is a DIV 1 alarm and the candidate may thing that all DIV 1 RHR pumps are affected. Aligning RHR loop II to LPCI would allow securing the 3A RHR pump. Part 2 is correct.

C. **Incorrect because –** Each RHR pump has a separate seal cooler to prevent overheating the seals.
Plausible because – the pump seals can be damaged by overheating and the RHR Pump Room Coolers are designed to maintain the ambient air temperature of the room at or below 148°F. The room cooler duct work does however blow on the RHR pump motors.

D. **Incorrect because –** Part 1 see B above. Part 2 see C above
Plausible because – Part 1 see B above. Part 2 see C above

Technical Reference(s): 3-OI-74 Rev 116, 3-ARP-9-3D Rev 28, OPL 171.044 Rev 19.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.044 R19 OBJ 10.e

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(5)

3.1 General Operating (continued)

I. The U-3 RHR Loops I & II RHR room coolers and seal heat exchangers do not have an auto backup supply of cooling water.

3.2 RHR Pumps (continued)

B. [NRC/C] The RHR pumps are considered to be operable without the seal cooler under the following conditions:

1. Always operable in the LPCI and Containment Cooling Mode.
2. During Shutdown Cooling, operable up to a suction temperature of 215°F.
3. Operable for an emergency with suction temperatures between 215°F and 400°F. Operation in this condition for more than two days will require an inspection of the seal surfaces. [NRC LER 296/83047 R1]]

4.0 PRESTARTUP/STANDBY READINESS REQUIREMENTS (continued)

[2] **VERIFY** the following support system requirements are satisfied:

Emergency Equipment Cooling Water System available to supply RHR pump room coolers. REFER TO 0-OI-67.

4. RHR Equipment Areas Room Coolers

a) The heat generated by the motors, pumps, and piping in the RHR quads is removed with air cooling units.

b) During system operation, the RHR Pump Room Coolers are designed to maintain the ambient air temperature at or below 148°F, when supplied with 95°F cooling water.

c) There is one air handling unit for each RHR pump

d) The cooled discharge from each air handling unit is ducted and directed across the RHR Pump Motor.

e) A reliable source of cooling water is provided to the RHR Pump Room Coolers by the EECW System.

f) Each room cooler that supports operation of an RHR Pump is designed to remove 405,000 Btu/hr.

g) The RHR Pump room coolers provide an important function relative to system operation:

(1) Their specific safety function is to maintain environmental cooling in the area of the pump motors <148oF.

(2) Room Coolers are addressed in the BFN TRM and have the potential to impact the operability of the RHR Pumps themselves.

h) Power supplies for the room coolers are from their respective 480VAC RMOV Board ‘A’ and 480VAC RMOV Board ‘B’

(1) ‘A’ 480V RMOV Board supplies the 1A/2A/3A Room Coolers for the ‘A’ and ‘C’ pumps

(2) ‘B’ 480V RMOV Board supplies the 1B/2B/3B Room Coolers for the ‘B’ and ‘D’ pumps

i) The room cooler will automatically start when, as previously described, auxiliary contacts inside the respective RHR Pump breaker close upon breaker closure.

j) The room cooler will also automatically start if room temperature for the room that it serves reaches 95°F.

k) The room cooler turns off at 95°F with no RHR Pump running.

l) The start function for the room coolers can be tested using the local hand switches (HS-64-68 (69/70/71), RHR PUMP MOTOR A (B/C/D) COOLER.

RHR/CS DIV I
TEMP HIGH

3-TA-56-2

29

(Page 1 of 1)

Sensor/Trip Point:

3-TR-56-2 160°F THRU 230°F

Sensor Location: Recorder is on Panel 3-9-47 in the Main Control Room.

Probable Cause: High temperature on one of the following:

RHR Pump	Motor windings 225°F, bearings 160°F.
CS Pump	Motor windings 225°F, bearings 230°F.

Automatic Action: None

Operator Action:

A. **IDENTIFY** the alarm source at Panel 3-9-47. ☐

B. **DISPATCH** personnel to the affected equipment to investigate the problem. ☐

C. **IF** temperature continues to rise, **THEN** **START** a spare pump or fan and **SHUT DOWN** the pump or fan in alarm. ☐

D. **VERIFY** room coolers are working and are in service. ☐

References:

3-45E620-2	3-47E610-75-1	3-47E2610-74-1
3-47E610-70-1	0-47E610-23-2 and 3	3-47E610-24-2

Tech Specs 3.5.1, ECCS - Operating
3.5.2, ECCS - Shutdown
3.6.2.3, Residual Heat Removal (RHR) Suppression Pool Cooling
3.6.2.4, Residual Heat Removal (RHR) Suppression Pool Spray
3.6.2.5, Residual Heat Removal (RHR) Drywell Spray

TRM 3.5.3, Equipment Area Coolers,
3.3.3.2, Low Pressure ECCS Area Cooler Instrumentation

TR 3.5 EMERGENCY CORE COOLING SYSTEMS

TR 3.5.3 Equipment Area Coolers

LCO 3.5.3 The equipment area cooler associated with each RHR pump and the equipment area cooler associated with each set of Core Spray pumps (A and C or B and D) must be OPERABLE at all times when the pump or pumps served by that specific cooler is considered to be OPERABLE.

APPLICABILITY: Whenever the associated subsystem is required to be OPERABLE

QUESTION 29 Rev 0

What is the power supply for RHR SYS II INBD INJECTION VLV, 2-FCV-74-67?

480 V RMOV Board…

A. 2A

B. 2B

C. 2D

D. 2E

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	205000 K2.02	
	Importance Rating	2.5	2.7

Shutdown Cooling System (RHR Shutdown Cooling Mode); Knowledge of electrical power supplies to the following: (CFR: 41.7) Motor operated valves

Justification for K/A match: To match the Tier 2 aspect of this K/A concerning RHR Shutdown Cooling Mode power supplies to motor operated valves, I chose an RHR Shutdown Cooling Valve and asked what is the power supply is to its motor.

Explanation: CORRECT D: 2-BKR-074-0067 RHR SYS II INBD INJECTION VLV 2-FCV-74-67 is located on 480 V RMOV Board 2E. As seen on the electrical lineup Attachment 3 of Electrical Lineup Checklist 2-OI-74/ATT-3 Rev. 0140 on Page 15 of 15.

A. **Incorrect because –** this is not the power supply to the 2-FCV-74-67 valve motor. **Plausible** because 480 V RMOV Bd 2A is the power supply to 2-BKR-074-0052 RHR OUTBOARD VALVE 2-FCV-74-52, outboard versus inboard.

B. **Incorrect because –** this is not the power to the 2-FCV-74-67 valve motor. **Plausible** because 480 V RMOV Bd 2B is the power supply to 2-BKR-074-0066 RHR OUTBOARD VALVE 2-FCV-74-66, outboard versus inboard.

C. **Incorrect because –** this is not the power to the 2-FCV-74-67 valve motor. **Plausible** because 480 V RMOV Bd 2D is the power supply to 2-BKR-074-0053 RHR SYS I INBD INJECTION VLV 2-FCV-74-53, different division, but on RHR system.

Technical Reference(s): 2-OI-74/ATT 3 Electrical Lineup Checklist Rev 140

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.044 Obj 3f.

Question Source:	Bank: Modified Bank: NRC 08-01 Q 29 New:
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41(b) 7

Reactor Bldg - 480V RMOV Bd 2E - El 621'

2C	2-BKR-074-0067 RHR SYS II INBD INJECTION VLV FCV-74-67	ON	____ ____

Reactor Bldg - 480V RMOV Bd 2D - El 593'

2C	2-BKR-074-0053 RHR SYS I INBD INJECTION VLV FCV-74-53	ON	____ ____

Control Bay - 480V RMOV Bd 2A - El 621'

2B	2-BKR-074-0052 RHR OUTBOARD VALVE FCV-74-52	ON	____ ____

Control Bay - 480V RMOV Bd 2B - El 593'

3A	2-BKR-074-0066 RHR OUTBOARD VALVE FCV-74-66	ON	____ ____

Modified Question 08-01 NRC Exm Question 29

HLT 0801 Written Exam

29. 205000 K2.02

Which ONE of the following completes the statement?

The power supply for RHR SYS II INBD INJECTION VLV, 2-FCV-74-67, is from 480 RMOV Board __**(1)**__.

The power supply for RHR SYS II MINIMUM FLOW VLV, 2-FCV-74-30, is from 480 RMOV Board __**(2)**__.

A. **(1)** 2B.
(2) 2B.

B. **(1)** 2B.
(2) 2E.

C. **(1)** 2E.
(2) 2B.

D. **(1)** 2E.
(2) 2E.

QUESTION 30 Rev 0

Given the following Unit 2 plant conditions:

- Reactor water level initially lowered to (-) 69 inches
- After Reactor water level was recovered to + 33 inches, HPCI was placed in pressure control in accordance with 2-EOI Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode

Subsequently:

- Condensate Storage Tank (CST) level dropped below 6800 gallons.

Which ONE of the following describes the current status of the HPCI system?

HPCI is…

A. operating in pressure control with suction from the CST.

B. pumping to the CST with suction from the Suppression Pool.

C. operating at shutoff head with suction from the Suppression Pool.

D. tripped on low suction pressure.

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	206000 K3.02	
	Importance Rating	3.8	3.8

High Pressure Coolant Injection System; Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE COOLANT INJECTION SYSTEM will have on following:(CFR: 41.7 / 45.4) Reactor pressure control

Justification for K/A match: This is a Tier 2, Systems type question, however this particular K/A deals with the affect a loss or malfunction will have on the HPCI system in Pressure Control (Emergency mode of operation) so the to keep the question a systems type question asks if you know how the system valves will respond an invalid momentary low CST level.

Explanation: CORRECT C: At the low level swap setpoint in the CST, HPCI auto swaps from CST suction to Suppression Pool (Torus) suction. When this occurs the CST Test Return Isolation valve receives a close signal from the Torus suction valves opening; to prevent pumping the Torus to the CST. Therefore, with the HPCI injection valve previously closed, HPCI would be operating at shutoff head without minimum flow protection. Note - the minimum flow valve would be closed due to lack of a valid initiation signal.

A. **Incorrect because** – HPCI suction will automatically swap to the Suppression Pool.
Plausible because – The candidate may remember that the HPCI suction swap can be initiated by high Suppression Pool level but forget that low CST level also initiates the swap

B. **Incorrect because** –The HPCI Test Valves will receive a close signal when the Torus suction valves open.
Plausible because – the candidate may forget the CST test valve interlock.

D. **Incorrect because** – HPCI will not trip on low suction pressure under this specific condition. The Torus suction valves open before the CST suction valve closes.
Plausible since closure of the suction path to HPCI would result in a low suction pressure trip and because of the reduced head pressure at the pump suction when aligned to the Suppression Pool.

Technical Reference(s): 2-OI-73 Rev 95, OPL 171.042 Rev 20

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.042 R20 OBJ V.B.5

Question Source:	Bank:	X
	Modified Bank:	
	New:	
Question History:	Previous NRC:BFN 1404Q 30	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(7)	

3.4 Initiation

B. The HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, will automatically open when system flow is at or below 900 gpm (lowering) if a system initiation signal is present, and will automatically close when system flow is at or above 1255 gpm (rising) regardless of presence of initiation signal.

3.7 Interlocks

A. When any of the following signals are received, HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27, and HPCI SUPPR POOL INBD SUCT VALVE, 2-FCV-73-26 automatically open, unless a HPCI isolation signal is present.
1. Suppression Pool Level High at +5.25 in.
2. HPCI Pump Suction Condensate Header Level Low at approximately 7000 gallons (El. 552'6" on 2-LS-73-56A and -56B).

B. When HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27 and HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26 are fully open, HPCI CST SUCTION VALVE, 2-FCV-73-40, automatically closes.

C. When either HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27, or HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26, is FULL OPEN, the HPCI/RCIC CST TEST VLV, 2-FCV-73-36, and HPCI PUMP CST TEST VLV, 2-FCV-73-35, will close.

BFN NRC Exam 1404

QUESTION 30

Given the following Unit 2 plant conditions:

- EOI-1, RPV Control, has been entered
- Reactor water level initially lowered to (-) 69 inches
- After Reactor water level was recovered to (+) 33 inches, HPCI was placed in pressure control in accordance with 2-EOI Appendix-11C, Alternate RPV Pressure Control Systems HPCI Test Mode

Subsequently:

- Condensate Storage Tank (CST) level dropped below 6800 gallons.

Which ONE of the following describes the status of the HPCI system (assume no other operator actions have occurred)?

A. HPCI would be operating in pressure control with suction from the CST.

B. HPCI would be pumping to the CST with suction from the Suppression Pool.

C. HPCI would be operating at shutoff head with suction from the Suppression Pool.

D. HPCI would trip on low suction pressure.

Correct Answer: **C**

QUESTION 31 Rev 0

In accordance with 1-OI-75, Core Spray System's Precautions and Limitations, how long can a Core Spray Pump run deadheaded with **no** Minimum Flow?

A. 0 minutes

B. 5 minutes

C. 4 hours

D. indefinitely

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001 G2.1.20	
	Importance Rating	4.6	4.6

Low Pressure Core Spray System; Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Justification for K/A match: Tier 2 Systems, tied to a generic conduct of operations K/A, to match both of these, the question asks about a precaution and limitation of the core spray system operating procedure OI-75.

Explanation: CORRECT A: 1-OI-75 section 3.3 B states how long a Core Spray Pump can be operated with minimum flow valves closed or manually isolated. There is no tolerance for a pump running without any flow indication, thus securing the pump immediately is the correct action to perform.

B. **Incorrect because** – this length of time is not allowed according to the P&L of OI-75, because in this case there is no minimum flow.
Plausible because this is a time specified in the P&L of OI-75 for running the core spray pump with reduced flows.

C. **Incorrect because** – this length of time is not allowed according to the P&L of OI-75, because in this case there is no minimum flow.
Plausible because this is a time specified in the P&L of OI-75 for running the core spray pump with reduced flows.

D. **Incorrect because** – this length of time is not allowed according to the P&L of OI-75, because in this case there is no minimum flow.
Plausible because this is a time specified in the P&L of OI-75 for running the core spray pump with reduced flows.

Technical Reference(s): 1-OI-75 rev 30

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.045 R17 OBJ 6

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)(10)

3.3 Equipment

A. Core Spray pump motor full-load current is 80 amps. This corresponds to rated flow conditions.

B. Core Spray pumps may be operated continuously on minimum flow without any impact to pump performance as long as the minimum flow is open. Minimum flow lines are orificed to pass about 20% rated pump flow. When operating pumps with the minimum flow valves closed or manually isolated, the following limits must be observed:

1. Core Spray pumps may be operated continuously for up to 5 minutes between 150 and 300 gpm per pump per loop. Operating Core Spray pumps at this flow rate greater than 5 minutes may cause pump degradation.

2. Core Spray pumps may be operated continuously for up to 4 hours between 300 and 600 gpm per pump per loop. Operating Core Spray pumps at this flow rate greater than 4 hours may cause pump degradation.

3. Core Spray pumps may be operated continuously at greater than 600 gpm per pump per loop with no time limitations.

QUESTION 32 Rev 1

What is the difference in time to inject Hot Shutdown Boron Weight by Standby Liquid Control (SLC) if **one squib primer fails** to fire compared to **both squib primers firing** during injection?

A. the same

B. twice as long

C. three times as long

D. four times as long

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000 K3.01	
	Importance Rating	4.3	4.4

Standby Liquid Control System; Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: (CFR: 41.7 / 45.4) Ability to shutdown the reactor in certain conditions

Justification for K/A match: Tier 2 systems question for the SLC system and how a loss or malfunction will affect the ability to shut down the reactor. To match the K/A, a question was designed to have the system injecting to shut down the reactor (ATWS) (Emergency condition) but ask a design question which requires the candidates to recall system configurations, to determine the amount of time to pump in the boron to shut down the reactor.

Explanation: CORRECT A: Each squib valve contains a primer subassembly. Each subassembly contains redundant (two) primers and firing circuits for high reliability. Both circuits (two individual primers) fire on a pump start. A squib firing circuit failure will not prevent the other squib firing circuit in the subassembly from opening the squib valve. Either squib valve is capable of 100% flow. RPV pressure and SLC pressure parameters are normal for injection.

B. **Incorrect because** – no matter which valve fires, it is designed to pass 100% flow.
Plausible if only one squib primer fired the second parallel path would be closed therefore it is conceivable, with only one path, that twice the time is required to inject Hot S/D Boron Weight.

C. **Incorrect because** – no matter which valve fires, it is designed to pass 100% flow.
Plausible if only one squib primer, of the four, fired it is conceivable that depending on the arrangement of the primers to Squib valves that each squib has a plunger to open its squib valve, and due to the redundancy of this safety system, four separate flow paths were possible the time required to inject Hot S/D Boron Weight would be three times longer.

D. **Incorrect because** – no matter which valve fires, it is designed to pass 100% flow.
Plausible if only one squib primer, of the four, fired it is conceivable that depending on the arrangement of the primers to Squib valves that each squib has a plunger to open its squib valve, and due to the redundancy of this safety system, four separate flow paths were possible that with the reduction of a presumed limited flow path could establish a much higher head to pump against and therefore quadrupling the time to achieve Hot S/D Boron Weight.

Technical Reference(s): OPL171.039 Rev 17, 1- EOI APPENDIX 3A Rev 0

Proposed references to be provided to applicants during examination: None

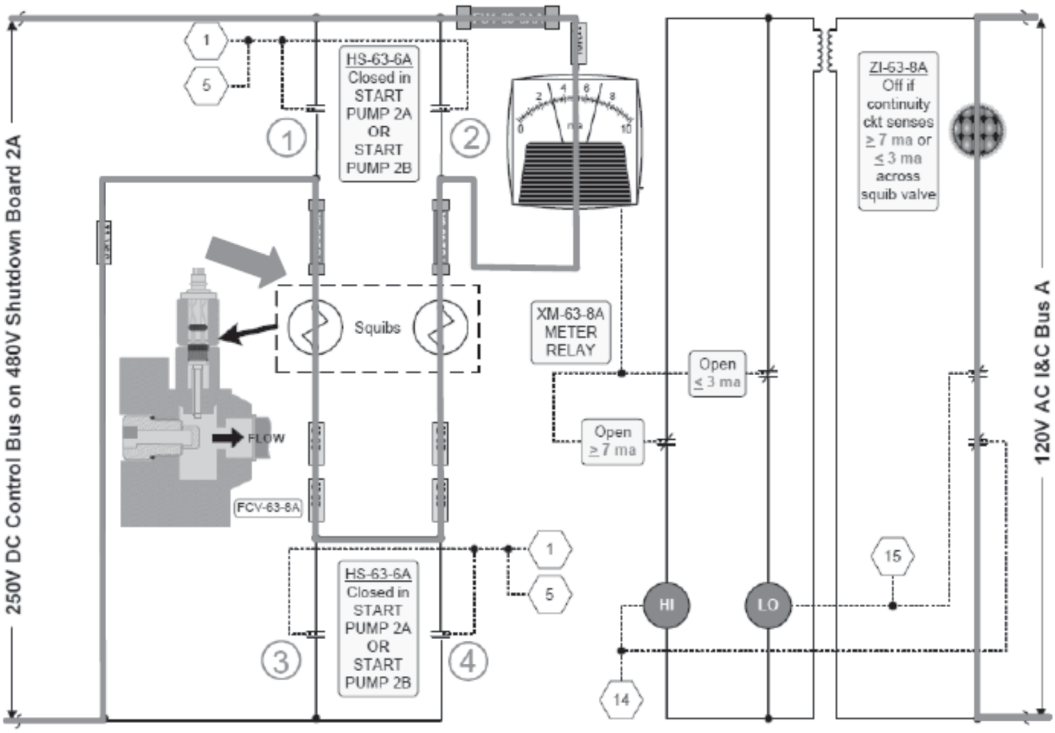
Learning Objective (As available): OPL 171.039 R17 ILT OBJ 5.d

Question Source:	Bank:	X
	Modified Bank:	
	New:	
Question History:	Previous NRC: BFN 1306 Q 33	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	41(b)(6)	

OPL171.039

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.



BFN 1306

QUESTION 33

Given the following conditions:

- A Unit 1 ATWS occurred
- During the performance of 1-EOI-Appendix 3A, SLC INJECTION, the Standby Liquid Control (SLC) pump control switch was placed in the START-A position
- RPV pressure is 1020 psig
- SLC discharge pressure is 1100 psig
- SQUIB VALVE A CONTINUITY blue light is illuminated
- SQUIB VALVE B CONTINUITY blue light is extinguished
- SLC SQUIB VALVE CONTINUITY LOST(Panel 1-9-5B, Window 20) is in alarm
- SLC INJECTION FLOW TO REACTOR (Panel 1-9-5B, Window 14) is in alarm

Which ONE of the following completes the statements below?

The SLC Squib valve ___(1)___ is OPEN.

The time to inject Hot Shutdown Boron Weight is ___(2)___ compared to the time with both squib valves open.

A. (1) B
(2) the same

B. (1) A
(2) longer

C. (1) B
(2) longer

D. (1) A
(2) the same

Answer: **A**

QUESTION 33 Rev 0

Unit 2 is operating at 40% Power.
2-SR-3.1.4.1, SCRAM Insertion Times, is in progress.

- Control Rod **26-43** has been selected and withdrawn to position 48.
- The UO in the Aux Instrument Room **inadvertently** places the SCRAM switch for Control Rod **22-43** which had been previously tested in the down position on Panel 2-9-16.

Which one of the following completes the statements below while the SCRAM switch is in the down position?

The blue SCRAM light on the Full Core Display for Control Rod **22-43** will be __ (1) __.

The Control Rod position indication on the Full Core Display for Control Rod **22-43** will have a __ (1) __ background.

A. (1) illuminated
   (2) red

B. (1) extinguished
   (2) red

C. (1) illuminated
   (2) green

D. (1) extinguished
   (2) green

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000 A4.13	
	Importance Rating	3.4	3.6

Reactor Protection System; Ability to manually operate and/or monitor in the control room:(CFR: 41.7 / 45.5 to 45.8) †Perform individual control rod SCRAM testing

Justification for K/A match: To match this K/A, the question sets up the conditions for Control Rod Scram Time Testing in accordance with the surveillance. To future match it, the question asks what the indications are for that scrammed rod in the control room.

Explanation: CORRECT C: When the individual SCRAM switch for Control Rod 22-43 is placed in the down position it will interrupt power to the SCRAM pilot valve solenoids, SCRAMMING the individual rod. When both scram valves open the blue scram valve light and the full in green light on the Rod Status Display on Panel 9-5 will illuminate.

A. **Incorrect because** – The red full out lit will extinguish when the rod scrams full in.
Plausible if the candidate thinks that the scram light indicates a scram signal for that rod but that the scram did not occur because the Control Rod was not selected (Control Rod 26-43 was selected and ready to test). Misconception of the interrelationship between the rod select matrix and the scram test panel.

B. **Incorrect because** – The blue scram valve light will illuminate when both scram valves open.
Plausible if the candidate thinks the Control Rod will not individually SCRAM unless it is selected, and the Control Rod is at position 48 (position of 22-43 not given in the stem).

D. **Incorrect because** – The blue scram valve light will illuminate when both scram valves open.
Plausible if the candidate thinks the Control Rod will not individually SCRAM unless it is selected and that the rod was already inserted (position of 22-43 not given in the stem).

Technical Reference(s): OPL 171.028 Rev 19, OPL 171.005 Rev 18, 2-SR-3.1.4.1 Rev 32.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.028 R 19 OBJ 13.f

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: 2009 Quad Cities Q33
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(7)

OPL171.028, Reactor Protection System, Revision 19 page 25

G. Individual Rod SCRAM Test Switch

1. Purpose - To provide individual rod SCRAM testing capability independent of RPS trip action
2. Function description
   a. Toggle switches on Panel 9-16 interrupt power to the SCRAM pilot valve solenoids, SCRAMMING individual rods.
   b. One switch per control rod trips both the Channel A and B SCRAM pilot solenoids for one drive.
3. To individually SCRAM a control rod, the toggle switch must be lowered. At the completion of the operation, the switch should be returned to the up position.

OPL171.005, Control Rod Drive (CRD) Hydraulics , Rev. 18 page 34

5) Scram inlet and outlet valves

f) Position indication is provided by means of spring-mounted position switches. When both valves open, position switches causing a blue rod scram light to be illuminated on the Rod Status Display, Panel 9-5.

OPL171.005, Control Rod Drive (CRD) Hydraulics , Rev. 18 page 42

6. Scram
   a. Following a scram, but before the SDV is full, the control rod will be in an overtravel-in position since there will still be a large differential pressure across the piston.
   b. Therefore, the green (full in) light on Panel 9-5 will be on but there will be no rod position readout displayed.
   c. After the SDV is full, there will be no differential pressure across the piston, and rod will settle into the 00 position.

EXAMINATION ANSWER KEY

U.S. Nuclear Regulatory Commission 2009 SRO Written Exam (Quad Cities)

33 ID: QDC.ILT.15465 Points: 1.00

Given the following:

- Unit 1 is at 75% power
- Control rod F-8 selected on the Rod Select Matrix
- Control Rod N-10 is at position 48

If the NSO places the Individual Rod Scram Test Switch on the 901-16 panel for Control Rod N-10 to the extreme UP position and leaves it there, which of the following indications will result ten (10) seconds after switch manipulation?

Indication for Control Rod N-10 on the Full Core Display will show a...

A. green 00 and the individual blue scram light will be lit.

B. green double-dash and the individual blue scram light will be lit.

C. red 48 and eight white pilot solenoid lights on the 901-5 panel will be lit.

D. red 48 and four white pilot solenoid lights on the 901-5 panel will be extinguished.

Answer: B

Answer Explanation:

Answer: The individual scram switch de-energizes both sides to scram that rod only. The blue scram light will be lit when the scram inlet and outlet valves are open (sensed by limit switches).

Distractor 1 is incorrect: Plausible if the candidate believes that the control rod will scram to position 00.
Distractor 2 is incorrect: Plausible if the candidate believes that the rod will not scram if it is not selected on the Rod Select Matrix.
Distractor 3 is incorrect: Plausible if the candidate believes that only a half scram for the control rod will result.

Reference: QCOS 0300-23 Rev 3, LIC-0280 Rev 11
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 2 Group: 1

Question Source: Modified from QDC.ILT.608712
Question History: N/A

10 CFR Part 55 Content: 41.7

QUESTION 34 Rev 0

A reactor plant startup is being conducted on Unit 2 in accordance with GOI-100-1A, Unit Startup and Power Operation.

- The reactor is critical and SRM/IRM overlap data has just been completed.
- All SRMs are reading between 5.0×10^3 and 1.0×10^4 cps.
- All IRMs are on mid scale on range 1.
- The operator has inadvertently selected both the SRMs and the IRMs for withdraw.

Which one of the following Control Rod Blocks will be the **first** automatic protective action to occur as the detectors are withdrawn?

A. SRM Detector Wrong position

B. IRM Detector Wrong position

C. SRM Downscale

D. IRM Downscale

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215003 K5.03	
	Importance Rating	3.0	3.1

Intermediate Range Monitor System; Knowledge of the operational implications of the following concepts as they apply to IRM SYS: (CFR: 41.5 / 45.3) Changing detector position.

Justification for K/A match: Tier 2 Systems question on IRMs to match K/A the question as set up with both the SRMs and IRMs being moved and asks to determine what will cause a rod block first, to test the knowledge of the operational implications of mis-positioning the IRMs now.

Explanation: CORRECT B: When the withdraw function is selected the first action will be the movement of the SRM and IRM detectors from the core. Immediately the IRMs will be detected as not fully inserted. This will generate a Detector Wrong Position rod block which is active whenever the mode switch is not in run.

A: **Incorrect because –** This would not be the first Rod block received.
Plausible in that this rod block will eventually occur however, it would require the SRMs to be moved significantly to lower their counts from 5 $X10^3$ and $1X10^4$ cps to less than 145 cps to produce the rod block.

C: **Incorrect because –** this is not the first condition that will cause a rod block.
Plausible in that this rod block may eventually occur however, it would require the SRMs to be moved significantly to lower their counts from 5 $X10^3$ and $1X10^4$ cps to less than 5 cps to produce the rod block.

D: **Incorrect because –** this will not cause a rod block because the IRMs are on range one, which bypasses the rod block.
Plausible in that IRMs will eventually go downscale, and would cause a Rod block if not on Range 1.

Technical Reference(s): 2-OI-92 Rev 22, 2-OI-92A Rev 29, OPL 171.019 Rev 13

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.019 R13 OBJ V.B.7 and V.B.8

OPL 171.020 R11 OBJ V.B.5

Question Source:	Bank:	
	Modified Bank:	X
Question History:	Previous NRC:	Nine Mile 2 2010
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	41(b)(2)

Illustration 1
(Page 1 of 1)

SRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
SRM High	≥ 6.8 X 10^4 counts per second	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Downscale	5 counts per second	Rod block unless IRMs on range 3 (or higher) or REACTOR MODE SWITCH in RUN
SRM Detector Wrong Position	145 counts per second	Rod block unless detector full-in, IRMs on range 3 (or higher), or REACTOR MODE SWITCH in RUN
SRM High-High	≥ 2 x 10^5 counts per second	Scram if shorting links removed

SRM OPL171.019 Revision 13 Page 23

(2) A Detector Wrong Position rod block will be generated if:

(a) Either SRM in that channel is not in the fully inserted position, AND

(b) The count rate on the associated SRM is less than or equal to 145 cps, AND

(c) Any of the four IRMs in that channel are not on range 3 or above, AND

(d) The reactor mode switch is not in RUN.

Illustration 1
(Page 1 of 1)

IRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
IRM High	> 90 ON 125 SCALE	Rod block unless REACTOR MODE SWITCH in RUN
IRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless REACTOR MODE SWITCH in RUN Reactor Scram unless REACTOR MODE SWITCH in RUN
IRM Downscale	< 7.5 on 125 SCALE	Rod block unless IRMs on range 1 unless REACTOR MODE SWITCH in RUN
IRM Detector Wrong Position	detector not full in	Rod block unless detector full-in, or REACTOR MODE SWITCH in RUN
IRM High-High	> 116.4 ON 125 SCALE	Reactor Scram unless REACTOR MODE SWITCH in RUN

Nine Mile 2 NRC Exam 2010

Question: RO #10

A reactor plant startup is being conducted in accordance with N2-0P-101A.

- The reactor is critical and SRM/IRM overlap data has just been completed.
- All SRMs are reading between 5 X 10E4 and 1 X 10E5 cps
- All IRMs are on mid scale on range 1
- The operator has selected both the SRMs and the IRMs for withdraw.

Which one of the following will be the first automatic protective action as the detectors are withdrawn?

A. SRM INOP trip

B. IRM Downscale rod block

C. SRM Downscale rod block

D. IRM Detector NOT fully inserted rod block

7.5.4.3 Power Generation Evaluation

Examination of the sensitivity of the SRM detectors (paragraph 7.5.4.2.3) and their operating ranges of 10^6 cps indicates that the IRMS is on scale before the SRM reaches full-scale (see Figure 7.5-25). Further overlap is provided by retraction of the SRM chambers to any position between full-in and full-out. SRM detector retraction is possible without the occurrence of a rod block only if the indicated SRM count rate remains above the rod block trip level (10_2 cps), or if the IRM has been ranged to the third or any less sensitive (higher) IRM range.

QUESTION 35 Rev 0

What are the power supplies to the SRM Channels/detectors?

SRM Channels/Detectors _____.

A. A & B are powered from the A channel ± 24VDC System and C & D are powered from the B channel ± 24VDC System.

B. A & C are powered from the A channel ± 24VDC System and B & D are powered from the B channel ± 24VDC System

C. A & B are powered from Division I, 250 VDC System and C & D are powered from Division II, 250 VDC System

D. A & C are powered from Division I, 250 VDC System and B & D are powered from Division II, 250 VDC System

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215004 K2.01	
	Importance Rating	3.1	3.1

Source Range Monitor (SRM) System; Knowledge of electrical power supplies to the following: (CFR: 41.7) SRM channels/detectors

Justification for K/A match: Simple match, Source Range Monitor (SRM) System and their electrical power supplies for SRM channels/detectors

Explanation: CORRECT B: The ± 24VDC Neutron Monitoring batteries/battery chargers supply the SRM drawers and detectors. Channel A supplies SRM A and C. Channel B supplies SRM B and SRM D.

A. **Incorrect because** – this is an incorrect pairing of the A and B channels into a division.
Plausible in that the A and B / C and D arrangement holds true for some divisionalized electric boards for example A and B 4KV Shutdown boards are DIV I and C and D are Div II.

C. **Incorrect because** – this is an incorrect pairing of the A and B channels into a division. Also the 250 VDC system is **not** used for the SRMs.
Plausible that the SRMs could be supplied by 250VDC since the SRMs do have a High Voltage Power supply however this is produced within the SRM drawer and is powered from the ± 24VDC system.

D. **Incorrect because** – The 250 VDC system is **not** used for the SRMs.
Plausible that the SRMs could be supplied by 250VDC since the SRMs do have a High Voltage Power supply however this is produced within the SRM drawer and is powered from the ± 24VDC system.

Technical Reference(s): OPL171.019 Rev 13 SRMs; 2-OI-92/ATT-3 Rev 22; 0-45E702-3; 2-730E237

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.019 R13 OBJ V.B 14

Question Source:	Bank:	X
	Modified Bank:	
	New:	
Question History:	Previous NRC: Cooper 2011 Q4	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41.b (7)	

OPL171.019 Rev 13

9. Power Supplies TP-1

a. The SRM power supplies receive unregulated ±24 VDC power from the neutron monitoring battery and convert it to regulated voltages of proper magnitude for use by the SRM detectors and logic circuits.

b. Three levels of voltage regulation:

(1) Takes the ±24 volt input and reduces it to a relatively constant + 20 volts for use by the voltage regulator.

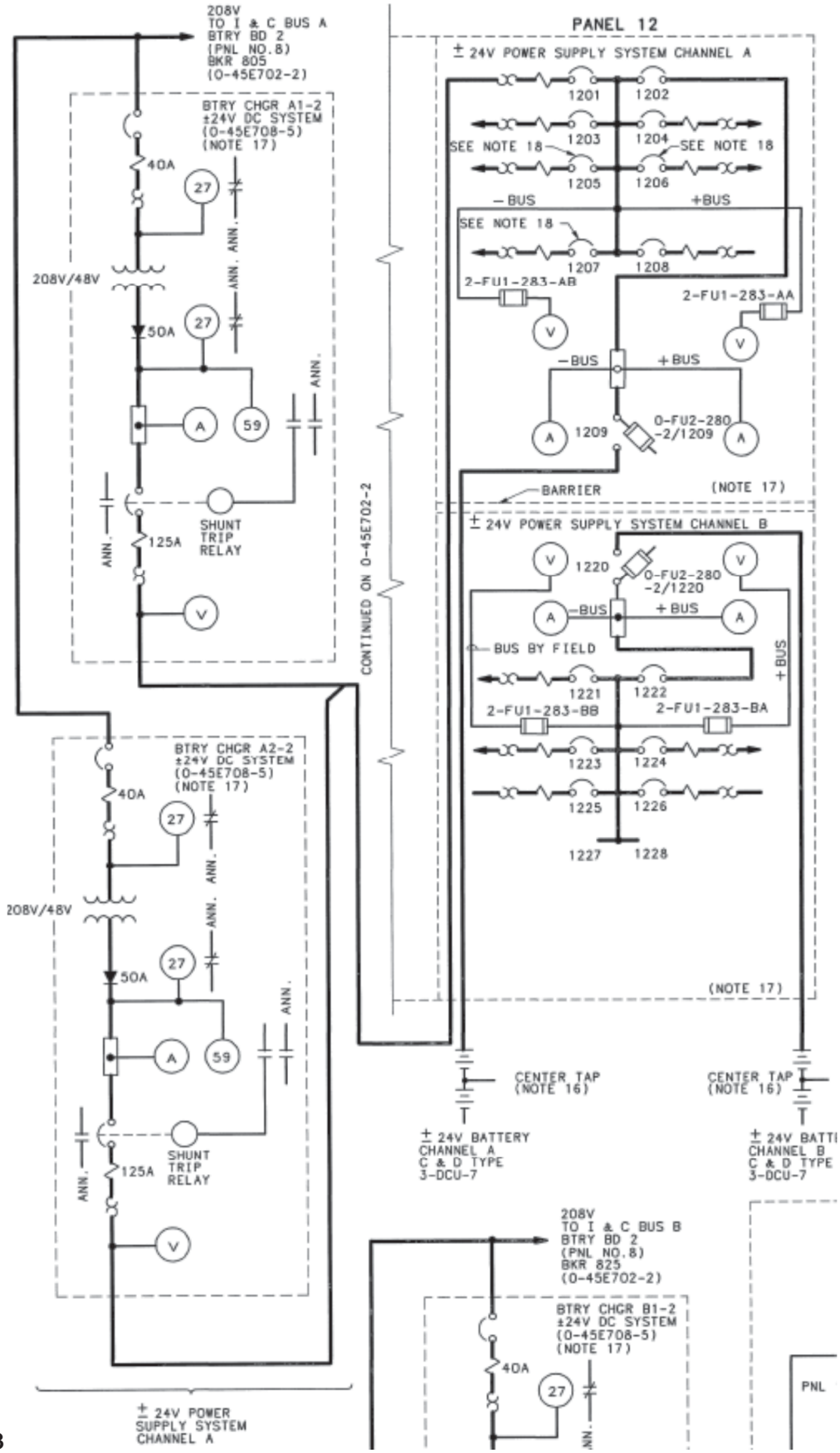
(2) Voltage regulator Takes the + 20 volt input from the pre-regulator and reduces it to a well regulated + 15 volts for use by the logic and high voltage power supply.

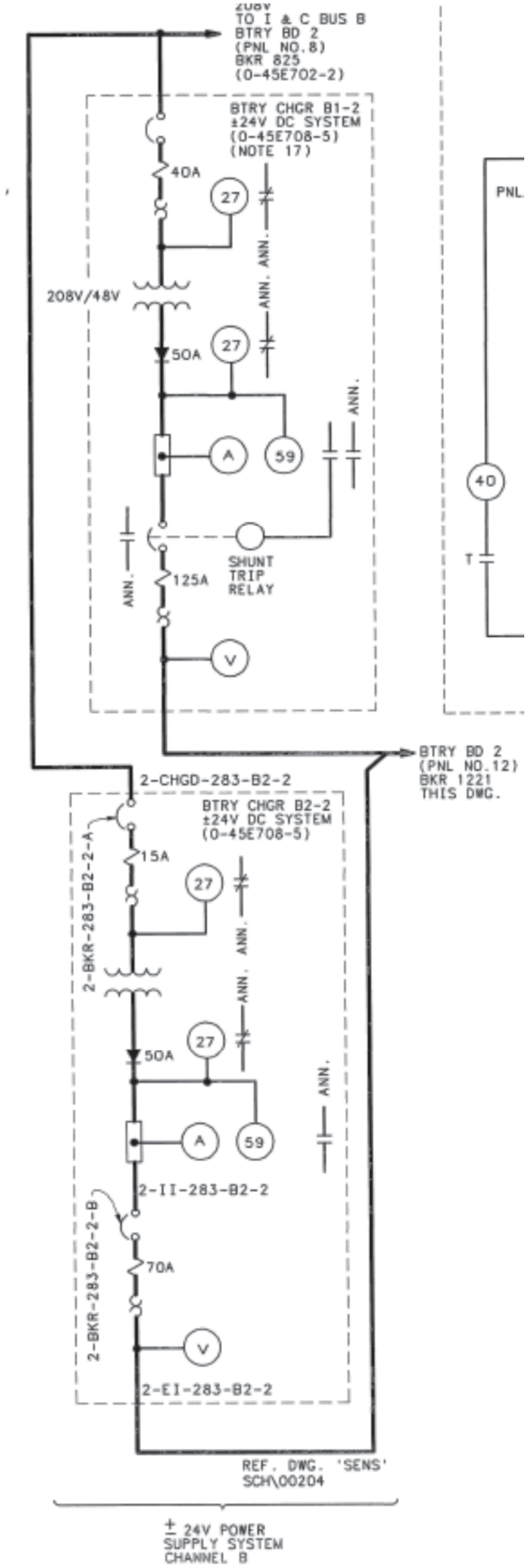
(3) High voltage power supply Takes the + 15 volt input from the voltage regulator and produces an adjustable voltage (100-350 volts) for use as the operating bias on the detector.

4.0 ATTACHMENT DATA

Performed On: _____

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Control Bay - Battery Board 2 - El 593'		
1203	0-BKR-280-002/1203 ±24V DC CHANNEL A TO PNL 2-9-12	ON	____ ____
1223	0-BKR-280-002/1223 ±24V DC CHANNEL B TO PNL 2-9-12	ON	____ ____

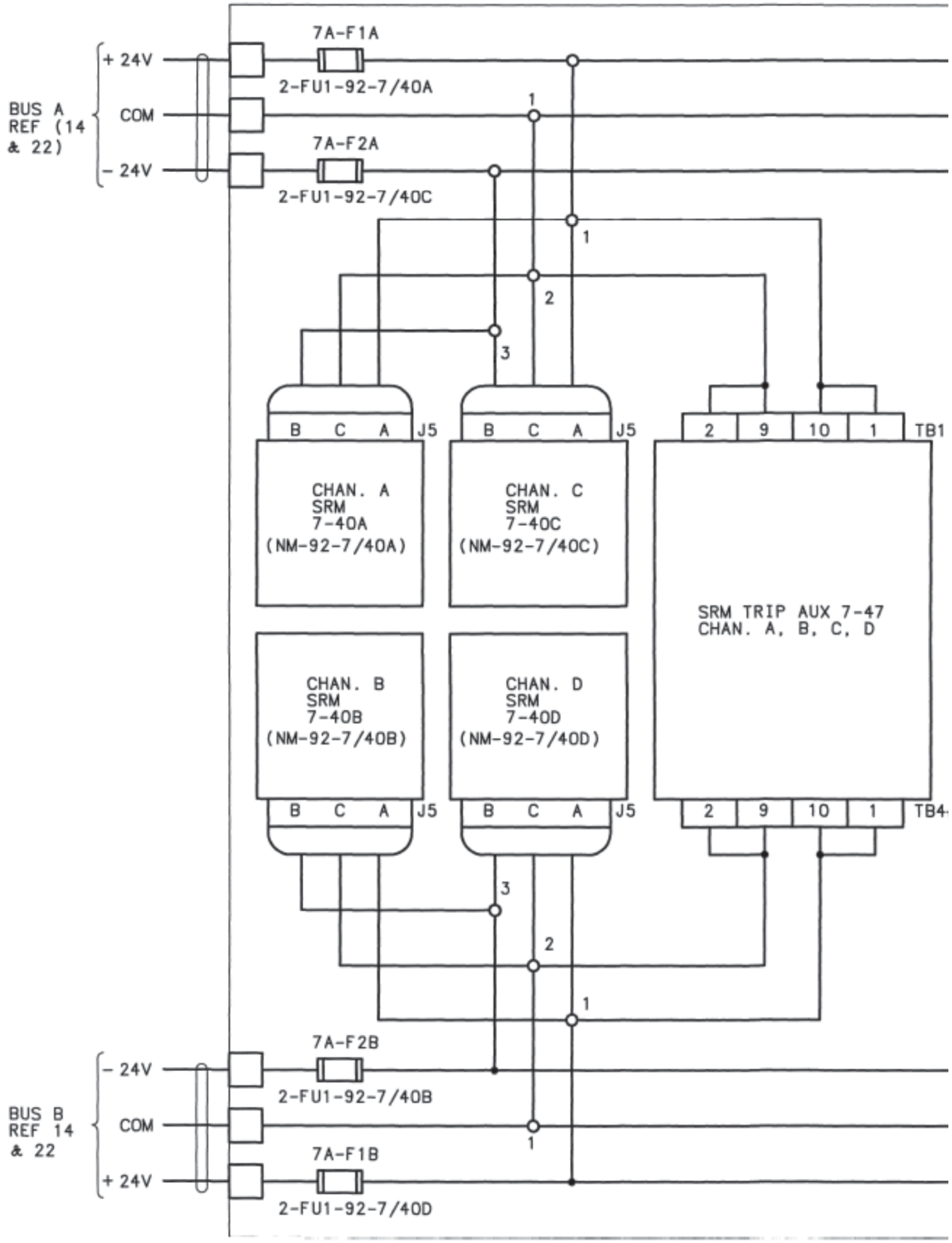

208V
TO I & C BUS A
BTRY BD 2
(PNL NO.8)
BKR 805
(0-45E702-2)
BTRY CHGR A1-2
±24V DC SYSTEM
(0-45E708-5)
(NOTE 17)
40A
27
ANN.
208V/48V
50A
A
59
125A
SHUNT
TRIP
RELAY
V
BTRY CHGR A2-2
±24V DC SYSTEM
(0-45E708-5)
(NOTE 17)
CONTINUED ON 0-45E702-2
PANEL 12
± 24V POWER SUPPLY SYSTEM CHANNEL A
1201
1202
1203
1204
SEE NOTE 18
1205
1206
– BUS
+BUS
1207
1208
2-FU1-283-AB
2-FU1-283-AA
0-FU2-280
-2/1209
1209
BARRIER
(NOTE 17)
± 24V POWER SUPPLY SYSTEM CHANNEL B
1220
0-FU2-280
-2/1220
BUS BY FIELD
1221
1222
2-FU1-283-BB
2-FU1-283-BA
1223
1224
1225
1226
1227
1228
CENTER TAP
(NOTE 16)
± 24V BATTERY
CHANNEL A
C & D TYPE
3-DCU-7
± 24V BATTI
CHANNEL B
C & D TYPE
3-DCU-7
208V
TO I & C BUS B
BTRY BD 2
(PNL NO.8)
BKR 825
(0-45E702-2)
BTRY CHGR B1-2
±24V DC SYSTEM
(0-45E708-5)
(NOTE 17)
PNL
± 24V POWER
SUPPLY SYSTEM
CHANNEL A


208V
TO I & C BUS B
BTRY BD 2
(PNL NO.8)
BKR 825
(0-45E702-2)
BTRY CHGR B1-2
±24V DC SYSTEM
(0-45E708-5)
(NOTE 17)
40A
27
ANN.
ANN.
208V/48V
50A
27
ANN.
A
59
ANN.
SHUNT
TRIP
RELAY
125A
V
PNL.
40
T
BTRY BD 2
(PNL NO.12)
BKR 1221
THIS DWG.
2-CHGD-283-B2-2
BTRY CHGR B2-2
±24V DC SYSTEM
(0-45E708-5)
2-BKR-283-B2-2-A
15A
27
ANN.
ANN.
50A
27
ANN.
A
59
2-II-283-B2-2
2-BKR-283-B2-2-B
70A
V
2-EI-283-B2-2
REF. DWG. 'SENS'
SCH\00204
± 24V POWER
SUPPLY SYSTEM
CHANNEL B

0-45E702-3

2-730E237

CONT ON SHEET 2 SHEET 1



COOPER 2011 NRC Q 4

ES-401 Sample Written Examination Form ES-401-5
Question Worksheet
K/A # 215004.K2.01

What are the power supplies to the SRM Channels/detectors?

SRM Channels…

a. A & B are powered from Division I + 24VDC System and C & D are powered from Division II + 24VDC System.

b. A & C are powered from Division I + 24VDC System and B & D are powered from Division II + 24VDC System.

c. A & B are powered from Division I 125 VDC System and C & D are powered from Division II 125 VDC System.

d. A & C are powered from Division I 125 VDC System and B & D are powered from Division II 125 VDC System.

ANSWER: 4
b. A & C are powered from Division I +24VDC System and B & D are powered from Division II +24VDC System.

Explanation:
From the Student Text for Lesson COR002-30-02 and GE Print 791E258 Sh.10, the Power Supplies for the SRM subsystem is powered from the + 24 VDC system. Channel A and C are powered from Div 1 and channel B and D are powered from Div 2.

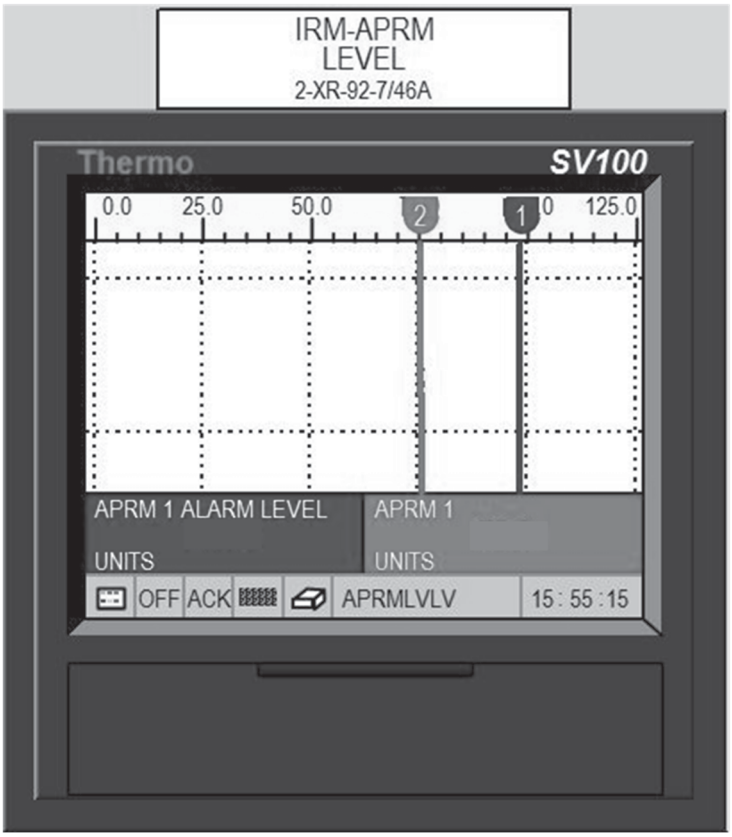
Distracters:
a. A & B are powered from Division I + 24VDC System and C & D are powered from Division II + 24VDC System is incorrect because SRM “B” and “D” are DIV II SRMs and “A” and “C” are the DIV I SRMs.

c. A & B are powered from Division I 125 VDC System and C & D are powered from Division II 125 VDC System. This is the normal power supply form 125 VDC DIV I components and logics, however the SRMs are powered from the 24 VDC system.

d. A & C are powered from Division I 125 VDC System and B & D are powered from Division II 125 VDC System. This is the normal power supply form 125 VDC DIV II components and logics, however the SRMs are powered from the 24 VDC system.

QUESTION 36 Rev 4

Unit 2 is operating at power with the following APRM/IRM Recorder selected:



While performing a channel check on the above recorder, the operator calculates the APRM 1 Alarm setpoint and compares it to the reading on the recorder.

Given that Core Flow is 59%, what is the value for that APRM 1 Rod Block setpoint and is the recorder indicating a correct reading?

A. 98; Yes

B. 98; No

C. 109; Yes

D. 109; No

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005A4.01	
	Importance Rating	3.2	3.1

Average Power Range Monitor/Local Power Range Monitor; Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) IRM/APRM recorder

Justification for K/A match: This is a Tier 2 Systems K/A concerning the APRM/IRM Recorder and the ability to monitor it in the control room. To match this K/A, a picture of the APRM/IRM Recorder was placed in the question and since there is really nothing to operate on it anymore, a monitoring question tied with what is actually being displayed and what happens when the alarm setpoint is exceeded has been asked.

Explanation:CORRECTA: The IRM/APRM Recorder on Panel 9-5 displays both the APRM reading and the APRM Alarm Level Readings. In this case both the APRM and Alarm Reading scale is obscured by the pointer number, the student has to know the scale and differentiate between the two readings along with recalling the automatic response when the alarm setpoint is exceeded.

B. **Incorrect because –**if the APRM exceeds the APRM Alarm reading of 98, then a Control Rod Withdrawal Block will be enforced, not a half scram.
Plausiblesince the alarm is a flow biased STP reading, it is conceivable that the trip displayed is the Scram Setpoint and not the Rod Block one.

C. **Incorrect because –**this is the APRM Alarm reading not the APRM reading.
Plausible since these two readings are opposite the digital display just below them.

D. **Incorrect because –**the APRM Alarm reading that is displayed is the Control Rod Block setpoint.
Plausible since the alarm is a flow biased STP reading, it is conceivable that the trip displayed is the Scram Setpoint and not the Rod Block one.

Technical Reference(s): OPL 171.148 rev 13,

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): N/A

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC:Perry 2003 Q #47

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

10 CFR Part 55 Content: 41(b)(7)

INSTRUCTOR NOTES

d. The process computer receives LPRM detector inputs for determination of core and LPRM performance.

Obj. V.B.9.d
Obj. V.D.5.d

C. Average Power Range Monitors (APRMs)

TP-7, 9, 11

1. Purpose

See OI-92B for current LPRM, APRM, set points.
OPRM setpoints are generated by nuclear fuels group and can be obtained from IM procedure for setting the set point and Rx Eng.
STP will be discussed in more detail in this section.

a. Monitors reactor power by calculating an average neutron flux signal representative of the reactor power from the LPRMs.

b. Calculates a simulated thermal power (STP) signal representative of the reactor thermal power.

c. Calculates flow biased upscale set points from the recirc flow monitor.

d. Generates reactor scram signals, rod blocks, and alarms if any of the following occur:

(1) The average neutron flux signal exceeds an upscale set point.

(2) The simulated thermal power (STP) signal exceeds the flow biased upscale set point.

(3) The APRM channel is inoperative.

		INSTRUCTOR NOTES
	(a) Control Rod block: 10% with mode switch not in RUN	
	(b) Reactor Trip signal: 14% w/ mode switch not in run OR 119% w/ mode switch in RUN	Obj. V.C.5.c
	(c) APRM Simulated Thermal Power (STP) Flow-Biased trip	
(2)	Each APRM instrument provides a rod block signal to the RMCS under for of the following conditions	
	(a) Simulated Thermal Power Upscale Alarm (as listed above).	Obj. V.B.15.b
	(b) Neuron Flux Downscale Alarm. <5% w/ mode switch in RUN	Obj. V.C.5.a

Illustration 1
(Page 6 of 6)

APRM/OPRM Trip Outputs and PRNMS Overview

4.0 RECIRCULATION FLOW PROCESSING

Each APRM powers and processes the Recirculation flow signals for one transmitter monitoring loop A and for one transmitter monitoring loop B. The PRNMS monitors the Recirculation System Flow in order to calculate a value for Total Recirculation Flow Rate. This value is then used by the APRM System to determine the flow-biased upscale trips and alarms for STP. Flow data is also provided to remote recorders and meters and to the Process Computer.

Most of the Recirculation System Flow signal processing is performed by the APRM instruments; however, the RBM instruments compare the Total Flow values generated by and received from each of the APRM instruments. The RBM generates a Flow Compare alarm if any two Total Recirculation Flow values differ by an amount greater than or equal to the setpoint.

Perry Nuclear Power Plant
NRC Written Examination
Data Sheets

QUESTION Common 047

The plant is operating at 75% power.

Which one of the following setpoints is displayed when the Operator at the Controls depresses the APRM ALARM LEVEL RECORD pushbutton for an IRM/APRM Recorder?

A. IRM high flux rod block.

B. IRM upscale scram.

C. APRM flow-biased rod block.

D. APRM flow-biased scram.

ANSWER: C.

QUESTION 37 Rev 0

How many LPRM strings are assigned to each APRM and how are they distributed in the core?

A. 22; symmetrically throughout the core

B. 43; symmetrically throughout the core

C. 22; asymmetrically throughout the core

D. 43; asymmetrically throughout the core

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005 K5.06	
	Importance Rating	2.5	2.6

215005 Average Power Range Monitor/Local Power Range Monitor System: K5.06 Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM : Assignment of LPRM's to specific APRM channels (CFR: 41.5 / 45.3)

Justification for K/A match: Since this is a Tier 2, Systems question it asks the candidate to recall the number of LPRMs that are assigned to each APRM, and how they are arranged though out the core, meeting both parts of the K/A.

Explanation: B is CORRECT: There are 43 LPRM strings assigned to each APRM. The LPRM strings are symmetrically distributed in the core.

A. **Incorrect because –** There are 43 LPRM strings assigned to each APRM.
Plausible because this is the number of LPRMs per APRM drawer however another 21 LPRMs input to the APRM from the LPRM drawer.

C. **Incorrect because –** There are 43 LPRM strings assigned to each APRM and they are distributed symmetrically in the core.
Plausible because **–** This is the number of LPRMs per APRM drawer however another 21 LPRMs input to the APRM from the LPRM drawer. The candidate may think the LPRMs are arranged asymmetrically.

D. **Incorrect because –** The LPRMs are not asymmetrical.
Plausible due to the definition of the word symmetrical, if the core is divided into quadrants, the LPRMs are assigned to cover open areas and each string although symmetrical they are offset to ensure each node is monitored.

Technical Reference(s): OPL 171.148 rev 13

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.148 rev 13 obj 7

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:	41(b)(6)

INSTRUCTOR NOTES

i. LPRM high and low lights on the full core display will illuminate when the set points are exceeded and will auto reset the condition clears. A white light illuminates for the downscale set point and an amber light illuminates for the upscale set point.

TP-6

Upscale= 100%
Downscale= 3%

6. APRM/LPRM Instruments (on panel 9-14)

TP-7, 8, 9

a. A total of 43 LPRM strings are assigned and monitored through each of the four APRM channels.

Table 1
Obj. V.B.6
Obj. V.D.4

(1) Each APRM instrument receives 22 LPRM inputs and the LPRM instrument receives 21 LPRM inputs

Q: How many detectors in an LPRM string?
A: four

(2) Each LPRM/APRM instrument can only accommodate up to 34 LPRM inputs.

Q: How many total LPRM detectors?
A: 172 (4 x 43)

(3) Since we have a total of 43 LPRM inputs, two instruments are necessary.

b. The flux level of each LPRM detector can be displayed on the APRM and LPRM instruments and on the operator's display assemblies (9-5), as well as the RBM (9-5).

	INSTRUCTOR NOTES
a. 172 total detectors in the core. Similar to SRM and IRM detectors with minor differences.	Obj. V.B.1/V.B.2 Obj. V.D.1/V.D.2
(1) =2.0" long (sensitive length)	
(2) .23" in diameter.	
(3) U3O8 (90 percent enriched) electroplated to outer electrode (case).	
(4) Argon filled to 1.3 atmospheres	
(5) Operate as a fission chamber (ionization).	
b. Neutron sensitivity decreases approximately 10 percent per 300 MWD/t average core exposure due to uranium depletion.	Obj. V.B.4
c. Change in argon pressure will affect sensitivity of detector to neutron flux.	
d. Gamma sensitivity – doesn't change with life, as it is not affected by uranium depletion, but is a function of argon pressure which does not change with life.	
3. Detector Assembly (LPRM String)	Obj. V.B.1 Obj. V.D.1
a. Houses four LPRM detectors, detector cables and calibration dry tube for TIP detector. LPRM assemblies are located at throughout the core area.	TP-1
b. Assemblies are positioned so that every location or its symmetrical counterpart in another quadrant is monitored.	TP-2
c. Detectors are positioned form the bottom of active fuel as follows, from top to bottom	Instructor: Emphasize with an unsymmetrical core and more questioning attitude is required when maneuvering the reactor
D 126"	
C 90"	
B 54"	
A 18"	36" apart
	TP-3

QUESTION 38 Rev 0

What Reactor Core Isolation Cooling (RCIC) design feature provides for the prevention of water hammer?

A. Suction head pressure provided by the CST

B. Minimum flow valve automatic operation

C. System snubbers

D. Low pressure isolation

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000 K4.01	
	Importance Rating	2.8	2.8

217000 Reactor Core Isolation Cooling System (RCIC) K4.01 Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: Prevent water hammer: Plant-Specific (CFR: 41.7)

Justification for K/A match: Tier 2 is a systems question K/A concerning RCIC System and its design features which provide for preventing water hammer. Keeping the discharge piping full will prevent water hammer and this is one of the design features of RCIC.

Explanation: CORRECT A: The RCIC System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve.

B. **Incorrect because -** Minimum flow valve – provides a flowpath for the pump when the discharge valve is closed. It provides cooling when running with the discharge valve closed.
Plausible this is a design feature of RCIC, but not for the prevention of water hammer.

C. **Incorrect because -** System snubbers provide pipe movement restraint, that might be the result of water hammer, but they do not prevent water hammer.
Plausible this is a design feature of the RCIC, but not for the prevention of water hammer.

D. **Incorrect because -** Low pressure isolation – Provides a way of preventing the system from running with too low of a supply steam pressure, and possible erratic operation.
Plausible in that this is a design feature of the RCIC, but not for the prevention of water hammer.

Technical Reference(s): 1-OI-71 Rev 19

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.040 rev 24 Obj 9

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X
10 CFR Part 55 Content:	41(b)(7)

OPL171.040, Reactor Core Isolation Cooling (RCIC) System

(3) Vent station for venting RCIC System located on El 565 Rx Bldg., outside steam vault. Vent lines located before Disch check valve FCV-71-40. RCIC may also be vented at RCIC pump. System vented to maintain piping filled and prevent water hammer on initiation.

The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge piping is kept full of water. The RCIC System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for RCIC is such that the water in the feedwater lines keeps the remaining portion of the RCIC discharge line full of water.

QUESTION 39 Rev 0

Unit 1 scrams due to a lowering Reactor water level and the US has entered 1-EOI-1, RPV Control.

At step RC/L-5 the US directs the UO to inhibit ADS.

The UO places ADS LOGIC INHIBIT switches 1-XS-1-159A and 1-XS-1-161A in inhibit then reports:

- 1-9-5 window 18 ADS LOGIC BUS A INHIBITED failed to alarm.
- 1-9-5 window 31 ADS LOGIC BUS B INHIBITED is in alarm.

Which one of the following completes the statement below?
In accordance with 1-ARP-9-3C window 18 the UO will direct an AUO to _____.

A. open the ADS System Logic Bus A breaker on 250V RMOV board 1A

B. pull 250V Logic A fuses in the Auxiliary Instrument room

C. place all ADS Backup Control Panel transfer switches in emergency at Panel 1-25-32

D. pull all ADS Solenoid power fuses at panel 1-25-32, Backup Control Panel

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	218000 G2.4.35	
	Importance Rating	3.8	4.0

ADS; Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)

Justification for K/A match: This is a Tier 2 systems question paired with a generic Emergency K/A so to match both parts of the Automatic Depressuration System K/A and knowledge of local Auxiliary Operator task during an emergency, The question requires it to be written on an Abnormal or Emergency aspect of the ADS system and the operational effects. Systems knowledge is required to know where electrical power is supplied to ADS and its effect when it is removed.

Explanation: CORRECT B: 1-ARP-9-3C window 18 states: **IF** this alarm fails to annunciate with 1-XS-1-159A in INHIBIT, **THEN PULL** (remove) the following fuses: □1-FU2-1-2E-K3 250V LOGIC A, 2E-F1A & 2A UV on Panel 1-9-30

A. **Incorrect because –** this is not where the procedure sends you to deenergize the logic. Wrong division of power is selected.
Plausible because opening the correct breaker would de-energize the logic however this would not be in accordance with the ARP and the A logic is powered by the 1B 250V RMOV board.

C. **Incorrect because –** this is not where the procedure sends you to deenergize the logic and it would not prevent all ADS valves from operating.
Plausible because placing an ADS Backup Control power switch in emergency will disable the ADS logic however there are only 4 ADS valves with Backup Control transfer switches on Panel 1-25-32.

D. **Incorrect because –** this is not where the procedure sends you to deenergize the logic and it would not prevent all ADS valves from operating.
Plausible because this would prevent the ADS valves from opening and is an option to close a stuck open relief valve IAW 1-AOI-1-1 however this would not be in accordance with 1-ARP-9-3C window 18

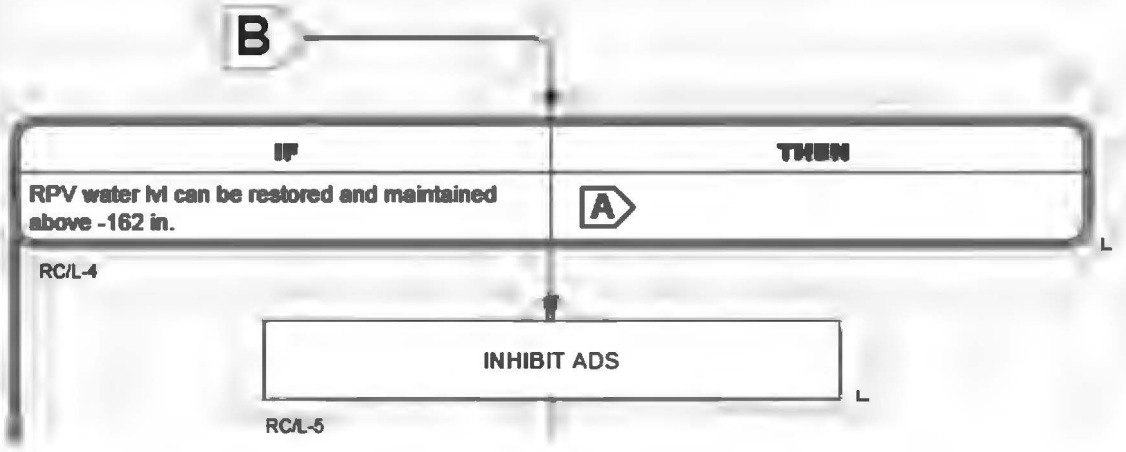
Technical Reference(s): 1-EOI-1 Rev 4, 1-ARP-9-3C Rev26, 1-AOI-1-1Rev4, 1-OI-1 Attachment 3 Rev10

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.043 OBJ 6

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:	41(b)(10)	

Alternate Level Control



ADS LOGIC BUS A INHIBITED	18

(Page 1 of 1)

Sensor/Trip Point:

1-XS-1-159A in INHIBIT position

Sensor Location: Panel 1-9-3 Main Control Room

Probable Cause: ADS LOGIC A INHIBIT Handswitch 1-XS-1-159A in INHIBIT.

Automatic Action: Automatic operation of the ADS valves is bypassed.

Operator Action:

A. **CHECK** position of 1-HS-1-159A on Panel 1-9-3. ☐

B. **IF** this alarm fails to annunciate with 1-XS-1-159A in INHIBIT, **THEN PULL** (remove) the following fuses:
   - 1-FU2-1-2E-K3 250V LOGIC A, 2E-F1A & 2A UV on Panel 1-9-30 ☐

References: 1-45E620-2-1 1-730E929-2
TRM 3.3.3

Attachment 1
(Page 2 of 4)

Unit 1 SRV Solenoid Power Breaker/Fuse Table, Panels 1-25-32 and 1-LPNL-925-0658

NOTES

1) Block AA is located in Bay 4 of Panel 1-25-32. Block EE is located in Bay 3 of Panel 1-25-32. (See Figure "Panel 1-25-32 Rear").
2) Located in Bay 4 of Panel 1-25-32, bottom right side. (See Figure " Panel 1-25-32 Rear").
3) Fuses are located in 1-LPNL-925-0658

Unit 1 SRV Solenoid Power Fuse Table, Panel 1-25-32					
SRV	Function	TRANS/DISC Switch	Fuse	Fuse Holder	Block (1)
1-4	Manual	1-XS-1-4	1-FU1-001-0004A 1-FU1-001-0004B	F3 F8	EE EE
1-42	Manual	1-XS-1-42	1-FU1-001-0042A 1-FU1-001-0042B	F17 F18	EE EE
1-23	Manual	1-XS-1-23	1-FU1-001-0023A 1-FU1-001-0023B	F4 F9	AA AA
1-41	Manual	1-XS-1-41	1-FU1-001-0041A 1-FU1-001-0041B	F3 F8	AA AA
1-180	Manual	1-XS-1-180	1-FU1-001-0180A 1-FU1-001-0180B	ABA (2) ABB (2)	N/A N/A
1-22	ADS	1-XS-1-22	1-FU1-001-0022A 1-FU1-001-0022B 1-FU1-001 0022C 1-FU1-001-0022D	F2 F7 F12 F15	EE EE EE EE
1-5	ADS	1-XS-1-5	1-FU1-001-0005A 1-FU1-000-0005B 1-FU1-001-0005C 1-FU1-001-0005D	F1 F6 F11 F14	AA AA AA AA
1-30	ADS	1-XS-1-30	1-FU1-001-0030A 1-FU1-001-0030B 1-FU1-001-0030C 1-FU1-001-0030D	F2 F7 F12 F15	AA AA AA AA
1-34	ADS	1-XS-1-34	1-FU1-001-0034A 1-FU1-001-0034B 1-FU1-001-0034C 1-FU1-001-0034D	F5 F10 F13 F16	AA AA AA AA

4.0 ATTACHMENT DATA (continued)

Performed On: _____

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Control Bay - 250V DC Reactor MOV Board 1B - El 593'		
1B2	1-BKR-001-0019 MN STM LINE B RELIEF VALVE	ON	____ ____
1C1	1-BKR-001-0022A MSL B RELIEF VALVE ALT FDR	ON	____ ____
1C2	1-BKR-001-0031 MN STM LINE C RELIEF VALVE	ON	____ ____
1F1	1-BKR-001-001B/1F1 AUTO BLOWDOWN SYSTEM LOGIC BUS A DIV I-1 PNL 9-30	ON	____ ____

4.0 ATTACHMENT DATA (continued)

Performed On: _____

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Control Bay - 250V DC Reactor MOV Board 1B - El 593'		
8F2	1-BKR-001-001B/8F2 AUTO BLOWDOWN DIV I BUS B PNL 9-30	ON	____ ____

QUESTION 40 Rev 0

Which one of the following completes the statements below?

Reactor Water Level Instruments, ___(1)___, provide a low-low-low Reactor Vessel water level signal to ADS initiation logic at less than or equal to __(2)__ inches.

NOTE: LIS-3-184 is Reactor Water Level A
LIS-3-185 is Reactor Water Level B
LIS-3-58A-D is Reactor Water Level A-D

A. (1) LIS-3-58A-D
(2) (-) 122

B. (1) LIS-3-58A-D
(2) + 2

C. (1) LIS-3-184 and LIS-3-185
(2) (-) 122

D. (1) LIS-3-184 and LIS-3-185
(2) + 2

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	218000 K1.03	
	Importance Rating	3.7	3.8

218000 Automatic Depressurization System: K1.03 Knowledge of the physical connections and/or cause effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Nuclear boiler instrument system (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Justification for K/A match: Tier 2 systems question on ADS and its physical tie to NBI. The match the K/A knowledge question was written on the recollection of which NBI level instruments provide certain logic functions for the ADS system.

Explanation: CORRECT A: The Reactor water level instruments LIS-3-58A, B, C & D provide the low-low-low reactor vessel water level signal to ADS initiation logic at (-) 122 inches.

B. **Incorrect because –** even though the NBI are correct the setpoint is not correct.
Plausible since +2 inches is the confirmatory setpoint for ADS.

C. **Incorrect because -** the NBI are not correct but the setpoint is correct.
Plausible since these are the two confirmatory level transmitters.

D. **Incorrect because -** the NBI are not correct
Plausible since these are the confirmatory level setpoint and transmitters.

Technical Reference(s): 3-OI-1 rev 41, 3-ARP-9-3C rev 28

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.043 rev 15 Obj 4

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: BFN 1404 Q #40
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)(7)

3.4 Main Steam Relief Valve (MSRV / ADS) (continued)

F. ADS will initiate when ALL of the following conditions are met:

1. A confirmatory low reactor water level signal (+2.0 inches), REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE, 3-9-3C Window 3, and

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, 3-9-3C Window 33 and RX WTR LVL LOW LOW LOW ECCS/ESF INIT 3-LA-3-58A, 3-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 3-LA-3-58A, 3-XA-55-9-3C Window 28, for 265 seconds (High drywell pressure bypass)

   c. When the above logic is satisfied, the 95 second timer starts (ADS BLOWDOWN TIMERS INITIATED, 3-XA-55-9-3C, Window 11).

3. ADS 95 second timer timed out.

4. One RHR pump OR two Core Spray pumps (A or B and C or D) running, RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 3-XA-55-9--3C Window 10.

RX WTR LVL
LOW LOW LOW
ECCS/ESF INIT
3-LA-3-58A

28

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-58A, B, C and D ≤ -122 inches (RPV low-low-low level)(Level 1)

REACTOR
LEVEL LOW
ADS BLOWDOWN
PERMISSIVE

3

(Page 1 of 1)

Sensor/Trip Point:

LIS-3-184	RPV level
LIS-3-185	≤ +2.0 inches

NRC BFN 1404 Q #40

QUESTION 40

Which ONE of the following completes the statement below?

The Reactor water level instrument __(1)__ provide a confirmatory low reactor vessel water level signal to ADS initiation logic at less than or equal to __(2)__ inches.

NOTE: LIS-3-184 is Reactor Water Level A
LIS-3-185 is Reactor Water Level B
LIS-3-58A-D is Reactor Water Level A-D

A. (1) LIS-3-58A-D
(2) (-) 45

B. (1) LIS-3-58A-D
(2) (+) 2

C. (1) LIS-3-184 and LIS-3-185
(2) (-) 45

D. (1) LIS-3-184 and LIS-3-185
(2) (+) 2

ANSWER: **D**

QUESTION 41 Rev 0

What is the design feature that allows testing of MSIV Reactor Water Level Instrumentation associated with Primary Containment Isolation System (PCIS) without causing a device actuation?

A. 1 out of 2 taken twice logic

B. 2 out of 3 selection logic

C. 2 out of 4 voter logic

D. mean select logic

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002 K4.02	
	Importance Rating	3.1	3.1

223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off K4.02 Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Testability (CFR: 41.7)

Justification for K/A match: Tier 2 Systems question on PCIS and the knowledge of design feature which provides testability. Matches the K/A by asking the difference in logic design and how PCIS is tested. That is, testing one channel or instrument will not cause an unwanted isolation.

Explanation: CORRECT A: PCIS Logic for MSIVs is arranged as follows A1 or A2 AND B1 or B2 which is 1 out of 2 taken twice. As referenced on 1-730E927-10 the Trip Logic is arrange with relays 16A-K7B and 16A-K7B in series and 16AK7C and 16A-K7A in series with each other respectfully. Both the AC and DC Solenoids must be de-energized to close MSIV. Relays on 16A-1A, B, C, D on print 1-730E927-7 &-8 input to 16A-K7A, B, C & D.

B. **Incorrect because -** This is not the correct logic scheme for PCIS. PCIS is a 1 out of 2 taken twice logic.
Plausible since this is the logic arrangement for Reactor Water Level High trip of Main Turbine and Reactor Feed Pumps.

C. **Incorrect because -** This is not the correct logic scheme for PCIS. PCIS is a 1 out of 2 taken twice logic.
Plausible since this is the APRM logic within RPS. RPS Powers PCIS Logic and student may have the misconception of this arrangement.

D. **Incorrect because -** This is not the correct logic scheme for PCIS. PCIS is a 1 out of 2 taken twice logic.
Plausible since this in how Reactor Water Level Control System works to control the speeds of Reactor Feed Pumps by using an average (mean) of Narrow Range Level Instruments.

Technical Reference(s): 1-OI-64 rev 18, 1-OI-3 rev 39, 1-OI-92B rev 10, 1-730E927-7,-8,-10, OPL 171.017 rev 17

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.017 rev 17 A.3a

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:	41(b)(7)

Appendix A
(Page 3 of 8)

Actions to Place PCIS in Tripped Condition

NOTE

Water level designators (1-8) are listed for relationship to the applicable device only.

(Tech Specs Table 3.3.6.1-1)

DEVICE	FUSE	RELAY	PANEL	PRINT	ALARM	REMARKS
1-LIS-003-0056A RX WATER LEVEL LOW **(Level 1)** **Function: 1a**	1-FU1-003-0056AA (16A-F1A)	16A-K1A	1-PNLA-009-0015	1-730E927-7 1-45E671-50	1-XA-55-5B-4 REAC VESSEL WTR LVL CH A LOW-LOW	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. A1 PCIS RED STATUS LIGHT EXTINGUISHES.
1-LIS-003-0056B RX WATER LEVEL LOW **(Level 1)** **Function: 1a**	1-FU1-003-0056BA (16A-F1B)	16A-K1B	1-PNLA-009-0017	1-730E927-8 1-45E671-62	1-XA-55-5B-5 REAC VESSEL WTR LVL CH B LOW-LOW	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. B1 PCIS RED STATUS LIGHT EXTINGUISHES.
1-LIS-003-0056C RX WATER LEVEL LOW **(Level 1)** **Function: 1a**	1-FU1-003-0056CA (16A-F1C)	16A-K1C	1-PNLA-009-0015	1-730E927-7 1-45E671-56	1-XA-55-5B-4 REAC VESSEL WTR LVL CH A LOW-LOW	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. A2 PCIS RED STATUS LIGHT EXTINGUISHES.
1-LIS-003-0056D RX WATER LEVEL LOW **(Level 1)** **Function: 1a**	1-FU1-003-0056DA (16A-F1D)	16A-K1D	1-PNLA-009-0017	1-730E927-8 1-45E671-68	1-XA-55-5B-5 REAC VESSEL WTR LVL CH B LOW-LOW	ALARMS AND GIVES 1/4 ISOLATION IN PCIS GROUP 1. NO PCIS DEVICES ACTUATE. B2 PCIS RED STATUS LIGHT EXTINGUISHES.

separate trip channels (A and B) are each provided with two sensor relay contacts (A/C and B/D). PCIS de-energizes to isolate (except HPCI/RCIC)

a. This arrangement creates trip sub-channels A1/A2 and B1/B2.

b. A trip of either sensor relay within a trip channel will cause opening of the associated contact and de-energization of the associated relay. This condition will create a "half Isolation" signal within both logic channels but NO VALVE MOVEMENT.

HPCI/RCIC are energize to actuate

ILT-3
LOR-3
NLO / NLOR-4

(a) These are systems which are not required for post-accident mitigation.

(b) These systems are either isolated immediately upon receipt of a PCIS Isolation signal, or are provided with manual valves that are locked closed when primary containment is required.

(c) Should a trip of either sensor relay in the
other trip channel occur, conditions will exist to de-energize the valve actuation relays in each logic channel, causing both Isolation valves to close.

PCIS logic is arranged as follows:

A1 **OR** A2 **AND** B1 **OR** B2	= **Inboard AND Outboard** valve closure

D. Group 1 (MSIV) Isolation Logic

1. TP-2 provides a simplified diagram of the Isolation logic for the "A" main steamline inboard Isolation valve (FCV-1-14). — 2-730E927-10

2. The MSIV is provided with both an AC-powered pilot solenoid (FSV-1-14C) and a DC-powered pilot solenoid (FSV-1-14B). — ILT-2 LOR-2

   Both of these pilot solenoids must be de-energized to cause the MSIV to close.

3. With the control handswitch in the AUTO/OPEN position, the associated HS-1-14A contacts will be closed.

   a. Should a Group 1 Isolation signal exist, the K7A,B,C,D relays will de-energize (see TP-3), causing the associated contacts to open.

   b. When these contacts open, the K13/K51 and K14/K77 relays de-energize, opening the associated contacts. This will cause the pilot solenoids to de-energize and the MSIV will close. — TP-2

4. Further detail regarding the MSIV Isolation and reset logic can be seen in TP-3. This is a simplified illustration of the A1 Isolation Sub-channel (relay K7A) — 2-730E927-7

   a. Note how relays associated with each Isolation signal are provided with contacts in the circuitry.

   b. Further information on this circuit will be provided below in section E (Isolation Reset Logic).

3.0 PRECAUTIONS AND LIMITATIONS (continued)

I. Each of the four APRM/OPRM channels input to the four Voters, such that when a signal is generated from an APRM/OPRM channel, all four Voters see and reflect that signal. Each Voter is directly associated with one RPS sub-channel.

When operating in a 2 out of 4 voting configuration, the first un-bypassed input will be seen as a single input with no trip outputs. When the second un-bypassed signal of the same type [The SAME TYPE inferring that one type is an APRM function and a different type is an OPRM function] is received it will also be seen by all four Voters resulting in a trip output from all four Voters consequently producing a full reactor scram.

3.0 PRECAUTIONS AND LIMITATIONS

F. Periodically monitor level instruments LEVEL A, 1-LI-3-208A, RX WATER LEVEL NARROW RANGE, 1-LI-3-208B, RX WATER LEVEL NARROW RANGE, 1-LI-3-208C, and LEVEL D, 1-LI-3-208D during normal operations and level transients. These instruments provide the high reactor water level trips for the Main Turbine and RFPTs at 55 inches as sensed by REACTOR WATER LEVEL 8,1-LT-003-0208A and REACTOR WATER LEVEL 8, 1-LT-003-0208C <u>or</u> REACTOR WATER LEVEL 8,1-LT-003-0208B and REACTOR LEVEL 8, 1-LT-003-0208D (two out of two taken once logic).

Illustration 4
(Page 2 of 3)

Single/Three Element Control

3.0 DESCRIPTION

SINGLE ELEMENT An automatic operating mode where one feedback input is used, Narrow Range Reactor water level. In SINGLE ELEMENT control, the operator adjusted Reactor water level setpoint is compared to the average of the valid Narrow Range level signals. SINGLE ELEMENT is used with automatic operation of the RFW Startup Level Control PDS and the Reactor Water Level Control PDS (with at least one individual RFPT Speed Control PDS in AUTO).

THREE ELEMENT An automatic operating mode using three feedback inputs; Narrow Range water level, Total Steam Flow, and Total Feedwater Flow. In THREE ELEMENT control, the operator adjusted Reactor level setpoint is compared to the average of the valid Narrow Range Reactor water level signals. A resultant level error signal is generated. This level error signal is summed with a Total Steam Flow signal and sent to the Steam Flow - Feed Flow Mismatch control block. The control block takes the demand signal and the Total Feedwater Flow signal and produces a Steam Flow/Feed Flow signal and is processed through the single element control logic. The demand signal is then sent to the RFPT Control logics, where it is combined with the automatic flow balance (flow biasing) inputs.

The RFW Control System will **NOT** allow THREE ELEMENT operation until all of the following permissives are met:

- RX WATER LVL CONT, 1-LIC-46-5, (PDS) in AUTO and at least one RFPT Speed Control (PDS) in AUTO.
- A **Valid Total Steam Flow** signal exists(Four good steam line flows, or at least two good steam line flow signals and a valid Turbine First Stage pressure signal).
- A **Valid Total Feedwater Line Flow** signal exists(Two good Feedwater line flow signals, or one good Feedwater line flow and three valid individual RFP flow signals).
- **Total Steam Flow >19%.**
- At least **one valid Narrow Range Level** signal.

QUESTION 42 Rev 0

Unit 3 is at 60% Reactor Power. A loss of Drywell Control Air (DWCA) has occurred due to a pipe rupture from the Containment Inerting System.

How is ADS MSRV 1-22 affected by this loss of Drywell Control Air (DWCA) in the short term?

ADS MSRV 1-22 will…

A. **NOT** operate.

B. operate in the Safety Mode ONLY.

C. operate in the Manual and Safety Modes ONLY.

D. operate in the ADS, Manual and Safety Modes.

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	239002 K6.02	
	Importance Rating	3.4	3.5

239002 Relief/Safety Valves K6.02 Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: Air (Nitrogen) supply

Justification for K/A match: To match this Tier 2 Systems K/A concerning the pneumatics to the SRVs, the question was written to set up a loss of normal pneumatics and then ask what effect this loss will have on the operation of the SRV.

Explanation: CORRECT D: each ADS MSRV has an accumulator designed to contain sufficient air for a minimum of 5 valve operations following a loss of Drywell Control Air. Each accumulator has a check valve to accomplish this. No electrical problem is stated in the stem therefore the ADS and Manual Functions will be available. Safety Function is always available since it is overcoming spring pressure.

A. **Incorrect because –** since the ADS SRVs have pneumatic accumulators, the valve will still function.
Plausible if misconception that pneumatic supply pressure is required to operate MSRV and student believes that accumulator pressure will be lost.

B. **Incorrect because –** since the ADS SRVs have pneumatic accumulators, the valve will still function.
Plausible since Safety Function is always available with a loss of pneumatics and electrical supplies. If the ADS MSRVs did not have accumulators, this loss would prevent some functions.

C. **Incorrect because –** since the ADS SRVs have pneumatic accumulators, the valve will still function.
Plausible since Safety Function is always available with a loss of pneumatics and electrical supplies. If the ADS MSRVs did not have accumulators, this loss would prevent some functions.

Technical Reference(s): OPL 171.043 rev 15 pg 9-13, 3-AOI-32-1 rev 13, Tech Spec Bases B3.5.1 rev 0, OPL 171.054 rev 16

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.043 Obj 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:	41(b)(7)

B. Component description

1. SRVs

a. Two-stage Target Rock

b. Will actuate on high reactor pressure via pressure switches, ADS logic initiation, or manual electric (via operator action)

c. Relieves 905,000 lb/hr at set pressure

d. Valve actuation modes

1) Safety Function (the DC solenoid valves do not operate in this mode) protects against nuclear over pressurization (pressure activated)

2) Relief Function provides automatic depressurization for small breaks in the primary coolant system so that the LPCI mode of RHR and the Core Spray System can operate to protect the fuel barrier

(a) This function is part of the Automatic Depressurization System (ADS)

(b) This function can be overridden with the use of two keylock switches (ADS LOGIC INHIBIT SWITCHES), or momentary break contact reset pushbuttons on panel 9-3

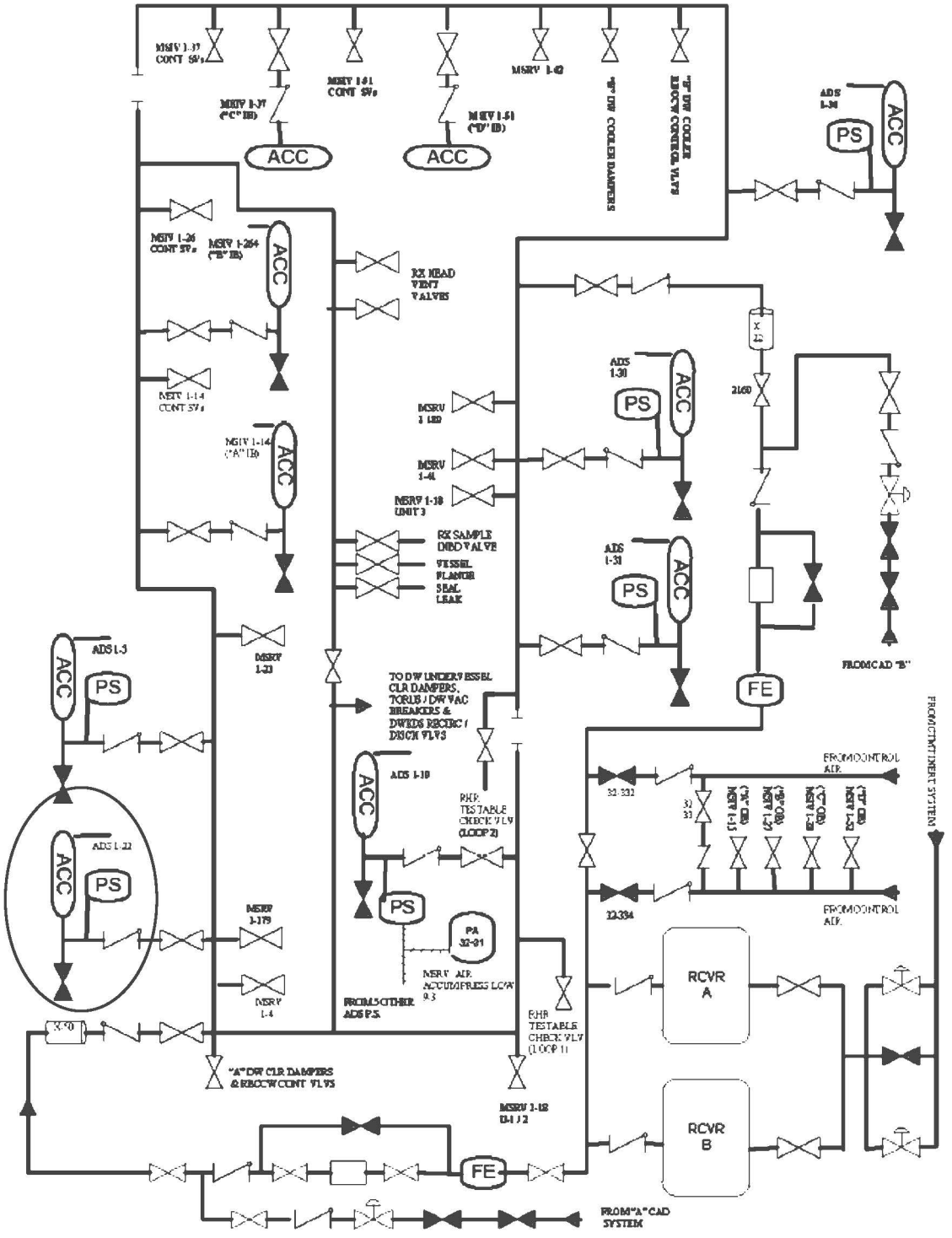
(c) Pressure switches are utilized to operate the MSRVs in the "relief mode' to back up the 'safety mode'

(d) The pressure switches setpoints are the same as the safety mode setpoints

Manual Mode is the manual operation (open/close) by the operator

3. Accumulator and check valve arrangement
    a. ADS valves are provided with accumulator arrangements
    b. Accumulators are provided to assure that the ADS valves can be held open for 30 minutes following a failure of the pneumatic supply to the accumulators
    c. Accumulators are sized to contain sufficient air for that minimum of five valve operations following a loss of Drywell Control Air. Extended loss of pneumatic supply and system leakage would result in failure of the 'relief mode' of the MSRVs
    d. EOI Appendix 8G crossties CAD to DWCA

OPL171.054 , Control and Service Air (Includes DWCA System), Rev# 16



TP-24: REACTOR BUILDING AND DRYWELL CONTROL AIR SYSTEM

4.0 OPERATOR ACTIONS

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 are sized to contain sufficient air for a minimum of five valve operations. Operations of the ADS MSRV should be limited to 5 times. Tech Spec 3.5.1 Bases. MSRVs 1-30 and 1-31 also have accumulators, however these MSRVs are not ADS assigned.
3) Nitrogen tanks supply pressuirzed nitrogen to the Drywell Control Air System via the DWCA SUPPLY REGULATORS 3-PREG-32-49A and 3-PREG-32-49B (lead requlator will be set at 108 psig and backup regulator set at 5 to 8 psig lower.
4) DWCA NITROGEN REG STATION BYPASS VLV, 3-BYV-032-0141 can be used to maintain approximately 98 psig in DWCA Receiver Tanks A & B when required by plant conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification every 31 days that ADS air supply header pressure is ≥ 81 psig ensures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 62.5% of design pressure plus three additional actuations at 0 psig drywell pressure (Ref. 10). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ 81 psig is provided by the Drywell Control Air System. The 31 day Frequency takes into consideration administrative controls over operation of the air system and alarms for low air pressure.

QUESTION 43 Rev 0

Unit 3 is operating at 100% power, with the following feedwater alignment:

- Reactor Water Level Master Controller in **MAN**
- A RFPT Speed Controller in **AUTO** at 5000 RPM
- B RFPT Speed Controller in **AUTO** at 4995 RPM
- C RFPT Speed Controller in **MAN** at 5005 RPM

How will Reactor Feed Pumps respond when the Reactor Water Level Master Controller raise pushbutton is depressed?

A. A, B and C RFPT speeds increase.

B. A, B and C RFPT speeds remain the same.

C. A and B RFPT speed increase; RFPT C speed remains the same.

D. A and B RFPT speed remain the same; RFPT C increases.

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	259002 A1.07	
	Importance Rating	2.6	

Reactor Water Level Control System; Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Turbine Driven Reactor Feed Pump speed

Justification for K/A match: To match the Tier 2 Systems K/A concerning the Reactor Water Level Control System and the operator's ability to predict and/or monitor changes in parameters associated with operating it. The question starts with a normal operation of the RWLCS when in MAN and then asks what happens to the speed of the feed pumps when the system is operated.

Explanation: CORRECT C: As the raise pushbutton on the Reactor Water Level Master Controller is depressed the only Reactor Feed Pumps to respond will be the ones in automatic, therefore A & B will speed up and C will remain at its current speed.

A. **Incorrect because –** not all three feed pumps will respond to this signal, because of their individual mode selection.
Plausible if student believes the Reactor Master Level Controller directly control all Reactor Feed Pumps regardless of their Speed Controller status.

B. **Incorrect because -** not all three feed pumps will fail to respond to this signal, because of their individual mode selection.
Plausible if student believes that the Reactor Master Level Controller has no controls in manual and that enables the individual RFPT PDS stations.

D. **Incorrect because -** This logic is 180 degrees out from what actually happens.
Plausible if student believes that the individual RFPT PDS station must be in manual to allow the master station to control the RFPT while it is in manual.

Technical Reference(s): 3-OI-3 rev 91

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.012 rev 12 obj VB8

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:	41(b)(5)

Illustration 3
(Page 3 of 4)

Reactor Water Level Master Controller Panel Display Station

3.0 DESCRIPTION

Reactor Water Level PDS controls RFPTs via any individual RFPT Speed Control PDSs placed in AUTO.

With Reactor Water Level PDS in MANUAL and Column 3 selected, Unit Operator has direct control over any RFPT Speed Control PDSs placed in AUTO. Reactor Water Level PDS output in MANUAL mode can be adjusted from 0% to 100% with Ramp Up/Ramp Down push-buttons. Any RFPT Speed Control PDSs placed in manual are independent of Reactor Water Level PDS.

With Reactor Water Level PDS in AUTO and Column 2 selected, PDS controls level setpoint is being maintained by RFW Level Control System in both SINGLE ELEMENT or THREE ELEMENT control. In SINGLE ELEMENT control, average of valid narrow range level signals is compared with level setpoint. The resultant level error signal is used by RFWCS to generate a speed demand signal for RFPT Woodward Governors. In THREE ELEMENT control, this level error signal is compared with average of valid Steam line Flow signals. The resultant error signal is compared with average of Feed Line Flows producing a Steam Flow/Feed Flow mismatch signal used to adjust RFPT speeds with their Woodward Governors.

Illustration 3
(Page 4 of 4)

Reactor Water Level Master Controller Panel Display Station

4.0 FAILURE MECHANISMS

When all four Narrow Range Level signals are declared bad or invalid, then Reactor Water Level Control PDS, 3-LIC-46-5, will trip to MANUAL. RFWCS FAILED TO MANUAL annunciation (3-XA-55-6C, window 21) will alarm.

On a loss of Unit Preferred, all indications are lost on the PDS. If the PDS was controlling RFPTs at the time, then system will continue to control RFPTs with last know system demand values.

QUESTION 44 Rev 0

Which one of the following (if any) identifies the suction source(s) for the Standby Gas Treatment Fans with respect to the Primary Containment System?

A. None

B. Drywell ONLY

C. Suppression Chamber ONLY

D. Drywell and Suppression Chamber

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	261000 K1.03	
	Importance Rating	3.1	3.1

Standby Gas Treatment System; Knowledge of the physical connections and/or cause-effect relationships between SBGT SYSTEM and the following: Suppression pool

Justification for K/A match: To match this Tier 2 Systems K/A for SBGT, the question was written to ask the candidate to recall the physical connection between SBGT and the Suppression Pool (Primary Containment).

Explanation: CORRECT D: 1-OI-64-1, Primary Containment System allows venting from either Drywell or Suppression Chamber using SGT.

A. **Incorrect because -** SGT can be aligned to take suction from the Drywell or Suppression chamber.
Plausible that SGT only takes suction from the secondary containment since it automatically aligns to Secondary Containment on an initiation signal.

B. **Incorrect because –** SGT can be aligned to take suction from the Suppression Chamber Air Space.
Plausible because – the normal venting procedures all vent from the Drywell.

C. **Incorrect because –** SGT can be aligned to take suction from the Drywell.
Plausible because this is the preferred suction source for emergency venting in the EOIs.

Technical Reference(s): 1-OI-64-1 Rev 18, 0-47E865-11, 1-47E865-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.018 OBJ VB2

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:	41(b)(8)

6.0 SYSTEM OPERATIONS

NOTE
All actions are performed from Panel 9-3 unless otherwise noted.

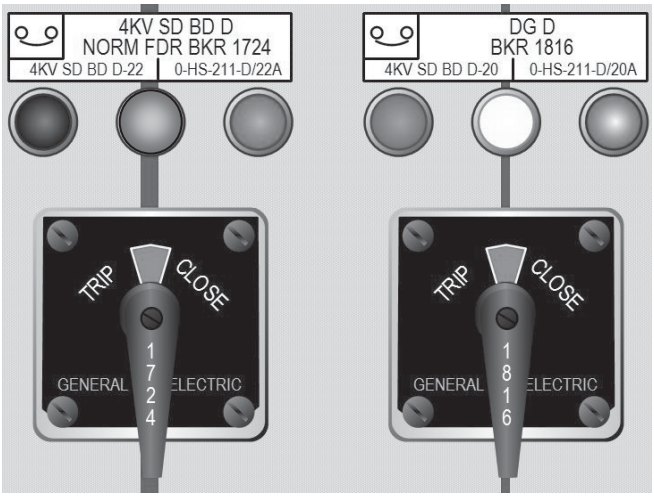
6.1 Venting the Drywell with Standby Gas Treatment Fan

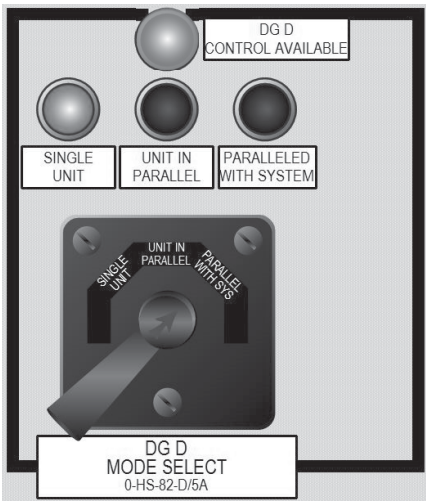
6.2 Venting the Suppression Chamber with Standby Gas Treatment Fan

QUESTION 45 Rev 1

Unit 2 is performing 0-SR-3.8.1.1(D), Diesel Generator D Monthly Operability; the Diesel has been loaded for 30 minutes.

The following indications **have just occurred**.





Which one of the following completes the statements below?

The white light above BKR 1816 is a __ (1) __.

Based on these conditions the first expected response is __ (2) __.

A. (1) disagreement light
   (2) DG D Breaker 1816 will trip open

B. (1) disagreement light
   (2) 4KV SD D Normal FDR BKR 1724 will trip open

C. (1) Diesel Generator Overload light
   (2) DG D Breaker 1816 will trip open

D. (1) Diesel Generator Overload light
   (2) 4KV SD D Normal FDR BKR 1724 will trip open

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262001 A3.01	
	Importance Rating	3.1	3.2

A.C. Electrical Distribution Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Breaker tripping

Justification for K/A match: To match this Tier 2 Systems K/A on the A/C Electrical Distribution system concerning the ability to monitor automatic operations of a breaker tripping, The question displays a set of lights for the Diesel Generator that is paralleled to the grid and a condition who's indication is displayed and asks the students to identify the cause and determine which breaker will trip.

Explanation: CORRECT D: The conditions shown indicate that the D D/G has failed to single unit. This will cause the D/G to pick up load and the white light indicates an overload condition which will trip the SD board Normal feeder breaker 1724.

A. **Incorrect because** – The white light is an overload indication and BKR 1724 will trip first to clear the condition.
Plausible because – A white light above a control switch typically indicates a disagreement (breaker tripped with control switch in normal after close). The candidate may believe that a disagreement exists because the mode switch is in Paralleled with system but the single unit light is light. It is plausible that the D/G breaker would trip first since it has the overload condition.

B. **Incorrect because** – The white light is an overload indication.
Plausible because – Part 1 see A above and Part 2 is correct.

C. **Incorrect because** – The overload condition will trip the Normal feeder breaker first.
Plausible because – Part 1 is correct and It is plausible that the D/G breaker would trip first since it has the overload condition.

Note: If tripping the Normal feeder breaker does not clear the overload condition the 51V relay would then trip the D/G breaker.

Technical Reference(s): 0-OI-57A rev 156,

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.036 rev 15 obj 15A

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:	41(b)(7)

d. Diesel Generator Overload (51X) (1) Picked up at 495 amps (current through generator stator at 2850 KW, 0.8 pf).
(2) Turns on white light above DG output breaker and activates annunciator.
(3) IF in "Parallel with System" mode, normal and alternate feeder breakers are tripped (but NOT locked out).

f. The diesel generator electrical lockout relay (86Gx) senses the following abnormal electrical conditions:
(1) Reverse Power (32).
(2) Loss of Excitation (40).
(3) Overcurrent with voltage restraint (51V) (trips at lower current setpoint when the generator output voltage is low than when the output voltage is normal.
(4) Differential Overcurrent 87G(x).

g. When the 86G(x) relay is tripped, the following takes place:
(1) Output breaker is tripped.
(2) Field breaker is tripped.
(3) Normal engine shutdown sequence occurs.

4160V BKR 1622 1816 OR 1826 OVERLOAD	30

(Page 1 of 1)

Sensor/Trip Point:

Aux. relay:	Relay:	
0-51X-211-000D/20A	0-51-211-000D/20B	A. Overload on diesel bkr 1816, 0-BKR-211-000D/020 ≥ 495 amps.
0-51X-211-000D/06A	0-51-211-000D/06B	B. Overload on tie bkr 1826, 0-BKR-211-000D/006 ≥ 600 amps.
0-51-211-000D/01A		A. Overload on Shutdown Bus 1 bkr 1622, 1500 amps 0-BKR-211-000D/001 ≥ 1500 amps.

Sensor Location: 4160V Shutdown Bd D
Elevation 593
Electric Bd Rm 2B

Probable Cause:
A. Overload.
B. Overcurrent.
C. Sensor malfunction.

Automatic Action: None: Possible white light ILLUMINATED for effected breaker.

Operator Action:
A. **CHECK** for white light ILLUMINATED at effected breaker. ☐
B. **IF** breaker(s) has tripped, **THEN**
HAVE board checked for abnormal conditions, relay target(s), smoke, burned paint, breaker position, etc. **REFER TO** 0-GOI-300-2. ☐

References: 0-45E765-25 0-45E765-21 GE 731E718 Series

MOTOR TRIPOUT
2-EA-57-116

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(Page 1 of 1)

Sensor/Trip Point:

Relay 30MT

Relay 30X on any of the following:

480V motor trip
4160V Common Bd motor trip
4160 Shutdown Bd motor trip
4160V Unit Bd motor trip

Sensor Location: Relay 30MT
Panel 2-9-36
Aux Instrument room

Probable Cause:

A. HS in NORMAL AFTER START and breaker open.
- Overload.
- Overcurrent.
- Ground.

B. Failure of Auto Transfer.
C. Safe stopped locally.
D. Sensor malfunction.

Automatic Action: Affected equipment trips.

NOTE

RHR, Core Spray, and CRD 1B Pumps do not bring in this alarm. These pumps bring in their own tripout alarm.

Operator Action:

A. **CHECK** Control Room for white disagreement light illuminated for affected equipment. ☐
B. **RESET** disagreement light. ☐
C. **DISPATCH** Personnel to CHECK: ☐
1. Relays at associated electrical bd. ☐
2. Equipment for abnormal conditions. ☐
3. Safe-stop locally reset, if necessary. ☐

D. **REFER TO** 0-GOI-300-2 if relay targets are present or for motor starting limits. ☐
E. **REFER TO** appropriate OI for recovery or realignment of equipment. ☐

References:

45N620-11	45N763-3	45N761-3
45N779-2	45N765-10	45N766-24

4160V SD BD D MOTOR OL or TRIP 26	Sensor/Trip Point:
	Relay 51X or 30X — Overload or tripout on any one of the following: CS pump 1D, 2D RHR pump 1D, 2D RHRSW pump D2, D3

(Page 1 of 1)

Sensor Location:
4160V Shutdown Bd. D
Elevation 593
Electric Bd. Rm. 2B

Probable Cause:

A. HS in "Normal After Start" and breaker open:
   1. Overcurrent.
   2. Ground.

B. Overload.
C. Sensor malfunction.

Automatic Action:
Effected equipment has tripped.

Operator Action:

A. **CHECK** control room for white light illuminated on effected equipment. ☐
B. **DISPATCH** personnel to check
   1. Relays at associated electrical bd. ☐
   2. Equipment for abnormal conditions, relay targets, smell, burned paint, breaker position, etc. ☐

References:

0-45E765-25	0-45E724-4	0-45E765-5
1-45E765-4,-7	2-45E765-4,-7	

QUESTION 46 Rev 1

All three Units are operating at 100% power.

- 240V Lighting Board 2A is tagged out of service for scheduled work.

An electrical fault causes 240 V Lighting Board 3B to deenergize.

Which one of the following completes the statements below?

The Plant Preferred MG will start __ (1) __ and energize __ (2) __.

A. (1) immediately
   (2) Panel 9-9 cabinet 4 on all 3 units

B. (1) immediately
   (2) Battery Board 2 Panel 14

C. (1) after a 6 second time delay
   (2) Panel 9-9 cabinet 4 on all 3 units

D. (1) after a 6 second time delay
   (2) Battery Board 2 Panel 14

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002 A1.02	
	Importance Rating	2.5	2.9

262002 Uninterruptable Power Supply (A.C./D.C.): A1.02 Ability to predict and/or monitor changes in parameters associated with operating the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) controls including: Motor generator outputs (CFR: 41.5 / 45.5)

Justification for K/A match: To match this Tier 2 Systems K/A on UPS (BFN Plant Preferred) the normal and alternate power failures are required for the MG to start and load. That is the set up in the question and then asks the candidate to predict when the MG starts and what loads it supplies.

Explanation: CORRECT C: The Plant Preferred MG will start when bus voltage lowers to 95% rated after a 6 second time delay. The Plant Preferred MG supplies all three units Panel 9-9 cabinet 4.

A. **Incorrect because –** There is a time delay of 6 seconds.
Plausible because – The candidate may remember that the MG starts on low bus voltage but forget the time delay and Part 2 is correct.

B. **Incorrect because –** Part 1 see A above and because BB2 Panel 14 will be de-energized.
Plausible because – Part 1 see A above and because Lighting board 2A or 3B normally supply BB2 Panel 14 via the Non-Preferred auto transfer switch. When power from this switch is lost the Plant Preferred MG starts but does not energize BB2 Panel 14.

D. **Incorrect because –** BB2 Panel 14 will be de-energized.
Plausible because – Part 1is correct and Part 2 see B above.

Technical Reference(s): 0-AOI-57-3 Rev 49, 0-AOI-57-6 Rev 19

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.102 rev 07 obj VB3b

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:	41(b)(7)

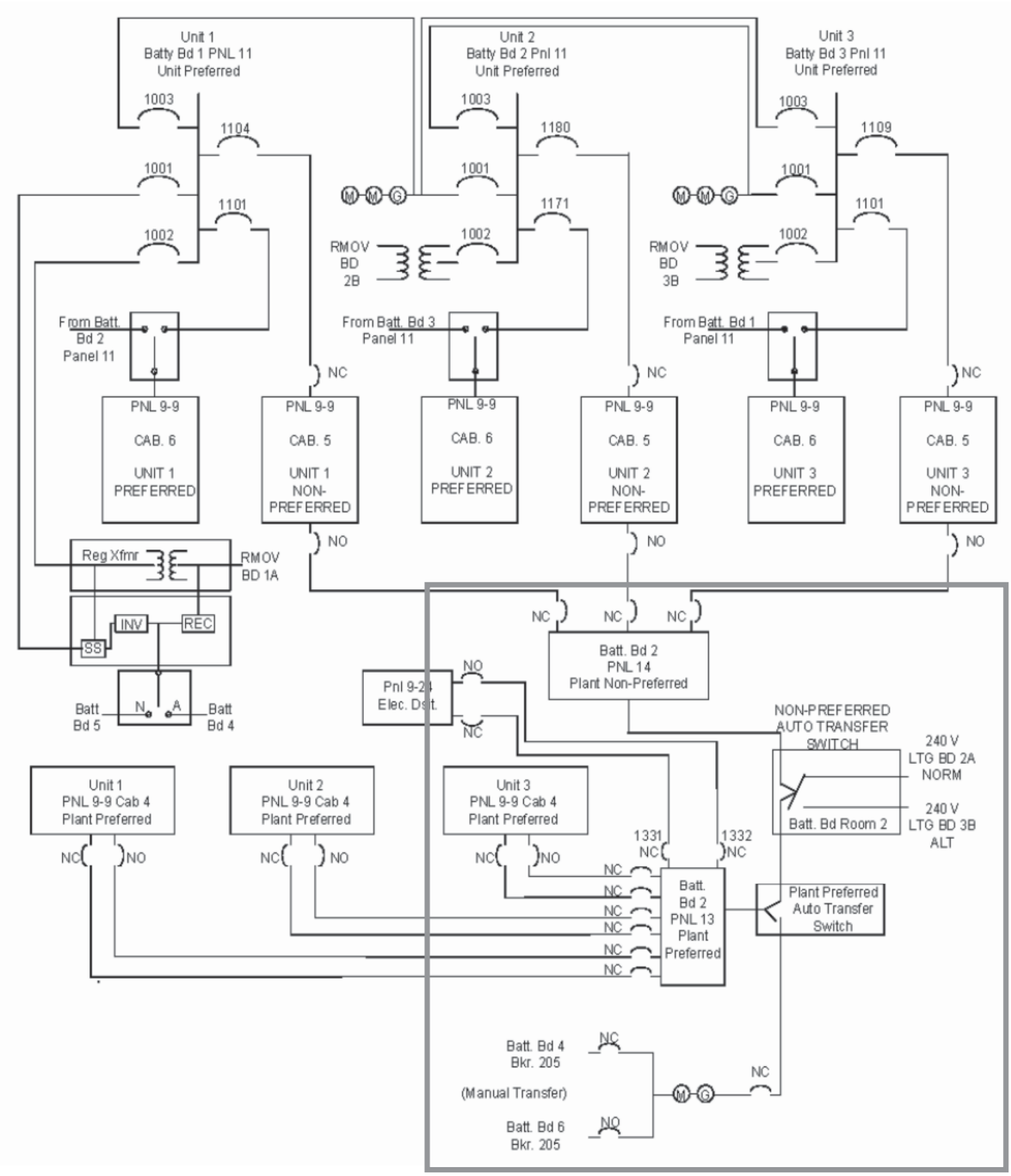
3.0 AUTOMATIC ACTIONS

NOTES

1) Battery Board 2 Panel 13 is normally supplied power from the Nonpreferred Transfer Switch (240V Lighting Board 2A or 3B). Upon loss of transfer switch power, the Plant Preferred MG should start and load to energize the Plant Preferred system.

2) The Plant Preferred MG starts when Plant Preferred bus voltage drops to 95% for 6 seconds. The Plant Preferred Auto Transfer Switch transfers from the lighting board to the Plant Preferred MG when Plant Preferred bus voltage drops to 95% for 6 seconds <u>and</u> MG set output is at 90% rated voltage and frequency. When the lighting board voltage reaches 97% rated for one minute (adjustable up to 30 minutes) the Plant Preferred Auto Transfer Switch transfers back to the lighting board and MG set is automatically disconnected.

Attachment 2
(Page 1 of 1)
Vital 120V AC Distribution



QUESTION 47 Rev 1

Unit 1 is operating at 100% Power.

1-9-8B window 35 UNIT PFD SUPPLY ABNORMAL alarms

The Control Bay AUO reports the following light illuminated at the Unit 1 Unit Preferred System Inverter:

1-IL-252-0001L (Red Lamp) Inverter Fuse Blown

No Operator actions have been performed.

Which ONE of the following completes both statements below?

Unit 1 Panel 9-9 cabinet 5 is on its __ (1) __ power supply.

Unit 1 Panel 9-9 cabinet 6 is on its __ (2) __ power supply.

A. (1) normal
   (2) alternate

B. (1) normal
   (2) normal

C. (1) alternate
   (2) alternate

D. (1) alternate
   (2) normal

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002 K6.03	
	Importance Rating	2.7	2.9

262002 Uninterruptable Power Supply (A.C./D.C.) K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : Static inverter (CFR: 41.7 / 45.7)

Justification for K/A match: To match this Tier 2 Systems K/A on UPS (BFN Plant/Unit Preferred) the question is written to give a malfunction of the Unit Preferred Static Inverter and asks which UPS power panels (9-9 Cab 5 and 6) will be supplied with power (the effect of the malfunction).

Explanation: CORRECT B: When the Unit 1 Unit Preferred inverter fails (blown fuse) the Static Switch will Auto transfer to alternate (powered by the Regulating Transformer). Both 9-9 cabinet 5 and 6 will continue to be powered without transferring to alternate.

A. **Incorrect because –** In this case the static switch will transfer and keep BB1 Panel 11 energized.
Plausible if the candidate does not remember that the Static Switch transfers to the regulating transformer.

C. **Incorrect because –** In this case the static switch will transfer and keep BB1 Panel 11 energized.
Plausible if the candidate does not remember that the Static Switch automatically transfers to the regulating transformer and forgets that 9-9 cabinet 5 must be manually transferred.

D. **Incorrect because –** In this case Cabinet 5 will still be energized from its normal source.
Plausible if the candidate does not remember that the Static Switch automatically transfers to the regulating transformer and confuses which one of the 9-9 cabinets auto transfer.

NOTE: The Unit 2 and 3 Regulating Transfer can supply Unit Preferred but only through a manual transfer. On U1 this transfer is automatic via the static switch.

Technical Reference(s): 0-OI-57C rev 124, 0-AOI-57-4 rev 33, 1-ARP-9-8B Rev 11

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.102 rev 07 Obj V.B.2.b

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: BFN 1404 NRC Exam Q#48
Question Cognitive Level:	Memory or Fundamental Knowledge : Comprehension or Analysis X
10 CFR Part 55 Content:	41(b)(7)

UNIT PFD SUPPLY ABNORMAL 35	Sensor/Trip Point: Relay SE - loss of normal DC power source. Relay TS - DC Xfer switch transfers to Emergency DC Power Source. Regulating Transformer Common Alarm. 1-INV-252-001, INVT-1 System Common Alarm.

(Page 1 of 1)

Sensor Location: EL 593' 250V DC Battery Board 2

Probable Cause:

A. Loss of normal DC power source
B. DC power transfer.
C. Relay failure
D. INVT-1 System Common Alarms
   1. Fan Failure Rectifier
   2. Over temperature Rectifier
   3. AC Power Failure to Rectifier
   4. Low DC Voltage
   5. High DC Voltage
   6. Low DC Disconnect
   7. Fan Failure Inverter
   8. Alternate Source Failure
   9. :Low AC Output Voltage
   10. High Output Voltage
   11. Inverter Fuse Blown
   12. Static Switch Fuse Blown
   13. Over Temperature Inverter

E. PFD Regulating XFMR Common Alarms
   1. Transformer Over temperature
   2. Fan Failure
   3. CB1 Breaker Trip
   4. CB2 Breaker Trip

Automatic Action:

A. Auto transfer to DC Power Source on Rectifier failure.
B. Auto transfer to Alternate AC supply (Regulated Transformer) on Inverter failure.

Operator Action:

A. **IF** 120V AC Unit Preferred is lost, **THEN** **REFER TO** 1-AOI-57-4. ☐
B. **REFER TO** appropriate portion of 0-OI-57C. ☐

References:

0-45E641-2	1-45E620-11	1-3300D15A4585-1
10-100467	0-20-100756	20-110437

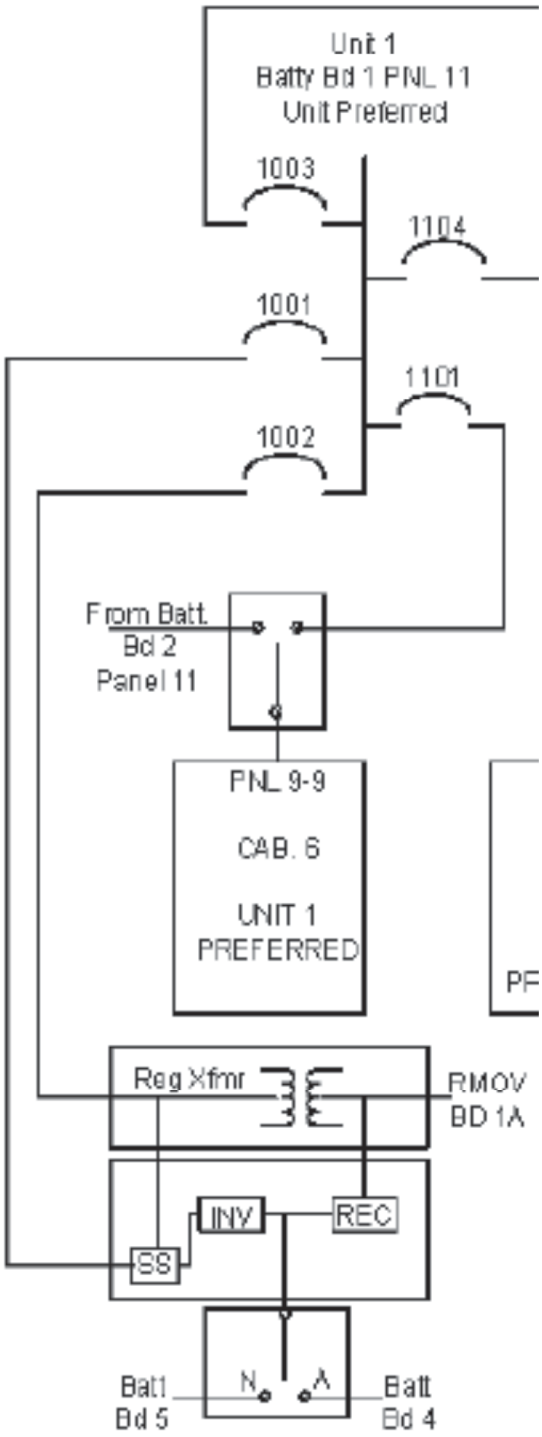
Illustration 4
(Page 1 of 2)

UNIT PREFERRED INVERTER LAMP INDICATORS

1-IL-252-0001L (Red Lamp) Inverter Fuse Blown	Auto Static Switch transfer to Alternate. Manual / Auto transfer back to inverter (normal) blocked.	
1-IL-252-0001D (Red Lamp) Alternate Source Supplying Load	Alternate AC source is powering the load through the static switch.	Requires a Manual transfer of the static switch back to the inverter.

Illustration 1
(Page 1 of 1)

Vital 120V AC Distribution



BFN 1404 NRC Exam Q#48

QUESTION 48

The Control Bay AUO reports that the Unit 1 Unit Preferred System Inverter, 1-INV-252-001, output has failed to zero.

Which ONE of the following completes both statements below?
(Consider each statement separately)

The Unit Preferred Inverter's normal DC Source is Battery Board __(1)__.

In accordance with 1-ARP 9-8B Window 35, UNIT PFD SUPPLY ABNORMAL, Battery Board 1, Cabinet 11 will automatically be powered from the __(2)__.

A. (1) Four
   (2) Unit 2 Unit Preferred MMG set

B. (1) Four
   (2) Unit Preferred Regulating XFMR1

C. (1) Five
   (2) Unit 2 Unit Preferred MMG set

D. (1) Five
   (2) Unit Preferred Regulating XFMR1

ANSWER: **D**

QUESTION 48 Rev 1

Unit 2 is operating at 100% power when the following annunciator alarms:

- 2-9-8C Window 18; 250V REACTOR MOV BD 2C UV

Multiple valve position indicating lights on Panel 2-9-3 for _____ being extinguished would confirm a loss of power on 2C 250V RMOV board?

A. RCIC

B. RHR Loop I

C. HPCI

D. CS Loop II

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	263000 A3.01	
	Importance Rating	3.2	3.3

DC Electrical Distribution, Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: (CFR: 41.7 / 45.7) A3.01 Meters, dials, recorders, alarms, and indicating lights

Justification for K/A match: To match this Tier 2 Systems K/A on DC Electrical Distribution, and to monitor automatic operations of the D.C. ELE DIST including: alarms, and indicating lights. The question is written, gives a condition that would cause an automatic system operation of the DC system and asks the candidate which system's lights will be illuminated or not (ability to monitor).

Explanation: Correct A: In accordance with 2-9-8C Window 18, Operator Actions, Verify alarm by checking light indication on the following equipment: Loss of RCIC and ADS indicating lights on Panel 2-9-3.

B. **Incorrect because** – RHR Loop I vlvs are not powered by 250V DC RMOV Board 2C
Plausible because RHR Loop I logic is powered by 250V DC Reactor MOV Board 2B

C. **Incorrect because** – HPCI valves are not powered by 250V DC RMOV Board 2C
Plausible because most HPCI valves are powered by 250V DC RMOV Board 2A

D. **Incorrect because** – CS Loop II vlvs are not powered by 250V DC RMOV Board 2C
Plausible because CS Loop II logic is powered by 250V DC RMOV Board 2A

Technical Reference(s): 2-ARP-9-8C Rev 16

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.102 OBJ B.2.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:	41(b)(7)

250V REACTOR
MOV BD 2C UV
2-EA-57-105

18

(Page 1 of 1)

Sensor/Trip Point:

72N-BA	Overcurrent on normal supply
72E-BA	Overcurrent on alternate supply
27EX	Undervoltage on supply voltage to MOV Bd.
27B	Undervoltage on MOV Bd (7 sec TDDO)

Sensor Location: 250V Rx MOV Bd 2C
El 565', R-8 Q-LINE
Rx Bldg

Probable Cause:
A. Loss of normal supply (250V Battery Bd 1, Pnl 3, Bkr 303).
B. Overcurrent on normal or alternate supply to the board.
C. Fuse failure.
D. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** alarm by checking light indication on the following equipment:
- Loss of RCIC indicating lights on Panel 2-9-3.
- Loss of ADS indicating lights on Panel 2-9-3.

B. **DISPATCH** Personnel to MOV board and **CHECK** for abnormal conditions: undervoltage, breaker tripped, etc.

NOTE

[II/C] If the mechanism resets (hear click and feel resistance when pushing in), this indicates that an overcurrent condition tripped the breaker.

C. **IF** Normal or Alternate feeder breaker tripped, **THEN** Manually **DEPRESS** mechanical trip/reset mechanism on breaker face to reset Bell Alarm lockout device.[NER/C II-B-92-069]

D. **VERIFY** bkr 303 closed at Panel 3, Battery Bd Room 1, El 593'.

E. **REFER TO** 0-OI-57D, to re-energize or transfer the board.

F. **REFER TO** appropriate OI for recovery or realignment of equipment.

G. **REFER TO** TS Sect. 3.8.7

250V REACTOR
MOV BD 2A
UV
2-EA-57-94

4

(Page 1 of 1)

Sensor/Trip Point:

72N-BA	Normal supply overcurrent.
72E-BA	Alternate supply overcurrent.
27EX	Normal supply undervoltage.
27B	MOV bd undervoltage (7sec TDDO)

Sensor Location: 250V RMOV Bd 2A, El 621', R-14 Q-LINE, Shutdown Bd Rm C

Probable Cause:
A. Loss of normal supply (250V Battery Bd 2, Pnl 3, Bkr 302).
B. Overcurrent on normal or alternate supply to the board.
C. Fuse failure.
D. Sensor malfunction.

Automatic Action: None.

Operator Action:

A. **VERIFY** conditions of alarm:
- Loss of HPCI indicating lights on Panel 2-9-3. ☐
- Loss of backup scram valve lights on Panel 2-9-5. ☐

B. **DISPATCH** Personnel to MOV board to check for abnormal conditions: undervoltage, breaker tripped, etc. ☐

NOTE

[II/C] If the mechanism resets (hear click and feel resistance when pushing in), this indicates that an overcurrent condition tripped the breaker.

C. **IF** Normal or Alternate feeder breaker tripped, **THEN** Manually **DEPRESS** mechanical trip/reset mechanism on breaker face to reset Bell Alarm lockout device.[NER/C II-B-92-069] ☐
D. **VERIFY** Bkr 302 closed at Battery Bd Room 2, Panel 3, El 593'. ☐
E. **REFER TO** TS Section 3.8.7. ☐
F. **REFER TO** 0-OI-57D to re-energize or transfer the board. ☐
G. **REFER TO** appropriate OI for recovery or realignment of equipment. ☐

References: 45N620-11 2-45E712-1 45N714-7
TS Section 3.8.7.

250V REACTOR
MOV BD 2B
UV
2-EA-57-100

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(Page 1 of 1)

Sensor/Trip Point:

72N-BA	Normal supply overcurrent
72E-BA	Alternate supply overcurrent
27EX	Normal supply undervoltage
27B	Rx 2B MOV bd undervoltage (7sec TDDO)

Sensor Location: 250V RMOV Bd 2B, El 593', R-14 Q-LINE, Shutdown Bd Rm B

Probable Cause:

A. Loss of normal supply (250V Battery Bd 3, Pnl 3, Bkr 303).
B. Overcurrent on normal or alternate supply to the board.
C. Fuse failure.
D. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** alarm by checking:
- Loss of HPCI and RHR indicating lights (Panel 2-9-3).
- Loss of backup scram valve lights (Panel 2-9-5).

B. **DISPATCH** Personnel to MOV board to check for abnormal conditions: undervoltage, breaker tripped, etc.

QUESTION 49 Rev 3

All three Units are operating at 100% power.

Subsequently,

4KV SD BUS 2 de-energizes.

Which of the following completes the statement below?

The __ (1) __ Diesel Generators will auto start.

In accordance with 0-OI-82, Standby Diesel Generator System, the Diesel Generator Maximum **Continuous** steady-state active power output (KW) is limited to __ (2) __.

A. (1) B and D
   (2) 2600 kW

B. (1) B and D
   (2) 2860 kW

C. (1) C and D
   (2) 2600 kW

D. (1) C and D
   (2) 2860 kW

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	264000 A2.06	
	Importance Rating	3.4	3.4

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

Justification for K/A match: This is a Tier 2 Systems K/A concerning the Emergency DGs and the ability of the student to predict the impacts of opening the normal or alternate power supply and use procedures to mitigate the consequences. The question is constructed to provide a situation where the Shutdown bus Feeder Breaker opens starting the DG for the associated board that was de-energized and asks the student an operational limit based on the conditions provided, matching the (b) part of the A2 K/A.

Explanation: CORRECT C: When 4kV Shutdown Bus 2 is de-energized. 4KV Shutdown Board C and D loses power and C and D DGs Start and output breakers close automatically once it reaches rated voltage and frequency. The maximum Continuous steady-state active power output is limited to 2600 KW.

A. **Incorrect because –** The C and D D/G will auto start
Plausible because – The B and D D/G supply Unit 2 480V Shutdown boards and the candidate may believe shutdown bus 2 supplies Unit 2 and Part 2 is correct

B. **Incorrect because –** Part 1 see A above and 2860KW is the Short-time (0-2 hours) limit.
Plausible because –Part 1 see A above and Part 2 because this is a D/G limit.

D. **Incorrect because –** 2860KW is the Short-time (0-2 hours) limit.
Plausible because – Part 1 is correct and 2860 KW is a D/G limit.

Technical Reference(s): 0-OI-82 rev 153, 1/2-ARP-9-23D rev 25

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.038 Rev 20 Obj. 1and 4

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:	41(b)(7)

3.0 PRECAUTIONS AND LIMITATIONS (continued)

J. Standby Diesel Generators are required to be operated at or below the following ratings:

Rating		Description	Time
Engine - Short-Time		Maximum steady-state active power output (running kW)	0 - 2 hours
2860 kW	2800 kW*		
Engine - Continuous		Maximum steady-state active power output (running kW)	greater than 2 hours
2600 kW	2550 kW*		

4160V SD
BD D
DEGRADED VOLTAGE
28

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Sensor/Trip Point:

0-27-211-000D/11D
or 0-27-211-000D/11C

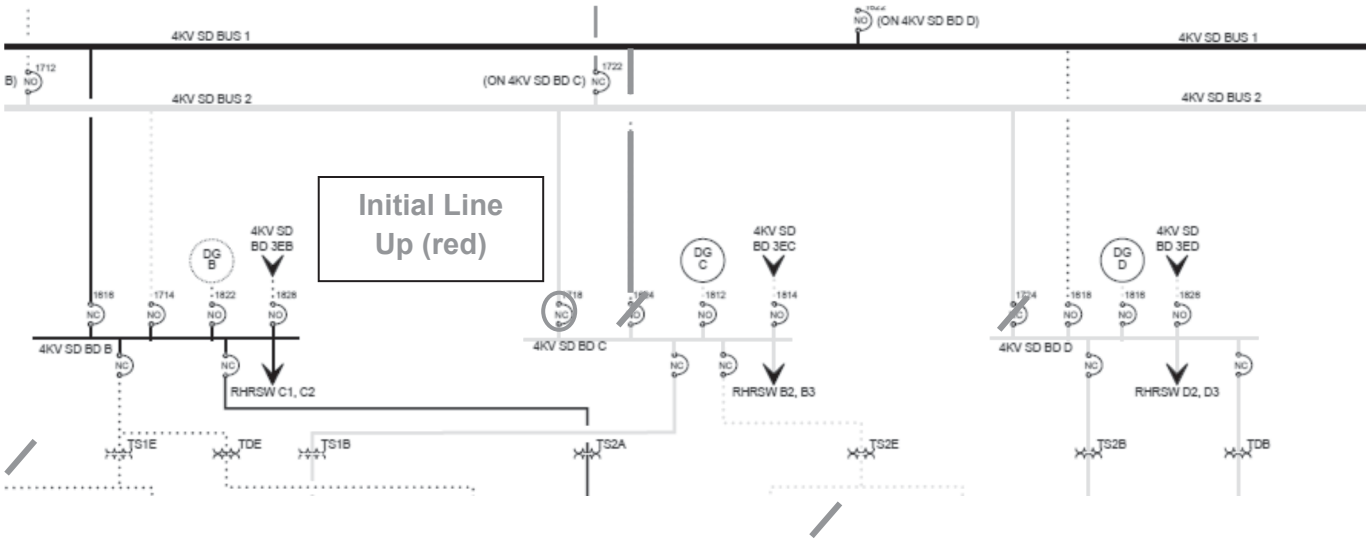
A. Energized by relays 0-27-211-000D/21A,21B, 21C a two out of three logic at 3920V lowering after ~4.3 sec delay.

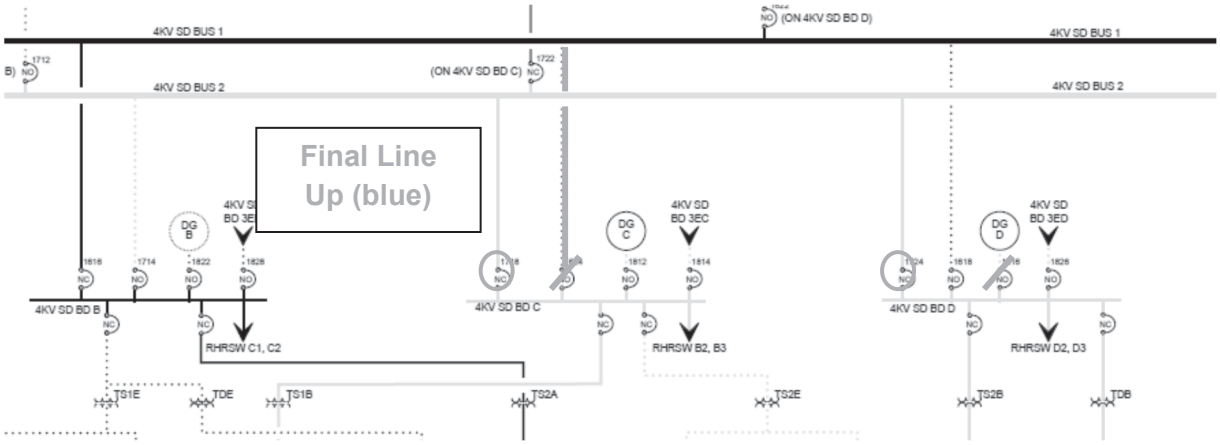
B. Energized by relays 0-27-211-000D/11A&11B at 0 Volts after ~1.5 sec delay.

Sensor Location: 4kV Shutdown Bd D
Elevation 593
Electric Bd Rm 2B

Probable Cause:
A. Supply voltage low.
B. Fuse failure.
C. Sensor malfunction.

Automatic Action:
A. Diesel generator starts.
B. Approximately 2.9 seconds later, load shed logic relay picks up (2-211-3D).
C. Approximately 1.3 seconds after load shed logic, diesel generator breaker logic picks up (relay 2-211-4D).


Initial Line Up (red)
4KV SD BUS 1
4KV SD BUS 2
(ON 4KV SD BD D)
(ON 4KV SD BD C)
4KV SD BD 3EB
4KV SD BD 3EC
4KV SD BD 3ED
DG B
DG C
DG D
4KV SD BD B
4KV SD BD C
4KV SD BD D
RHRSW C1, C2
RHRSW B2, B3
RHRSW D2, D3
TS1E
TDE
TS1B
TS2A
TS2E
TS2B
TDB


Final Line Up (blue)
4KV SD BUS 1
4KV SD BUS 2
(ON 4KV SD BD D)
(ON 4KV SD BD C)
4KV SD BD 3EC
4KV SD BD 3ED
DG B
DG C
DG D
4KV SD BD B
4KV SD BD C
4KV SD BD D
RHRSW C1, C2
RHRSW B2, B3
RHRSW D2, D3
TS1E
TDE
TS1B
TS2A
TS2E
TS2B
TDB

QUESTION 50 Rev 1

Which one of the following completes the statement below?

When Control Air is lost, the Drywell Control Air System…

A. loses its only backup source of pneumatics.

B. loses one of two backup sources of pneumatics.

C. slowly depressurizes.

D. remains pressurized due to installed accumulators only.

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000 K3.01	
	Importance Rating	2.7	2.9

Instrument Air System; Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: Containment air system

Justification for K/A match: This is a Tier 2 Systems K/A on the Instrument Air System (BFN Control Air) and the effects that a loss or malfunction of the (Control Air System) has on Containment air system (BFN Drywell Control Air). To match this K/A, the question just simply asks the candidate to recall the interrelationship between the two system and which is normally in service and the effect that a backup system loss will have on that pneumatic supply.

Explanation: Correct B. There are two backup pneumatic sources to Drywell Control Air (Control Air and CAD).

A. **Incorrect because –** There are two backup pneumatic sources to Drywell Control Air (Control Air and CAD).
Plausible because – Control air is not a preferred back up when at power due to the introduction of O_2 to the Drywell.

C. **Incorrect because –** Drywell Control Air is normally supplied by Nitrogen storage tanks that will maintain pressure.
Plausible because – The candidate may think the Nitrogen isolation valves are control air operated valves.

D. **Incorrect because –** Drywell Control Air is normally supplied by Nitrogen storage tanks that will maintain pressure.
Plausible because – There are installed accumulators that will maintain functionality of some valves in the Drywell for a period of time.

Technical Reference(s): 0-OI-32 Rev 134, OPL171.054 Rev 15

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.054 Objective 7

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:	41(b)(7)	

D. **Relationships to Other Systems

The plant / unit air system serves virtually every other system in the plant. Some of the major systems and their interdependencies include:

1. Drywell Control Air System Drywell Control Air is normally supplied from the Nitrogen Storage Tank, but can be supplied from Plant Control Air or CAD **A** (**B**).

   a. Whenever the plant control air system is being used to supply the drywell control air system, any leak or break inside the drywell will directly introduce oxygen into the inerted drywell.

   b. Any leakage into the drywell from the drywell control system will also contribute to a higher containment pressure and subsequent additional venting requirements.

   c. A calculation to evaluate the rupture of the DCA headers inside the drywell during an accident determined that injecting all the nitrogen from the 6000 gallon Containment Inerting system liquid nitrogen storage tank would take approximately 12 days and would only raise the drywell and suppression pool pressure to approximately 33 psig.

   d. When CAD is aligned to drywell control air only nitrogen will be introduced to the system.

      (1) If a break of the drywell control air system occurs while CAD is aligned the only net effect is a higher containment pressure and a depletion of the nitrogen supply.

      (2) The CAD to drywell crosstie provides long term MSRV accumulator gas supply in order to fulfill Appendix R fire requirements. It can also be used during short periods as a backup to drywell control air.

   e. Drywell control air can be cross-tied to supply the outboard MSIVs on all three units.

   f. Drywell Control Air supplies normal air supply to the inboard MSIVs, MSRVs, and other pneumatically operated equipment inside the drywell.

   g. Loss of DWCA without a backup source available will force all pneumatically controlled components within the drywell to go to their fail position. For example, see items 2 and 6 below.

QUESTION 51 Rev 1

The G Control Air compressor is in service with Control Air Compressors A, B, C, and D in Standby in accordance with 0-OI-32, Control Air System.

Subsequently,

The G Control Air compressor trips due to low oil pressure.

Control Air pressure lowered to 90 psig and then recovered.

Which one of the following describes the response of the Control Air compressors?

A. Compressors A, B, C, and D started simultaneously.

B. Only compressors selected as lead compressors started.

C. The lead compressors started first then the lag compressors started with 2 psig offset between them.

D. The compressors started one at a time with 2 psig offset between each start.

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000 K5.01	
	Importance Rating	2.5	2.5

300000 Instrument Air System (IAS): Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Air compressors (CFR: 41.5 / 45.3)

Justification for K/A match: This is a Tier 2 Systems K/A on the Instrument Air System and the operational implications of the Air Compressors. The question matches this K/A because it asks systems knowledge concerning the operational implications of tripping the normally in-service air compressor.

Explanation: CORRECT C: In accordance with 0-OI-32 as control air pressure lowers
The two Lead standby compressors will start at 98psig followed by the first lag at 96psig and second lag at 94psig. All four compressors start and load by 94 psig.

A. **Incorrect because – T**he air compressors do not start simultaneously.
Plausible because – The two lead compressors both start at 98psig

B. **Incorrect because –** The Control air pressure lowered below the lag compressor set value of 96 and 94 psig.
Plausible because – If control air pressure was maintained above 96 psig this would be correct.

D. **Incorrect because –** Both lead compressors start at 98 psig.
Plausible because – The lag compressors start with a 2 psig offset between starts.

Technical Reference(s): 0-OI-32 rev 134

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.054 obj 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:	41(b)(7)

3.0 PRECAUTIONS AND LIMITATIONS

C. The Control Air Compressors are normally lined up as follows:

1. Compressor G running and loaded, maintaining compressor discharge pressure at approximately 105 psig

2. Two compressors (A, B, C, or D) in LEAD with ONLINE pressure setpoint of 98 psig and OFFLINE pressure setpoint of 108 psig.

3. Two Compressors (A, B, C, or D) with LAG OFFSET of 2 psig for the first lag compressor and LAG OFFSET of 4 psig for the second lag compressor depending upon the desired sequence.

QUESTION 52 Rev 1

All three units are operating at 100% power.
The A3 RHRSW pump is tagged for motor replacement.

Subsequently:

The C-3 EECW pump trips and the AUO reports that the pump is hot to the touch.

In accordance with 0-OI-67, Emergency Equipment Cooling Water System, which ONE of the following completes the statements below?

RHRSW pump __ (1) __ can be aligned to EECW in place of the C-3 RHRSW pump.

This pump __ (2) __ the same AUTO start signals as the C-3 RHRSW pump.

A. (1) C-1
   (2) has

B. (1) C-1
   (2) does NOT have

C. (1) C-2
   (2) has

D. (1) C-2
   (2) does NOT have

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000 A2.01	
	Importance Rating	3.3	3.4

CCW Sys; Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Loss of CCW pump

Justification for K/A match: This is a Tier 2 Systems K/A, however it is an A2, which asks the candidate to, use procedures to correct, control, or mitigate the consequences of those abnormal operation. The question sets up a loss of EECW North header pumps and asks the candidate to determine IAW 0-OI-67 which RHRSW pump can be aligned to EECW in place of the C-3 pump. The question also asks systems knowledge concerning the pump auto start signals matching the Tier 2 part of the K/A.

Explanation:CORRECT B: IAW 0-OI-67 section 8.3 C-1 RHRSW pump may be aligned to EECW in place of C-3 RHRSW. IAW 0-OI-67 step 3.0.D The number 1 RHRSW pumps do not have the same auto start signals as the associated number 3 pump.

A. **Incorrect because**–D The number 1 RHRSW pumps do not have the same auto start signals as the associated number 3 pump.
Plausiblebecause–Part 1 is correct and because some of the auto start signals are the same for example common accident signals and low RCW header pressure signals are the same for both pumps however, C-1 starts from a Unit 1 or 2 CS or D/G start while C-3 starts from a Unit 3 CS or D/G start.

C. **Incorrect because** – 0-OI-67 does not allow aligning the C-2 pump to EECW and the pumps do not have the same auto start signals.
Plausiblebecause– The C-2 RHRSW pump is in the same room as the C-3 pump and it is possible to align it to the EECW header.

D. **Incorrect because** – Part 1 see C above.
Plausiblebecause – Part 1 see C above and Part 2 is correct.

Technical Reference(s): 0-OI-67 Rev108, 1-47E859-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: BFN NRC Exam 1205 Q#7
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis: X
10 CFR Part 55 Content:	41(b)(4)

8.3 Operation of RHRSW Pump C1 (for EECW in place of C3)

NOTES

1) RHRSW Pump C1 may be aligned for service by this section when:
   - It is used to meet the minimum number of Tech. Spec. operable pumps; or
   - At the discretion of the Unit Supervisor, it is needed to replace another pump's operation; or
   - At the discretion of the Unit Supervisor, it is needed to assist in supplying header flow/pressure demand.
2) If used to meet EECW requirements, RHRSW pump C1 must be aligned to EECW, the pump started, and should remain running. RHRSW Pump C1 does **NOT** have the same auto start signals as RHRSW Pump C3.

3.0 PRECAUTIONS AND LIMITATIONS

E. RHRSW Pumps A3, B3, C3, and D3 as well as A1, B1, C1, and D1, when lined up for EECW operation, will auto-start when either:
   1. Any unit Common Accident Signal Relay is energized. (High Drywell Pressure in conjunction with low reactor pressure, or Low-Low-Low Reactor Water Level.)
   2. Low Raw Cooling Water header pressure at control air compressor (less than 30 psig).
   3. Low Raw Cooling Water pressure at RBCCW heat exchanger (less than 15 psig).

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.

QUESTION 53 Rev 0

The following plant conditions exist for Unit 1 RBCCW System.

- A loss I&C A has deenergized the RBCCW Surge Tank fill valve.

Which way does 1-FCV-70-1, RBCCW Surge Tank fill valve, fail and where do you send someone to control level in the RBCCW Surge Tank?

A. Open; Reactor Building EL 639 foot.

B. Open; Reactor Building EL 717 foot.

C. Closed; Reactor Building EL 639 foot.

D. Closed; Reactor Building EL 717 foot.

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000 K6.01	
	Importance Rating	2.7	2.8

400000 Component Cooling Water System (CCWS): K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Valves (CFR: 41.7 / 45.7)

Justification for K/A match: This is a Tier 2 Systems K/A on RBCCW concerning a loss or malfunction of valves and how that affects the system. The question asks the candidate to recall how the head tank fill valve fails (matches malfunction of valve in system) and then where to control the tank level (knowledge of local control of head tank level matches effect on the system).

Explanation: C is CORRECT: The 1-FCV-70-1 fails closed on a loss of I&C A and the surge tank level must be controlled in the reactor building at the Surge Tank located at EL RB 639 foot.

A. **Incorrect because –** the valve does not fail open
Plausible because – 1-FCV-70-1 failing open would preserve RBCCW operation and other RBCCW control valves such as TCV-70-49 do fail open on a loss of power.

B. **Incorrect because –** Part 1 see A above and Part 2 is the incorrect elevation.
Plausible because – Part 1 see A above because the Condensate head tank is located on the Reactor Bldg Roof EL 717

D. **Incorrect because –** the location is incorrect
Plausible because – the Condensate head tank is located on the Reactor Bldg Roof EL 717.

Technical Reference(s): 1-OI-70 Rev 54;; OPL171.047 Rev 12.;
0-OI-2C/ATT-1A Rev 64.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): N/A

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:	41(b)(7)

6.2 Adding Water to Surge Tank

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

A. RBCCW in operation. ☐

B. Demineralized Water System in service in accordance with 0-OI-2C. ☐

[2] **STATION** an operator to monitor RBCCW Surge Tank level at the local RBCCW System Surge Tank sight glass. ☐

NOTE
Excessive makeup to the RBCCW Surge Tank may indicate a leak.

[3] **ADD** water to the RBCCW Surge Tank using the following:

- **OPEN** RBCCW SYS SURGE TANK FILL VALVE, 1-FCV-70-1 using 1-HS-70-1 on Panel 1-9-4, OR ☐
- **LOCALLY OPEN** manual FCV-70-1 BYPASS VLV, 1-BYV-002-1369. ☐

[4] **WHEN** RBCCW Surge Tank is filled to desired level (normal level band is from 4 inches below to 4 inches above tank centerline), **THEN**

- **CLOSE** RBCCW SYS SURGE TANK FILL VALVE, 1-FCV-70-1 using 1-HS-70-1 on Panel 1-9-4, **OR** ☐
- **LOCALLY CLOSE** manual FCV-70-1 BYPASS VLV, 1-BYV-002-1369. ☐

[5] **RECORD** addition of water to the RBCCW Surge Tank in the Narrative Log. ☐

[6] **IF** a significant amount of water has been added to the RBCCW System, **THEN**

NOTIFY Chemistry. ☐

5. Process radiation monitoring system provides monitoring of RCW flow from the RBCCW heat exchangers to detect leakage of potentially contaminated systems. Another radiation monitor in the RBCCW return line just prior to the heat exchangers monitors for leakage into the RBCCW system. A failure of either of these monitors would require alternate sampling methods be established.
6. Demineralized water system provides a source of make up water to the RBCCW system. A loss of this system may lead to pump cavitation and possible loss of RBCCW system flow. An alternate make up source may be required to maintain RBCCW system operation.
7. 480V AC distribution provides power to RBCCW system pumps and valves/dampers. A loss 480 volt distribution may result in a partial or complete loss of RBCCW flow.
8. I&C A and B provides power to the drywell cooling dampers. A loss of this power causes the dampers to fail open. In addition, I&C A provides power to the RBCCW surge tank fill valve. A loss of this power supply will cause the fill valve to fail closed.
9. Plant preferred (9-9 cabinet 4) provides power to the RBCCW pump suction header temperature indicator TIS- 70-3. A loss of this power supply will cause a loss of temperature indication and annunciation.

8. Temperature control on RBCCW Heat Loads

d. TCV-70-49 controls the RBCCW flow through the RWCU Non-Regenerative Heat Exchange based on the RWCU temperature leaving the heat exchanger. TCV-70-49 fails open.

e. An RBCCW chemical feeder is provided for addition of chemicals for rust inhibition and pH control of RBCCW.

f. A surge tank is located above the highest point in system. It allows for water expansion within the system and the addition of make up water.

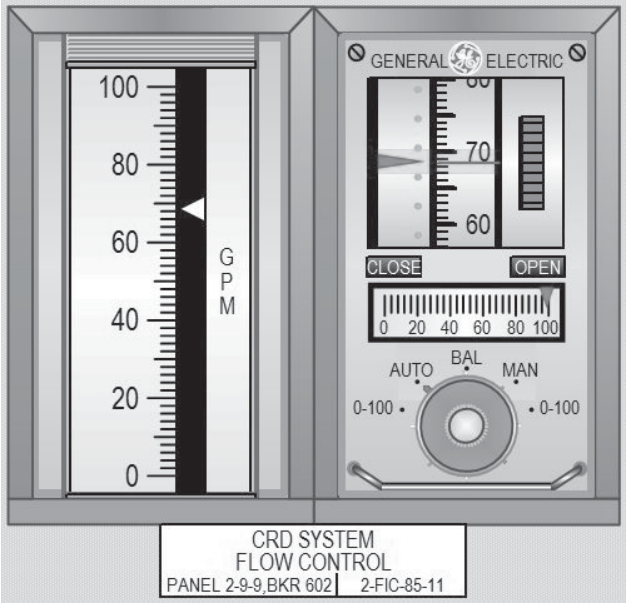
2.0 ATTACHMENT DATA (continued)

Performed On: _____

Valve Number	Valve Description	Required Position	Initials
	Reactor Bldg - El 639'		
1-SHV-002-1208	SLC DEMIN WTR SPLY VLV (R2P)	CLOSED	_____
1-SHV-002-1368	RBCCW SURGE TANK MAKEUP VLV (R2T)	OPEN	_____
1-BYV-002-1369	FCV-70-1 BYPASS VLV (R1S)	CLOSED	_____
1-SHV-002-1189	FCV-70-1 INLET VLV (R1S)	OPEN	_____
	Reactor Bldg - El 621'		
1-SHV-002-1366	RWCU DEMIN WTR SUPPLY (R6T)	CLOSED	_____
1-SHV-002-1677	DEMIN WTR TO SAMPLE PANEL VLV (R2T)	OPEN	_____
1-SHV-002-1678	DEMIN WTR TO HDR VLV (R6T)	OPEN	_____
1-SHV-002-1679	DEMIN WTR TO PANEL 25-9 VLV (R6T)	OPEN	_____

QUESTION 54 Rev 2

U2 is operating at 100 % power.
The following conditions exist on U2:



Which of the following completes the statement below?

In accordance with 2-OI-85, Control Rod Drive System, The Control Rod Drive system flow __ (1) __ in the normal band.

The next time the UO adjusts CRD system flow, Calculated Thermal Power __ (2) __ be affected.

A. (1) is
(2) will

B. (1) is
(2) will NOT

C. (1) is NOT
(2) will

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

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201001 A3.06	
	Importance Rating	2.8	2.8

201001 Control Rod Drive Hydraulic System: A3.06 Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: Reactor power

Justification for K/A match: This is a Tier 2 Systems K/A concerning the CRD hydraulic system and the ability to monitor automatic operations tied to reactor power. To match the CRDH automatic operation, an automatic valve operation was selected, and to match how this change in CRDH flow will affect reactor power the question then asks how actual power is affected.

Explanation: CORRECT C: In accordance with 2-OI-85, Control Rod Drive System, The Control Rod Drive system flow is normally maintained between 40gpm and 65gpm . In accordance with 0-TI-248, Station Reactor Engineer, the primary inputs into the nuclear heat balance calculation are the Reactor Feed Water (RFW) flow rates, the RFW temperatures, the control rod drive (CRD) flow rate,…

A. **Incorrect because –** The normal band is between 40gpm and 65gpm.
Plausible because – 2-OI-85 section 6.10 allows operation at elevated flow up to 80gpm and Part 2 is correct.

B. **Incorrect because –** The normal band is between 40gpm and 65gpm and CRD flow will affect the calculated thermal power.
Plausible because – Part 1 see A above and because the candidate may not think about CRD flow affecting the calculated power since CRD flow is not directly injected into the core via the feedwater lines but through the CRD mechanisms.

D. **Incorrect because –** CRD flow will affect the calculated thermal power.
Plausible because – Part 1 is correct and Part 2 see B above.

Technical Reference(s): 0-TI-248 Rev104, 2-OI-85 Rev141

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.233 Obj. 8, OPL 171.005 Obj. 33

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:	41(b)(7)

5.1 Control Rod Drive Hydraulic System Startup (continued)

[7.13] **ESTABLISH** the following by alternately adjusting tape setpoint of CRD SYSTEM FLOW CONTROL, 2-FIC-85-11 and throttled position of CRD DRIVE WATER PRESS CONTROL VLV, 2-HS-85-23A:

- CRD CLG WTR HDR DP, 2-PDI-85-18A, between 10 psid and 20 psid.
- CRD DRIVE WTR HDR DP, 2-PDI-85-17A, between 250 psid and 270 psid.
- CRD SYSTEM FLOW, 2-FIC-85-11, between 40 gpm and 65 gpm.

6.10 CRD Pump Operation At Elevated Flow

[1] **VERIFY** CRD System in service. Refer to Section 5.1. ☐

[2] **REVIEW** all Precautions and Limitations in Section 3.6. ☐

CAUTIONS

1) Elevated flow rates are likely to reduce drive water D/P and raise CRD cooling water D/P.
2) Elevated flows for extended periods result in CRD Graphitar Seal erosion and should only be used to maintain temperatures below 350°F.

NOTE

PUMP DISCH THROTTLING valve, 2-85-527, has been set to supply 1500 psig charging water pressure and Unit Supervisor authorization is required prior to changing valve position.

[3] **PERFORM** the following steps concurrently, as required, to establish a maximum of 80 gpm, as indicated on CRD SYSTEM FLOW CONTROL, 2-FIC-85-11

Appendix Q
(Page 1 of 15)

Nuclear Heat Balance Monitoring and Trending

The core thermal power is calculated by a heat balance across the RPV. This calculation is performed automatically by the process computer and core monitoring system to provide regular updates to the Reactor Operators. The Reactor Operators are expected to maintain core thermal power as close to 100% RTP as possible without exceeding 100% RTP. The primary inputs into the nuclear heat balance calculation are the Reactor Feed Water (RFW) flow rates, the RFW temperatures, the control rod drive (CRD) flow rate, the reactor water cleanup (RWCU) flow rates and inlet and outlet temperatures, the reactor recirculation pump powers and the reactor pressure. The parameters which have the largest impact on the nuclear heat balance calculation are the RFW flows and the RFW temperatures.

F. Powerplex Heat Balance Calculation (NHB)

Powerplex calculates the core thermal power (CTP) by obtaining a flow and enthalpy for the terms in parenthesis (heat credited to the reactor) and subtracting the power input provided by the Recirc Pumps.
The core gets credit for heating up everything in parenthesis.
(Steam flow is not used because it is difficult accurately measure.)

$$Q_{Rx} = [Q_{FW} + Q_{RWCU} + Q_{CRD} + Q_{Amb}] - Q_{Recirc}$$

QUESTION 55 Rev 0

Which ONE of the following identifies how the collet fingers are released from the notch in the index tube during a normal control rod withdrawal, and if the Control Rod is difficult to withdraw how does the operator adjust Drive Water differential pressure?

A short insert signal ports __ (1) __ gpm of Drive Water to the P-under port, under the drive water piston to release the collet fingers.

Drive water differential pressure is raised by throttling __ (2) __ the CRD Drive Water pressure control valve.

A. (1) 2
   (2) open

B. (1) 2
   (2) closed

C. (1) 4
   (2) open

D. (1) 4
   (2) closed

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	201003 A1.02	
	Importance Rating	3.1	3.1

201003 Control Rod and Drive Mechanism: A1.02 Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: CRD drive pressure (CFR: 41.5 / 45.5)

Justification for K/A match: This is a Tier 2 Systems K/A concerning the CRDM and the ability to monitor changes in parameters, such as drive flow and differential pressure when operating the CRDM.

Explanation: CORRECT D: The insert signal ports 4gpm of drive water flow to the P-under port, under the drive water piston. Since the Drive Water pressure Control valve is downstream of the Drive Water Header throttling it closed will raise Drive Water pressure.

A. **Incorrect because** – this is the wrong drive flow, it's the drive out flow, not the insert flow.
Plausible if the candidate thinks the Drive Water taps off downstream of the Drive Water Pressure Control valve.

B. **Incorrect because** – this is the wrong drive flow, it's the drive out flow, not the insert flow.
Plausible if the candidate thinks the directional control valves for the CRD for withdrawal are the only ones that actuate on a drive out.

C. **Incorrect because** – This is the wrong direction to throttle the valve, this would lower drive pressure delta pressure.
Plausible in that on most systems to raise discharge pressure, you throttle the downstream valve open, this is backwards for the CRD system for drive pressure.

Technical Reference(s): OPL 171.005 Rev 20, OPL 171 006 Rev 10, 47E820-2 Rev12

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.005 rev 20 obj 4

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: VY 2010 Q #60
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)(6)

OPL171.005, Control Rod Drive (CRD) Hydraulics Rev. 20

e. Driving water section

4) Operation of the directional control valves is accomplished by energizing two of the four valves simultaneously, so that the drive water header is connected to either the under- or overpiston area while the exhaust header is simultaneously connected to the opposite side of the drive piston.

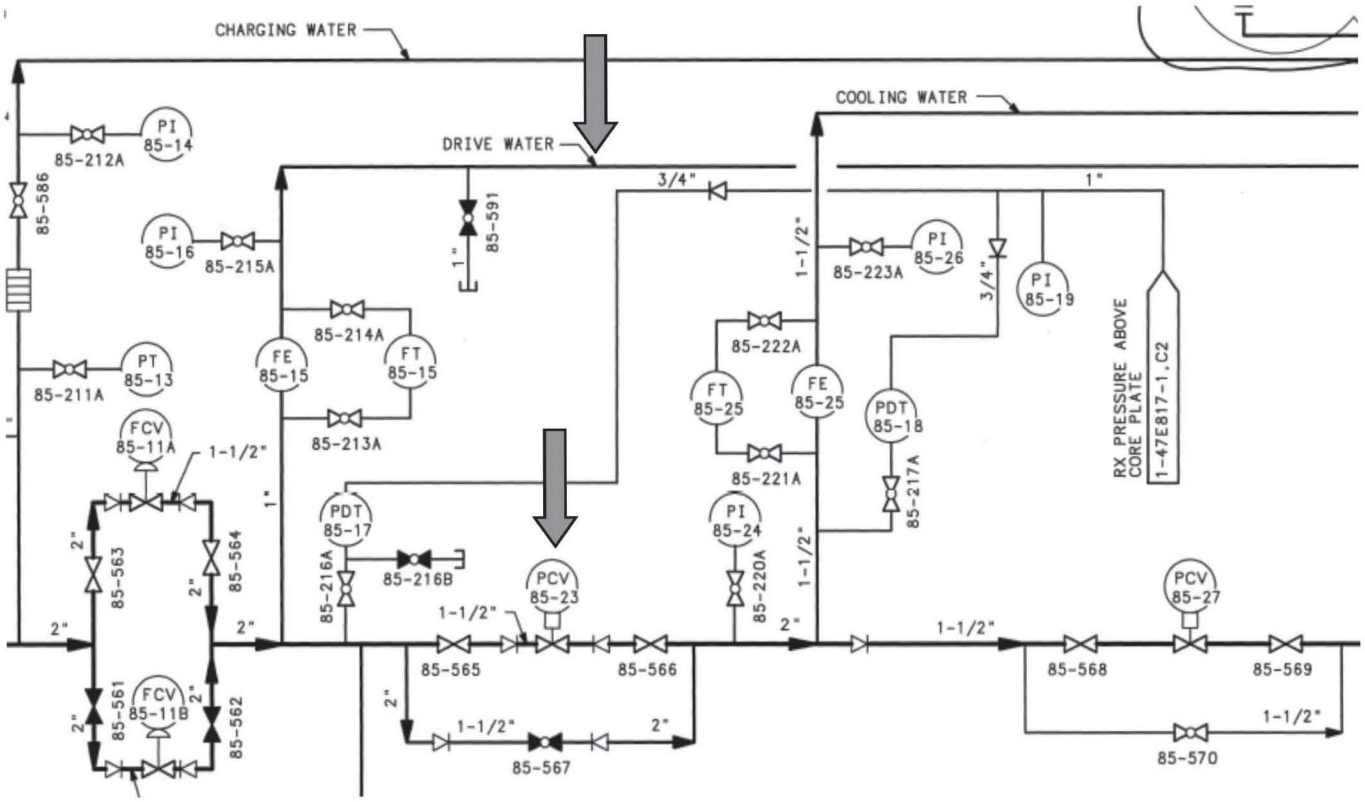
8) Flow rate
a) Flow is approximately 4 gpm when the drive is being inserted at 3 inches per second, and 2 gpm during drive withdraw operation.

OPL171.006, Control Rod Blade and Drive Mechanism, Rev. 10

C. Operational Characteristics
1. Flow Paths for Drive Operation
a. Insert function
(1) Water from drive water header enters at the P-under port under drive piston.
(2) At the same time, water in the drive is discharged to the exhaust water header via the P-over port.
(3) Flow is up between index tube and piston tube through buffer holes, and down between piston tube and indicator tube out to P-over port.

c. Withdraw function
(1) The weight of the control rod blade and the index tube is approximately 280 pounds.
(a) For withdrawal, therefore, the friction between the index tube notch and the collet fingers is too great to allow a pressure force from the collet piston to extend the fingers.
This is accomplished by first inserting the index tube a small distance, prior to applying a withdraw signal.

(2) Insert pressure is first applied, as discussed above.
(a) Raises drive about 2 to 3 inches
(b) Cams collet fingers out via tapered surface on index tube


CHARGING WATER
COOLING WATER
DRIVE WATER
PI 85-14
85-212A
85-586
PI 85-16
85-215A
85-591
1"
3/4"
1"
1-1/2"
85-223A
PI 85-26
3/4"
PI 85-19
RX PRESSURE ABOVE CORE PLATE
1-47E817-1,C2
85-214A
85-222A
PT 85-13
85-211A
FE 85-15
FT 85-15
85-213A
FT 85-25
FE 85-25
PDT 85-18
85-221A
85-217A
FCV 85-11A
1-1/2"
PDT 85-17
85-216A
85-216B
PI 85-24
85-220A
PCV 85-23
PCV 85-27
2"
85-563
85-564
85-565
85-566
85-568
85-569
85-561
FCV 85-11B
85-562
85-567
85-570

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 A1.02	
	Importance Rating	2.8	

(K&A Statement) A1.02- Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD AND DRIVE MECHANISM controls including: CRD Drive Pressure.

Proposed Question: RO 60

Which ONE of the following identifies how the collet fingers are released from the notch in the index tube during a control rod withdrawal as observed by the Operator at the Controls (OATC) on CRP 9-5?

A short insert signal ports ___(1)___ drive pressure flow to the ___(2)___ of the drive water piston.

A. (1) 2 gpm
(2) bottom

B. (1) 2 gpm
(2) top

C. (1) 4 gpm
(2) bottom

D. (1) 4 gpm
(2) top

Proposed Answer: C

A. INCORRECT: Insert signal flow is 4 gpm
B. INCORRECT: Insert signal flow is 4 gpm to the top
C. CORRECT:
D. INCORRECT: Insert signal flow is to the top

Technical Reference(s): CRD System DBD (page 60 of 83) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

QUESTION 56 Rev 0

U3 is operating at 100% power when the following occurs:

- The 3C RFPT tripped
- Reactor Water level lowered to 25 inches on the Normal Range instruments.

What is the expected response of the Reactor Recirc System?

Recirc Pumps speed lower to…

A. 480 rpm

B. 1130 rpm

C. a core flow of 60 Mlbm/hr

D. a steam flow of 10.9 Mlbm/hr

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	202002 A3.03	
	Importance Rating	3.1	

Recirc Flow Control System; Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: VFDs

Justification for K/A match: This is a Tier 2 Systems K/A on the Reactor Recirculation Flow Control System (BFN VFDs) and the ability to monitor automatic operation, To match the K/A, the question asks the candidate to identify the automatic flow changes for a condition satisfying the 75% Recirc Pump Runback.

Explanation: CORRECT B: The conditions for a 75% Runback are satisfied therefore, Recirc Pump speeds Lower speed to 1130 rpm **.**

A. **Incorrect because –** 480 RPM corresponds to the 28% limiter and the conditions to initiate this limiter are not satisfied.
Plausible because – The 28% limiter is also initiated based on Reactor Feed Water signals and it too initiates a Recirc Pump Runback.

C. **Incorrect because –** A Core Flow of 60Mlbm/hr corresponds to the core flow runback which is manually initiated.
Plausible because – Several AOIs call for inserting a Core flow runback prior to initiating a Reactor Scram.

D. **Incorrect because –** A Steam Flow of 10.9 Mlbm/hr corresponds to the Mid Power runback which is manually initiated.
Plausible

Technical Reference(s): 3-OI-68 rev 88

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.007 Obj 14

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: 41(b)(6)

Illustration 7
(Page 1 of 1)

Recirc Drive Speed Control and Acceleration Rates

Recirc Drive Speed Demand	Acceleration Rate
Drive Start	Accelerate to 345 rpm at 100 rpm per second
Raise Slow	Accelerate 1 rpm at 1 rpm per second
Raise Medium	Accelerate 5 rpm at 1 rpm per second
Lower Slow	Decelerate 1 rpm at 1 rpm per second
Lower Medium	Decelerate 5 rpm at 1 rpm per second
Lower Fast	Decelerate 50 rpm at 15 rpm per second
75% Runback	Lowers speed to 1130 rpm at 15 rpm per second and limits speed to 1130 rpm if rpm is less than 1130 rpm.
28% Limiter	Lowers speed to 480 rpm at 15 rpm per second and limits speed to 480 rpm if rpm is less than 480 rpm. If the 28% limiter resets while the Recirc Drive is decelerating the rate of change will lower to 1 rpm/sec. When the 28% limiter resets selecting a RAISE and LOWER SPEED demand push button will stop the Recirc Drive from decelerating
Drive Shutdown	Decelerate to 345 rpm at 25 rpm per second, once 345 rpm is reached drive will shutdown.
Drive Minimum Speed	345 rpm
Drive Maximum Speed	Adjustable up to ~1700 rpm.
High Power Runback	Lowers speed to a steam flow of 12.7 Mlbm/hr or 700-750 rpm at 15 rpm per second. When Speed is within ~1% of desired setpoint the rate of change changes to 1 rpm per second.
Mid Power Runback	Lowers speed to a steam flow of 10.9 Mlbm/hr or 700-750 rpm at 15 rpm per second. When Speed is within ~1% of desired setpoint the rate of change changes to 1 rpm per second.
Core Flow Runback	Lowers speed to a core flow of 60 Mlbm/hr or 700-750 rpm at 15 rpm per second. When Speed is within ~1% of desired setpoint the rate of change changes to 1 rpm per second.

QUESTION 57 Rev 0

When Rx Power reaches 25%, what provides the signal to the Rod Block Monitor (RBM) to begin enforcing Control Rod Blocks?

A. Total Steam Flow transmitter

B. Turbine 1st Stage Shell Pressure transmitter

C. Reactor Recirculation Flow signals

D. Reference APRM signals

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	215002 K4.03	
	Importance Rating	2.9	3.0

215002 Rod Block Monitor System: K4.03 Knowledge of ROD BLOCK MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Initiation point (30%): BWR-3,4,5 (CFR: 41.7)

Justification for K/A match: This is a Tier 2 Systems K/A concerning the RBM and its design features / interlocks that provide initiation point. To match the K/A a question was written to ask the candidate to recall what parameter causes the RBM to initialize.

Explanation: CORRECT D: The Rod Block Monitor receives its initiation signal from the reference APRM Channel for that RBM.

A. **Incorrect because –** The RBM uses the reference APRM.
Plausible because – The RWM utilizes Total Steam Flow to determine when to enforce Rod blocks and the two are often confused with one another.

B. **Incorrect because –** The RBM uses the reference APRM.
Plausible because – The Reactor Protection System utilizes Turbine 1st Stage Pressure to determine whether or not to enforce a Reactor scram on a Turbine trip.

C. **Incorrect because –** The RBM uses the reference APRM
Plausible, Recirculation Flow signals input into the APRMs for flow biased rod blocks and Scram signals.

Technical Reference(s): 1-OI-92B Rev 10, 1-OI-92C Rev 10, 1-OI-85 Rev 40, 1-GOI-100-1A Rev 43

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.148 rev 13 obj 26

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:	41(b)(7)

3.0 PRECAUTIONS AND LIMITATIONS

G. Each RBM channel can select from a designated hierarchy of normal and alternate APRM channels to use as their reference. If an Alternate was selected by the RBM and the Primary becomes available, it will automatically be selected as the reference APRM channel.

H. RBM reference signals from APRM channels:

RBM Channel	Reference APRM Channel
A	CHANNEL 1 (PRIMARY)
A	CHANNEL 3 (FIRST ALTERNATE)
A	CHANNEL 4 (SECOND ALTERNATE)
B	CHANNEL 2 (PRIMARY)
B	CHANNEL 4 (FIRST ALTERNATE)
B	CHANNEL 3 (SECOND ALTERNATE)

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Z. Recirculation Flow signals are used to produce the flow biasing used in the APRMs only. To operate with a single recirculation flow loop, the Δ flow value in the flow biased flow equation will be different for single loop operation verses two loop operation. This function is only manually initiated by the operator. This Δ Flow takes into account any backflow that might exist in the idle loop during single loop operation. When operating with two loops there will be no Δ Flow to offset the flow biased setpoints. During operation, if a recirculation pump trips, the operator can operate in compliance with Technical Specifications and apply the Δ flow factor into the flow biased setpoints. Flow biased equation is:

SETPOINT = (0.66 (Flow - ΔFlow)) + OFFSET

3.8 Rod Worth Minimizer (RWM)

E. For RWM to enforce, Total Feedwater Flow or Total Steam Flow must be <23%. To take RWM out of service automatically, Low Power Set Point (LPSP), Total Steam Flow and Total Feedwater Flow must be >23%. The Low Power Alarm Point (LPAP) for the RWM is (27%) as sensed by Total Steam Flow. When the RWM is operating in the transition zone, between the LPSP (23%) and the LPAP (27%), no rod blocks will be applied as a result of insert or withdraw errors, but the RWM will continue to provide alarm indications and error displays.

5.4 Withdrawal of Control Rods while in Mode 2 (continued)

G. Turbine bypass valves closed.

(R) _____ _____ _____
Initials Date Time

H. RFW START-UP LEVEL CONTROL, 1-LIC-3-53 is available and aligned to the RPV as needed. (If CRD system cannot maintain Reactor water level RFP SU Bypass Valve should be utilized.)

(R) _____ _____ _____
Initials Date Time

CAUTION

If turbine first stage pressure exceeds 147 psig, a reactor scram may result.

The following Shell Warming interlocks exist at given Turbine First Stage pressures:

- 100 psig - RAISE pushbutton, 2-HS-47-78C, is disabled.
- 105 psig - EHC/TSI SYSTEM TROUBLE (Panel 2-9-7B Window 6) annunciation.
- 115 psig - shell warming is disabled.

QUESTION 58 Rev 0

Unit 3 is operating at 100% Rx Power with the following indications and alarms present:

- DRYWELL NORM OPERATING PRESS HIGH, Panel 9-3B window 19
- DRYWELL TO SUPPR CHAMBER DIFF PRESS ABNORMAL, Panel 9-3B widow 26
- DRYWELL TO SUPPR CHAMBER DIFF PRESS, 3-PDS-64-137C is reading 1.41 psid

It is noted that the Drywell DP Compressor is running and additional Drywell Blowers are available.

What actions are **required** by the Unit Operator?

A. Start additional Drywell Blowers.

B. Secure Drywell DP Compressor.

C. Vent the Drywell using normal ventilation path.

D. Verify open RBBCW Primary Containment Outlet Valve.

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	223001 G2.4.45	
	Importance Rating	4.1	4.3

223001 Primary Containment System and Auxiliaries: G2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.

Justification for K/A match: This is a Tier 2 Systems K/A for Primary Containment system paired with an emergency generic K/A for the prioritization and interpretation of annunciators. To match this set of K/As, a question was written with a set of Pri. Cont. parameter annunciators to address and to interpret their significance.

Explanation: B is CORRECT. As DRYWELL TO SUPPR CHAMBER DIFF PRESS rises to 1.28 psid the Drywell DP Compressor automatically stops. Since pressure is reading 1.41 psid the compressor should have stopped. Since it has not the action to take is to secure Drywell DP Compressor per 3-ARP-9-3B Alarm Response 9-3B window 26.

A. **Incorrect because –** Primary Containment does not have a high temperature alarm in so starting additional blowers is not needed.
Plausible since this action will reduce Drywell temperature, P_{sat}-T_{sat} relationship, but this action is not warranted with the above alarms indicated.

C. **Incorrect because –** one of the first steps for venting has you check that the DW DP compressor is not running.
Plausible since this action is warranted under 3-ARP-9-3B window 19, but not until after securing Drywell DP Compressor.

D **Incorrect because –** again the lack of Pri. Cont. high temperature alarms, makes this a wrong answer.
Plausible since this action will reduce Drywell temperature, P_{sat}-T_{sat} relationship, but this action is not warranted with the above alarms indicated.

Technical Reference(s): 3-ARP-9-3B rev 21, 3-OI-64 rev 58

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): None

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:	41(b)(7)

DRYWELL TO SUPPR CHAMBER DIFF PRESS ABNORMAL 3-PDA-64-137	26

(Page 1 of 1)

Sensor/Trip Point:

3-PDS-64-137A or 64-138A	1.13 psid low
3-PDS-64-137C or 64-138C	1.34 psid high

Sensor Location:
Panel 9-19
Elevation 593'
Auxiliary Instrument Room

Probable Cause:
A. Drywell DP compressor malfunction.
B. Excessive N2 makeup to the drywell/torus.
C. Excessive venting from drywell/torus.
D. Sensor malfunction.

Automatic Action:
A. 3-PDS-64-137B and/or 138B.
1. Starts compressor on lowering DP.
2. Stops compressor on rising DP.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

V. Drywell D/P Air Compressor Interlocks

1. The Drywell D/P Air Compressor will auto start on lowering Drywell D/P (<1.15 psid), provided ALL of the following are met:

   a. Local control switch in REMOTE.

   b. DRYWELL DP COMP AND VALVES CONTROL, 3-HS-64-142A, in AUTO.

   c. No group 6 isolation signal present.

   d. Control Air discharge temperature <150°F.

   e. 3-FCV-64-31 is OPEN.

   f. 3-FCV-64-34 is OPEN.

   g. Compressor oil pressure ≥10 psig (signal bypassed for 15 seconds on compressor start).

   h. 3-FCV-64-140 NOT fully closed.

2. The Drywell D/P Air Compressor will stop on rising Drywell D/P (1.28 psid) OR IF ANY of the conditions in Step 3.0V.1 are NOT met.

QUESTION 59 Rev 0

During Refueling Operations with the Reactor mode switch in the refuel position, the following events occur:

- A fuel bundle is pulled to full up from its spent fuel pool location.
- The bridge is then driven over the core to its new location and the Refueling Bridge operator starts lowering the fuel bundle into the core.
- **NO** Rod Block alarm is received during this evolution.

Based on the events that just occurred what action(s) is/are **required** by Tech Specs immediately?

A. Insert a control rod withdrawal block only.

B. Verify all control rods are fully inserted only.

C. Place the reactor mode switch in the shutdown position.

D. Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	234000 A2.01	
	Importance Rating	3.3	3.7

234000 Fuel Handling Equipment A2.01 Ability to (a) predict the impacts of the following on the FUEL HANDLING EQUIPMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Interlock failure

Justification for K/A match: This is a Tier 2 Systems question for the RO about fuel handling equipment and the ability to predict the impacts and use procedures to correct the abnormal condition. To match this K/A for an RO, it was written to question the ROs ability to recall when Control Rod Blocks should or should not be received during fuel movements. To match the A2 part b. for using procedures, Tech Specs were selected and the RO is responsible for less than one hour specs.

Explanation: CORRECT D: IAW T.S. 3.9.1 Refueling Equipment Interlocks Required Action A.1. Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s) immediately.

A. **Incorrect because –** This is part of T.S. 3.9.1 required action A.2 but is incomplete.
Plausible because – This is partially correct and if coupled with answer B would be correct.

B. **Incorrect because –** This is part of T.S. 3.9.1 required action A.2 but is incomplete.
Plausible because – This is partially correct and if coupled with answer A would be correct.

C. **Incorrect because -** The insertion of an RPS trip and an intentional mode change is not directed..
Plausible because – This would be a way of inserting any withdrawn Control Rod and inserting a rod block accomplishing Required action A.2.

Technical Reference(s): 0-GOI-100-3C rev 82 Illustration 2

Proposed references to be provided to applicants during examination: none

Learning Objective (As available): OPL 171.053 obj 5

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:	41(b)(7)

Illustration 2
(Page 2 of 2)

Rod, Bridge, and Hoist Blocks - Block Diagram

3.9 REFUELING OPERATIONS

3.9.1 Refueling Equipment Interlocks

LCO 3.9.1 The refueling equipment interlocks shall be OPERABLE.

APPLICABILITY: During in-vessel fuel movement with equipment associated with the interlocks.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required refueling equipment interlocks inoperable.	A.1 Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).	Immediately
	OR	
	A.2.1 Insert a control rod withdrawal block.	Immediately
	AND	
	A.2.2 Verify all control rods are fully inserted.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Perform CHANNEL FUNCTIONAL TEST on each of the following required refueling equipment interlock inputs: a. All-rods-in, b. Refuel platform position, c. Refuel platform main hoist, fuel loaded, d. Refuel platform fuel grapple fully retracted position, e. Refuel platform frame mounted hoist, fuel loaded, f. Refuel platform monorail mounted hoist, fuel loaded, and g. Service platform hoist, fuel loaded.	7 days

QUESTION 60 Rev 1

The following plant conditions exist on Unit 2:

- Main Turbine Shell Warming is in progress
- The UO is pulling Control Rods in accordance with 3-GOI-100-1, Unit Startup

3-OI-47, Turbine-Generator System section 5.2 Turbine Shell Warming cautions the Operators not to exceed _____ psig Main Turbine First stage pressure to prevent a Reactor Scram.

A. 100

B. 105

C. 115

D. 147

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	241000 K5.05	
	Importance Rating	2.8	

Reactor Turbine Pressure Regulating System Knowledge of the operational Implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM : Turbine inlet pressure vs. turbine load

Justification for K/A match: This is a Tier 2 Systems K/A for Rx Turbine Pressure Regulating System and the operational implications of turbine inlet pressure and turbine load. To match the K/A, a question was written asking what pressure would result in a Reactor SCRAM (operational implication) while in Shell Warming.

Explanation: CORRECT D: 3-OI-47 Caution on page 43 states: 4) If turbine first stage pressure exceeds 147 psig, a reactor scram may result.

A. **Incorrect because –** The Reactor SCRAM on Main Turbine Trip is bypassed below 30% power as sensed by 147 psig Turbine First stage pressure.
Plausible because – At this pressure the Shell Warming Raise pushbutton is disabled.

B. **Incorrect because –** The Reactor SCRAM on Main Turbine Trip is bypassed below 30% power as sensed by 147 psig Turbine First stage pressure.
Plausible because – At this pressure while in Shell Warming the EHC/TSI SYSTEM TROUBLE (Panel 3-9-7B Window 6) annunciation will alarm.

C. **Incorrect because –** The Reactor SCRAM on Main Turbine Trip is bypassed below 30% power as sensed by 147 psig Turbine First stage pressure.
Plausible because – At this pressure shell warming is disabled.

Technical Reference(s): 3-OI-47 Rev 108

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.010 Rev 13 OBJ 17b

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:	41(b)(5)	

5.2 Turbine Shell Warming (continued)

NOTES

1) MSV-2 PILOT POSITION, 3-ZI-47-78, will normally indicate approximately zero demand when the OFF pushbutton is selected. When the SHELL WARMING pushbutton is selected, 3-ZI-47-78 will indicate a negative demand (approximately -63%). Steam will start to be admitted when 3-ZI-47-78 indicates approximately 0%. The position demand is limited to a maximum of 70%.
2) If steam flow to the turbine is raised too rapidly, the turbine will roll off the turning gear.
3) The following Shell Warming interlocks exist at the given Turbine First Stage pressures:
   - 100 psig - RAISE pushbutton, 3-HS-47-78C, is disabled.
   - 105 psig - EHC/TSI SYSTEM TROUBLE (Panel 3-9-7B Window 6) annunciation.
   - 115 psig - shell warming is disabled.
4) The Shell Warming RAISE/LOWER pushbuttons change valve position by a 1% increment every time the associated pushbutton is momentarily depressed. If the associated pushbutton is depressed for more than 3 seconds, then the valve position will change at a rate of 2% per second.
5) To prevent a rapid lowering of the RPV water level on initial shell warming, turbine first stage shell pressure should not exceed ~10 psig for about the first 15 to 30 minutes. After ~15 to 30 minutes of warming time, shell pressure can be raised.

CAUTIONS

1) **DO NOT** exceed 150°F/hr heat up rate on Turbine first stage shell. .
2) **DO NOT** exceed 75°F Turbine first stage shell inner to outer surface temperature differential..
3) Operating the turbine on turning gear with rotor differential expansion operating in the red band may result in turbine damage.
4) If turbine first stage pressure exceeds 147 psig, a reactor scram may result.

QUESTION 61 Rev 0

The Unit 1 Main Generator synchronization is in the progress IAW 1-GOI-100-A, Unit Startup.

The following indications are observed on panel 1-9-8:

- VOLTAGE REGULATOR MAN/AUTO in MAN
- GEN SYNC REF VOLTAGE, 1-E-57-54 is reading 27 V
- SYSTEM SYNC REF VOLTAGE is reading 28 V
- SYNCHROSCOPE 1-XI-57-55 is stopped at the 6:00 position

Before the Generator PCB 214 can be closed, the operator must go to __ (1) __ on the Voltage Regulator Lower/Raise Adjust Switch to match voltages.

The operator must also go to __ (2) __ on the Turbine Generator Sych Speed INC/DEC Adjust Switch until the Synchroscope is moving slowly in the clockwise direction.

A. (1) raise
   (2) INC

B. (1) raise
   (2) DEC

C. (1) lower
   (2) INC

D. (1) lower
   (2) DEC

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	245000 A4.02	
	Importance Rating	3.1	

245000 Main Turbine Generator and Auxiliary Systems A4.02 Ability to manually operate and/or monitor in the control room: Generator controls

Justification for K/A match: This is a Tier 2 Systems K/A on the Main Turbine Generator and Aux Systems tied to the ability to manually operate and/or monitor in the control room, Generator Controls. To match this K/A, the main generator is being synced and the question asks the candidate which way to turn certain switches (manually operate generator controls)

Explanation: CORRECT A: With voltage on Gen Sync Ref Voltage at 27.5 V and on the System Sync Ref Voltage at 28 V, the Voltage Regulator Lower/Raise Adjust Switch needs to be placed in RAISE position to cause Gen Sych Ref Voltage to rise. With the synchroscope stopped, the generator frequency exactly matches the grid frequency. The Turbine Generator Sych Speed Adjust needs to be raised by taking handswitch to INC position.

B. **Incorrect because –** This would cause speed of the Main Generator will slow down and frequency would be < grid. Frequency needs to be slightly above the grid so the generator will pick up some initial load.
Plausible because – There are cases when a sync scope has to be rotating counter clockwise.

C. **Incorrect because –** This action will cause the GEN SYNCH REF VOLTAGE to lower further from the SYSTEM SYNCH REF VOLTAGE.
Plausible because – The candidate may confuse which voltage indicator is affected by the raise/lower switch.

D. **Incorrect because –** Part 1 see C above Part 2 see B above.
Plausible because – Part 1 see C above Part 2 see B above.

Technical Reference(s): 1-OI-47 Rev 48

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171G001 obj 10

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: Perry NRC Exam 2013 Q# 71
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis: X
10 CFR Part 55 Content:	41(b)(7)

5.5 Generator Synchronization and Loading (continued)

NOTE

In Step 5.5[17.3], the Synchroscope is adjusted for a rotation slow in the fast direction (clockwise) at one revolution every 15-20 seconds.

[17.3] **PERFORM** the following concurrently to obtain both conditions at the same time:

- **OPERATE** VOLTAGE REGULATOR LOWER/RAISE ADJUST, 1-HS-57-26 to match SYSTEM SYNC REF VOLTAGE, 1-EI-57-56 to GEN SYNC REF VOLTAGE, 1-EI-57-54. ☐

- **OPERATE** TURBINE GENERATOR SYNC SPEED ADJUST, 1-XS-57-20, to obtain SYNCHROSCOPE, 1-XI-57-55, rotation slow in the FAST direction (clockwise, one revolution every 15-20 seconds). ☐

Perry NRC Exam 2013 Q# 71

NRC Exam 2013

QUESTION RO 71

The Main Generator synchronization is in the progress IAW IOI-0003, Power Changes.

The following indications are observed on panel H13-P680:

- SYNC SELECT SWITCH is in the S610-PY-TIE position
- MAIN TRANSFORMER (incoming) S11-R013 346 KV
- PY-EL-LINE (running) N41-R120 344 KV
- Synchroscope is stopped at the 6:00 position

Before the S610-PY-TIE breaker can be closed, the operator must go to __(1)__ on the Auto Voltage Regulator to match voltage. He must also go to __(2)__ on the Load Selector pushbuttons until the Synchroscope is moving slowly in the clockwise direction.

	__(1)__	__(2)__
A.	LOWER	INCREASE
B.	LOWER	DECREASE
C.	RAISE	DECREASE
D.	RAISE	INCREASE

QUESTION 62 Rev 0

What is the effect on the Reactor Feedwater System with a loss of **120V I&C Bus A**?

A. RPFT 2B Woodward Governor loses power.

B. RFP 2C Minimum Flow Valve fails open.

C. RFW Start-up Level Control PDS controls are rendered inoperative.

D. RFPT/RFP 2A Vibration Monitoring Equipment loses indication.

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	259001 K6.03	
	Importance Rating	3.1	3.1

259001 Reactor Feedwater System: K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: A.C. electrical power.

Justification for K/A match: This is a Tier 2 Systems K/A on the Reactor Feedwater System concerning the effect of a loss of A.C. power. Simply put, the question asks the candidate to recall the effect that a loss of 120 V AC will have on the feedwater system.

Explanation: B is CORRECT. On a loss of I&C A the RFP 2C Minimum Flow Valve will fail OPEN.

A. **Incorrect because –** The governor is not powered from 120V I&C Bus A.
Plausible because – Failure of I&C B will cause the RFP 2B Woodward Governor to lose 1 out 2 auctioneered power sources. The other source of power is ICS.

C. **Incorrect because –** RFW Start-up Level Control PDS are not powered by I&C A
Plausible because – Loss of Unit Preferred, a different 120 V power source will cause a loss of PDS controls to RFW Start-up Level Control Valve.

D. **Incorrect because –** the RFPT/RFP Vibration Monitoring Equipment is not powered by I&C A.
Plausible because – Loss of I&C B will cause a loss of RFPT/RFP Vibration Monitoring Equipment to all 3 RFPTs/RFPs.

Technical Reference(s): 2-AOI-57A-5A rev 58, 2-AOI-57A-5B rev 49, 2-OI-99 rev 99

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.102 obj V.B.1e

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:	41(b)(7)

3.0 AUTOMATIC ACTIONS

A. Panel 2-9-9 Cabinet 2 transfers to Alt. Power Supply.

B. Short cycle valves FCV-2-29A and FCV-2-29B fail CLOSED.

C. SJAE "A" isolates if in service.

D. RFP 2C Minimum Flow Valve (2-FCV-3-6) fails OPEN.

E. Reactor/refuel zone ventilation System static switch isolates.

F. Drywell Radiation Monitor isolates.

G. +24V DC neutron monitoring battery chargers A1-2 & A2-2 trip.

H. Drywell to suppression chamber ΔP compressor starts due to of PS-64-137B.

3.0 AUTOMATIC ACTIONS

A. Panel 9-9, Cabinet 3, Transfers to Alt Pwr Supply.

B. RFPT 2A and 2B Minimum Flow Valves (2-FCV-3-20, 13) fail OPEN.

C. Loss of one out of two auctioneered power sources to RFPT 2B Woodward Governor and Final Driver (ICS is the other power source).

D. Loss of RFPT/RFP Vibration Monitoring Equipment for all three RFPT/RFPs.

E. SJAE "B" Isolates if in Service.

F. Reactor/Refuel Zone Ventilation System static switch Isolates.

Illustration 1
(Page 3 of 3)

RFW Startup Level Controller Panel Display Station

FAILURE MECHANISMS

Loss of Control Air -	When Control Air pressure drops to ≤ 65 psig, the Startup Bypass Valve fails as is. The valve cannot be controlled with the PDS until Control Air pressure returns to > 65 psig.
Loss of Unit Preferred -	All indications are lost on the PDS. If the PDS was controlling in Auto, the system will continue to control automatically. The controls on the PDS are rendered inoperative.

QUESTION 63 Rev 0

Where do the Reactor Building Equipment Drain Sump Pumps **normally** discharge their contents to for Unit 1?

A. Waste Collector Tank

B. Waste Sample Tanks

C. Floor Drain Collector Tank

D. Floor Drain Sample Tank

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	268000 K1.03	
	Importance Rating	2.6	2.9

268000 Radwaste: K1.03 Knowledge of the physical connections and/or cause effect relationships between RADWASTE and the following: Reactor building equipment drains: Plant-Specific (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Justification for K/A match: This is a Tier 2 Systems K/A concerning Radwaste and the physical connections relationship to the Reactor Building Equipment Drains system. To match this K/A, a simple question was written that asks the candidate to recall the physical connection between Rx Bldg EDs to the receiving tanks in Radwaste.

Explanation: CORRECT A: IAW DWG 1-47E852-2 and 0-47E830-2 Reactor Building Equipment Drain Sumps discharge to Waste Collector Tank.

B. **Incorrect because –** The discharge of the Rx Bldg Equip. Drains is pumped directly to the Waste Collector Tank so it can be processed, not the Waste Sample Tank.
Plausible because – The equipment drains are considered clean Radwaste and the candidate may think it is clean enough to go to the Waste Sample Tanks without processing.

C. **Incorrect because –** The discharge of the Rx Bldg Equip. Drains is pumped directly to the Waste Collector Tank.
Plausible because – Reactor Building floor drains go to FDCT prior to processing. The Floor Collector Tank can be pumped to Waste Collector Tank per drawing 0-47E830-3 (E7).

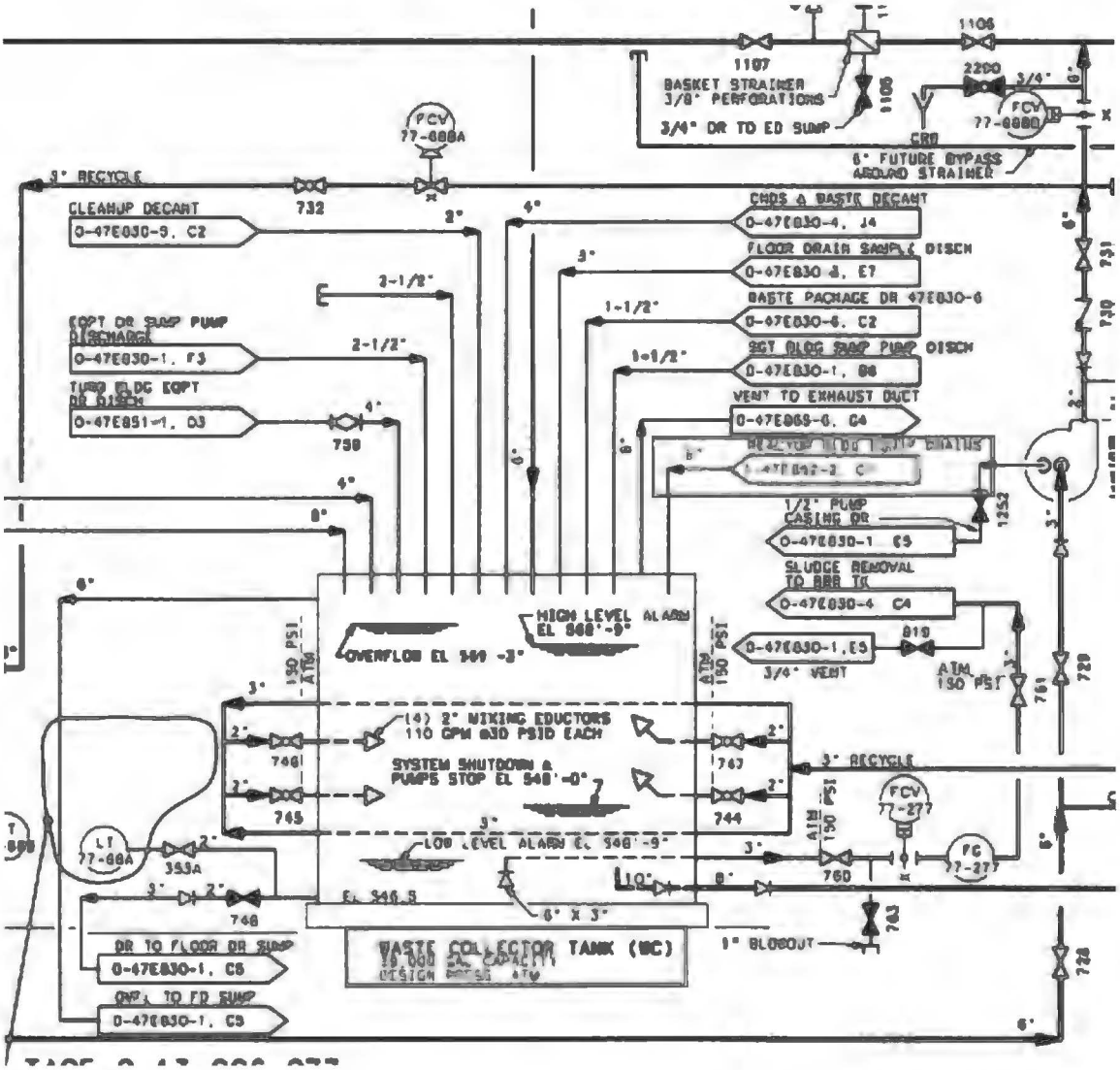
D. **Incorrect because –** The discharge of the Rx Bldg Equip. Drains is pumped directly to the Waste Collector Tank so it can be processed.
Plausible because – Reactor Building floor drains go to FDCT prior to processing and because the Floor Drain Sample Tank can be pumped to the Waste Collector Tank.

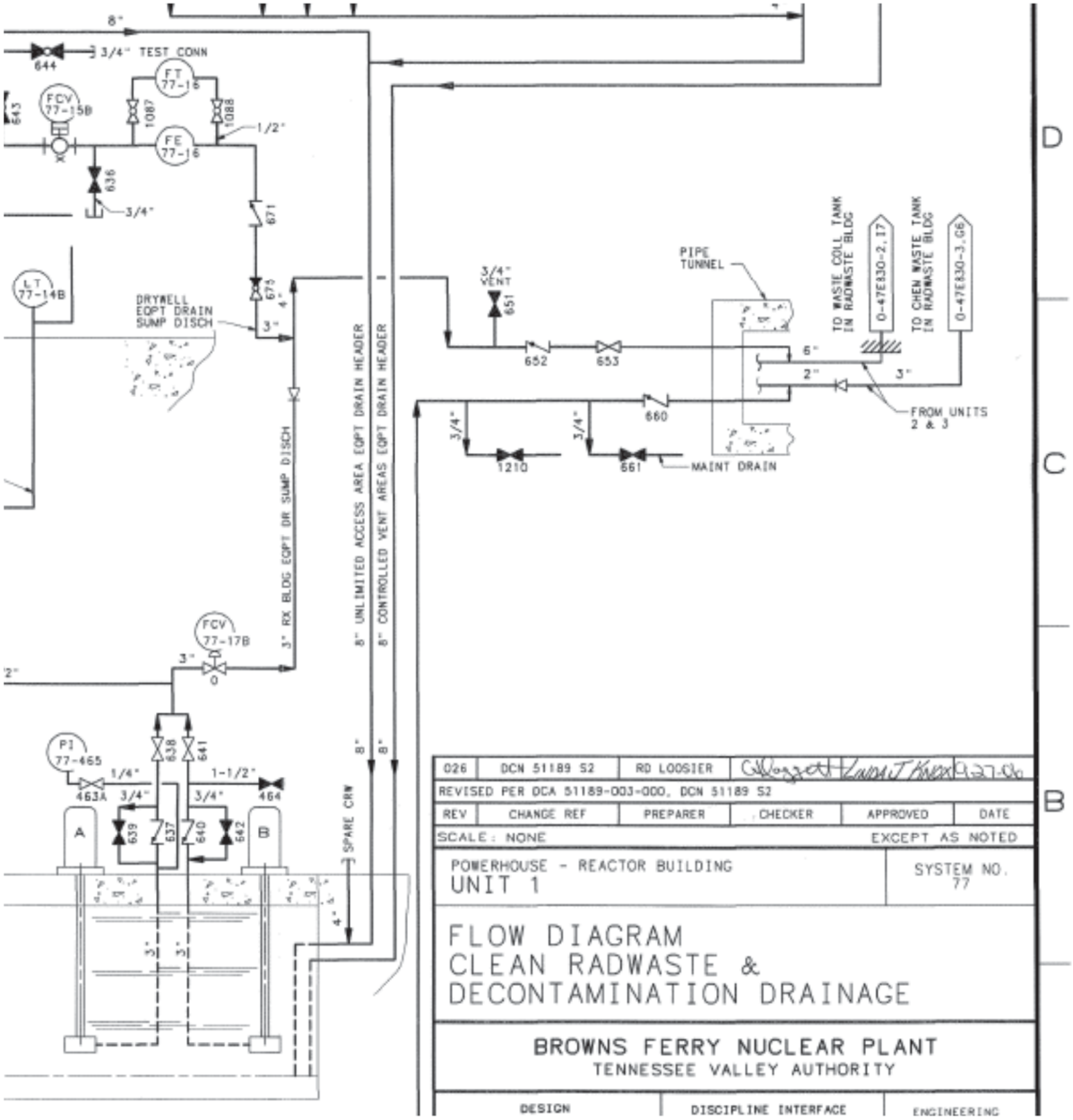
Technical Reference(s): 1-47E852-2, 0-47E830-2, 0-47E830-2, OPL 171.084 rev 8

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.084 obj 4b

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:	41(b)(4)


BASKET STRAINER
3/8" PERFORATIONS
3/4" DR TO ED SUMP
6" FUTURE BYPASS
AROUND STRAINER
3" RECYCLE
CLEANUP DECANT
0-47E030-9, C2
CHDS & WASTE DECANT
0-47E030-4, J4
FLOOR DRAIN SAMPLE DISCH
0-47E030-8, E7
WASTE PACKAGE DR 47E030-6
0-47E030-6, C2
EQPT DR SUMP PUMP
DISCHARGE
0-47E030-1, F3
SGT BLDG SUMP PUMP DISCH
0-47E030-1, B6
TURB BLDG EQPT
DR DISCH
0-47E051-1, D3
VENT TO EXHAUST DUCT
0-47E063-6, C4
1/2" PUMP
CASING DR
0-47E030-1 C5
SLUDGE REMOVAL
TO BRR TK
0-47E030-4 C4
0-47E030-1, E5
3/4" VENT
AIR
150 PSI
OVERFLOW EL 560'-3"
HIGH LEVEL ALARM
EL 548'-9"
(4) 2" MIXING EDUCTORS
110 GPM @30 PSID EACH
SYSTEM SHUTDOWN &
PUMPS STOP EL 548'-0"
LOW LEVEL ALARM EL 548'-9"
EL 546.5
6" X 3"
WASTE COLLECTOR TANK (WC)
1" BLOWOUT
DR TO FLOOR DR SUMP
0-47E030-1, C5
DRS TO FD SUMP
0-47E030-1, C5
FCV
77-277
FC
77-277
LI
77-88A
FCV
77-888A
FCV
77-8880


8"
3/4" TEST CONN
644
FT 77-16
FCV 77-15B
1087
1088
1/2"
FE 77-16
636
3/4"
671
675
4"
LT 77-14B
DRYWELL EQPT DRAIN SUMP DISCH
3"
3" RX BLDG EQPT DR SUMP DISCH
8" UNLIMITED ACCESS AREA EQPT DRAIN HEADER
8" CONTROLLED VENT AREAS EQPT DRAIN HEADER
3/4" VENT
651
652
653
PIPE TUNNEL
TO WASTE COLL TANK IN RADWASTE BLDG
0-47E830-2,I7
TO CHEM WASTE TANK IN RADWASTE BLDG
0-47E830-3,G6
6"
2"
3"
660
FROM UNITS 2 & 3
1210
661
MAINT DRAIN
FCV 77-17B
0
PI 77-465
638
641
1/4"
1-1/2"
463A
464
639
637
640
642
A
B
SPARE CRN
026
DCN 51189 S2
RD LOOSIER
REVISED PER DCA 51189-003-000, DCN 51189 S2
REV
CHANGE REF
PREPARER
CHECKER
APPROVED
DATE
SCALE: NONE
EXCEPT AS NOTED
POWERHOUSE - REACTOR BUILDING
UNIT 1
SYSTEM NO. 77
FLOW DIAGRAM
CLEAN RADWASTE &
DECONTAMINATION DRAINAGE
BROWNS FERRY NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY
DESIGN
DISCIPLINE INTERFACE
ENGINEERING

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
2. Waste Collector System a. Receives high purity (Low conductivity) equipment drain liquid waste. b. Liquid sources which go to Waste Collector System 1) Equipment sump pump discharge from following locations: a) Turbine Building b) Reactor Building c) Drywell d) SBGT Building e) Radwaste Building f) Evaporator Building 2) RHR System 3) RWCU Blowdown	Objective 4.b

QUESTION 64 Rev 1

What is the power supply to the Stack-Gas Radiation Monitor (0-RM-90-147 & 148) scintillation detectors?

A. Unit 1 (±) 24 VDC Neutron Monitoring Battery System

B. Unit 2 (±) 24 VDC Neutron Monitoring Battery System

C. Unit 3 (±) 24 VDC Neutron Monitoring Battery System

D. (±) 48 VDC Annunciator Battery System

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	272000 K2.03	
	Importance Rating	2.5	2.8

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

Justification for K/A match: This is a Tier 2 Systems K/A concerning the Radiation Monitoring System and the knowledge of the electrical power supplies to the stack gas radiation monitoring system. Simply asks what the power supply is to the Stack Radiation Monitors.

Explanation: CORRECT A: Stack-Gas Radiation monitors are powered from the Unit 1 ±24 VDC Neutron Monitoring Battery System.

B. **Incorrect because -** Stack-Gas Radiation monitors are powered from Unit 1 ±24 VDC **Plausible** since the scintillation detectors are powered from a ±24 VDC Neutron Monitoring Battery System.

C. **Incorrect because -** Stack-Gas Radiation monitors are powered from Unit 1 ±24 VDC **Plausible** since the scintillation detectors are powered from a ±24 VDC Neutron Monitoring Battery System.

D. **Incorrect because -** Stack-Gas Radiation monitors are powered from Unit 1 ±24 VDC **Plausible** since the scintillation detectors are powered from a low voltage DC power source and the Annunciator Battery System is common equipment as are the Stack-Gas radiation monitors.

Technical Reference(s): 0-OI-57D rev 152,

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.033 V.B.3b

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)(7)

7.7 Removing Unit 1 ±24V Neutron Monitoring Battery A(B) from Service

[1] **REVIEW** Precautions and Limitations.
REFER TO Section 3.0. ☐

[2] **REVIEW** the following loads for the ± 24V DC Bus to be deenergized:

[2.1] ± 24V Neutron Monitoring Channel A

A. Panel 9-5
- IRM recorders A, C, E, and G
- SRM Channels A and C indicators, recorders and period meters

B. Panel 9-10
- 0-RM-90-147B, STACK GAS CH1 RAD MON RTMR
- 1-RM-90-266A, OG POST-TREATMENT CH A RAD MON RTMR

[2.2] ± 24V Neutron Monitoring Channel B

A. Panel 9-5
- IRM recorders B, D, F, and H
- SRM Channel B and D indicators, recorders and period meters

B. Panel 9-10
- 1-RM-90-265A, OG POST-TREATMENT CH B RAD MON RTMR
- 0-RM-90-130B, RADWASTE EFFLUENT RAD MON RTMR
- 0-RM-90-148B, STACK GAS CH2 RAD MON RTMR

QUESTION 65 Rev 0

A fire has been reported in Unit 2 Auxiliary Instrument Room and the CO_2 System failed to automatically or manually initiate.

The Unit Supervisor has ordered the AUO to manually initiate CO_2 using the Pilot Control Valve Station(s).

How will the CO_2 System respond when the pilot valve lever is placed in the OPEN position?

CO_2 will be dispensed ___ (1) ___ and the evacuation alarm ___ (2) ___ sound.

A. (1) immediately
   (2) will

B. (1) immediately
   (2) will **NOT**

C. (1) after 60 sec time delay
   (2) will

D. (1) after 60 sec time delay
   (2) will **NOT**

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	286000 K3.02	
	Importance Rating	3.2	3.4

286000 Fire Protection System: K3.02 Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on following: Personnel protection (CFR: 41.7 / 45.4)

Justification for K/A match: The question asks systems knowledge concerning the safety implications of operating the CO2 system from the Local Pilot Control Stations.

Explanation: CORRECT B: When the manual pilot valve lever is taken to the open position it bypasses the evacuation alarm and the 60 second time delay for discharging CO_2.

A. **Incorrect because** – The evacuation alarm will not sound.
Plausible because – Part 1 is correct and because the manual pushbutton will sound the alarm.

C. **Incorrect because** – The time delay is bypassed and the alarm will not sound.
Plausible because – The manual pushbutton will dispense CO_2 after a time delay and the alarm will sound.

D. **Incorrect because** – The time delay is bypassed.
Plausible because – The manual pushbutton will dispense CO_2 after a time delay and Part 2 is correct.

Technical Reference(s): 0-OI-39 Rev 32

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.049 Obj V.B.6

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:	41(b)(7)

3.0 PRECAUTIONS AND LIMITATIONS

I. Prior to operating the CO_2 System from the pilot control valve station, make sure the area that CO_2 will discharge into is evacuated, as manual operation of the pilot control valve will bypass the alarm and 60 second time delay (20 second time delay for Lube Oil Purification Room).

5.2 Manual CO_2 Initiation (continued)

[8] **VERIFY** CO_2 initiation is required and discharge area is evacuated.
[9] **PULL** the hinged cover of the pushbutton station (this safely breaks the glass covering).
[10] **DEPRESS** the initiation pushbutton to actuate CO_2.
[11] **VERIFY** the following actions occur:
- A. Pushbutton station light for hazard area extinguishes.
- B. Alarm warning bell sounds.
- C. After at least 20 seconds CO_2 is discharged (discharge pressure provides a means to cause doors and dampers to close and isolate the fire).
- D. After a predetermined discharge period the CO_2 supply valves to the affected area close.
- E. Pushbutton station light for hazard area illuminates.

8.2 CO_2 Initiation - Pilot Control Valve Station

WARNINGS

1) This section is only performed at the direction of the Unit Supervisor/SRO.
2) Manual operation of the pilot control valve bypasses the evacuation alarm and the 60 (or 20) second time delay.

NOTE

The pilot valve should never be operated manually except when there is a failure to operate from normal automatic or manual control.

[1] **VERIFY** CO_2 initiation is required and discharge area is evacuated.
[2] **BREAK** the glass front of the pilot valve station with the attached hammer.
[3] **PLACE** the pilot valve lever in OPEN for the time in seconds specified below:

Hazard Area	Time	Hazard Area	Time
Dsl Gen Rooms	85	Aux Inst Rm 1	124
480V Dsl Aux Bd Rooms	127	Aux Inst Rm 2	135
Fuel Oil Trans Rooms	88	Aux Inst Rm 3	198

NOTE

The Master Supply Pilot Control valves are located in the CO_2 tank rooms.

[4] **IF** it is determined CO_2 has **NOT** discharged, **THEN DISPATCH** personnel to the applicable CO_2 tank room to perform the following:

[4.1] **BREAK** the glass front of the master pilot valve station specified below with the attached hammer:

Hazard Areas	Master Pilot Valve
U1&2 Diesel Generator Building	0-FSV-039-0004
Aux Inst and Computer Rooms	0-FSV-039-0011

[4.2] **PLACE** the master pilot valve lever in OPEN for the time in seconds specified in step 8.2[3].

QUESTION 66 Rev 0

What is the frequency of panel walk downs in accordance with OPDP-1, Conduct of Operations?

The Unit Operator is to perform a panel walk down a minimum of once _____ (with a 25% grace period).

A. per hour

B. every 2 hours

C. every 4 hours

D. every 6 hours

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #	N/A	
	K/A#	G2.1.1	
	Importance Rating	3.8	4.2

Knowledge of conduct of operations requirements

Justification for K/A match: The question asks for the time frame for Control Board monitoring IAW OPDP-1, Conduct of Operations.

Explanation: A is CORRECT: per OPDP-1 a minimum walkdown of the reactor Control Area is once per hour with grace period of 25%.

B. INCORRECT: Plausible due to the requirement to perform 1-SR-2, Instrument Checks and Observations, for Core Thermal Power every 2 hours.

C. INCORRECT: Plausible due to the requirement to perform 1-SR-2, Instrument Checks and Observations, for DW Leakage every 4 hours.

D. INCORRECT: Plausible due to requirement to walk panels outside the Main Control Room twice a shift.

Technical Reference(s): OPDP-1 Rev 34, 1-SR-2 Rev 29

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.071 obj 9

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:	41(b)(10)

3.4.1 Control Board Monitoring (continued)

C. A walk down of the Reactor Control Area panels is to be performed a minimum of once per hour (with a 25% grace period) to ensure that indications are within established bands.

1. The walk down of the panels in the Reactor Controls Area shall be conducted by any licensed operator assigned to the unit.

2. The walk down of the MCR panels outside the Reactor Controls Area will either be conducted by the assigned Control Room Operator or the OATC will be temporarily relieved by another licensed individual prior to leaving the Reactor Controls Area. This walk down is required twice per shift.

Attachment 2
(Page 1 of 102)

Surveillance Procedure Data Package - Modes 1, 2, & 3

TABLE 1.1 CORE THERMAL POWER AND CORE POWER DISTRIBUTION **DAY SHIFT** WEEK: _____ to _____

APPLICABILITY:	Mode 1 when ≥ 25% RTP **RECORD** the readings as soon as possible after the generator breaker has been closed.										
Criteria Source:	3.2.1.1; 3.2.2.1; 3.2.3.1; DEFINITIONS SECTION 1.1 - FSAR 3.7.7										
LOCATION:	ICS Computer (Case Summary - CSUM)									Review Initials	
DAY	TIME Note 2	Core Thermal Power (MWt)	Percent Power (% RTP)	LIMIT (AC)	MFLCPR Note 3	MAPRAT Note 3		MFDLRX Note 3	LIMIT (AC)	Unit Operator	Unit Supvr
Friday	0800										
	1000										
	1200										
	1400										
	1600										
	1800										
	0800										
	1000										

Attachment 2
(Page 5 of 102)

Surveillance Procedure Data Package - Modes 1, 2, & 3

TABLE 1.2 DRYWELL UNIDENTIFIED LEAKAGE **DAY SHIFT** WEEK: _____ to _____

APPLICABILITY: Modes 1, 2 & 3 Readings are required at all times.

Surveillance Requirements: 3.4.4.1 LOCATION: Panel 1-9-4, 1-FR-77-6

Preferred reading times are 0800, 1200 and 1600	Col. A.1	Col. B.1	Col. C.1	Col. D.1	Col. E.1	Col. F.1	Col. G.1	Col. H.1	Col. I.1		Review Init	
	Current Point 3 (1-FQ-77-6) Reading (gals) Notes 1, 2	Previous Days 1-FQ-77-6 Reading from Col. A.1 (gals) Note 2	Gallons Pumped Col. A.1 - Col. B.1 Note 2	Current Time Note 2	Previous Days Time from Col. D.1 Note 2	Elapsed Time Col. D.1 - Col. E.1 (min) Note 2	Current Leakrate Col. C.1 ÷ Col. F.1 (gpm) Note 2, 5	Previous Days Leakrate from Col. G.1 (gpm) Note 2	Change in Leakrate Col. G.1 - Col. H.1 (gpm) Note 2, 3, 5	LIMITS (AC)	UO	Unit Supvr Note 4
Tuesday												
Wednesday												
Thursday												

(1) Manually pump down sump per 1-OI-64 prior to reading. Record gallons as displayed on recorder point 3 indication. Record right most five digits as gallons of flow. Example: Record 00654321 as 54321.
(2) May be N/A'd if Surveillance Requirement is being met with the performance of 1-SR-3.4.4.1 or 1-SR-3.4.4.1-A and a note stating such shall be made in the remarks section of this SR. When initial TOTALIZE reading is taken and no previous reading exists, all other entries except for Col. A.1 and D.1 should be N/A'd.
(3) Acceptance Criteria for Col. I.1 is only applicable when in Mode 1 for > 24 hours.
(4) Unit Supervisor shall Independently Verify Inleakage Calculations and verify Inleakage Acceptance Criteria.
(5) **NOTIFY** Chemistry to initiate 1-TI-275E when Drywell Unidentified leakage reaches the following trigger values: Drywell Unidentified leakage has reached 0.50 gpm **OR** Drywell Unidentified leakage has risen by 0.25 gpm in 24 hours.

QUESTION 67 Rev 0

In accordance with NPG-SPP-1.2, Administration of Site Technical Procedures, Identification of Critical Steps are to be annotated in which types of procedure?

A. Emergency Operating Instructions (EOIs)

B. Abnormal Operating Instructions (AOIs)

C. Operating Instructions (OIs)

D. Annunciator Response Procedures (ARPs)

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.1.20	
	Importance Rating	4.6	4.6

Ability to interpret and execute procedure steps.

Justification for K/A match: This is a Tier 3 generic Conduct of Operations K/A, about the ability to interpret and execute steps. The question asks about how critical steps are denoted and in which procedures they apply.

Explanation: Correct C: Steps in technical procedures which meet the definition of a critical step should be flagged as critical steps as described in Attachment 2, Identification of Critical Steps. Sections of procedures or entire procedures can be identified as critical.

A. Incorrect: Emergency Operating Instructions (EOIs) are always critical, by their very nature, and will not be revised to note that they are critical procedures.

B. Incorrect: Abnormal Operating Instructions (AOIs) are always critical, by their very nature, and will not be revised to note that they are critical procedures.

D. Incorrect: Annunciator Response Instructions (ARIs) are always critical, by their very nature, and will not be revised to note that they are critical procedures.

Technical Reference(s): NPG-SPP-1.2 Rev 11

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): None

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:	41(b)(10)

3.2.23 Identification of Critical Steps

A. Steps in technical procedures which meet the definition of a critical step should be flagged as critical steps as described in Attachment 2, Identification of Critical Steps. Sections of procedures or entire procedures can be identified as critical.

B. All pre-job briefings require identification of critical steps, by use of a designated stamp or equivalent marking, in conjunction with the pre-job briefing. The individual identifying the critical step should request a change to the affected procedure in accordance with Section 3.2.7 of this procedure.

C. The following types of procedures are always critical, by their very nature, and will not be revised to note that they are critical procedures:

1. Emergency Operating Procedures/Instructions (EOPs/EOIs), which are not within the scope of this procedure (see Section 3.2.22).

2. Abnormal Operating Procedures/Instructions (AOPs/AOIs)

3. Annunciator Response Instructions/Alarm Response Procedures (ARIs/ARPs)

4. Safe Shutdown Instructions (SSIs)

QUESTION 68 Rev 1

In accordance with ODM-4.5, Operator Aids and Operator Information System, how does the Unit Operator determine during the panel walk down, that a system is aligned correctly?

The normally running pumps shall have a lit ___ (1) ___ red lens cover.

The normally opened valves shall have an extinguished ___ (2) ___ green lens cover.

A. (1) clear
   (2) clear

B. (1) diffused
   (2) diffused

C. (1) diffused
   (2) clear

D. (1) clear
   (2) diffused

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G 2.1.29	
	Importance Rating	4.1	4.0

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc. (CFR: 41.10)

Justification for K/A match: This is a Tier 3 Generic K/A about the conduct of lineups such as valves, breakers, switches. To match this generic K/A an Operations Directive Manual question concerning the use of different lens covers for the lights in the control room to indicate normal and abnormal alignment by looking at the control board.

Explanation: Correct C: Per BFN-ODM-4.5 Operator Aids and Operator Information Systems E. Light Lens Cover Convention to help the operator light convention has been developed to aid the process: For Pumps: • If the pump is normally running, it shall have a lit diffused red lens cover. The unlit green light for this pump shall be a clear green lens cover. If the pump is normally off it shall have a lit diffused green lens cover. The unlit red light for this pump shall be a clear red lens cover. For Valves: If the valve is normally open it shall have a lit diffused red lens cover. The green light for this valve shall be a clear unlit green lens cover. If the valve is normally closed it shall have a lit diffused green lens cover. The red light for this valve shall be a clear unlit red lens cover.

A. **Incorrect because -** clear lens, if lit, indicate non standard position or running.
Plausible since this is the other selection for lens covers, its either one or the other.

B. **Incorrect because -** because the valve lens would be clear not diffused.
Plausible since this is the other selection for lens covers, its either one or the other.

D. **Incorrect because -** because the pump lens would be diffused. not clear.
Plausible since this is the other selection for lens covers, its either one or the other.

Technical Reference(s): BFN-ODM-4.5, Rev 05

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): None

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:	41(b)(10)

E. Light Lens Cover Convention

1. OPDP-1 currently requires a walk down of the control panels to determine plant status. During the walk downs he/she is to check switch positions, instrumentation readings, and status lights.

2. To help the operator as he/she walks the board down, the following light convention has been developed to aid the process:

For Pumps:

- If the pump is normally running, it shall have a lit diffused red lens cover. The unlit green light for this pump shall be a clear green lens cover.

- If the pump is normally off it shall have a lit diffused green lens cover. The unlit red light for this pump shall be a clear red lens cover.

For Valves:

- If the valve is normally open it shall have a lit diffused red lens cover. The green light for this valve shall be a clear unlit green lens cover.

- If the valve is normally closed it shall have a lit diffused green lens cover. The red light for this valve shall be a clear unlit red lens cover.

- Information (General Use) (Hot Pink)

For status lights:

- If the status light is normally lit it shall have a diffused lens cover of the appropriate color.

- If the status light is normally unlit it shall have a clear lens cover of the appropriate color.

3. As the operator walks down the board he/she can look for the lit clear colored lights or a diffused lens cover which is not lit, which indicates an abnormal condition.

4. When equipment is alternated the lens covers shall be changed to match the lens cover convention.

5. This light convention is also being applied to the electrical boards in the plant. The same rules apply as for the control room. If a lens cover is normally lit on the electrical boards it shall be a diffused lens cover. The remainder shall be clear colored lenses. By using this light convention on the electrical boards a person

can quickly look at the board and tell if something is abnormal. After verifying light bulbs are good, the Outside Unit Supervisor should be contacted for any abnormalities identified.

6. As equipment is alternated the AUOs shall change the lens covers on the electrical boards to match what is normally running. These lens covers are located in the black Operations boxes located near the boards.

QUESTION 69 Rev 0

Unit 1 is performing a startup per 1-GOI-100-1A, Unit Startup.

During control rod withdrawal, prior to critically, the following conditions are noted:

- SRM PERIOD, (1-9-5A, Window 20), in alarm
- SRM period indicates 25 seconds on 1-XI-92-7/44A

Which ONE of the following completes the statement below?

The Unit Operator is required to _____.

A. **PAUSE** Control Rod withdrawal until a stable period of greater than 100 seconds is observed

B. **REINSERT** the last Control Rod withdrawn to obtain a stable period greater than 60 seconds

C. **INSERT** Control Rods **AND** verify the Reactor is brought subcritical

D. **SHUT DOWN** the Reactor until a thorough assessment has been performed

Answer: C

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G. 2.2.1	
	Importance Rating	4.5	4.4

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (CFR: 41.5 / 41.10)

Justification for K/A match: This is a Tier 3 Generic K/A concerning the performance of pre-startup procedures that control reactivity. To match this K/A the question is written for the generic start up plant procedure and the guidance in there for how to operate with short periods. The answers are all actions that are addressed in that General Operating Instruction.

Explanation: CORRECT C: INSERT Control Rods **AND** verify the Reactor is brought subcritical. These are the step that have to be taken for a period less than 30 seconds and are specified in the GOI and in the ARP.

A. **Incorrect because -** a 100 second period is the optimum period to achieve during startup to criticality.
Plausible since this would be correct if you are pulling rods in the heating range and negative reactivity was turning power

B. **Incorrect because –** the trigger for this alarm is a period 30 seconds. This action is for indication of < 60 but >30 second period.
Plausible since this is an action that must be performed if the period was greater and 30 seconds but less than 60.

D. **Incorrect because -** this action is based on receiving a 5 second period indication.
Plausible since it is rare that a period this short is experienced during normal rod withdrawal, so shutting down could be warranted, but the period did not meet that threshold in accordance with the procedure.

Technical Reference(s): 1-GOI-100-1A, Unit Startup Rev 44, 1-ARP-9-5A Window 20 Rev 20

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.059 ILT Obj 1 and 7

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:	41(b)(5)

5.4 Withdrawal of Control Rods while in Mode 2 (continued)

NOTE

The following steps apply for all Control Rod Withdrawals and do not require an Operator signoff for the steps. The actions should be reviewed by all personnel involved with withdrawing control rods.

[5] **MONITOR** Reactor Power during rod withdrawals and perform the following for the associated conditions.

[5.1] **IF** single-notch withdrawals result in a Reactor period of less than 60 seconds, **THEN**

PERFORM the following:

[5.1.1] **REINSERT** the last control rod withdrawn to obtain a stable period greater than 60 seconds.

[5.1.2] **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal.

[5.2] **IF** a Reactor period of less than 30 seconds is observed, **THEN**

PERFORM the following:

[5.2.1] **INSERT** control rods in accordance with 1-SR-3.1.3.5(A).

[5.2.2] **ENSURE** Reactor subcritical.

[5.2.3] **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal.

[5.3] **IF** a Reactor period of less than 5 seconds is observed, **THEN**

SHUT DOWN the Reactor until a thorough assessment has been performed (reference 1-GOI-100-12A).

Initials	Date	Time

SRM PERIOD 20	Sensor/Trip Point: Relay K21 30 seconds period

(Page 1 of 1)

Sensor Location: Panel 1-9-12, MCR.

Probable Cause:
A. Electrical noise.
B. Rx power rising on a period of ≤ 30 sec.
C. SI (or SR) in progress.
D. Malfunction of sensor.

Automatic Action: None

Operator Action:

A. **CHECK** reactor period meter reading and amber indicating light illuminated on Panel 1-9-5.

B. **IF** withdrawing control rods and a period less than 30 seconds is observed, **THEN**
INSERT rods until subcriticality is observed and **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission before pulling any more rods.

QUESTION 70 Rev 0

Which one of the following is **NOT** an approved method of maintaining system status?

A. Clearances

B. Approved procedures

C. Temporary Modifications (T-Mods)

D. Off Normal Equipment Alignments (ONEAs)

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G 2.2.14	
	Importance Rating	3.9	4.3

Knowledge of the process for controlling equipment configuration or status. (CFR: 41.10)

Justification for K/A match: This is a Tier 3 Generic K/A about the equipment configuration or status process. To match the K/A a question was written to have the candidate recall which is not an approved method of maintaining system status.

Explanation: CORRECT D: Each responsible individual is required to ensure the all activities that change the status of plant equipment are authorized by an approved plant procedure, clearance, work order or TACF. Also if not maintained by one of the four listed processes, that a "Mispositioned Component" is identified for any active component found out of the expected position for existing plant conditions when the component's required position is tracked by one or more of the following plant status control methods:
• Procedures, • Clearances, • Work Orders, • TACFs

A. **Incorrect because -** Clearances are an approved method.
Plausible since this is normally just used when maintenance is needed for the piece of equipment, not that it can be used as a way of controlling status.

B. **Incorrect because -** Approved procedures are an approved method.
Plausible since the procedure is used to manipulate the system it could be thought of as a temporary activity and not a controlling status activity.

C. **Incorrect because -** Temporary Modifications are an approved method.
Plausible since in a lot of cases a temporary mod is used until a procedure change is made, making it a temporary activity that could be thought not to control status of a system.

Technical Reference(s): NPG-SPP-10.1 Rev 7. NPG-SPP-10.2 Rev 13.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.113 Obj 9

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: 2011 Duane Arnold NRC Q 69
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis
10 CFR Part 55 Content:	41(b)10

3.1.4 Responsible Individuals

Ensure procedures and work documents restore systems and equipment to the correct status.

Ensure all activities that change the status of plant equipment are authorized by an approved plant procedure, clearance, work order or TACF.

Identify any discrepancy between the actual field status of a system or component and the status assumed by the prerequisites of a procedure.

Notify the SM / SRO of any discrepancy identified in this procedure.

1.0 PURPOSE

This procedure establishes a standardized method of control for temporary modifications (TMods). It provides the necessary requirements to ensure that consistent evaluations and plant operational impacts are performed and documented for the temporary modifications to Systems, Structures, and Components (SSCs). The TVA terminology is “T-Mod” but historically has been called Temporary Alterations, Temp Alts, or TACFs. The INPO terminology is “Temporary Configuration Changes".

Duane Arnold NRC Exam 2011 RO Question #69

Examination Outline Cross-reference: Level RO SRO
Tier # 3
Group # 2
K/A # G2 2.2.14
Importance Rating 3.9

Equipment Control: Knowledge of the process for controlling equipment configuration or status.

Question: RO Question # 69

Which one of the following are approved methods of deviating from the Locked Valve List?

1. Component clearance
2. An approved procedure
3. Work Control Supervisor direction
4. Operations Shift Manager direction

A. 1, 2, 4

B. 1, 3, 4

C. 2, 3, 4

D. 1, 2, 3

Proposed Answer: A

QUESTION 71 Rev 0

Which ONE of the following meets the requirements to be considered a "Complex Infrequently Performed Test or Evolution" (CIPTE) per NPG-SPP-06.9.1, Conduct of Testing?

A. Switching Order to remove the West Point 500KV line

B. 0-SR-3.8.1.9(A) Diesel Generator A Emergency Unit 1 Load Acceptance Test

C. 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test (IST Data)

D. 1-SR-3.5.1.6(RHR I) Quarterly RHR System Rated Flow Test Loop I

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G 2.2.7	
	Importance Rating	2.9	3.6

Knowledge of the process for conducting special or infrequent tests. (CFR: 41.10)

Justification for K/A match: This is a Tier 3 K/A to evaluate knowledge on the special or infrequent test. To match this K/A, a question was written that asks of the four choices provided which is or is not categorized as a special or infrequent test in the SPP-06.9.1.

Explanation: Correct B. 0-SR-3.8.1.9(A) Diesel Generator A Emergency Unit 1 Load Acceptance Test. This is a once per refuel cycle (24 month) surveillance i.e Infrequent.

A. **Incorrect because -** Switching Order to remove the West Point 500KV line, is a common switching order that is or could be frequently.
Plausible since switching is a very important to safety activity it might be thought to be a special test.

C. **Incorrect because -** 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test (IST Data) surveillance is a quarterly surveillance.
Plausible since HPCI Comprehensive Pump Test it might be thought to be a Complex, Infrequently Performed Tests or Evolutions.

D. **Incorrect because -** 1-SR-3.5.1.6(RHR I) Quarterly RHR System Rated Flow Test Loop I is not frequent, but it does not fit the category of infrequent.
Plausible since Quarterly RHR System Rated Flow Test Loop I it might be thought to be a Complex, Infrequently Performed Tests or Evolutions.

Technical Reference(s): NPG-SPP-06.9.1 Rev 9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.078, Conduct of Testing Lesson Plan. Objective V.B.8

Question Source: Bank: BFN 1108 Q #69
Modified Bank:
New:

Question History: Previous NRC: BFN 1108 Q #69

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

10 CFR Part 55 Content: 41(b)(10)

3.2.7 Complex, Infrequently Performed Tests or Evolutions

A. Implementing documents that are classified as CIPTEs shall include the following key elements:

   1. conditions that warrant stopping the test or evolution and steps to follow when that action is necessary (such as restoration of equipment and systems to stable conditions and guidance for removal of temporary modifications)

   2. verification that adequate margins of safety are maintained when interlocks and protection systems are bypassed

   3. engineering reviews (broad and specific) identifying conditions where the limits and controls in normal operating procedures are changed or exceeded

   4. clearly identified areas of deviation from, exceptions to, and applicability of other station operating procedures

   5. expected plant responses

   6. complete guidance concerning required actions to properly establish the conditions for the test or evolution, including installation and removal of temporary modifications

   7. contingency actions to address unexpected conditions or responses that may be encountered

B. Responsible supervisors shall consider the temporary assignment of additional personnel to assist in the conduct of these types of tests or evolutions. This includes assignment of personnel to exercise continuous responsibility for the oversight of a particular test or evolution, including controlling the pace and resolving problems. [C.3]

C. Responsible supervisors shall consider the need for just-in-time training. If just-in-time training is required, the training should be conducted in a simulator, if applicable.

D. Plant operators shall be properly prepared prior to the start of a special test. Such preparation shall include:

   1. Briefing of operators on the test objectives, initial conditions, anticipated plant performance, termination guidance, and risks involved

   2. Validation of the procedure by walk-throughs and trials on a plant simulator when feasible

   3. Establishment of clearly delineated responsibilities for the plant staff during the test [C.16]

E. Responsible supervisors shall consider the temporary assignment of additional personnel under the direction of the shift manager to augment the shift personnel (for example assignment of an engineer or coordinator for the test or evolution, assignment of an additional senior reactor operator during control rod manipulations, or assignment of additional data takers when data is not readily available to the assigned shift at their normal shift location). The duties, authority, and responsibilities of extra personnel should be included on the organization chart and made clear in the test briefings. [C.3]

F. Responsible supervisors shall ensure these types of tests or evolutions have been reviewed by individuals knowledgeable of the test or evolution before performance of the test or evolution.

G. Pre-test/evolution briefings, as a minimum, should include one briefing (for example a general test/evolution overview) before the test/evolution crew assumes shift duties (usually at the Operations shift turnover meeting), and a second briefing (for example detailed) before commencing the test.

H. The test coordinator/lead should ensure that the OE discussed at the CIPTE briefing is from significant operating experience (SOER, IER, Level 1 and 2), if available. Consideration should be given to reviewing the lessons learned from INPO SER 18-90, Problems Experienced During a Special Test of the Main Turbine. If no significant OE is available, ensure the relevant industry or internal OE is provided. The test coordinator/lead should also ensure that the OE lessons learned and how they will be applied in the task performance are included in the brief to the performers. [C.13]

I. Site OE managers should be utilized to review and discuss pertinent operating experience for the test/evolution at the necessary briefings. REFER TO NPG-SPP 22.202, Human Performance Tools.

J. A senior line organization manager shall be designated for the CIPTE and shall have the necessary authority and experience to exercise continuous responsibility for the oversight of the particular test or evolution. The senior line organization manager, which may be designated by the plant manager, shall be senior to the shift manager, but shall not interfere with, or reduce, the shift manager's ultimate responsibility for the performance of the test/evolution. The senior line organization manager's responsibilities and authority shall include:

   1. ensuring that test/evolution conduct meets NPG standards

   2. discussing the topic areas prescribed by the CIPTE pre-test briefing checklist (form TVA 40682) at the necessary briefings [C.2]

   3. controlling the pace of the test/evolution, if necessary

   4. appropriation and direction of the necessary resources to resolve issues or problems encountered during performance of the test or evolution [C.4]

K. Post-test briefings, as may be required by the assigned senior manager, shall discuss lessons learned, procedure improvements, and any necessary changes to training.

5.0 DEFINITIONS

Complex Infrequently Performed Tests or Evolutions (CIPTE) - Infrequently performed tests or evolutions that have the potential to significantly degrade the plant's margin of safety that warrant additional management oversight and control. The following criteria shall be used to identify these types of tests/evolutions:

A. Tests/evolutions not specifically covered by existing normal or abnormal operating procedures.

B. Tests/evolutions that are seldom performed even though covered by existing normal or abnormal procedures (for example, plant startup after a prolonged outage or after any outage that involves significant changes to systems, equipment, or procedures related to the core, reactivity control, or reactor protection).

C. Special, infrequently performed surveillance testing that involves complicated sequencing or placing the plant in an unusual configuration.

D. Tests/evolutions that require the use of special test procedures in conjunction with existing procedures.

This definition shall be reviewed and updated as part of the normal review cycle, as specified within the review cadence, to ensure that subsequently identified tests or evolutions receive the additional management attention prior to their performance. [C.1]

BFN 11-08 Question #69.

Which ONE of the following meets the requirements of a "Complex Infrequently Performed Test or Evolution" (CIPTE) per NPG-SPP-06.9.1, Conduct of Testing?

A. Switching Order to remove the West Point 500KV line in accordance with 0-GOI-300-4, Switchyard Manual.

B. 1-SR-3.5.1.7(COMP), HPCI Comprehensive Pump Test.

C. 0-SR-3.8.1.9(A) Diesel Generator A Emergency Unit 1 Load Acceptance Test.

D. 1-SR-3.5.1.6(RHR I) Quarterly RHR System Rated Flow Test Loop I.

CORRECT ANSWER C

Tier 3: Generic.
2.2.7. Knowledge of the process for conducting special or infrequent tests.
(CFR: 41.10/43.3/45.13) (RO – 2.9)
G2.2.7 NEW/L

Supporting References
NPG-SPP-06.9.1, Conduct of Testing

OPL171.078, Conduct of Testing Lesson Plan
Lesson Plan Objectives

Enabling Objectives of OPL171.078, Conduct of Testing include, “Describe the situations that require the Test Director to notify Operations personnel” (objective V.B.8).

QUESTION 72 Rev 0

Which ONE of the following completes the statement below?

The Wide Range Gaseous Effluent Radiation Monitor System (WRGERMS) consists of __ (1) __ ranges, AND can be monitored remotely from __ (2) __.

A. (1) TWO
   (2) all three Units Control Room

B. (1) TWO
   (2) the UNIT 2 Control Room

C. (1) THREE
   (2) all three Units Control Room

D. (1) THREE
   (2) the UNIT 2 Control Room

Answer: D

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.3.15	
	Importance Rating	2.9	

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Justification for K/A match: This is a Tier 3 Generic K/A about radiation monitoring systems. To match the K/A and to stay away from simple rad worker knowledge, a question about the plant's Wide Range Gaseous Effluent Radiation Monitor System and it's common display in the control rooms.

Explanation: CORRECT D: Normal, Intermediate and high ranges are supplied. Units 1 & 3 only receive common alarms. 1/3-9-3A windows 6 &13. The only remote monitoring is from Unit 2.

A. **Incorrect because –** there are three ranges not just two and all three are only displayed in the Unit 2 Control Room.
Plausible in that there is a Normal and a High range, and that each control room would be able to monitor WRGRM radiation levels.

B. **Incorrect because –** there are three ranges not just two and all three are only displayed in the Unit 2 Control Room.
Plausible in that there is a Normal and a High range, and that each control room would be able to monitor WRGRM radiation levels.

C. **Incorrect because –** there are three ranges not just two and all three are only displayed in the Unit 2 Control Room.
Plausible in that there is a Normal and a High range, and that each control room would be able to monitor WRGRM radiation levels.

Technical Reference(s): 2-OI-90, OPL171.033

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.033 V.B.2

Question Source: Bank: BFN 1205 NRC #72
Modified Bank:
New:

Question History: Previous NRC: BFN 1205 NRC #72

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

10 CFR Part 55 Content: 41(b)(11)

BFN 1501 NRC Q 72

Which ONE of the following completes the statement below?

The Wide Range Gaseous Effluent Radiation Monitor System (WRGERMS) consists of __ (1) __ ranges, AND can be monitored remotely from __ (2) __.

A. (1) TWO
(2) all three Units Control Room

B. (1) TWO
(2) the UNIT 2 Control Room

C. (1) THREE
(2) all three Units Control Room

D. (1) THREE
(2) the UNIT 2 Control Room

Answer: D

QUESTION 73 Rev 0

Which ONE of the following completes the statements below in accordance with RCI-9.1, Radiation Work Permits?

The Shift Manager has authorized immediate entry to a radiation area in emergency situations, Radiation Protection __ (1) __ be required to escort personnel entering the area.

When the area has been exited and the emergency situation is over an RWP __ (2) __ required to be completed for this entry.

A. (1) will
   (2) is

B. (1) will
   (2) is **NOT**

C. (1) will **NOT**
   (2) is

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

Answer: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.3.7	
	Importance Rating	3.5	

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Justification for K/A match: This is a Tier 3 Generic K/A concerning the ability to comply with RWPs either normal or abnormal. To match this K/A and to stay away from general rad worker knowledge, a question was written to place the unit in an abnormal radiological condition, and then asks about the Rad Con administrative requirements of emergency entry.

Explanation: CORRECT A: RP escort is required IAW RCI-9.1 section 3.2.20 and the RWP will be completed after the entry is completed or when the emergency is over.

B. **Incorrect because –** an RWP does have to be filled out after the emergency entry. **Plausible** that the dose received would be documented by means other than an RWP since RCI-9.1 3.2.3.G states: The RWP is the primary means by which RP documents and controls work in radiologically hazardous areas.

C. **Incorrect because –** an RP is required to escort the individual into the high rad area. **Plausible** since all entries into high radiation areas require supplemental monitoring in addition to the secondary dosimeter. Supplemental monitoring consists of the use of remote monitoring, a Personal External Alarm (PEA or similar device) or continuous Rad Ops coverage.

D. **Incorrect because –** First part Incorrect –
Plausible in that, RCI-9.1 section 3.2 states: All entries into high radiation areas require supplemental monitoring in addition to the secondary dosimeter. Supplemental monitoring consists of the use of remote monitoring, a Personal External Alarm (PEA or similar device) or continuous Rad Ops coverage. Second part incorrect – see B above.

Technical Reference(s): RCI-9.1 Rev 77.

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 1501 NRC Q 73
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis :
10 CFR Part 55 Content:	41(b)(12)

3.2.18 TEDE ALARA Assessment for Respirator Use Determination (continued)

2. Radiation Protection Manager for all TEDE ALARA evaluations in which the standard guidance is to be changed due to other mitigating factors (e.g., heat stress, different inefficiency factor, etc.).

F. For accountability purposes, obtain a control number from the TEDE ALARA Assessment control number log. The log is located on the Radcon drive under S:\Radcon\TEDE ALARA LOGBOOK.xls. Record the control number on Attachment 4 "Form FO-20 - TEDE ALARA Assessment for Respirator Use Determination."

3.2.19 RWPs Involving Movement of Special Nuclear Material (SNM)

A. Prior to the issuance of a radiation work permit (RWP) for the movement of SNM either into or out of an ICA, a copy of Form NPG-SPP-5.8-1, "Nonfuel Special Nuclear Material (SNM) Transfer Form" shall be submitted to Radiation Protection.

B. Form NPG-SPP-5.8-1, copy shall be kept with the RWP or at the applicable RP control point until completion of the job.

3.2.20 Emergency Situations

In emergency situations where the Shift Manager authorizes immediate entry to an area, the prior approval requirements of a RWP will be waived. If the RWP approval requirement is waived, Radiation Protection or the personnel escorted by RP must comply with radiation protection procedures for entry into high radiation areas (i.e., RP individual is equipped with radiation dose rate monitoring device and provides positive control over activities within the area to include protective recommendations for the personnel being escorted for the duration of the emergency). Radiation surveillance by virtue of RP escort is considered to be continuous coverage in this situation. The RWP must be completed when the emergency entry is completed or the emergency is over.

3.2.21 Safety Situations/ALARA Considerations

A. In certain instances it may be necessary for personnel to enter areas that do NOT have current survey data to perform personnel or nuclear safety related functions such as install lighting verify oxygen content, operate valves, perform leak detection, etc.

B. With the approval of the RP Shift Supervisor, the RP technician may enter the area with the work team and perform the survey with the work team. If any unacceptable radiological conditions are encountered, the technician will inform the team and all will exit the area.

BFN 1501 NRC Q 73

Which ONE of the following completes the statements below in accordance with RCI-9.1, Radiation Work Permits?

The Shift Manager has authorized immediate entry to a radiation area in emergency situations, Radiation Protection __ (1) __ be required to escort personnel entering the area.

When the area has been exited and the emergency situation is over an RWP __ (2) __ required to be completed for this entry.

A. (1) will
   (2) is Not

B. (1) will
   (2) is

C. (1) will Not
   (2) is Not

D. (1) will Not
   (2) is

Answer: B

QUESTION 74 Rev 3

Which of the following is an **ENTRY CONDITION** into the Emergency Operating Instructions (EOI) and what is the overall mitigating strategy as directed by that EOI for that parameter?

A. Primary Containment H_2 concentration above 4%;
Initiate CAD to the Drywell.

B. Suppression Pool Level above (-) 1 inches;
Maintain Suppression pool level below the suppression chamber-to-drywell vacuum breaker penetrations and within the safe area of the SRV tail pipe level limit.

C. Secondary Containment D/P at or above (-) 0.25 inches of water;
Restart the Reactor Zone ventilation to maintain habitability and differential pressure control.

D. Spent Fuel Pool Water Temperature above 125 °F;
Lower the Spent Fuel Pool temperature using Supplemental Fuel Pool Cooling with the B RHR Drain Pump.

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.4.1	
	Importance Rating	4.6	4.8

Knowledge of **EOP** entry conditions and immediate action steps.

Justification for K/A match: This K/A is not really applicable to BWRs, since they do not have immediate action steps in the EOPs. At the request of the NRC we developed a questions using the entry condition and the mitigating strategies in lieu of the immediate action steps portion of the K/A.

Explanation: CORRECT B: The EOI-2 Entry Condition for high Suppression Pool Level is Above (-) 1 inch and IAW EOIPM 0-V-E the overall mitigating strategy is Maintain Suppression pool level below the suppression chamber-to-drywell vacuum breaker penetrations and within the safe area of curve 4.

A. **Incorrect because** – The EOI-2 Entry Condition for high H_2 concentration is 2.4% and EOI-2 does not direct initiating CAD.
Plausible because – Tech Spec LCO 3.6.3.2 for O_2 is less than 4% and CAD is used to control H_2 and O_2 in SAMG.

C. **Incorrect because** – The EOI-3 Secondary Containment D/P entry condition is at or above (-) 0 .17 inches of water.
Plausible because – Restarting Reactor Zone ventilation (appendix 8F) is correct.
OI-30B directs entry into 2-AOI-30B-1 if Reactor Zone D/P is not between (-) 0.25 inches and (-) 0.40 inches of water. The design basis for SGT is that 2 trains will draw down the secondary containment to ≥ (-)0.25 inches of water.

D. **Incorrect because** – EOI-3 does not direct using Supplemental Fuel Pool Cooling with an RHR Drain Pump B.
Plausible because – The EOI-3 Entry Condition on Spent Fuel Pool Water Temperature is above 125 °F and OI-74, Residual Heat Removal System, section 8.14 is Initiation of Supplemental Fuel Pool Cooling with RHR Drain Pump B.

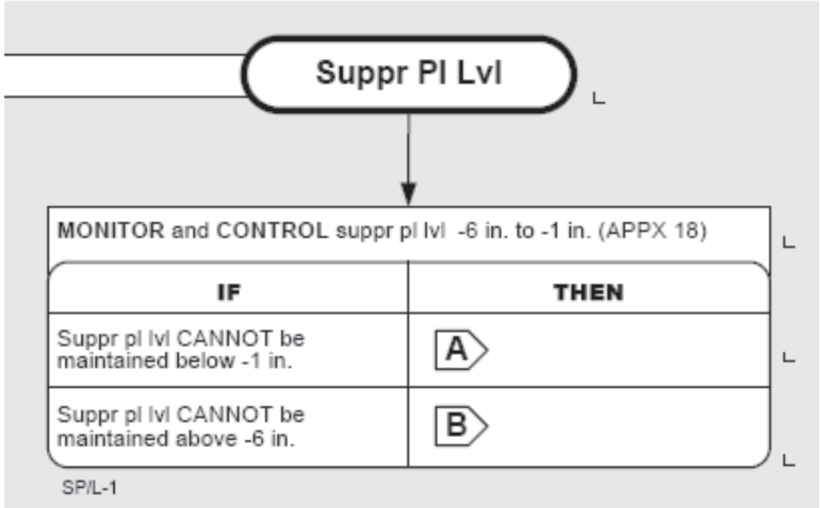
Technical Reference(s): 2-EOI-2 rev 15, 2-EOI-3 rev 16, EOIPM 0-V-D Rev 2, 0-III-E Rev 2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.201 rev 08 obj 2

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:	41(b)(10)	

2-EOI-2 PRIMARY CONTAINMENT CONTROL



1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

DISCUSSION: SP/L-3, SP/L-4, SP/L-5

If the suppression chamber-to-drywell vacuum breaker penetrations are submerged, the vacuum breakers cannot function as designed to relieve noncondensibles into the drywell and equalize drywell and suppression chamber pressures. Suppression pool water level must therefore be maintained below the bottom of the vacuum breaker openings to permit initiation and operation of drywell sprays.

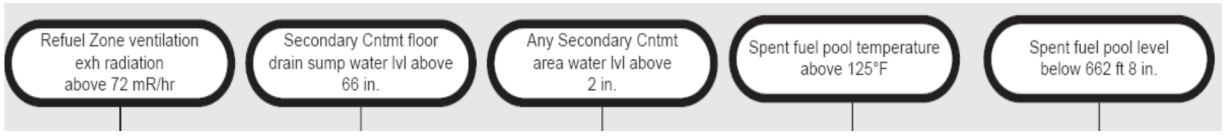
1.0 EOI-2, PRIMARY CONTAINMENT CONTROL BASES (continued)

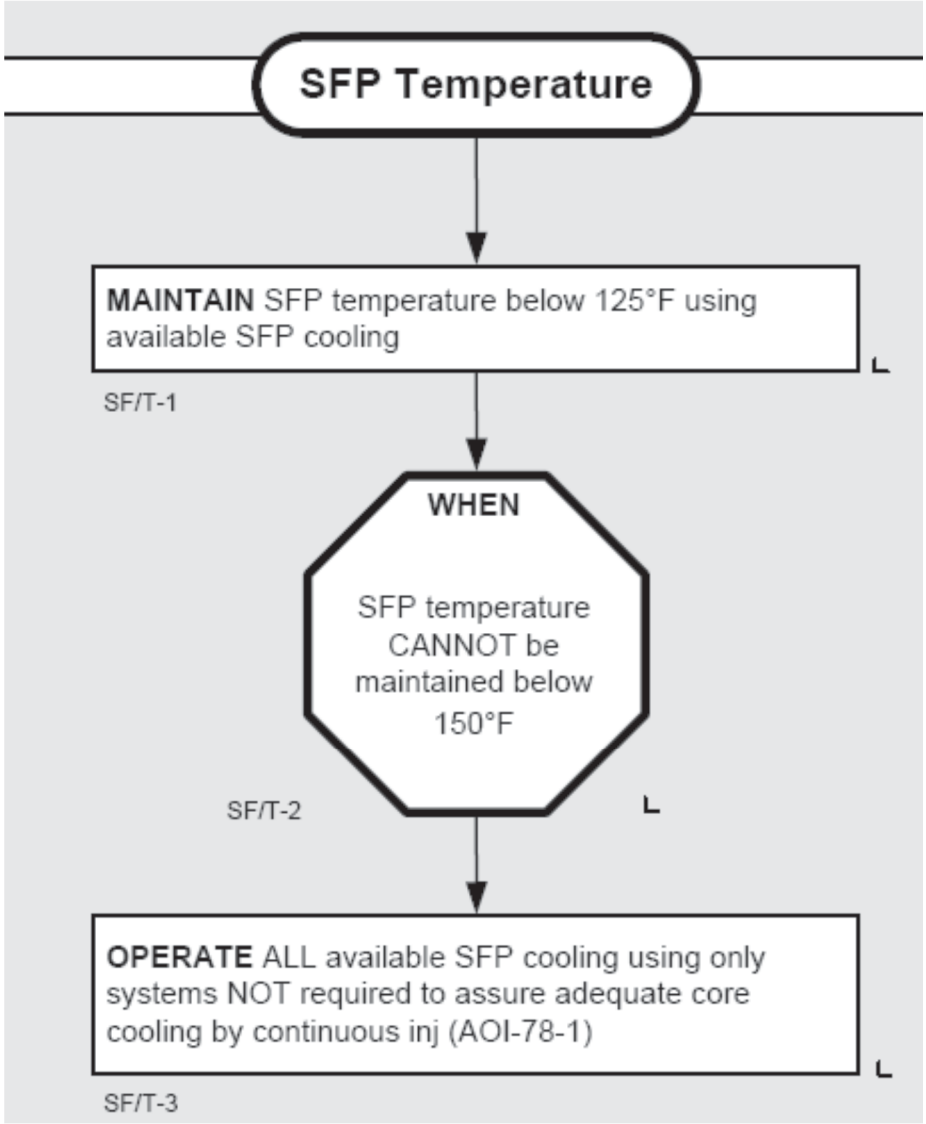
DISCUSSION: SP/L-6

The SRV Tail Pipe Level Limit (STPLL, Curve 4) is calculated in EOIPM Section VI-K and is the lesser of:

- The Maximum Pressure Suppression Primary Containment Water Level.
- The highest suppression pool water level at which opening an MSRV will not result in exceeding the code allowable stresses in the MSRV tail pipe, tail pipe supports, quencher, or quencher supports. The STPLL is a function of RPV pressure. MSRV operation with suppression pool water level above the STPLL could damage the MSRV discharge lines. This, in turn, could lead to containment failure from direct pressurization and damage to equipment inside the containment (ECCS piping, RPV water level instrument runs, wetwell-to-drywell vacuum breakers, etc.) from pipe-whip and jet-impingement loads.

2-EOI-3 SECONDARY CONTAINMENT CONTROL





1.0 PSTG TO EOI-3 CROSS REFERENCE (continued)

PSTG/SATG Step

SF/T Monitor and control spent fuel pool temperature below **A.37** (Spent fuel pool high temperature alarm setpoint) using available spent fuel pool cooling.

If spent fuel pool temperature cannot be maintained below **A.36** (Spent fuel pool design temperature).

- Operate all available spent fuel pool cooling, using only systems not required to assure adequate core cooling by continuous injection.

Discussion

1. The PSTG phrase "Monitor and control spent fuel pool temperature below" has been changed to MAINTAIN SFP temperature below" for simplification. It is evident from the context of this flowchart that SFP temperature must be monitored and controlled.
2. The PSTG phrase "If spent fuel pool temperature . . ." has been changed to "WHEN SFP temperature. . ." The logic term WHEN is the proper term because there is no alternative action to be performed once this step is reached other than the subsequent substep.

8.0 INFREQUENT OPERATIONS

8.14 Initiation of Supplemental Fuel Pool Cooling with RHR Drain Pump B

2-EOI-3 SECONDARY CONTAINMENT CONTROL

Any Secondary Cntmt area temp above Max Normal value of Table SC-1	Any Secondary Cntmt area radiation lvl above Max Normal value of Table SC-2	Secondary Cntmt dp at or above -0.17 in. of water	Rx Zone ventilation exh radiation above 72 mR/hr

Rx Zone ventilation is isolated AND Rx Zone ventilation exh radiation lvl is below 72 mR/hr	**RESTART** Rx Zone ventilation (APPX 8F) ➢ OK to defeat isolation interlocks if necessary (APPX 8E)
Refuel Zone ventilation is isolated AND Refuel Zone ventilation exh radiation lvl is below 72 mR/hr	**RESTART** Refuel Zone ventilation (APPX 8F) ➢ OK to defeat isolation interlocks if necessary (APPX 8E)

SC-1

BFN Unit 2	Reactor Zone Ventilation System	2-OI-30B Rev. 0028 Page 14 of 41

5.1 Startup of Reactor Zone Supply and Exhaust Fans (continued)

- **IF** REACTOR ZONE PRESS DIFFERENTIAL Indicator, 2-PDIC-064-0002, is not between -0.25 inches and -0.40 inches H_2O, **THEN**

  REFER TO 2-AOI-30B-1, Reactor Building Ventilation Failure. ☐

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[14] **IF** reactor building pressure CANNOT be maintained more negative than -0.25 inch H_2O, **THEN START** Stand-by Gas Treatment. REFER TO 0-OI-65.

SURVEILLANCE SR 3.6.4.1.3 and SR 3.6.4.1.4
REQUIREMENTS
To ensure that all fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that two SGT subsystems will draw down the secondary containment to $\geq$ 0.25 inches of vacuum water gauge in $\leq$ 120 seconds.

QUESTION 75 Rev 2

Unit 1 is operating at 100% power.

Which one of the following completes the statement below?

When assessing the EOI Exclusion Plot Status Boxes on the Safety Parameter Display System (SPDS):

__ (1) __ is expected to be colored “red” because current plant operation __ (2) __ within the “Safe” region of the curve.

A. (1) Curve 5, DW Spray Init Limit
(2) is

B. (1) Curve 5, DW Spray Init Limit
(2) is **NOT**

C. (1) Curve 6, Press Suppr Press
(2) is

D. (1) Curve 6, Press Suppr Press
(2) is **NOT**

Answer: B

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G 2.4.21	
	Importance Rating	4.0	4.6

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7)

Justification for K/A match: SPDS, although not qualified to make operational decisions, will assist the operator in assessing the status of safety functions. Therefore having them assess the status of containment conditions during normal operation is satisfactory to match this K/A.

Explanation: CORRECT B: Even though the SPDS system is not a qualified system, it can be and is used to aid the operator in making decisions if those parameters are verified with qualified instruments. In this case, during normal operation the drywell pressure is below the lowest limit of the Drywell Spray Initiation Curve, therefore turning its block red. All other parameters are in the satisfactory regions of their curves and the remainder of the status blocks will be green during normal operation.

A. **Incorrect because** – Curve 5, DW Spray Initiation Limit is **NOT** within the safe region of the curve at normal operating pressure and therefore would be colored Red not Green.
Plausible because the candidate may think all blocks should be green if there are no emergency conditions in the plant, but due to its design, this one is an anomaly.

C. **Incorrect because** – Curve 6, Press Suppr Press is within the safe region of the curve at normal operating pressure and therefore would be colored Green not Red.
Plausible because there are two states for each curve, either Safe or Not Safe, which toggles the colors on the screen, remembering which, is what color during certain conditions is a skill that a competent operator would have.

D. **Incorrect because** – Curve 6, Press Suppr Press is within the safe region of the curve at normal operating pressure and therefore would be colored Green not Red.
Plausible because there are two states for each curve, either Safe or Not Safe, which toggles the colors on the screen, remembering which, is what color during certain conditions is a skill that a competent operator would have.

Technical Reference(s): OPL 171.099 Integrated Computer System, Rev 9.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.099 Integrated Computer System, Obj 6

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis: X
10 CFR Part 55 Content:	10 CFR 55.41b(7)

INSTRUCTOR NOTES

13. High Pressure Heaters (HPHTR)
14. Meteorological Data (METDATA) Met data is obtained off the BOP Summary menu. (Go to Dose Assessment). This will get you to the Wind Speed/ direction and stability class. There are also links to MET Tower data and river data. This info is utilized for Emergency Plan response
15. Rx Feed Pump Turbs (RFPTURB)
16. Main Turbine Bearings (TURBBRG)
17. Turbine EHC system (EHC)
18. Demin System Flows (DEMFLO)
19. Condemin Alarms (CONDALA)
20. Demin System Overview (DSYS1)
21. Variable Frequency Drives VFDPMPA and VFDPMPB
22. Generator Hydrogen alarms (GENHYDA) and Stator Coolant Water alarms (STATCWA)

D. Safety Parameter Display System (SPDS)

1. Purpose

Obj. V.B.4

a. The SPDS generates the information necessary for rapid detection of abnormal or emergency operating conditions.

Instructor detail why this came about (TMI accident)

b. The SPDS monitors the plants response to corrective actions.

2. Component Description

a. The SPDS is part of the Integrated Computer System (ICS).

b. The input points were chosen to support the EOI entry conditions.

3. Relation of SPDS to EOIs — Obj. V.B.5

INSTRUCTOR NOTES

a. The SPDS was designed to integrate fully with the BFN EOI's by monitoring all entry conditions to the EOI's by providing trend graphs of the monitored parameters and by providing assistance with monitoring of several of the "parameter-vs-parameter" or "exclusion" plots in the EOI's.

Instructor discuss INPO SER 3-05 and 'monitoring of plant conditions and indications closely

b. Unit 2 and Unit 3 EOI-3 (Secondary Containment Control) entry conditions are not monitored by the SPDS.

c. SPDS is not Qualified Instrumentation; therefore, SPDS cannot be used as the sole guide in operating the plant. Qualified Instrumentation must be checked to backup SPDS before decisions or plant manipulations are made.

Obj. V.B.3.a

Query? How are EQ instruments labeled in MCR?

Management expectation/ Conservative Decision Making

4. Use of colors on the SPDS displays.

Colors have been used as sparingly as possible for the SPDS, and where possible matches the use of the same color elsewhere in the control room.

Stress use of redundant indication

a. RED indicates a value which exceeds the EOI entry condition.

Obj. V.B.7.a-f

b. GREEN indicates a value which does not exceed the EOI entry condition.

c. CYAN indicates a value which has been SUBSTITUTED by the computer operator. This should rarely, if ever, occur on the SPDS. CYAN is also used for trend lines so they will stand out against the scale lines.

Cyan is pale/light blue.

d. BLUE indicates BAD data. The SPDS will normally display the "last good value" for a parameter whenever it loses track for some reason. The dark blue characters are somewhat difficult to read, by design.