Examination Outline Cross-reference:

295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

G2.1.7 (10CFR 55.41.5)

Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295001G2.1.7	
Importance Rating	4.4	------

Proposed Question: **# 1**

Unit 1 is recovering from a trip of Recirc Pump 1A **AND** while executing the actions of 1-AOI-68-1A, “Recirc Pump Trip/Core Flow Decrease OPRMs Operable,” the Unit Operator (UO) has just reported that the RECIRC PUMP 1A DISCHARGE VALVE, 1-FCV-68-3, has been **MANUALLY** opened.

The Balance of Plant (BOP) Operator then reports that Recirc Pump 1B has tripped **AND** the Unit has entered Region I of the Power to Flow Map.

Which ONE of the following completes the statement?

For these conditions, the required action in accordance with 1-AOI-68-1A is to ________.

A. insert a manual Reactor Scram.

B. commence a normal Reactor shutdown / cooldown.

C. close the Discharge Valve on the outlet of Recirc Pump 1B.

D. insert Control Rods on the “Shove Sheet” to exit Region I of the Power to Flow Map.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** AOI-68-1A is entered; BOTH Recirc Pumps are now tripped. From this condition an immediate Rx Scram is directed iaw 1-AOI-68-1A.

B INCORRECT: Although this is a required action contained in 1-AOI-68-1A, it is an IF/THEN, which is dependent upon the “IF” of a single Recirc pump trip due to dual seal failure.

C INCORRECT: Although this is a required action contained in 1-AOI-68-1A, it is an IF/THEN, which is dependent upon the “IF” of a single Recirc pump trip.

D INCORRECT: Although this is a required action contained in 1-AOI-68-1A, in this case it would be overridden by the dual Recirc pump trip actions. Region I is an Immediate Exit Region, but in this case you would NOT wait to manually drive Control Rods on the “Shove Sheet” (standard terminology).

RO Level Justification: Tests candidate’s ability to evaluate plant performance, i.e., knowing the status of Recirc Pump 1A and where they are in the recovery process and how the tripping of Recirc Pump 1B impacts this recovery. The candidate must then make an operational judgment based upon previous data.

Technical Reference(s): 1-AOI-68-1A Rev 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:

Bank #	**X** 0801 #1
Modified Bank #	
New	

(Note changes or attach parent)

Question History: HLT 0801

Last NRC Exam (07 / 2009)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease OPRMs Operable	1-AOI-68-1A Rev. 0003 Page 6 of 12

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **IF** both Recirc Pumps are tripped in modes 1 or 2, **THEN** (Otherwise N/A)

[1.1] **SCRAM** the Reactor. ☐

> **CAUTION**
>
> [NER/C] Failure to restart Reactor Recirculation pumps in a timely manner may result in exceeding the differential temperature limit for pump start and subsequently require plant depressurization to avoid exceeding pressure-temperature limits for the reactor vessel. [SER 93-005]

[1.2] **RESTART** affected Reactor Recirculation pumps. Refer to 1-OI-68 Section 8.0. ☐

[1.3] **IF** the ΔT between the Rx vessel bottom head temperature and the moderator temperature precludes restart of a Recirc pump, **OR** forced Recirculation flow CANNOT be established for any reason, **THEN** (Otherwise NA)

[1.3.1] **INITIATE** a plant cooldown to prevent exceeding the pressure limit for the Rx vessel bottom head temperature indicated on REACTOR VESSEL METAL TEMPERATURE, 1-TR-56-4 pt. 10 (Panel 1-9-47) and based on Tech Specs Figure 3.4.9-1. ☐

[1.3.2] **INFORM** the Unit Supervisor, Tech Spec 3.4.1 requires the Reactor be placed in Mode 3 in 12 hours. **REFER TO** 1-GOI-100-12A and Tech Specs 3.4.1.B. ☐

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease OPRMs Operable	1-AOI-68-1A Rev. 0003 Page 7 of 12

4.2 Subsequent Actions (continued)

NOTE

1) Step 4.2[2] through Step 4.2[17.3] apply to any core flow lowering event.

2) Power To Flow Map is maintained in 0-TI-248, Station Reactor Engineer and on ICS.

[2] **IF** a single Recirc Pump has tripped, **THEN**

CLOSE tripped Recirc Pump discharge valve. ☐

[3] **IF** Region I or II of the Power to Flow Map is entered, **THEN** (Otherwise N/A)

IMMEDIATELY take actions to insert control rods to less than 95.2% loadline AND **REFER TO** 0-TI-464, Reactivity Control Plan Development and Implementation. ☐

[4] **RAISE** core flow to greater than 45% in accordance with 1-OI-68. ☐

[5] **INSERT** control rods to exit regions if **NOT** already exited AND **REFER TO** 0-TI-464, Reactivity Control Plan Development and Implementation. ☐

NOTE

The remaining subsequent action steps apply to a single Reactor Recirc Pump trip.

[6] **MAINTAIN** operating Recirc pump flow less than 46,600 gpm in accordance with 1-OI-68. ☐

[7] [NER/C] **WHEN** plant conditions allow, **THEN**, (Otherwise N/A)

MAINTAIN operating jet pump loop flow greater than 41 x 10^6 lbm/hr (1-FI-68-46 or 1-FI-68-48). [GE SIL 517] ☐

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease OPRMs Operable	1-AOI-68-1A Rev. 0003 Page 8 of 12

4.2 Subsequent Actions (continued)

CAUTION

The temperature of the coolant between the dome and the idle Recirc loop should be maintained within 75°F of each other. If this limit cannot be maintained, a plant cool down should be initiated. Failure to maintain this limit and **NOT** cool down could result in hangers and/or shock suppressers exceeding their maximum travel range. [GE SIL 251, 430 and 517]

[8] **IF** Recirc Pump was tripped due to dual seal failure, **THEN** (Otherwise N/A)

[8.1] **VERIFY TRIPPED**, RECIRC DRIVE 1A(1B) NORMAL FEEDER, 1-HS-57-17(14). ☐

[8.2] **VERIFY TRIPPED**, RECIRC DRIVE 1A(1B) ALTERNATE FEEDER, 1-HS-57-15(12). ☐

[8.3] **CLOSE** tripped recirc pump suction valve using, RECIRC PUMP 1A(1B) SUCTION VALVE, 1-HS-68-1(77). ☐

[8.4] **IF** it is evident that 75°F between the dome **AND** the idle Recirc loop cannot be maintained, **THEN**

COMMENCE plant shut down and cool down in accordance with 1-GOI-100-12A. ☐

[9] **NOTIFY** Reactor Engineer to perform Reactor Recirculation System Single Loop Operation, 1-SR-3.4.1(SLO) AND to refer to Station Reactor Engineer, 0-TI-248 and Tech Specs 3.4.1 as necessary. ☐

[10] [NER/C] **WHEN** the Recirc Pump discharge valve has been closed for at least five minutes (to prevent reverse rotation of the pump) [GE SIL-517], **THEN** (N/A if Recirc Pump was isolated in Step 4.2[8])

OPEN Recirc Pump discharge valve as necessary to maintain Recirc Loop in thermal equilibrium. ☐

Examination Outline Cross-reference:	Level	RO	SRO
295003 Partial or Complete Loss of AC / 6	Tier #	1	------
G2.1.27 (10CFR 55.41.7)	Group #	1	------
Knowledge of system purpose and/or function.	K/A #	295003G2.1.27	
	Importance Rating	3.9	------

Proposed Question: **# 2**

Which ONE of the following completes the statement?

480 Volt Load Shed Logic receives a Common Accident Signal generated from **__(1)__ AND** is divisionalized on **__(2)__** in regards to which board loads will actually be shed.

A. **(1)** RHR System instrumentation
(2) Unit 1

B. **(1)** Core Spray System instrumentation
(2) Unit 1

C. **(1)** RHR System instrumentation
(2) Unit 3

D. **(1)** Core Spray System instrumentation
(2) Unit 3

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: See below for Part 1 on RHR vs. Core Spray. RHR and Core Spray have identical initiation signals, which lends to plausibility. Unit 1 and 2 share 480 Volt Load Shed Logic and are dependent upon each other to function. This fact, in and of itself, lends plausibility to the logic being divisionalized on a priority basis.

B INCORRECT: First part correct (see below). Second part incorrect but plausible based on the premise discussed above.

C INCORRECT: First part incorrect for reasons discussed in 'A' above. Second part is correct.

D **CORRECT:** (See attached excerpts) Common Accident Signal Logic is generated from Core Spray Logic, as detailed in Lesson Plan OPL171.072. Only Unit 3 is divisionalized

RO Level Justification: Tests whether the candidate has knowledge of 480 Volt Load Shed Logic as it applies to a Loss of Offsite Power concurrent with a LOCA; both of which are required for it to function. This question also incorporates unit differences in the fact that Units 1 and 2 share Load Shed Logic and Unit 3 does not. Additionally, the question plays on the divisional aspects of load shed logic.

Technical Reference(s): OPL171.072, Rev. 11 (Attach if not previously provided)
3-AOI-57-1D, Rev. 6

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _V.B.2/ 3____________ (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

INSTRUCTOR NOTES

X. Lesson Body

A. The 480V Load Shedding Logic System removes selected loads from 480V boards which are powered from the 4kV Shutdown Boards

Obj. V.B.1/V.D.1
TP-1, 2

1. The load shedding is initiated by an accident signal on Unit 1 or 2 with <u>a</u> **diesel generator supplying one 4kV Shutdown Board as its only source of power**

Obj. V.B.3/ V.D.3
Obj. V.C.2

AND

2. The accident signal is generated in the Core Spray System logic

TP-3
Obj. V.B.2./V.D.2
Obj. V.C.1

a. Low-low-low reactor water level (-122"/Level 1)

OR

CASA signal

b. High drywell pressure (2.45 psig) with low reactor pressure (450 psig)

c. For load shed signal on U1 or U2, the accident is for either unit

d. Unit 3 accident signal won't cause Unit 1 or 2 load shed or vice versa

3. The signal representing "diesel generator supplying a 4KV shutdown board" is called "DGVA"

TP-4

4. For DGVA logic to be satisfied, both conditions must be present:

a. The DG output breaker or the U2 tie breaker to U3 being closed

b. The normal and alternate feeder breaker must be open

5. All Unit 1-2 DGVA contacts are in parallel

6. Any D/G tied to its Shutdown Board with an accident signal present will initiate U1-2 load shed logic

7. All unit 1and 2 components which are affected by 480v load shed receive signals from both divisions of load shed logic

8. U3 DGVA input to 480v load shed comes from the 4Kv shutdown board which is providing power to the 480v shutdown board — TP-5

9. U3 480v load shed is divisionalized; such that if the following conditions exist, load shed will occur: — Obj V.B.5/ V.C.4 Obj V.D.5

   a. Condition 1

      (1) 3A 480v S/D Bd normal feeder Bkr is closed

      (2) 3EA D/G is the only source of power to 3EA 4Kv S/D Bd (DGVA-A)

      (3) U3 accident signal is present

      OR

   b. Condition 2

      (1) 3A 480v S/D Bd alternate feeder Bkr is closed

      (2) 3EB D/G is the only source of power to 3EB 4Kv S/D Bd (DGVA-B)

      (3) U3 accident signal is present

   c. Load shed will then occur for the loads on the following boards:

      (1) 3A 480v S/D Bd

      (2) 3A 480v RMOV Bd

      (3) 3A 480v D/G Aux Bd

   d. (Division 2 load shed logic is similar)

OPL171.072
Revision 11
Appendix C
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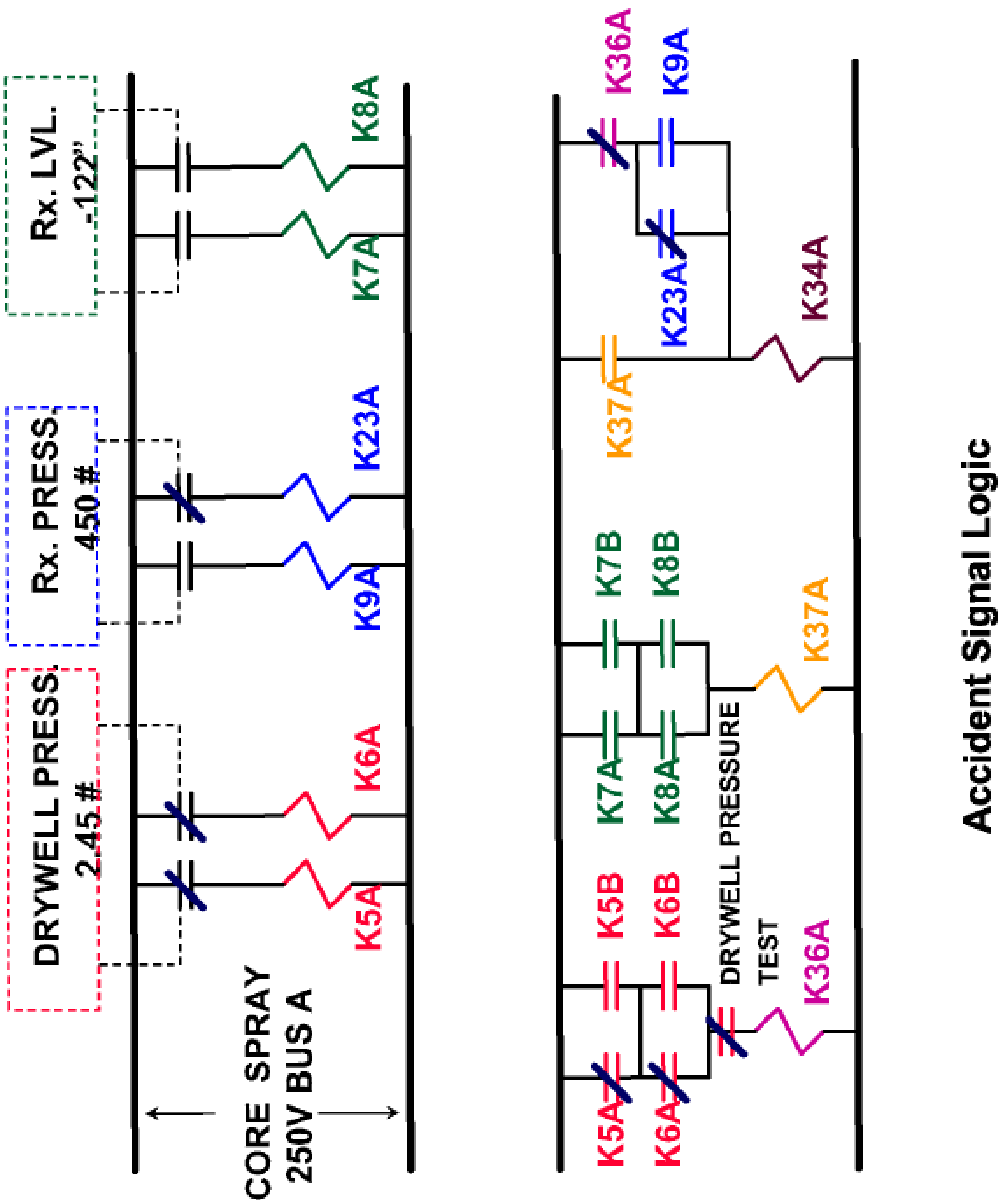

TP-3: Accident Signal Logic

BFN Unit 3	480V Load Shed	3-AOI-57-1D Rev. 0006 Page 4 of 21

1.0 PURPOSE

The 480V Load Shedding Logic System removes selected loads from 480V boards which are powered from the 4kV Shutdown Boards. Unit 3 Load Shed Logic is divisionalized. If a Unit 3 accident signal is present, **(122" Reactor Water Level or 2.45 psig High Drywell Pressure with 450 psig Low Reactor Pressure) and the appropriate conditions listed below:

DIVISION 1

3EA D/G or the U-1/2 crosstie breaker is the source of power to 3EA 4KV S/D BD (DGVA-A) and
3A 480V S/D BD normal feeder Bkr is closed

OR

3EB D/G or the U-1/2 crosstie breaker is the source of power to 3EB 4KV S/D BD (DGVA-B) and
3A 480V S/D BD alternate feeder Bkr is closed

THEN

Load Shed will occur for the loads on the following boards:

3A 480V S/D Bd
3A 480V RMOV Bd
3EA 480V D/G AUX Bd

DIVISION 2

3EC D/G or the U-1/2 crosstie breaker is the source of power to 3EC 4KV S/D BD (DGVA-C) and
3B 480V S/D BD normal feeder Bkr is closed

OR

3EB D/G or the U-1/2 crosstie breaker is the source of power to 3EB 4KV S/D BD (DGVA-B) and
3B 480V S/D BD alternate feeder Bkr is closed

THEN

Load Shed will occur for the loads on the following boards:

3B 480V S/D Bd
3EB 480V D/G AUX Bd
3B 480V RMOV Bd
3C 480V RMOV Bd *

* Normal Feeder Breaker for 480V RMOV Bd 3C is enabled for manual closure after 40 seconds and will auto CLOSE in 11 minutes

** -122 low-low-low level(Level 1)

4. Test Mode — TP-4
   a. Manual operation - can be performed during normal plant operation.
   b. Test valve sized for maximum flow (same valve as Suppression Pool Cooling).
   c. Isolates on LPCI injection signal.
5. LPCI Mode — TP-3
   a. Automatic, accident mode of RHR — Obj. V.B.18
   b. Initiated by a "one-out-of-two-twice" logic.
      (1) Low vessel water level, Level 1(-122 inches) , OR
      (2) High drywell pressure ($\geq$2.45 psig) and reactor pressure permissive ($\leq$450 psig)
   c. This signal starts the RHR pumps.
   d. Reactor Recirculation pumps are tripped at -45 inches. (Level 2)
   e. All valves not needed for LPCI injection automatically isolate and are interlocked shut as previously described.
   f. Sends permissive to ADS when RHR pump pressure is sensed >100 psig.
   g. Minimum flow valves open if injection flow in loop is <5800. Automatically shut as injection valves open and injection flow increases to >5800 gpm. — IE Bulletin 86-01
   h. Reactor pressure decreases through the break in conjunction with either HPCI or ADS.
      (1) As reactor pressure decreases to <450 psig the LPCI outboard injection valves open and are interlocked full open. They can then be throttled. Interlocking the injection path isolated until less than 450 psig protects the RHR piping from over-pressurization.
      (2) As pressure reaches <230 psig both Reactor Recirculation pump discharge valves shut and interlock shut.

Examination Outline Cross-reference:	Level	RO	SRO
295004 Partial or Total Loss of DC Pwr / 6	Tier #	1	------
AA1.03 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:	K/A #	295004AA1.03	
• A.C. electrical distribution	Importance Rating	3.4	------

Proposed Question: **# 3**

Unit 2 was operating at 100% Reactor Power.

A ground **AND** subsequent fire in Shutdown Board 250V DC Distribution Panel SB-B resulted in de-energization of the SB-B panel **AND** trip of 4kV Shutdown Board B Normal Feeder Breaker.

Which ONE of the following completes the statements?

480V Shutdown Board 2A is __**(1)**__.

4kV Shutdown Board B __**(2)**__ automatically transfer to its alternate source.

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

B. **(1)** de-energized
(2) will

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

D. **(1)** de-energized
(2) will **NOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - 480v Shutdown Board 2A is de-energized with the loss of 4kV Shutdown Board B. The transfer to alternate power is manual. Plausible in that Unit 1 and 3 480v Shutdown Board A normal power supply is from 4kV Shutdown Board A. Part 2 incorrect - Each Shutdown Battery system supplies its respective 4KV Shutdown Board and 480V Shutdown Board. All control power transfers are manual. Plausible in that if control power transfer is automatic as board power supply is or control power was not from SB-B DC Distribution Panel, this would be the correct answer.

B INCORRECT: Part 1 correct – See explanation D. Part 2 incorrect – See explanation A.

C INCORRECT: Part 1 incorrect – See explanation A. Part 2 correct – See explanation D.

D **CORRECT**: Part 1 correct - 480v Shutdown Board 2A is deenergized with the loss of 4kV Shutdown Board B. It is the normal feeder to the 480v S/D Bd 2A and the transfer to alternate power is manual. Part 2 correct - Each Shutdown Battery system supplies its respective 4KV Shutdown Board and 480V Shutdown Board. All control power transfers are manual. With the loss of control power, normal automatic transfer to alternate power supply will not occur.

RO Level Justification: To successfully answer this question, candidate must recognize the impact of partial loss of DC (SB-B Distribution Panel) will have on control power to 4 kV Shutdown Board B and the impact of loss of 4kV Shutdown Board B will have on 480v Shutdown Board 2A. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.036 Rev 12 (Attach if not previously provided)
OPL171.037 Rev 12
0-OI-57B Rev 184

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.037 V.B.1 (As available)
OPL171.036 V.B.6/8

Question Source: Bank #
Modified Bank # OPL171.037 #51 (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

d. Synchronizing System

(1) All four breakers feeding the unit 1/2 shutdown boards require the use of synchroscope to parallel supplies or perform manual transfer.

(2) The SYNC switch must be on to complete the closing circuit for any board feeder unless the Board is dead as sensed by the Board's residual voltage relay.

J. 480VAC Standby Distribution Substations

1. 480V Shutdown Boards

a. Each unit has two 480V Shutdown Boards, A and B. Their normal and alternate power supplies are from their associated 4kV Shutdown Boards, as follows:

Obj. V.B.6.e
Obj V.D.5
Obj. V.D.6.e
Obj. V.C.1.e
Obj. V.B.6.f
Obj. V.C.1.f
Obj. V.D.6.f

480V Board		4kV Board U1/U3	4kV Board U2
A	Normal	A	B
	Alternate	B	C
B	Normal	CD	
	Alternate	B	C

b. All transfers are manual. The Board may be transferred from the Control Room by operating the transfer selector switch on panel 9-8. Manual transfer at the Shutdown Board is accomplished by (1) placing the normal/emergency switches (both normal and alternate breakers) in EMERGENCY, (2) placing the alternate breaker control switch in CLOSE and holding until (3) the normal breaker control switch is operated to TRIP. After the transfer operation, the normal/emergency switches should be returned to NORMAL so the breakers can be controlled from the Control Room.

Obj. V.B.8.e
Obj. V.C.2.e
Obj. V.D.8.e
Obj. V.B.8.f
Obj. V.C.2.f
Obj. V.D.8.f

c. The 480V Shutdown Boards feed safety-related loads, either directly or via feeder breakers to MCC boards. (In general, motors rated between 40 and 200 hp are served directly.)

Examples: SLC, RWCU, RBCCW, & FPC

d. Supply breakers are provided with relay overcurrent protection which will trip and lockout the associated breaker and lockout its alternate.

2. 480V Diesel Auxiliary Boards

a. Diesel Auxiliary Boards A, B, 3EA, and 3EB principally serve loads associated with the operation of the diesel generators. Other essential small loads are also served from these boards. Loss of any single diesel auxiliary board will not negate the effectiveness of standby core cooling. (Standby Gas Treatment System Trains A and B are served by Diesel Auxiliary Boards A and B. Train C is served by the 480V Standby Gas Treatment Board, which is connected through a transformer to 4kV Shutdown Board 3ED.)

Obj V.D.5

Excerpt from OPL171.037 Rev 12

d. Distribution

Each Shutdown Battery system supplies its respective 4KV and 480V Shutdown Board. All control power transfers are manual.

BFN Unit 0	480V/240V AC Electrical System	0-OI-57B Rev. 0184 Page 105 of 111

Illustration 1
(Page 7 of 9)

Auxiliary Power Supplies and Bus Transfer

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE 1	ALTERNATE 2	REMARKS
12	480V Turbine Building Vent Boards				
	A. Board A (Unit 1,2,3)	480V Unit Board A (Unit 1,2,3)	480V Common BD 1 (Unit 1 only) 480-V Com. BD 3 (Unit 2 and 3)		Automatic transfer from normal to alternate source is initiated by time-undervoltage on the normal source. Return to normal source is automatic upon return of voltage to normal source. The normally closed, manually operated bus tie breaker provides for maintenance on one bus section while keeping the other bus section energized and in operation.
	B. Board B (Unit 1,2,3)	480V Unit Board B (Unit 1,2,3)	480V Common Board 2		
13	480V Shutdown Boards				
	A. Unit 1, 480V Shutdown BD 1A	4kV Shutdown Board A	4kV Shutdown Board B		Transfer from normal to alternate source is manual. Interlocking is provided to prevent manually transferring to a faulted board and to prevent paralleling two sources. 480V Load Shed Relay Time Delay Setting is set at 1.8 secs per DCN-W14030.
	B. Unit 1, 480V Shutdown BD 1B	4kV Shutdown Board C	4kV Shutdown Board B		
	C. Unit 2, 480V Shutdown BD 2A	4kV Board B	4kV Shutdown Board C		
	D. Unit 2, 480V Shutdown BD 2B	4kV Shutdown Board D	4kV Shutdown Board C		
	E. Unit 3, 480V Shutdown BD 3A	4kV Shutdown Board 3EA	4kV Shutdown Board 3EB		
	F. Unit 3, 480V Shutdown BD 3B	4kV Shutdown Board 3EC	4kV Shutdown Board 3EB		

1. OPL171.037 51

Unit 2 was operating at 100% power (ALL ELECTRICAL SYSTEMS IN NORMAL ALIGNMENT) when a ground and subsequent fire in 250v DC distribution panel SB-B resulted in deenergization of the SB-B panel and trip of 4kv Shutdown board B normal feeder breaker.

ASSUME NO OPERATOR ACTIONS HAVE BEEN TAKEN

Which ONE of the following is correct?

A. 4kv Shutdown Board B is energized from D/G B.

B. 4kv Shutdown board B is energized from its alternate feeder.

C. 480v RMOV board 2D is energized from its alternate feeder.

D. 480v Shutdown Board 2A is energized from its alternate feeder.

Examination Outline Cross-reference:

295005 Main Turbine Generator Trip / 3

AA2.08 (10CFR 55.41.10)

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP:

- Electrical distribution status

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295005AA2.08	
Importance Rating	3.2	------

Proposed Question: **# 4**

Unit 2 is operating at 100% Reactor Power when the following alarm **AND** indications occur:

- EHC HYD FLUID HDR PRESS LOW, (2-9-7B, Window 1)
- EHC Header Pressure is lowering
- STANDBY EHC Pump is isolated and Red Tagged

Which ONE of the following completes the statements?

A Main Turbine TRIP occurs at EHC Control Oil Pressure of __**(1)**__.

Following Main Turbine Generator TRIP, Shutdown Bus 2 board loads will be supplied by __**(2)**__.

A. **(1)** 1300 psig.
(2) 161 kV sources.

B. **(1)** 1300 psig.
(2) 500 kV sources.

C. **(1)** 1100 psig.
(2) 161 kV sources.

D. **(1)** 1100 psig.
(2) 500 kV sources.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part incorrect – Plausible in that this is the EHC HYD FLUID HDR PRESS LOW alarm set point. Second part incorrect – Plausible in that Shutdown Buses have the capability of being powered from the 161 kV sources (via the Start Buses),

B INCORRECT: First part is incorrect as detailed in 'A' above. Second part correct as detailed in 'D' below.

C INCORRECT: First part is correct as detailed in 'D' below. Second part incorrect as detailed in 'A' above.

D **CORRECT:** First part correct – EHC Header Pressure of 1100 psig results in a Main Turbine Trip. Second part is correct - Although the Shutdown Buses have the capability of being powered from the 161 kV sources (via the Start Buses), there is nothing in the stem to indicate that 500 kV is locked out/unavailable. Therefore, in this particular case, Shutdown Buses will be powered from 500 kV backfeed.

RO Level Justification: Tests the candidate's ability to determine electrical distribution status following a Main Turbine Generator Trip. Additionally, tests the candidate's knowledge of Main Turbine Trip on low EHC Pressure. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-ARP-9-7B, Rev. 22 (Attach if not previously provided)
OPL171.010, Rev. 12
OPL171.036, Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.134, V.B.9 (As available)
OPL171.010, V.B.12

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BFN Unit 3	Panel 9-7 3-XA-55-7B	3-ARP-9-7B Rev. 0022 Page 4 of 47

EHC HYD FLUID
HDR PRESS
LOW
3-PA-47-1

1

(Page 1 of 1)

Sensor/Trip Point:
3-PS-47-1C (PS-30) 1300 psig

Sensor Location: EHC pump unit
El 586'
Turbine Bldg

Probable Cause:
A. Inservice pump trip.
B. Inservice filter plugged.
C. Hyd fluid piping break.
D. Sensor malfunction.

Automatic Action:
A. Standby pump starts.
B. Red light above EHC PUMP 3B(3A) TEST pushbutton 3-HS-47-5A(4A) for selected pump extinguishes at 1300 psig lowering, illuminates at 1500 psig rising.

Operator Action:
A. **VERIFY** Standby EHC PUMP 3B(3A), 3-HS-47-2A(1A) running. ☐
B. **CHECK** EHC HEADER PRESSURE, pressure indicator, 3-PI-47-7 between 1550 and 1650 psig. ☐
C. **DISPATCH** personnel to inspect EHC pump unit. ☐

NOTE
On EHC Hydraulic System failure, accumulator and check valve arrangement will provide approximately one minute bypass valve operation.

D. **IF** EHC Hydraulic system fails, **THEN** **VERIFY** turbine trips at or below 1100 psig. ☐

References: 3-45E620-10 0-45E602-6 3-45E602-3

d. Even though the valves are combined in a single casing each of the distinct valves has separate operating mechanisms and controls.

e. The purpose of the intercept valves is TO PROTECT THE TURBINE FROM EXCESSIVE OVERSPEED in the event of a generator trip or loss of load (load reject). Obj V.E.11

f. Any time the main turbine is tripped, the generator is separated from the system grid. Obj.V.E.12 Obj.V.B.9, V.D.6

(1) When the turbine trips, a large amount of energy is trapped between the high pressure turbine inlet and the low pressure turbine inlet. The energy is in the form of saturated steam and water in the cross-around piping and the moisture separators.

(2) If this steam were allowed to expand to the lowest pressure area it can find after the turbine main control and stop valves have been tripped closed, the path the steam would take is the normal flow path through the low pressure turbines to the main condenser.

(3) If this were to occur with no load on the generator and thus on the turbine rotor, the rotor would accelerate to dangerous speeds for which it was not designed.

(4) The purpose of the combined intermediate valves is to block this steam flow path (two valves for redundancy) and prevent excessive turbine overspeed.

3. By paralleling two diesels to a 4kV shutdown bus (UNITS IN PARALLEL mode), a CCW Pump and Condensate Pump can be started on a Unit Board to provide a heat sink and to supply makeup to the reactor vessel.

M. Integrated Plant Operations

1. During normal operation, station auxiliary power is taken from the main generator through the Unit Station Service Transformers. During startup and shutdown, auxiliary power is supplied from the 500-kV system through the main transformers to the Unit Station Service Transformer with the main generators isolated by the main generator breakers. Auxiliary power is also available through the two Common Station Service Transformers (CSSTs) and two Cooling Tower Transformers (CTTs). Standby (onsite) power is supplied by eight diesel-generator units (four for Units 1 and 2, and four for Unit 3).

2. In the event of a main generator trip during normal operation, the generator breaker opens and auxiliary power is supplied from the 500-kV system through the main transformer. Failure of a preferred offsite circuit from the 500-kV switchyard to the main power transformer for Units 1 and 2 brings about an automatic transfer for both safety and non-safety-related buses. The non-safety-related buses transfer to the CSSTs. The safety-related buses transfer to the alternate units' USSTs if voltage is available. If this supply subsequently fails, only the safety-related buses are automatically transferred to the standby onsite electric power sources.

3. Concerning Unit 3, failure of the preferred offsite circuit from the 500-kV switchyard to the main power transformer brings about an automatic transfer of the 4-kV Unit Boards with their connected Shutdown Boards to the Start Buses. If this supply is unavailable (or subsequently fails), the 4kV Shutdown Boards are automatically energized by the diesel generators.

d. Discharge flow path

(1) Two steam packing exhausters (SPEs)

(a) Normally maintain 10-12 inches of water vacuum on condenser by throttling discharge valve
(b) Standby unit auto starts if vacuum falls to 5" H2O vacuum (the inservice exhauster will trip)
(c) Drains through loop seal to condensate drain tank

(2) Discharges to 1.75 minute holdup volume

5. Turning Gear (OPL171.118) TP-1 and

a. Located between the C low pressure turbine and the generator.

b. Used to rotate the turbine and 3-5 rpm following shutdown. This is necessary to prevent rotor bowing caused by uneven turbine cooldown.

N. Turbine Protection and Reactor Scram Instrumentation

1. Turbine Trips

Obj. V.B.12
Obj. V.C.5
Obj. V.D.4
Obj. V.E.20

	Trip	Setpoint	Reason for Trip
a.	High reactor water level	+55" Level 8 2/3 logic	To prevent moisture carryover from the reactor into the turbine
b.	Low EHC control oil pressure (FAS)	≤1100 psig 2/3 Logic	Prevent loss of control of the turbine

Examination Outline Cross-reference:	Level	RO	SRO
295006 SCRAM / 1	Tier #	1	------
AK3.03 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to SCRAM: • Reactor pressure response	K/A #	295006AK3.03	
	Importance Rating	3.8	------

Proposed Question: **# 5**

Unit 3 is operating at 100% power conditions with EHC in the **PREFERRED** mode of operation. The reactor scrams on low water level with the following timeline:

- The Scram report notes Reactor Pressure at 980 psig and lowering slowly.
- One minute after the scram, the operator reports Reactor Pressure has turned at a low of 955 psig and is now rising.
- Two minutes after the scram, the operator reports Reactor Pressure at 980 psig and rising at approximately 2 psi/sec.

Which ONE of the following completes the statement?

EHC is in __**(1)**__ Pressure Control **AND** the Main Turbine Bypass Valves __**(2)**__.

A. **(1)** Header
(2) have failed to operate at their setpoint.

B. **(1)** Reactor
(2) have failed to operate at their setpoint.

C. **(1)** Header
(2) will open once they reach their setpoint.

D. **(1)** Reactor
(2) will open once they reach their setpoint.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: first part incorrect – second part correct as detailed in ‘B’ below.

B **CORRECT:** (See attached excerpts) While either mode of operation is permissible, BFN operates in REACTOR Pressure control to eliminate single failure vulnerabilities. But, if header pressure drops below 700 psig, EHC will auto swap to HEADER Pressure control. Stem conditions do not warrant an auto-swap. Thus, EHC remains in REACTOR Pressure control with the setpoint at 955 psig (approximate by procedure, but where we set it exactly). Normally, at 100% power, there is a delta between this setpoint and actual reactor pressure (1036 psig). But, following a scram, setpoint and actual reactor pressure converge to within a couple of psig; with Bypass Valves cycling to control. With Pressure rising past 980 psig two minutes after the scram, the Bypass Valves must have failed.

C INCORRECT: first part incorrect – second part incorrect as detailed in 'B' above.

D INCORRECT: first part correct – second part incorrect as detailed in 'B' above.

RO Level Justification: Tests the candidate's knowledge of EHC's Reactor Pressure Control response following a Reactor Scram and the reason for the specific pressure response driven by the stem of the question. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.228, Rev. 4 (Attach if not previously provided)
3-OI-47, Rev. 89

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.1.b / 9.a (As available)

Question Source:			
	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

	INSTRUCTOR NOTES
c. Selecting SPEED HOLD above 100-RPM will hold turbine speed at its current position and flash the **synch speed** indicator on panel 9-7 and the EHC Workstation on and off.	Right Unit, Train, Component
12. Speed Regulation	
a. Functions to modulate the control valves and intercept valves to prevent an overspeed condition both offline and online.	DCN 69410 (U2) Alarm is received when overspeed regulation is being enforced.
b. Offline, speed regulation occurs if turbine speed increases above the speed reference.	
c. Online, speed regulation occurs if bus frequency increases above 60 Hertz.	
d. Control valve regulation is set at 5% whereas the intercept valve regulation is set at 2%	
e. Whenever speed regulation takes effect, the SPEED CONTROL indicator will be illuminated.	Obj.V.B.9.q Parameter in Control
C. STEAM PRESSURE CONTROL	Obj.V.B.1.b
1. Steam Pressure Control maintains either the Header Pressure or the Reactor Pressure at a pressure reference setpoint. Steam pressure control, as a reactor pressure controlling parameter is utilized at all times, either by the Bypass Valve Control or the Turbine Control. The Bypass Valve Control System, however, is always in pressure control	TP-2B

		INSTRUCTOR NOTES
2.	During turbine start-up and for a brief time following synchronization, the bypass valve control also maintains the reactor steam pressure. Once all the bypass valves are closed, then the turbine control maintains reactor steam pressure either in Header Pressure or Reactor Pressure Control depending on which operating mode is selected.	Monitor Plant parameters for expected response
3.	Steam pressure control is selectable from either panel 9-7 or the EHC Workstation by selecting HEADER PRESSURE CONTROL or REACTOR PRESSURE CONTROL.	Obj.V.B.9.a
4.	Header Pressure Control Input Signal	
a.	Two redundant pressure transmitters sense header pressure at the main steam throttle just upstream of the main turbine stop valves.	Powered from within the EHC system
b.	Both signals are monitored for low, high, difference, and hardware failures.	
c.	The higher of the two signals when no failures are detected is selected as the input.	
d.	A maximum difference setpoint of 10-PSI is also established to detect a fault and/or transmitter drift from either of the inputs.	
e.	In the event a fault is detected, the channel is prohibited from being used in the signal processing and the appropriate BYPASS pushbutton light will illuminate on 9-7 and on the HMI operator interface.	Obj.V.B.9.c
f.	Once the failed signal is corrected, depressing the BYPASS pushbutton will reset the BYPASS logic and both input signals will then be processed.	

INSTRUCTOR NOTES

g. This mode IS NOT single failure proof - one of the two pressure sensors failing upscale can, and generally will be selected by the logic to control. This will open the TCV's and BPV's to depressurize the header to the MSIV isolation setpoint of 852 psig in RUN Mode.

h. In the unlikely event that both inputs signals are detected as failed, the control logic will automatically switch to reactor pressure control.

i. If header pressure drops below 700-PSI, and reactor pressure control is the controlling mode of operation, the control logic will automatically transfer to header pressure control. If desired, the operator may re-select reactor pressure control after the transfer has been made even though header pressure is below 700-psi. The automatic transfer logic will re-engage if header pressure rises above 725-psi.

5. Reactor Pressure Control Input Signal — TP-3

a. Four (4) redundant pressure transmitters (PT- 204a-d) grouped in pairs with "A" and "B" constituting one pair and "C" and "D" the other pair.

b. A pressure-biasing algorithm determines the lagged high-median value of the four (4) inputs and biases the remaining three (3) input signals to that high median value.

c. The high-median signal is then averaged with the other three signals and is used as "Actual Rx Pressure". — Four biased signals are averaged.

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3.9 EHC Controls

A. The EHC Workstation is normally in the VIEW mode. While in this mode, disabled function lettering is grey The only function available to the operator in this mode is the ability to reset an alarm on the Alarm screen.

Performing other functions at the workstation requires that the operator go to the Menu screen, select LOGIN button, type in as user OPS and password OPS. The enabled function lettering is now black. When the extended functions are no longer necessary the operator may return to VIEW mode by typing into user VIEW and password VIEW. Otherwise, the system will automatically revert to VIEW mode after approximately one hour of inactivity.

B. The EHC Control System can be used in either Reactor Pressure control or Header Pressure control. While in Header Pressure control, a single header pressure input failing high could cause the bypass valves to open. While in Reactor Pressure control, a single Reactor Pressure input failing high will not affect the bypass valves. For this reason, Reactor Pressure control is the preferred mode of operation for the EHC Control System.

C. The following pertain to the Max Combined Flow Limit:

1. Maximum combined flow limit setting of 150% (upper limit) precludes exceeding Thermal Limits during a single turbine control valve closure.

2. The max combined flow upper and lower setting limits are 50% and 150%. Normally it is set at 125%.

3. The Maximum Combined Flow Limit setting is adjustable only on the EHC WORK STATION computer (Panels 3-9-7 and 3-9-31).

4. Max Combined Flow Limit setpoint can be found on the following computer screens:

   a. On ICS, EHC TURBINE CONTROL (EHCTC) screen.

   b. On EHC WORK STATION, TURBINE CONTROL screen.

D. [NER/C] Complete failure of the Push Rod-Spring Guide Coupling Bolts on a Control Valve (CV) or a Combined Intermediate Valve (CIV) will give indication of a closed CV or CIV on Panel 3-9-7 even though the valve may actually still be full open. Should this event be believed to have occurred, and the unit appears to be running satisfactorily, DO NOT TRIP the turbine. Notify Operations management and the GE Representative for assistance. (Note: Maintenance will have these bolts completely changed out

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5.4 Turbine Roll (continued)

NOTE
The Motor Suction Pump may be placed in service, as needed, to assist in oil warm up prior to turbine roll.

[15] **VERIFY** MOTOR SUCTION PUMP, 3-HS-47-12A, is in service. ☐

[16] **IF** any Turbine lift pump motor has been disabled, **THEN**

RETURN the disabled Lift Pump(s) to service. **REFER TO** 3-OI-47B.(Otherwise N/A.) ☐

NOTE
The EHC Control System can be used in either Reactor Pressure control or Header Pressure control. While in Header Pressure control, a single header pressure input failing high could cause the bypass valves to open. While in Reactor Pressure control, a single Reactor Pressure input failing high will not affect the bypass valves. For this reason, Reactor Pressure control is the preferred mode of operation for the EHC Control System.

[17] **VERIFY** the following on EHC TURBINE CONTROL panel:

- EHC SETPOINT, 3-PI-47-162, indicates approximately 955 psig in Reactor Pressure Control, if available. ☐
- Either REACTOR PRESSURE CONTROL, 3-HS-47-204, or HEADER PRESSURE CONTROL, 3-HS-1-16, is ILLUMINATED. ☐
- BPV DEMAND, 3-ZI-47-130, indicates zero. ☐
- TURBINE TRIPS NORMAL green light, 3-IL-47-87, is illuminated. ☐
- VACUUM TRIP BYPASSED, 3-IL-47-72, is extinguished. ☐
- BPV VACUUM INHIBIT, 3-IL-47-73, is extinguished. ☐
- ALL VALVES CLOSED pushbutton backlight, 3-HS-47-77D, is illuminated. ☐
- CV POSITION LIMIT, 3-XI-47-157, set at approximately 66%. ☐

Examination Outline Cross-reference:	Level	RO	SRO
295016 Control Room Abandonment / 7	Tier #	1	------
AA1.08 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT:	K/A #	295016AA1.08	
• Reactor pressure	Importance Rating	4.0	------

Proposed Question: **# 6**

Due to toxic gas intrusion, the Unit 1 Control Room is being abandoned in accordance with 1-AOI-100-2, "Control Room Abandonment." **ALL IMMEDIATE** Operator Actions have just been completed.

Which ONE of the following completes the statements?

UNTIL control is established at the Backup Control Panel, Reactor Pressure will be controlled by the **__(1)__**.

During depressurization / cooldown efforts at the Backup Control Panel, with Reactor Pressure at 58 psig the Operator will **__(2)__**.

A. **(1)** SRVs in Safety Mode.
(2) monitor RCIC operation while injecting.

B. **(1)** Turbine Bypass Valves.
(2) monitor RCIC operation while injecting.

C. **(1)** SRVs in Safety Mode.
(2) verify that RCIC has automatically isolated.

D. **(1)** Turbine Bypass Valves.
(2) verify that RCIC has automatically isolated.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: (See attached excerpts) Part 1 of 'A' and 'C' distractors plays on a misconception. There are many Immediate Actions in AOI-100-2. Occasionally, in the heat of the moment during execution of Abandonment actions (JPMs and Scenarios), candidates will close the MSIVs before leaving the Control Room because they remember that they do get closed; but, in reality, it is not until they are closed from the Backup Control Panel. Part 2 is correct in that RCIC is not shut down until 50 psig.

B **CORRECT:** (See attached excerpts) Nothing in AOI-100-2 actions disables Turbine Bypass Valves –i.e., like putting EHC Pumps in Pull-to-Lock. Thus, until MSIVs are closed, Turbine Bypass Valves will control pressure. Once the Subsequent Actions are conducted on location at the Backup Control Panel, the MSIVs get closed, transitioning pressure control to a subset of the total available SRVs. RCIC would have normally isolated by now (at 60 psig – Units 1 &3); but, this isolation is defeated upon transferring control to the Backup Control Panel. Because RCIC is below its full-flow design spectrum of 150 – 1120 psig (Rx Pressure), system operation can be erratic and should be monitored closely.

C INCORRECT: Part 1 is incorrect for the same reason detailed in 'A' above. Part 2 is incorrect because the normal RCIC isolation at 60 psig is defeated upon transferring control to the Backup Control Panel.

D INCORRECT: Part 1 is correct. Part 2 is incorrect for the same reason detailed in 'C' above.

RO Level Justification: Tests the candidate's knowledge of Reactor Pressure Control as it relates to Control Room Abandonment. Along with this, Immediate Operator Actions, system interrelationships, and intricacies are also tested as they relate to the Control Room Abandonment procedure. This question is rated as C/A due to the requirement to assemble, sort, and integrate two distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-AOI-100-2, Rev. 17 (Attach if not previously provided)
1-OI-71, Rev. 9
SSD 1P-071-0001A-00-02, Rev. 2
OPL171.040, Rev. 23

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.1 / 4 / 9 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BFN Unit 1	Control Room Abandonment	1-AOI-100-2 Rev. 0017 Page 7 of 81

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**

LOWER core flow to between 50-60%. (Otherwise N/A) ☐

[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] **CHECK ALL** control rods fully inserted. ☐

[5] **CHECK** all eight SCRAM SOLENOID GROUP A/B LOGIC RESET lights extinguished. ☐

[6] **TRIP** Reactor Recirc Pumps. ☐

[7] **ISOLATE** RWCU. ☐

[8] **VERIFY** Main Turbine tripped. ☐

[9] **TRIP** Reactor Feed Pumps as necessary to prevent tripping on high water level. ☐

[10] **START** Emergency Diesel Generators. ☐

[11] **VERIFY** each EECW header has at least one pump in service. ☐

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4.1 Immediate Action (continued)

[12] ANNOUNCE to all plant personnel

- Unit 1 Main Control Room is being evacuated ☐
- All operations personnel report to your assigned backup control stations ☐

[13] OBTAIN hand-held radios from Unit 2 Control Room. ☐

[14] PROCEED to Backup Control Panel 1-25-32. ☐

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4.2 Unit 1 Subsequent Actions (continued)

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

[6] **CLOSE** MSIVs using the following switch sequence at Panel 1-25-32:

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

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

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

BFN Unit 1	Control Room Abandonment	1-AOI-100-2 Rev. 0017 Page 21 of 81

4.2 Unit 1 Subsequent Actions (continued)

NOTE

With RCIC steam supply line isolation valves 1-FCV-71-2 and 1-FCV-71-3 transfer switches in EMERG position, RCIC will **NOT** automatically isolate when Reactor Pressure drops to 50 psig.

[19] **WHEN** Reactor Pressure is less than 50 psig, **THEN**

SHUT DOWN RCIC, at Panel 1-25-32, as follows:

[19.1] **DEPRESS** RCIC TURBINE TRIP pushbutton, 1-HS-71-9C. ☐

[19.2] **CHECK** that the turbine comes to a complete stop using, RCIC TURBINE SPEED, 1-SI-71-42B. ☐

[20] **INITIATE** RHR Shutdown Cooling as follows:

[20.1] **VERIFY** REACTOR PRESSURE B, 1-PI-3-79, less than 50 psig at Panel 1-25-32. ☐

[20.2] IF RHR pumps are operating in Suppression Pool Cooling or RHR LPCI, **THEN**

PERFORM the following: (Otherwise N/A) ☐

[20.2.1] **VERIFY CLOSED** 1-FCV-074-0073, using RHR SYS II SUPPR POOL CLG/TEST VLV, 1-HS-074-0073C at 480V RMOV Bd 1B, Compt. R11C. ☐

[20.2.2] **VERIFY CLOSED** 1-FCV-074-0071, using RHR SYS II SUPPR CHBR/POOL ISOL VLV, 1-HS-074-0071C at 480V RMOV Bd 1B, Compt. 11C. ☐

[20.2.3] **VERIFY CLOSED** 1-FCV-74-0066, using RHR SYS II LPCI OUTBD INJECT VLV, 1-HS-074-0066C at 480V RMOV Bd 1B, Compt. 3A. ☐

Excerpt from Setpoint and Scaling Document 1P-071-0001A-00-02 (Rev. 2)

SETPOINT AND SCALING DOCUMENT FORM

Loop ID **1-P-71-1A**
Sheet 3 c/o 4

Setpoint And Scaling Document						
Loop Number **1-P-71-1A**			Quality-Related? Yes ☒ No ☐		Safety-Related? Yes ☐ No ☒ (Note 2)	
Loop Function: To isolate RCIC steam supply valves 1-FCV-71-2 and 1-FCV-71-3 and to trip the RCIC turbine whenever reactor vessel pressure is below the useful pressure range of the RCIC turbine for the purpose of preventing radioactive steam and gas leakage through the RCIC turbine shaft seals and into the reactor building.						
Loop Accuracy — Function			Allowable Value		Acceptable As Found	Acceptable As Left
Permissive to trip and isolate RCIC			76.0 psig (Note 1)		79.1 to 92.9 psig (Note 1)	83.6 to 88.4 psig (Note 1)
Loop Components No.	UNID No.	Process Range/Setpoint	Calibration Value Input	Calibration Value Output	Acceptable As Found	Acceptable As Left
1	**1-PS-71-1A**	60.0 psig setpoint decreasing	86.0 psig decreasing	Contacts Close	± 6.9 psig	± 2.4 psig
Component Description No.	Manufacturer / Model Number				Original Contract No.	Location
1	SOR / 5N6-B3-U8-C1A-JJTTNQ				DCN 51236	Pnl 25-7A, T-R1, 544' 5"

Additional Calibration Requirements:
Calibration Frequency Requirements: **Once every 24 months plus 25%**

Calibration Equipment Accuracy Requirements:
For pressure switch input M&TE: ICTe = ± 1.0 PSIG and ICRe = ± 1.0 PSIG; No output M&TE required

Static Head Value: 16.0 PSIG + 10.0 PSIG = + 26.0 PSIG (See Note 1)

Technical Specifications/TRM/ODCM Sections Applicable: 3.3.6.1

References:
No. ED-N0071-930113 No.

X. LESSON BODY

A. General Description — DCN's 51149, 51196, 51220, 51236 Make U1, U2, U3 the same.

1. The purpose of the RCIC System is to provide a source of high pressure coolant makeup to the reactor vessel in case of a loss of feedwater flow. The system is used to maintain the reactor water level and for reactor pressure control under MSIV isolation conditions and loss of normal feedwater. — TP-1 Obj. V.B.1. Obj. V.E.1

2. Safety Design Basis

RCIC operates automatically to maintain sufficient coolant in the vessel so that the fuel will not overheat in the event of reactor isolation and loss of feedwater flow. The system is a consequence limiting system rather than an ECCS system.

B. The RCIC System consists of: — Obj. V.D.1 Obj. V.E.2

1. Turbine-driven pump located in basement of Reactor Building (Elev. 519)

2. Turbine is driven by steam from Main Steam Line C and exhausts to the suppression pool. — Obj. V.B.2.

3. Pump is normally lined up to take suction from the Condensate Storage Tank (CST), but can take suction from suppression pool (only done manually). — Obj. V.E.3

4. Pump discharges to reactor via feedwater line B.

a. Turbine — Obj. V.B.1. TP-1 & TP-2

(1) 100% capacity

(2) Delivers full pump design flow at reactor pressures of 150 to 1120 psig

(3) 500 hp at 1200 psig to 80 hp at 225 psig

BFN Unit 1	Reactor Core Isolation Cooling System	1-OI-71 Rev. 0009 Page 10 of 71

3.0 **PRECAUTIONS AND LIMITATIONS (continued)**

F. [NER/C] If RCIC Turbine is tripped by mechanical overspeed trip device, RCIC TURB TRIP/THROT VLV, 1-FCV-071-0009, is required to be manually reset at the turbine. [INPO SOER 82-008] If there is NO positive indication in Control Room that 1-FCV-071-0009 is reset, personnel should verify position and document in rounds sheet.

G. RCIC Turbine operation below 2100 rpm may result in unstable system operation and equipment damage.

H. RCIC SUPP CHBR TURB EXH VAC RELIEF VLV, 1-FCV-071-0059, is normally de-energized in the open position and is required to be re-energized and closed to minimize leakage from primary containment following a LOCA when HPCI and RCIC are shut down and NO longer required.

I. Technical Specification 3.5.3 requires RCIC System operability be determined within 12 hours after RPV pressure is above 150 psig, or prior to startup using auxiliary steam.

J. RCIC Turbine oil drain and sample valves should not be operated without permission from the Unit Supervisor.

K. Injection of Suppression Pool water into the RPV should be avoided whenever possible to prevent degradation of primary system water quality.

L. [NER/C] Failure to manually trip the RCIC Turbine if speed exceeds 5700 rpm may result in equipment failure. [IE notice 90-045] Operation of the RCIC turbine can be stopped using the RCIC TURBINE TRIP pushbutton, 1-HS-71-9A. Section 8.4 is used to restore the turbine to operation if required.

M. When operable, RCIC SYSTEM FLOW/CONTROL, 1-FIC-71-36A, should be in AUTO in order to provide more stable system operation.

N. When RCIC SYSTEM FLOW/CONTROL, 1-FIC-71-36A, is operated in MANUAL, turbine speed should be raised as rapidly as possible to prevent turbine exhaust check valve chatter.

O. When RCIC STEAM LINE INBD ISOLATION VLV, 1-FCV-071-0002 or RCIC TURB STM SUP OUTBD ISOL VLV, 1-FCV-071-0003, are closed, MN STM LINE DRAIN INBD ISOL VLV, 1-FCV-001-0055 and MAIN STEAM LINE OUTBD ISOL VLV, 1-FCV-001-0056, should be open to drain RCIC Stm Line.

P. [NRC/C] When RCIC Steam Line has been isolated with 1-FCV-071-0003 only, MN STM LINE DRAIN INBD ISOL VLV, 1-FCV-001-0055 and MAIN STEAM LINE DRAIN OUTBD ISOL VLV, 1-FCV-001-0056, should be opened for at least one minute before closing 1-FCV-071-0002 and repressurizing the steam line. This action drains accumulated moisture from the upstream side of 1-FCV-071-0002. [RPT 84-52-06]

Examination Outline Cross-reference:	Level	RO	SRO
295018 Partial or Total Loss of CCW / 8	Tier #	1	------
AK3.07 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:	K/A #	295018AK3.07	
• Cross-connecting with backup systems	Importance Rating	3.1	------

Proposed Question: **# 7**

Unit 3 is operating at 100% Reactor Power. A grid disturbance results in Loss of Offsite Power; **BUT**, power is quickly restored. The following conditions currently exist:

- Raw Cooling Water (RCW) can **NOT** be restored
- **ONLY** Reactor Building Closed Cooling Water (RBCCW) Pump 3A could be started

Which ONE of the following identifies the reason Emergency Equipment Cooling Water (EECW) is cross-connected to RBCCW?

To provide cooling to _____.

A. RWCU Non-Regenerative Heat Exchanger to avoid resin damage.

B. Drywell Coolers to ensure Drywell Temperature limits **NOT** exceeded.

C. Reactor Recirculation Pump Sample Cooler to protect chemistry sample probes.

D. Spent Fuel Pool Cooling Heat Exchanger to ensure adequate decay heat removal.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT**:** 1-FCV-70-48 closes automatically on RBCCW Pump discharge header pressure at or below 57 psig. RWCU non-regenerative heat exchangers are on non-essential loop which is isolated with 1-FCV-70-48 closing.

B **CORRECT:** RBCCW SECTIONALIZING VLV, 1-FCV-70-48, closes automatically on RBCCW Pump discharge header pressure at or below 57 psig. Drywell Atmospheric Coolers are essential loads and remain cooled by RBCCW after 1-FCV-70-48, closes. Since RBCCW flow can only be restored to 1 Pump, 1-FCV-70-48 is not re-opened.

C INCORRECT: 1-FCV-70-48 closes automatically on RBCCW Pump discharge header pressure at or below 57 psig. Recirc Pump sample cooler is a non-essential load.

D INCORRECT: 1-FCV-70-48 closes automatically on RBCCW Pump discharge header pressure at or below 57 psig. Fuel Pool Cooling heat exchangers are on non-essential loop which is isolated with 1-FCV-70-48 closing.

Justification: To correctly answer this question, the candidates must demonstrate knowledge of the reason for cross-connecting EECW to RBCCW with plant conditions such that there is a complete loss of RCW and a partial loss of RBCCW. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.016 Rev 17 (Attach if not previously provided)

OPL171.047 Rev 12

0-AOI-24 Rev. 3

3-AOI-70-1 Rev. 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.047_V.B.3/6_ (As available)

Question Source:

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

Question History: Last NRC Exam RBS 08 #7

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

(1) Drywell accesses (not shown)

(2) Drywell penetrations (not shown)

(3) Pressure suppression chamber and pool

(4) Vent pipes, header and downcomer

(5) Torus - drywell vacuum breakers

(6) Primary auxiliary systems — TP-13

(a) Primary containment vent, purge and inerting system.

(b) Drywell air cooling system

1. Drywell coolers are designed to maintain drywell air temperature between 135°F and 150°F. — Obj. V.B.10.b & V.B.10.e; Obj. V.B.19.b; TP-27

a. Ten (10) cooling units are arranged in two (2) banks of five (5) units .

b. Units use RBCCW as cooling medium off the essential loop.

c. Powered from: — Obj. V.B.18.a

Units 1, 2, & 3 use a similar lineup. (# except U-1, B-2 is on 480V RMOV BD 1B)

A-1, A-2	A 480V S/D Bd	**DCN: 51090**
B-1, B-2 #	B 480V S/D Bd	See 1/2-OI-70
A-3, A-4	A 480V RMOV Bd	Illustration 2
B-3, B-4	B 480V RMOV Bd	Table 1
A-5, B-5	C 480V RMOV Bd	

d. Drywell blowers can be started using the control switch on panel 9-25. Any operating drywell cooler will trip following a load shed signal (Unit 1 or 2 accident signal and any Unit 1/2 diesel generator tied to its respective 4KV shutdown board). — Obj. V.B.12 & V.B.19.b; Obj. V.C.8 & V.C.10.b; See 1 & 2-OI-70 Ill 2, table 1

(1) The load shed trip signal will clear after a 40 second time delay and all previously operating drywell coolers on the **non-accident unit** will automatically start provided their respective control switch is still in the normal-after-start position.

BFN Unit 0	Degraded Raw Cooling Water Capability	0-AOI-24 Rev. 0003 Page 8 of 25

4.2 Subsequent Actions (continued)

[7] **IF** additional cooling is required, **THEN**

[7.1] **PLACE** EECW in service to Control Air Compressors using 0-OI-67. ☐

NOTE
Steps 4.2[7.2] and 4.2[7.3] may be performed as time and manpower permit.

[7.2] **VERIFY CLOSED** 0-SHV-024-1504 ☐

[7.3] **VERIFY CLOSED** 0-SHV-024-0562 ☐

[8] **IF** additional cooling is required, **THEN**

[8.1] **ALIGN** RBCCW Heat Exchangers to EECW using 1,2,3-OI-70 ☐

PROCEDURAL EXCERPTS FROM 3-AOI-70-1

BFN Unit 3	Loss of Reactor Building Closed Cooling Water	3-AOI-70-1 Rev. 0016 Page 4 of 13

1.0 PURPOSE

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

2.0 SYMPTOMS

A. Annunciator in alarm:

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

3.0 AUTOMATIC ACTIONS

RBCCW SECTIONALIZING VLV, 3-FCV-70-48, closes automatically on RBCCW Pump discharge header pressure $\leq$ 57 psig.

BFN Unit 3	Loss of Reactor Building Closed Cooling Water	3-AOI-70-1 Rev. 0016 Page 6 of 13

4.2 Subsequent Actions (continued)

[3] **IF** RBCCW Pump(s) has tripped and it is desired to restart the tripped RBCCW pump, **THEN**

PERFORM the following (otherwise N/A):

[3.1] **INSPECT** the tripped RBCCW pump and its associated breaker for any damage or abnormal conditions. ☐

[3.2] **IF** no damage or abnormal conditions are found, **THEN**

ATTEMPT to restart tripped RBCCW pump(s). ☐

[4] **IF** unable to restart a tripped pump, **THEN:** (Otherwise N/A)

PLACE Spare RBCCW Pump in service. REFER TO 3-OI-70. ☐

[5] **IF** RBCCW flow was restored to two pump operation by placing the Spare RBCCW pump in service in the preceding step, **THEN**

PERFORM the following (Otherwise **N/A**):

[5.1] **REOPEN** RBCCW SECTIONALIZING VLV, 3-HS-70-48A ☐

[5.2] **RESTORE** the RWCU system to operation. (REFER TO 3-OI-69) ☐

[6] **IF** RBCCW loss is partial, **THEN:**

CONTROL OR REDUCE Drywell and Recirc Pump Temperatures.(Otherwise N/A) REFER TO 3-GOI-100-12A

[6.1] **REDUCE** Recirculation flow as necessary until 50-60% core flow is reached. ☐

[6.2] **INSERT** control rods per 3-SR-3.1.3.5(a) as directed by Reactor Engineer until below 66.7% rod line. ☐

[6.3] **REDUCE** Recirculation flow as necessary until minimum core flow is reached. ☐

[6.4] **INSERT** control rods per 3-SR-3.1.3.5(a) as directed by Reactor Engineer. ☐

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

2. RBCCW Heat Loads

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

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

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

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

3. RBCCW Heat Exchangers

a. These provide the means for heat removal from RBCCW by RCW with Emergency Equipment Cooling Water (EECW) as a backup. — DCN 51195, replaced HX1A & 1B, HX 1C NOT replaced. OPL171.051

b. They are counter-flow type, 50% capacity each.

- RBCCW flow makes one pass through the shell side.
- RCW makes one pass through the tube side.

2008 River Bend Station
Initial NRC License Examination
Reactor Operator

QUESTION 7 Rev 1

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295018 AK3.07
	Importance Rating	3.1

Knowledge of the reasons for the cross connecting of backup systems as it applies to the partial or complete loss of component cooling water.

Proposed Question:

AOP-0011, Loss of CCP, provides guidance to supply certain CCP loads with Standby Service Water.

Why is it desirable to do this during a loss of CCP?

A. To provide cooling to the Reactor Recirculation Pumps to avoid seal degradation.

B. To provide cooling to the RWCU Non Regenerative Heat Exchanger to avoid RWCU resin damage.

C. To provide cooling to the Spent Fuel Pool Cooling Heat Exchanger to ensure adequate decay heat removal from the Spent Fuel Pool.

D. To provide cooling to the drywell sample cooler to protect chemistry sample probes from high temperature conditions.

Proposed Answer: C.

Explanation (Optional): When SSW is cross connected to CCP loads, SFC HXs, CRD pumps and RHR pump A&B seal cooler receive cooling. Recirc pumps, RWCU non regen HXs and RWCU pumps and the drywell sample cooler do not receive cooling from SSW when it is aligned to the CCP header.

Technical Reference(s): STM-115, Rev 4, AOP-0011 Rev 16

Proposed references to be provided to applicants during examination: NA

Learning Objective: Obj 2b, 3e, 5b, 11a

Question Source: New

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ☒3
Comprehension or Analysis ☐

Examination Outline Cross-reference:	Level	RO	SRO
295019 Partial or Total Loss of Inst. Air / 8	Tier #	1	------
AA2.02 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:	K/A #	295109AA2.02	
Status of safety-related instrument air system loads (see AK2.1 – AK2.19)	Importance Rating	3.6	------

Proposed Question: **# 8**

Given the following plant conditions:

- Unit 2 was at 100% Reactor Power when a transient occurred which resulted in a Reactor Scram
- After stabilizing the unit, the scram signal is RESET
- **ALL** eight (8) Scram Solenoid Group lights are ON
- Approximately ten minutes later, the following conditions are present:
  - RCW DISCH HDR PRESS LOW, (2-9-20A, Window 34), in alarm
  - CRD ACCUM CHG WTR HDR PRESS HIGH, (2-9-5A, Window 10), in alarm
  - Outboard MSIVs are CLOSED
  - Inboard MSIVs are OPEN
  - Scram Discharge Volume (SDV) Vent **AND** Drain Valves are CLOSED
  - Scram Inlet **AND** Outlet Valves are OPEN

Which ONE of the following describes the cause for the event?

A. Loss of Control Air

B. Loss of Both RPS Buses

C. Loss of Drywell Control Air

D. Loss of 9-9 Cabinet 5, Unit Non-Preferred

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** On loss of control pressure, all TCVs except for one TCV on each of the RBCCW heat exchangers fail open, resulting in low RCW pressure. 85-11A/B fail closed resulting in high charging pressure. Outboard MSIVs drift closed, Inboard MSIV are unaffected due to drywell control air. High steam line flow will not occur due to the unit scram on low scram pilot air pressure before MSIVs drift closed. SCRAM can be reset. SCRAM INLET and OUTLET VALVEs, 2-FCV-85-39A(B) fail OPEN. CRD SCRAM DISCH VOL VENT and DRAIN VLVs fail CLOSED on loss of air.

B INCORRECT: Loss of RPS would not result in RCW pressure low alarm. This is plausible because the scram would occur as well as scram valves open and SDV vents and drains closed, however these indications would NOT be appropriate AFTER the scram was reset. In fact, the scram could NOT be reset without RPS available.

C INCORRECT: This is plausible because a loss of Drywell Control Air would cause MSIVs to close, however the INBOARD valves would close.

D INCORRECT: This is plausible because loss of 9-9 Cabinet 5, Unit Non-Preferred affects several of the indications given. See attached excerpts from 2-AOI-57-4. However, outboard MSIVs would not close and SDV vents and drains would not fail to re-open.

RO Level Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of a loss of Control Air on safety related loads. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-AOI-32-2 Rev 32 (Attach if not previously provided)
OPL171.054 Rev 14

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _OPL171.054 V.B.9_ _ (As available)

Question Source:

Bank #	0610 #47
Modified Bank #	
New	

(Note changes or attach parent)

Question History: Last NRC Exam BFN 0610 RO

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

BFN Unit 2	Loss of Control Air	2-AOI-32-2 Rev. 0032 Page 19 of 25

Attachment 1
(Page 1 of 6)

Expected System Responses

1.0 MAIN STEAM

A. If the loss of control air is instantaneous, when control air pressure drops to < 45 psig, the MSIV accumulator air will be routed to close the MSIVs.

B. If the loss of control air is slow or gradual, a high probability exists for the accumulator air to be vented to atmosphere due to the slow realignment of the 4-way valve. This will prevent accumulator air from assisting in MSIV closure.

C. ADS relief valves have accumulator sized to supply air volume for five valve operations or to maintain the valve open for 30 minutes (valve will still operate for Reactor vessel pressure relief function regardless of Drywell Control Air pressure and accumulator air pressure).

D. Non-ADS relief valves will operate for Reactor Vessel pressure relief only.

E. STEAM SEAL REGULATOR, 2-PCV-1-147 steam seal regulating valve will fail closed on loss of air. Steam seal pressure can be controlled using STEAM SEAL REG BYPASS VALVE, 2-HS-1-145.

F. Pressure control valve 2-PCV-1-175A(B) for Off Gas Preheater A(B) fails open on loss of air. 2-FCV-1-176A and 2-FCV-1-176B are to be closed after loss of air.

G. Pressure control valves 2-PCV-1-151(153) for SJAE A(B) first and second stage fail closed on loss of air.

H. Pressure control valves 2-PCV-1-166(167) for SJAE A(B) third stage fail closed on loss of air.

5.0 RCW

A. All RCW temperature control valves fail open except for 2-TCV-24-80B and 2-TCV-24-85B on 2A and 2B RBCCW heat exchangers and 2-TCV-024-0075B on the Main Turbine Oil Coolers (4" line) which fail CLOSED.

20.0 CRD

A. SCRAM INLET and OUTLET VALVEs, 2-FCV-85-39A(B) fail OPEN.

B. EAST & WEST CRD SCRAM DISCH VOL VENT CONT VLVs A & B, 2-FCV-85-83(83A)(82)(82A) fail CLOSED on loss of air.

C. EAST & WEST CRD SCRAM DISCH VOL DRAIN CONT VLVs A & B, 2-FCV-85-37C(D)(E)(F) fail CLOSED on loss of air.

D. CRD SYSTEM FLOW CONTROL VALVEs, A & B 2-FCV-85-11A and 2-FCV-85-11B fail CLOSED on loss of air. Valves can be manually opened if required.

OPL171.054
Revision 14

2. Main Steam System — Obj. V.B.6

   a. MSIVs and MSRVs-The air accumulators for the MSIVs contain enough air for one closing actuation. When control air pressure drops to < 45 psig, the MSIV accumulator air will be routed to the MSIVs. — Obi. V.C.8

      (1) Control Air supplies normal air supply to the outboard MSIVs — Normal lineup

4. Control Rod Drive System
   a. FCV-85-11 will fail closed, but can be manually opened
   b. Scram valves will fail open
   c. SDV vent and drains will fail closed
5. Raw Cooling Water System TCVs fail open on loss of Control Air except for: TCV-24-80B and TCV-24-85B on the RBCCW heat exchangers and the Turbine Lube Oil TCV-24-75B-these fail closed. — Secondary effect on RCW header pressure drops.
6. RBCCW System NRHX outlet TCV fails open
   a. Drywell Cooler valves (RBCCW side and air side) fail open
   b. Cooling unit fan dampers fail closed

BFN Unit 2	Reactor Scram	2-AOI-100-1 Rev. 0092 Page 4 of 67

1.0 INTRODUCTION

1.1 Purpose

This instruction provides symptoms, automatic actions, and operator actions for a reactor scram.

2.0 SYMPTOMS

Any Reactor Scram.

3.0 AUTOMATIC ACTIONS

A. Any withdrawn control rods fully insert.

B. Scram Discharge Volume Vent and Drain valves close.

C. Reactor Water Level Control swaps to Single Element (3 sec delay), when either RPS A or B Backup Scram channel activates.

D. Recirc Pump 75% Runback initiated.

E. Programmed Scram Response of RFPTs if initiated.

1. For Scram Response logic to initiate, ALL of the following conditions are required:

    a. Scram Response Logic is not inhibited (amber light at SCRAM RESPONSE INHIBIT/RESET switch, 2-HS-46-5 on Panel 2-9-5, is extinguished).

    b. REACTOR WATER LEVEL CONTROL PDS, 2-LIC-46-5 on Panel 2-9-5, is in AUTO and at least one individual RFPT Speed Control PDS in AUTO.

    c. Either RPS A or B Backup Scram channel activates.

    d. Reactor Level (narrow range) falls below 0 inches within 60 seconds of first Backup Scram channel activating.

BFN Unit 2	Loss of Drywell Control Air	2-AOI-32A-1 Rev. 0021 Page 4 of 9

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for the loss of Drywell Control Air System for causes other than Group 6 Isolation. The loss of Drywell Control Air caused by a Group 6 Isolation is addressed in 2-AOI-64-2d.

2.0 SYMPTOMS

A. DRYWELL CONTROL AIR PRESS LOW (2-XA-55-3E, Window 35) at ≤ 87 psig.

B. MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW (2-XA-55-3D, Window 18) at ≤ 82 psig.

C. Inboard MSIV's close or start to close.

D. Drywell cooler dampers close.

3.0 AUTOMATIC ACTIONS

None

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0041 Page 4 of 32

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for Loss of Unit Preferred (Battery Board 2 Panel 11, Control Room Pane l2-9-9 Cabinet 5 Non-preferred and Unit Preferred to Panel 2-9-9 Cabinet #6).

NOTE

A loss of the Unit Preferred power source results in the following boards being de-energized:

- Battery Board 2 Panel 11
- Panel 2-9-9 Cabinet 5 Non-preferred
- Panel 2-9-9 Cabinet 6 Unit Preferred (if panel does **NOT** auto transfer to alternate)

2.0 SYMPTOMS

A. Loss of indication to CRD rod control indicator lights, Panel 2-9-5 (i.e. rod out permissive, timer switch malfunction - select block, rod withdraw, insert and settle, notch override and stabilizing valve indicator lights.)

B. The following recorders fail downscale (Panels 2-9-6, 2-9-7, 2-9-8):

1. Turbine Generator Vibration (2-XR-47-15),
2. TURB GEN ECC/SPEED/VALVE POSN (2-XR-47-16),
3. TURB EXP AND TEMP MN XFMR Winding Temp (2-XR-47-20/2-TR-47-20)
4. RFPT/RFP VIB and ECC (2-XR-3-177).

C. PANEL 2-9-9 CABINET 1, 2,3 or 6 CONTROL POWER TRANSFER (2-XA-55-7A, Window 15), on Panel 2-9-7, is in alarm on transfer of Unit Preferred Cabinet 6.

D. Unit 2 Panel 2-9-9, Cabinet 6 UNIT PFD 120V AC NORMAL and ALTERNATE SUPPLY lights 2-XI-57-600A and 2-XI-57-600B extinguish.

E. UNIT PFD SUPPLY ABNORMAL (2-XA-55-8B, Window 35) on Panel 2-9-8 is in alarm.

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0041 Page 7 of 32

3.0 AUTOMATIC ACTIONS

A. Panel 2-9-9, Cabinet 6, Unit Preferred automatically transfers to the alternate power source.

B. RFWCS Panel Display Stations are disabled and the system continues controlling on the last known signal.

C. CRD SYS FLOW CONTROL VLV 1A / B, 2-FCV-85-11 A / B CLOSES on loss of Cabinet 6 and OPENS if only Cabinet 5 is de-energized.

D. 2-FCV-1-58, UPSTREAM MSL DRAIN TO CONDENSER, auto opens.

E. Loss of one out of two auctioneered power sources for EHC Control System (ICS is the other power source).

F. Loss of power to Turbine Supervisory Instrumentation Panel 2-9-46.

G. Loss of one out of two auctioneered power sources for RFPT 2C Woodward Governor and Final Driver (ICS is the other power source).

H. Loss of one out of two auctioneered power sources for Recirc Flow Control System (ICS is the other power source).

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0041 Page 20 of 32

Attachment 1
(Page 2 of 10)

Power Supplies

2.0 UNIT 2 PANEL 9-9 CABINET 5 UNIT NON-PREFERRED

A. BKR 501, CONTROL AND SERVICE AIR VALVES

Secondary Containment Control Air isolation valves FSV-32-28B, FSV-32-29B, FSV-32-91B; Drywell Service Air valve FCV-33-10. Valves have parallel air supply solenoids, each solenoid having separate power supplies (Non-preferred and I&C Bus). Valve remains open if open.

1. BKR 502, PANEL 2-25-114 HYDROGEN GAS DRYER

   Loss of power to 500W Hydrogen gas heater

2. BKR 503, PANEL 2-9-13 TIP SYSTEM

   Loss of power to Panel 2-9-13 Tip System Channel A & B components:

   - Drive control units for channel A and B
   - Valve assemblies and guide tubes
   - Drive and indexing mechanisms
   - Flux probing monitors
   - Drive control units from channels C, D and E

3. BKR 504, UNIT PAGING SYSTEM

   Loss of power to unit paging system (communications intercom and paging system).

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0041 Page 21 of 32

Attachment 1
(Page 3 of 10)

Power Supplies

2.0 UNIT 2 PANEL 9-9 CABINET 5 UNIT NON-PREFERRED (continued)

4. BKR 505, PANEL 2-9-28 CRD SELECT RELAYS

Control Rod Drive Relay Panel 2-9-28

Loss of power to the following CRD components:

a. Rod Withdrawal Block Relays

b. Rod Control and Interlock Relays (Rod Overtravel, All Rods In, RPIS INOP)

c. Rod Control Indicator Lights (rod out permissive, timer switch malfunction - select block, rod withdraw, rod insert, rod settle, CRD notch override, refuel mode one rod permissive and stabilizing valves A and B lights)

d. Refuel Interlock Relays

Refuel and Startup Mode Auxiliary relays, refuel mode one rod out permissive, service platform hoist loaded, refuel equipment rod out block, refuel platform **NOT** over core, service platform jib crane, west scram discharge volume high level, east scram discharge volume high level

e. Rod Control and Interlock Relays

RWM Rod withdraw permissive, RBM failure to null withdraw switch operating relay, rod selected and driving relay, select withdraw, insert and timer relays, timer switch and unlatch relay, rod insert and withdraw block, continuous withdraw, insert block and settle relay

f. Rod Drift Sets 1, 2, 3, 4 and Power Supply for Drift Alarms

Loss of Rod Select and Rod Drift lights for each rod on the full-core-display

Loss of Rod Drift Test and Reset Relays, rod drift alarm relays for each rod

g. Scram Timing Test jacks on Panel 2-9-16.

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0041 Page 24 of 32

Attachment 1
(Page 6 of 10)
Power Supplies

2.0 UNIT 2 PANEL 9-9 CABINET 5 UNIT NON-PREFERRED (continued)

8. BKR 513, ALTEREX DCCT AND DCPT AC SUPPLY
   - Alterex System
   - Loss of power to the following Alterex System components:
   - DC current and DC potential transformers feeding field current and ampere meters and data logger
   - Regulator cubicle
   - Field Ground Detection Circuit
   - Generator Field Temperature Recorder

9. BKR 514, PANEL 2-9-19 CRD HYDRAULIC SYS

   Loss of power to the following CRD instrumentation:

PNL 25-18 Designation	PNL 2-9-5 Designation	Instrument Description
FI-85-25B	FI-85-25A	CRD Cooling Water Header Flow CRD
FI-85-15B	FI-85-15A	Drive Water Flow
FI-85-30B	FI-85-30A	CRD Exhaust Flow
PI-85-13B	PI-85-13A	Charging Water Header Pressure Flow Indicating Controller CRD System Flow
FIC-85-11B	FIC-85-11	

10. BKR 516, AIR PARTICULATE MON 2-RM-90-59 AND 51

    Loss of the following radiation monitors:
    - RE-90-51
    - RE-90-59

Examination Outline Cross-reference:

295021 Loss of Shutdown Cooling / 4

AK2.01 (10CFR 55.41.7)

Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following:

- Reactor water temperature

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295021AK2.01	
Importance Rating	3.6	------

Proposed Question: **# 9**

Unit 3 is in Mode 4 with the following conditions:

- Reactor Level band is (+) 70 to (+) 80 inches to support testing
- **ALL** Reactor Recirc **AND** RWCU Pumps are isolated and tagged
- RHR Loop I in Shutdown Cooling experiences an inadvertent Group 2 Isolation **AND** can **NOT** be restored

In accordance with 3-AOI-74-1, "Loss of Shutdown Cooling," which ONE of the following completes the statements?

Accurate Reactor Water Temperature __**(1)**__ available.

Reactor Coolant Stratification is indicated by __**(2)**__ .

A. **(1)** is
(2) Feedwater Sparger temperature **GREATER THAN OR EQUAL TO** 200°F on any Vessel Feedwater Nozzle indication.

B. **(1)** is **NOT**
(2) Feedwater Sparger temperature **GREATER THAN OR EQUAL TO** 200°F on any Vessel Feedwater Nozzle indication.

C. **(1)** is
(2) Reactor pressure **GREATER THAN** 0 psig with any Reactor Coolant temperature indication **GREATER THAN** 212°F.

D. **(1)** is **NOT**
(2) Reactor pressure **GREATER THAN** 0 psig with any Reactor Coolant temperature indication **GREATER THAN** 212°F.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that Reactor Level is high enough to establish natural circulation. Candidate may believe natural circulation is adequate to provide accurate level indication. Part 2 correct – See explanation B.

B **CORRECT**: Part 1 correct – In accordance with "Loss of Shutdown Cooling," 3-AOI-74-1, accurate coolant temperatures will not be available if forced circulation is lost. Part 2 correct – Reactor Coolant Stratification is indicated by Feedwater Sparger temperature GREATER THAN OR EQUAL TO 200°F on any Vessel Feedwater Nozzle indication.

C INCORRECT: Part 1 incorrect – See explanation A. Part 2 incorrect – Plausible in that in accordance with “Loss of Shutdown Cooling,” 3-AOI-74-1, with the Reactor in Cold Shutdown Condition (Mode 4 or Mode 5) coolant stratification may be indicated by Reactor pressure above 0 psig with any reactor coolant temperature indication reading at or below 212 F

D INCORRECT: Part 1 and 2 incorrect as explained above.

RO Level Justification: To successfully answer the question, the candidate must demonstrate knowledge of the interrelationship between loss of shutdown cooling and Reactor Water Temp. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.074 Rev 8 (Attach if not previously provided)

3-AOI-74-1 Rev 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.074 V.B.6 (As available)

Question Source:	Bank #		
	Modified Bank #	0707 Audit #49	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 3	Loss of Shutdown Cooling	3-AOI-74-1 Rev. 0017 Page 9 of 26

4.2 Subsequent Actions (continued)

NOTES

1) With the Reactor in Cold Shutdown Condition (Mode 4 or Mode 5), reactor coolant stratification may be indicated by one of the following:

- Reactor pressure above 0 psig with any reactor coolant temperature indication reading at or below 212°F.
- Differential temperatures of 50°F or greater between either RX VESSEL BOTTOM HEAD (FLANGE DR LINE) 3-TE-56-29 (8) temperatures and RX VESSEL FW NOZZLE N4B END (N4B INBD)(N4B END)(N4D INBD) 3-TE-56-13(14)(15)(16) temperatures from the REACTOR VESSEL METAL TEMPERATURE recorder, 3-TR-56-4.
- With recirculation pumps and shutdown cooling out of service, a Feedwater sparger temperature of 200°F or greater on any RX VESSEL FW NOZZLE (N4B END (N4B INBD)(N4D END)(N4D INBD) 3-TE-56-13(14)(15)(16) temperatures from the REACTOR VESSEL METAL TEMPERATURE recorder, 3-TR-56-4.

2) [NER/C] For purposes of thermal stratification monitoring, the bottom head drain line is more representative as long as there is flow in the line. [GE SIL 251 and 430]

BFN Unit 3	Loss of Shutdown Cooling	3-AOI-74-1 Rev. 0017 Page 13 of 26

4.2 Subsequent Actions (continued)

CAUTION

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

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

PERFORM the following:

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

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

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

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

PLACE the alternate loop of RHR in Shutdown Cooling. **REFER TO** 3-OI-74. ☐

[16] **IF** no Unit 3 RHR loop can be placed in Shutdown Cooling, **THEN** (Otherwise **N/A**)

OBTAIN Shift Manager approval and **PLACE** Unit 2 RHR loop in service, CROSS-TIED with Unit 3, for Shutdown Cooling. **REFER TO** 3-OI-74. ☐

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

INSTRUCTOR NOTES

C. 1/2/3 AOI-100-1, Reactor Scram

1. The reactor scram AOI-100-1 provides guidance regarding immediate operator actions required to stabilize the plant in the areas of controlling and monitoring reactor power, level, and pressure.

NOTE: Immediate Operator Actions are to be performed from memory.

2. The subsequent actions provide guidance for long term stabilization and recovery of both RPV and balance-of-plant parameters.

3. The following subsequent action sections should be studied in detail:

    a) Actions to stabilize Reactor power, level, and pressure

    b) Verification of all rods fully inserted

    c) Actions to secure the Main Generator and Turbine

    d) Resetting the scram and PCIS

    Obj. V.B.3
    Obj. V.B.4

    e) Scram Report (Attachments 1-3)

D. 1/2/3-AOI-74-1, Loss of Shutdown Cooling

1. This instruction provides the symptoms and operator actions for a Loss of Shutdown Cooling.

2. Accurate coolant temperatures will not be available if all forced circulation is lost.

3. Reactor vessel stratification may occur until Shutdown Cooling is restored or a Reactor Recirculation Pump is placed in service.

4. With the reactor in Cold Shutdown Condition (Mode 4 or Mode 5), reactor coolant stratification may be indicated by one of the following:

INSTRUCTOR NOTES

a) Reactor pressure above 0 psig with any reactor coolant temperature indication reading at or below 212°F. — Obj. V.B.5

b) Differential temperatures of 50°F or greater between either RX VESSEL BOTTOM HEAD (FLANGE DR LINE) 1/2/3-TE-56-29 (8) temperatures and RX VESSEL FW NOZZLE N4B END (N4B INBD) (N4B END) (N4D INBD) 1/2/3-TE-56-13(14) (15) (16) temperatures from the REACTOR VESSEL METAL TEMPERATURE recorder, 1/2/3-TR-56-4. — Obj. V.B.6

c) With recirculation pumps and shutdown cooling out of service, a Feedwater sparger temperature of 200°F or greater on any RX VESSEL FW NOZZLE (N4B END) (N4B INBD) (N4D END) (N4D INBD) 1/2/3-TE-56-13(14) (15) (16) temperatures from the REACTOR VESSEL METAL TEMPERATURE recorder, 1/2/3-TR-56-4.

5. For purposes of thermal stratification monitoring, the bottom head drain line is more representative as long as there is flow in the line. {GE SIL 251 and 430]

6. IF forced circulation has been lost and vessel cavity is less than 80 inches, THEN, RAISE RPV water level to 80 inches as indicated on 1/2/3-LI-3-55.

7. Maintain RPV water level between +70 inches to +90 inches as indicated on RX WTR LEVEL FLOOD-UP, 1/2/3 3-55.

0707 AUDIT
#49 295021 SDC LOSS 25
The following conditions exist on Unit 3:

- Unit in Mode 5 with Shutdown Cooling aligned to RHR Loop I.
- RPV Temperature band of 80°F to 100°F.
- RPV level band of +50 to +70 inches to support testing.
- Day 2 of a scheduled 29 day outage.
- RHR Loop II is tagged.
- BOTH Reactor Recirc Pumps secured.
- Reactor Water Cleanup operating.

The following annunciators are received 1 hour after taking the shift:

- RHR SYS I PUMP A TRIPPED, 3-XA-55-3D window 13.
- RHR SD CLG FLOW LOW, 3-XA-55-3D window 11.

Based on the above indications, which ONE of the below indications would represent Reactor Coolant Stratification _(1)_ and how could this be mitigated _(2)_?

A. (1) Differential temperature of $\geq$ 40°F between Vessel Flange Drain Line and Vessel Feedwater Nozzle indications.
(2) Maximize Reactor Water Cleanup operation.

B. (1) Reactor pressure > 0 psig with any Reactor Coolant temperature indication $\geq$ 212°F.
(2) Raise RPV water level.

C. (1) Feedwater Sparger temperature $\geq$ 200°F on any Vessel Feedwater Nozzle indication.
(2) Raise RPV water level.

D. (1) Differential temperature of $\geq$ 40°F between Vessel Bottom Head and Vessel Head Flange indications.
(2) Maximize Reactor Water Cleanup operation.

Examination Outline Cross-reference:

295023 Refueling Acc / 8

AA1.03 (10CFR 55.41.7)

Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS:

- Fuel handling equipment

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295023AA1.03	
Importance Rating	3.3	------

Proposed Question: **# 10**

Unit 1 is in a Refueling Outage. The Refueling Supervisor reports that an **IRRADIATED** fuel assembly has been seated in the **WRONG** location in the core. The grapple remains engaged on the bundle.

The following conditions are then noted:

- Rising count rates on SRMs
- SRM Period lights illuminated
- Rising dose rates on the Refuel Floor

Which ONE of the following describes an **IMMEDIATE** Operator action to be taken?

A. Verify Secondary Containment is intact.

B. If any CRD Pump is in service stop the CRD Pump

C. If unexpected criticality is observed following control rod withdrawal, manually SCRAM the reactor.

D. Traverse the refueling bridge **AND** fuel assembly away from the reactor core, preferably to the area of the cattle chute.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This is a required subsequent action of 1-AOI-79-1, "Fuel Damage During Refueling."

B INCORRECT: This is plausible because it is a required subsequent action of 1-AOI-79-2, not immediate action.

C INCORRECT: This is an immediate action per 2-AOI-79-2, however, only appropriate to perform if ALL Control Rods cannot manually be inserted. NO indication given to suspect otherwise, therefore this action would be inappropriate to take.

D **CORRECT:** In order to answer this question correctly the candidate must determine the appropriate condition and Immediate Action required by 1-AOI-79-2.

Justification: This question satisfies the *KIA* statement by requiring the candidate to analyze specific plant conditions to determine appropriate actions to take with fuel handling equipment in response to inadvertent criticality. This question is rated as *CIA* due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-AOI-79-2 Rev. 0 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.060 V.B.3 (As available)

Question Source:			
	Bank #		
	Modified Bank #	BFN 0610 #49	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	BFN 2008	

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 1	Inadvertent Criticality During Incore Fuel Movements	1-AOI-79-2 Rev. 0000 Page 6 of 9

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** unexpected criticality is observed following control rod withdrawal, **THEN**

REINSERT the control rod. ☐

[2] **IF** all control rods can NOT be fully inserted, **THEN**

MANUALLY SCRAM the Reactor. ☐

[3] **IF** unexpected criticality is observed following the insertion of a fuel assembly, **THEN**

PERFORM the following:

[3.1] **VERIFY** fuel grapple latched onto the fuel assembly handle AND **IMMEDIATELY REMOVE** the fuel assembly from the Reactor core. ☐

[3.2] **IF** the Reactor can be determined to be subcritical **AND** no radiological hazard is apparent, **THEN**

PLACE the fuel assembly in a spent fuel storage pool location with the least possible number of surrounding fuel assemblies and **LEAVE** the fuel grapple latched to the fuel assembly handle. ☐

[3.3] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

TRAVERSE the Refueling Bridge and fuel assembly away from the Reactor core, preferably to the area of the cattle chute and **CONTINUE** at Step 4.1[4]. ☐

[4] **IF** the Reactor can NOT be determined to be subcritical **OR** adverse radiological conditions exist, **THEN**

EVACUATE the refuel floor. ☐

BFN Unit 1	Fuel Damage During Refueling	1-AOI-79-1 Rev. 0000 Page 6 of 9

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **STOP** all fuel handling.

[2] **EVACUATE** all non-essential personnel from Refuel Floor.

4.2 Subsequent Actions

CAUTION

The release of IODINE is of major concern. If gas bubbles are identified at any time, Iodine release should be assumed until RADCON determines otherwise.

[1] **VERIFY** Secondary Containment is intact. **REFER TO** Tech Spec 3.6.4.1. ☐

[2] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). ☐

[3] **VERIFY** automatic actions. ☐

[4] **NOTIFY** RADCON to perform the following:

- **EVALUATE** the radiation levels. ☐
- **MAKE** recommendation for personnel access. ☐
- **MONITOR** around the Reactor Building Equipment Hatch at levels below the Refuel Floor for possible spread of the release. ☐

[5] **REFER TO** EPIP-1 for proper notification. ☐

BFN Unit 1	Inadvertent Criticality During Incore Fuel Movements	1-AOI-79-2 Rev. 0000 Page 7 of 9

4.2 Subsequent Actions

[1] **NOTIFY** the Shift Manager and Reactor Engineer. ☐

[2] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOIs. ☐

[3] **VERIFY** all control rods are inserted. ☐

[4] **IF** criticality is still evident **AND** at the direction of the Unit Supervisor, **THEN**

PERFORM the following:

[4.1] **IF** the CRD pump is in operation, **THEN**

STOP the CRD pump. ☐

[4.2] **IF** the RWCU system is in service, **THEN**

ISOLATE RWCU as follows:

[4.2.1] **CLOSE** 1-FCV-069-0001 using RWCU INBD SUCT ISOLATION VALVE, 1-HS-69-1. ☐

[4.2.2] **CLOSE** 1-FCV-069-0002 using RWCU OUTBD SUCT ISOLATION VALVE, 1-HS-69-2A. ☐

[4.3] **IF** SLC is operable, **THEN**

UNLOCK and **PLACE** SLC PUMP 1A/1B, 1-HS-63-6A control switch in START A or START B. ☐

[5] **NOTIFY** RADCON to conduct surveys to determine radiation levels on Refuel Floor. ☐

[6] **NOTIFY** Chemistry to sample and analyze the Reactor water. ☐

[7] **REFER TO** EPIP-1 for proper notifications. ☐

[8] **NOTIFY** NRC. **REFER TO** SPP-3.5. ☐

[9] **NOTIFY** Plant Manager. ☐

0610 NRC RO EXAM

49. RO 295023AK1.02 001/C/A/T1G1/79-2/V.B.3.B/295023AK1.02//RO/SRO/MODIFIED 11/17/07

Fuel loading is in progress on Unit 1 when you notice an unexplained rise in Source Range Monitor (SRM) count rate and an indicated positive reactor period.

Which ONE of the following actions is an appropriate response?

A. Immediately EVACUATE all personnel from the refuel floor.

B. If unexpected criticality is observed following control rod withdrawal, manually SCRAM the reactor.

C.✔ If the reactor cannot be determined to be subcritical, traverse the refueling bridge and fuel assembly away from the reactor core, preferably to the area of the cattle chute.

D. If all rods are not inserted/cannot be inserted, verify the fuel grapple is latched onto the fuel assembly handle and immediately remove the fuel assembly from the reactor core.

K/A Statement:

295023 Refueling Acc Cooling Mode / 8
AK1.02 - Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS : Shutdown margin

K/A Justification: This question satisfies the K/A statement by requiring the candidate to analyze specific plant conditions to determine a reduction in Shutdown Margin has occurred and the actions required to address that condition.

References: 1-AOI-79-2

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

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MODIFIED FROM OPL171.060 #1

0610 NRC RO EXAM

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The appropriate condition and Immediate Action required by 1-AOI-79-2.

C - correct:

A - incorrect: This is plausible because the evacuation of the Refuel Floor MAY be directed, but other actions to mitigate the problem take precedence until personnel safety is compromised.

B - incorrect: This is plausible because the condition is correct, but the action to scram is incorrect. Reinserting the control rod is required.

D - incorrect: This is plausible because the required action is correct, but the condition is NOT correct. This action is based on unexplained criticality following insertion of a fuel assembly.

Examination Outline Cross-reference:

295024 High Drywell Pressure / 5

EK1.01 (10CFR 55.41.9)

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE:

- Drywell integrity: Plant-Specific

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295024EK1.01	
Importance Rating	4.1	------

Proposed Question: **# 11**

Which ONE of the following completes the statement?

The MAXIMUM Containment Pressure that Primary Containment Vent Valves can be opened **AND** closed is______.

A. 50 psig.

B. 55 psig.

C. 62 psig.

D. 65 psig.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausibility based on recognizable value - Calculated maximum pressure after blowdown (no pre-purge) Drywell is 49.6 psig (PUR 50.6 psig)

B **CORRECT**: Primary Containment Pressure Limit is defined to be the lesser of either: pressure capability of the containment, (~100 psig) or maximum containment pressure at which vent valves can be opened and closed to reject all decay heat from primary containment (55 psig), or maximum primary containment pressure at which MSRVs can be opened and will remain open (65 psig), or maximum containment pressure at which vent valves can be opened and closed to vent the RPV.

C INCORRECT: Plausibility based on recognizable value - The Drywell is ASME rated 62 psig, 110% of design pressure of 56 psig.

D INCORRECT**:** Plausibility based on recognizable value maximum primary containment pressure at which MSRVs can be opened and will remain open (65 psig)

RO Level Justification: To successfully answer this question, candidate must have knowledge of the effect of High Drywell Pressure on Drywell Integrity.

Technical Reference(s): OPL171.016 R. 17 / OPL171.203 R. 7 (Attach if not previously provided)

EOI Program Manual 0-V-D Rev 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.016 V.B.2 (As available)

Question Source: Bank #

Modified Bank # Bruns 08 #50 (Note changes or attach parent)

New

Question History: Last NRC Exam Brunswick 2008

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

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a. If primary containment pressure cannot be maintained below Pressure Suppression Pressure, emergency depressurization of the RPV is required to terminate, or reduce as much as possible, any continued primary containment pressure increase. The operator continues at Step PC/P-11.

2. Step PC/P-11

a. This signal step informs the operator that actions to control RPV pressure must immediately change because of present plant conditions. — Procedure Use, change in strategy

b. When emergency RPV depressurization is required, the operator shall transfer RPV pressure control actions to C2, Emergency RPV Depressurization.

3. Step PC/P-12

a. Calculations have determined that primary containment integrity can be assured, and core damage (that might be caused by inability to vent the RPV to permit injection into the RPV) is prevented when suppression chamber pressure is maintained below 55 psig, Primary Containment Pressure Limit.

b. Primary Containment Pressure Limit is defined to be the lesser of either:

(1) pressure capability of the containment, or — Around 100#

(2) maximum containment pressure at which vent valves can be opened and closed to reject all decay heat from primary containment, or — 55 #
Obj V.B.8/ V.C.8

(3) maximum primary containment pressure at which MSRVs can be opened and will remain open, or — 65#

OPL171.016
Revision 17

c. Design Specifications — Obj. V.B.2, Obj. V.C.2

(1) Drywell internal design pressure
+56 PSIG(ASME rated 62 psig, 110% of design)
-2 PSIG

B. Hardened Wetwell Vent System

During emergency operation, to vent the Suppression Chamber with Suppression Chamber Pressure above 55 psig it is necessary to utilize the hardened wetwell vent system in accordance with the EOI's. This system will provide assurance of pressure relief through a path with scrubbing of fission products and an exhaust line to the stack.

e. Pressure in the drywell reaches a peak of 49.6 psig approximately 10 seconds after the accident, (Point B - Figure 8) after which the rate of blowdown is decreasing and steam condenses faster than the reactor is blowing down. — Point B - TP-8 (encircled)

Excerpt from EOI Program Manual 0-V-D

The maximum containment pressure that primary containment vent valves can be opened and closed is <A.61>.

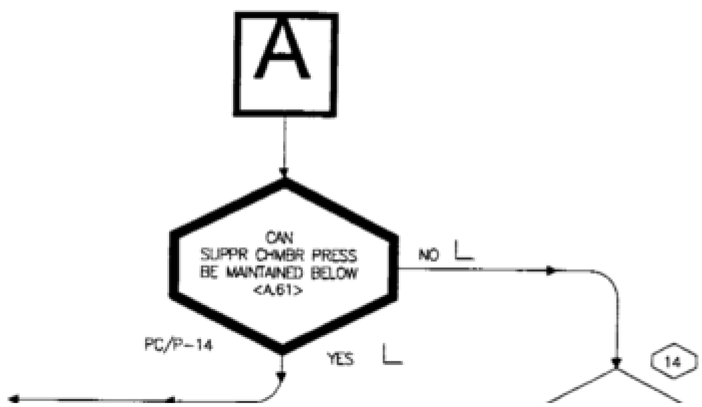
DISCUSSION: STEP PC/P-13

This action step directs the operator is to use all available containment pressure control systems to maintain suppression chamber pressure below Primary Containment Pressure Limit.

Engineering calculations have determined that primary containment integrity can be assured, and core damage (that might be caused by inability to vent the RPV to permit injection into the RPV) is prevented when suppression chamber pressure is maintained below <A.61>, Primary Containment Pressure Limit. This value is rounded off in the EOI to use the closest, most conservative value that can be accurately determined on available instrumentation.

Primary Containment Pressure Limit is defined to be the lesser of either: 1) pressure capability of the containment, or 2) maximum containment pressure at which vent valves can be opened and closed to reject all decay heat from primary containment, or 3) maximum primary containment pressure at which MSRVs can be opened and will remain open, or 4) maximum containment pressure at which vent valves can be opened and closed to vent the RPV. At BFN, this limit is a function of primary containment vent valve operability (item 2). The maximum containment pressure that primary containment vent valves can be opened and closed is <A.61>.

The operator is then directed to follow the branch designator, A, to Step PC/P-14.



Brunswick 2008 #50

50. Which one of the following is the primary containment pressure limit and the required action before this limit is reached in accordance with PCCP?

A. 62 psig
Vent primary containment irrespective of offsite release rate

B. 62 psig
Vent primary containment only if offsite release rate do not exceed ODCM limits

C✓ 70 psig
Vent primary containment irrespective of offsite release rate

D. 70 psig
Vent primary containment only if offsite release rate do not exceed ODCM limits

REFERENCE:
SD-04 page 7/8

EXPLANATION: The calculated peak containment pressure is 49.4 psig which is increased by 25% to establish the drywell design pressure of 62 psig. the PCPL-A graph shows this limit is 70 psig.

CHOICE "A" Incorrect. PCPL-A graph shows this limit is 70 psig.

CHOICE "B" Incorrect. PCPL-A graph shows this limit is 70 psig and venting irrespective is required.

CHOICE "C" Correct answer.

CHOICE "D" Incorrect. when PCPL-A is exceeded venting irrespective is required.
295024 High Drywell Pressure

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : (CFR: 41.8 to 41.10)

EK1.01 Drywell integrity: Plant-Specific.................... 4.1 / 4.2*

SOURCE: new

LESSON PLAN/OBJECTIVE:
CLS-LP-04, Obj. 2. State the PC design bases, including temperature and pressure limits for the DE and Suppression Chamber, as given in the FSAR.

COG LEVEL: low

Examination Outline Cross-reference:	Level	RO	SRO
295025 High Reactor Pressure / 3	Tier #	1	------
EK3.01 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE:	K/A #	295025EK3.01	
• Safety/relief valve opening	Importance Rating	4.2	------

Proposed Question: **# 12**

Which ONE of the following completes the statements?

The Limiting Condition for Operation (LCO) statement for Unit 1 Safety Relief Valves (SRVs), LCO 3.4.3, requires operability of the safety function for __**(1)**__ SRVs in Modes 1, 2, and 3.

This ensures that the ASME Code Limit of __**(2)**__ is **NOT** exceeded following the Design Basis Event of simultaneous closure of **ALL** Main Steam Isolation Valves (MSIVs) at 100% Reactor Power.

A. **(1)** 12
(2) 1250 psig

B. **(1)** 12
(2) 1375 psig

C. **(1)** 13
(2) 1375 psig

D. **(1)** 13
(2) 1250 psig

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The first part is correct – See 'B' below. Second part is incorrect in that 1250 psig is Vessel Design Pressure versus the ASME Code Limit of 110% of Vessel Design Pressure, or 1375 psig.

B **CORRECT:** (See attached excerpts) Tech Spec LCO 3.4.3 requires only the safety function of 12 SRVs to be operable. The Bases Section on Applicable Safety Analyses discusses the reason why in relation to the ASME Code Limit.

C INCORRECT: The first part is incorrect in that the LCO does NOT require operability of ALL 13 SRVs, ONLY 12 SRVs. 13 is plausible in that most safety systems are required, as a whole, to be operable. The second part is correct.

D INCORRECT: Both parts are incorrect – See 'C' for first part and 'A' for second part.

RO Level Justification: Tests the candidate's knowledge of Technical Specifications and Bases for SRV operability. Embedded in the Tech Spec Bases is the reason for SRVs opening in relative to transient analyses for the BFN Facility.

Technical Reference(s): Unit 1 Tech Spec 3.4.3, Amendment 234 (Attach if not previously provided)
Unit 1 Tech Spec 3.4.3 Bases, Rev. 0
OPL171.009, Rev. 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _V.B.2 / 14.e _________ (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

LCO 3.4.3 The safety function of 12 S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 4.	12 hours 36 hours

BASES (continued)

ACTIONS

A.1 and A.2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of one or more required S/RVs is inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the 12 required S/RVs open at the pressures assumed in the safety analysis of Reference 1. The setpoint groups for all 13 S/RVs are listed. The demonstration of the S/RV safe lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint tolerance is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

(6) The worst over pressure transient is:

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

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

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

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

(1) There are 13 Safety / Relief valves.

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

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

Obj. V.B.6
Obj. V.C.4

BASES (continued)

APPLICABLE SAFETY ANALYSES	The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 12 S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event. Reference 2 discusses additional events that are expected to actuate the S/RVs. From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. S/RVs satisfy Criterion 3 of the NRC Policy Statement (Ref. 4).
LCO	The safety function of 12 S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2). The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function). The S/RV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the

(continued)

3. Safety / Relief Valves

a. Design basis

(1). The safety / relief valves are designed to prevent over pressurizing the nuclear steam supply system to prevent failure of the nuclear system process barrier due to high Rx Vessel pressure.

(2) Design pressure is 1250 psig to protect the Rx Vessel integrity.

(3) ASME code allows a transient over pressure condition of 10%.

1250 psig + 10% (1250) =
1250 psig + 125 psig = 1375 psig

(4) The highest pressure in the primary system will be at the lowest elevation due to system pressure + the static water head. The highest pressure point will occur at the bottom of the vessel. Because the pressure is not monitored at this point; it cannot be directly determined if this design pressure has been exceeded.

(5) To ensure adequate margin to the vessel design pressure limit, the Tech Spec Safety Limit for Rx pressure is 1325 psig steam dome pressure. Tech Spec 2.0

Examination Outline Cross-reference:	Level	RO	SRO
295026 Suppression Pool High Water Temp. / 5	Tier #	1	------
EA2.01 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:	K/A #	295026EA2.01	
• Suppression pool water temperature	Importance Rating	4.1	------

Proposed Question: **# 13**

Unit 3 is currently testing Safety Relief Valves (SRVs) at 10% Reactor Power with the following plant conditions:

- Suppression Pool Single Element Temperature is 112 ºF
- Suppression Pool Bulk (Average)Temperature is 96 ºF

Which ONE of the following describes the required action relative to Suppression Pool Temperature?

A. Suspend testing of the SRVs **IMMEDIATELY** in accordance with Tech Specs.

B. Testing can continue **UNTIL** bulk temperature exceeds 110 ºF in accordance with Tech Specs.

C. Operate **ALL** available Suppression Pool Cooling as directed by EOI-2 “Primary Containment Control.”

D. Enter EOI-1 “RPV Control” from EOI-2 “Primary Containment Control” **AND** SCRAM the reactor as directed by the EOIs.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Bulk temp above 105 ºF would require testing to be suspended. However the bulk temp is the determining factor and not a single element.

B INCORRECT: Bulk temp above 110 ºF would require the Mode Switch be placed in Shutdown IMMEDIATELY in accordance with Tech Specs. Testing would have to be stopped IMMEDIATELY upon exceeding 105 ºF.

C **CORRECT:** Bulk temperature is used and not a single point. (See attached excerpts) Entry into EOI-2 is warranted at 95 ºF and it dictates that if Suppression Pool Temperature cannot be maintained below this value, ALL available cooling is operated when adequate core cooling is assured.

D INCORRECT: Bulk Temp above 110 ºF would require entry into EOI 1 for reactor scram. However only a single point is above 110 ºF and not bulk, or average, temperature.

RO Level Justification: Tests the candidate's knowledge of High Suppression Pool Temperature. Determination of appropriate temperature indication to use and the required actions for EOI strategies is required for the operator. The use of LCO limits and the actions directed by the EOIs make this a discriminating question for an operator. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-EOI-2, Rev. 8 (Attach if not previously provided)

EOIPM Section 0-V-D, Rev. 0

Unit 3 Tech Spec 3.6.2.1 Amm 212

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _V.B.10 / 11 ________ (As available)

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

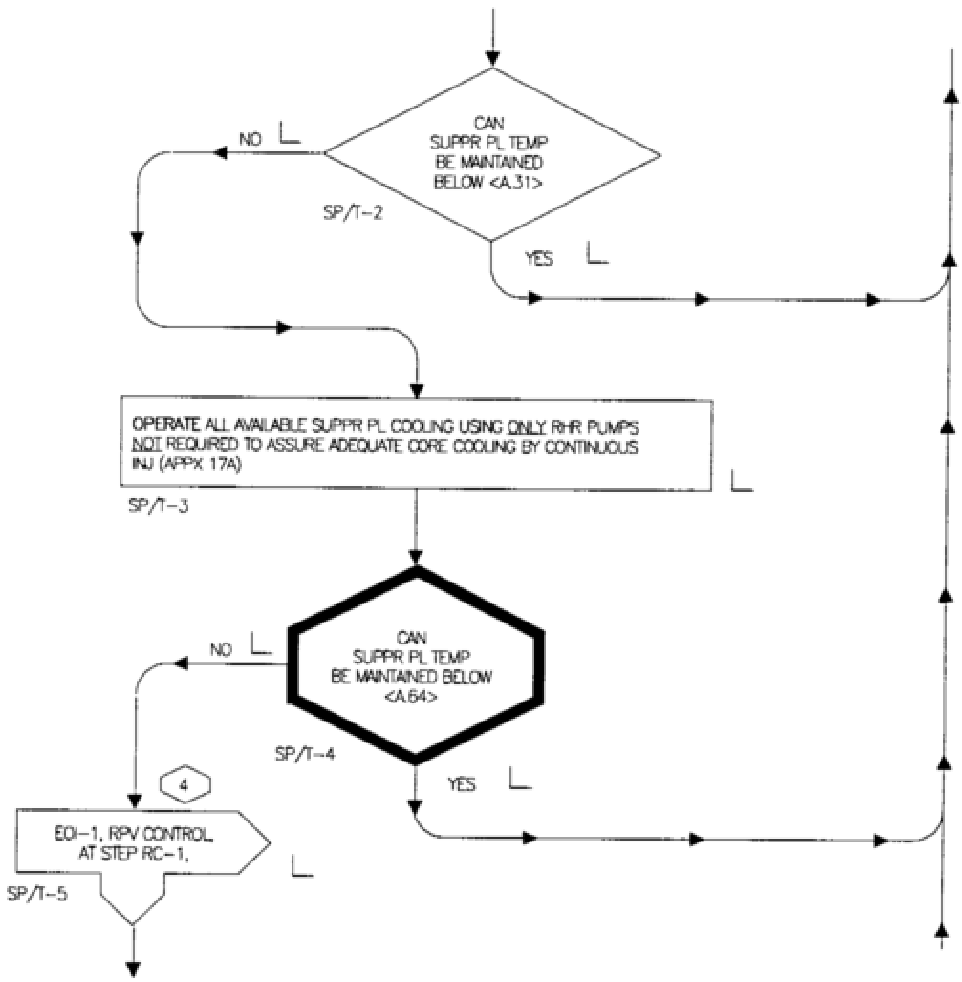
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

STEP: SP/T-3

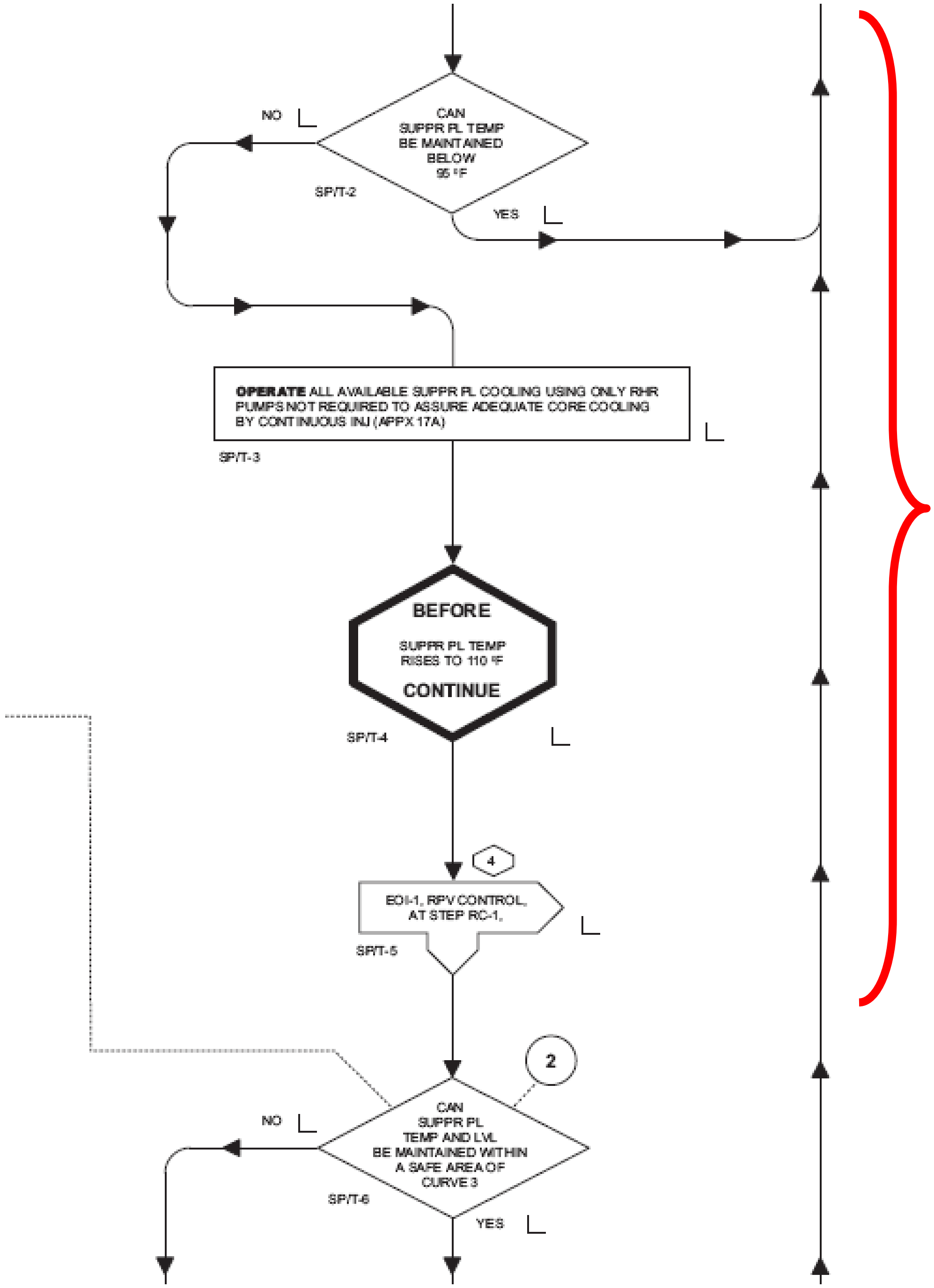

CAN SUPPR PL TEMP BE MAINTAINED BELOW <A.31>
NO
YES
SP/T-2
OPERATE ALL AVAILABLE SUPPR PL COOLING USING ONLY RHR PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING BY CONTINUOUS INJ (APPX 17A)
SP/T-3
CAN SUPPR PL TEMP BE MAINTAINED BELOW <A.64>
NO
YES
SP/T-4
4
EOI-1, RPV CONTROL, AT STEP RC-1.
SP/T-5

DISCUSSION: STEP SP/T-3

This action step directs the operator to manually place all available RHR pumps, not required in LPCI mode, in suppression pool cooling mode. EOI Appendix 17A provides step-by-step guidance for operating RHR in suppression pool cooling.

Since this step is reached only when suppression pool temperature cannot be maintained below <A.31> (a conclusion that may be reached in advance of suppression pool temperature actually reaching this value), explicit instruction is given to operate all available methods of suppression pool cooling.

Maintaining adequate core cooling takes precedence over maintaining suppression pool temperature below the LCO value since catastrophic failure of primary containment is not expected to occur at this suppression pool temperature. In addition, further action is still available for reversing the increasing suppression pool temperature trend. Therefore, only if continuous operation of RHR pump in LPCI mode is not required to assure adequate core cooling, is it permissible to use that pump for suppression pool cooling. This step, however, does permit alternating use of RHR pumps between LPCI injection and suppression pool cooling modes, as the need for each occurs and as long as adequate core cooling can be maintained.



3-EOI-2	PAGE 1 OF 1
PRIMARY CONTAINMENT CONTROL	
UNIT 3 BROWNS FERRY NUCLEAR PLANT	
REV: 8	

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;

b. ≤ 105°F when any OPERABLE IRM channel is > 70/125 divisions of full scale on Range 7 and testing that adds heat to the suppression pool is being performed; and

c. ≤ 110°F when all OPERABLE IRM channels are ≤ 70/125 divisions of full scale on Range 7.

APPLICABILITY: MODES 1, 2, and 3.

Suppression Pool Average Temperature
3.6.2.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Suppression pool average temperature > 105°F. AND Any OPERABLE IRM channel > 70/125 divisions of full scale on Range 7. AND Performing testing that adds heat to the suppression pool.	C.1 Suspend all testing that adds heat to the suppression pool.	Immediately
D. Suppression pool average temperature > 110°F but ≤ 120°F.	D.1 Place the reactor mode switch in the shutdown position. AND D.2 Verify suppression pool average temperature ≤ 120°F. AND D.3 Be in MODE 4.	Immediately Once per 30 minutes 36 hours

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
295028 High Drywell Temperature / 5	Tier #	1	------
EK2.01 (10CFR 55.41.5)	Group #	1	------
Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: • Drywell spray: Mark-I&II	K/A #	295028EK2.01	
	Importance Rating	3.7	------

Proposed Question: **# 14**

Unit 1 has scrammed from 100% Reactor Power with the following conditions:

- A small, high pressure steam leak in the Drywell has caused a Drywell Temperature of 280 ºF and rising
- Drywell Pressure is 15 psig and rising

Which ONE of the following completes the statement?

IMMEDIATELY following the initiation of Drywell Sprays under the superheated conditions listed above, the Drywell is expected to have a **__(1)__** pressure drop indicative of **__(2)__** cooling being the dominant mechanism of heat transfer.

A. **(1)** slow, steady
(2) convective

B. **(1)** slow, steady
(2) evaporative

C. **(1)** rapid, large
(2) convective

D. **(1)** rapid, large
(2) evaporative

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This is incorrect because it is based upon initiating Drywell Sprays with saturation conditions present in containment. At some point during the Drywell pressure drop, conditions will result in a transition *to* convective cooling as the dominant mechanism; but, not immediately following initiation of sprays.

B INCORRECT: First part incorrect – See 'A' above. Second part is correct in relation to the stem, but not in relation to the first part of the question.

C INCORRECT: First part is correct in relation to the stem, but not in relation to the first part of the question.. Second part is incorrect – See 'A' above.

D **CORRECT:** (See attached excerpt) Small, high pressure steam leaks in the Drywell result in high temperatures with superheated conditions. Initiating Drywell Sprays under these conditions results in an immediate, rapid, large pressure reduction due to evaporative cooling per the EOI Program Manual Bases. At some point during the Drywell pressure drop, conditions will reach saturation and result in a transition to convective cooling, but not immediately following initiation of sprays.

RO Level Justification: Tests the candidate's knowledge of the relationship of Drywell Temperature to the initiation of Drywell Sprays and the resulting effects. It also requires overall knowledge of the environmental effects of sprays on containment in relation to superheated vs. saturated conditions. This question is rated as Memory or Fundamental Knowledge based upon the requirements to recall technical information.

Technical Reference(s): EOI Program Man. Sect. 0-V-D, Rev. 0 (Attach if not previously provided)
OPL171.201, Rev. 7
OPL171.203, Rev. 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _OPL171.201 V.B.12__ _ (As available)
_OPL171.203 V.B.4 / 13__

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

DISCUSSION: STEP PCC-1

This retainment override step applies throughout performance of EOI-2, Primary Containment Control. Whenever any of the conditions in this retainment override step are met, the operator is directed to perform the specified actions. Any one of the two retainment override statements require specific primary containment control actions.

The first retainment override statement directs the operator to terminate suppression chamber sprays, if operating, when suppression chamber pressure reaches the high drywell pressure scram setpoint.

The second retainment override statement directs the operator to terminate drywell sprays, if operating, when drywell pressure reaches the high drywell pressure scram setpoint.

The high drywell pressure scram setpoint is <A.22>. This value has been rounded off in the EOI to use the closest, most conservative value that can be accurately determined from available control room indication.

Drywell and suppression chamber spray operation reduces primary containment pressure and temperature through combined effects of evaporative and convective cooling. During evaporative cooling, water spray undergoes a change of state, liquid to vapor, whereas convective cooling involves no change of state.

Evaporative cooling occurs when water is sprayed into a superheated atmosphere. Water at the surface of each droplet is heated and flashes to steam, absorbing heat energy from the atmosphere until the atmosphere reaches saturated conditions. In the drywell, with typical drywell spray flowrate, the evaporative cooling process results in an immediate, rapid, large reduction in pressure. This pressure reduction occurs at a rate much faster than can be compensated for by the primary containment vacuum relief system. Unrestricted operation of drywell or suppression chamber sprays could cause an excessive negative differential pressure to occur between the drywell and suppression chamber, large enough to cause a loss of primary containment integrity.

Convective cooling occurs when water is sprayed into a saturated atmosphere. Sprayed water droplets absorb heat from the surrounding atmosphere through convective heat transfer (sensible heat from the atmosphere is transferred to the water droplets). This effect reduces ambient temperature and pressure until equilibrium conditions are established. The convective cooling process occurs at a rate much slower than the evaporative cooling process. An operator can effectively control the magnitude of a containment temperature/pressure reduction from convective cooling by terminating operation of drywell or suppression chamber sprays.

	INSTRUCTOR NOTES
a. This decision step has the operator evaluate the present status of drywell pressure and drywell temperature to determine if conditions are favorable for drywell spray operation. Because this step is prioritized with the miniature before decision step DW/T-6 symbol, this evaluation must be made before drywell temperature reaches 280°F, drywell design temperature.	
b. Drywell spray operation reduces drywell pressure and temperature through combined effects of evaporative and convective cooling. During evaporative cooling, water spray undergoes a change of state, liquid to vapor, whereas convective cooling involves no change of state.	
c. Evaporative cooling occurs when water is sprayed into a superheated atmosphere. Water at the surface of each droplet is heated and flashes to steam, absorbing heat energy from the drywell atmosphere until the atmosphere reaches saturated conditions.	SER 03-05 Questioning Attitude
d. In the drywell, with typical drywell spray flow rate, the evaporative cooling process results in an immediate, rapid, large reduction in pressure. This pressure reduction occurs at a rate much faster than can be compensated for by the primary containment vacuum relief system.	
e. Unrestricted operation of drywell sprays could cause an excessive negative differential pressure to occur between the drywell and suppression chamber, large enough to cause a loss of primary containment integrity.	Situational Awareness

		INSTRUCTOR NOTES
	f. Convective cooling occurs when water is sprayed into a saturated atmosphere. Sprayed water droplets absorb heat from the surrounding atmosphere through convective heat transfer (sensible heat from the atmosphere is transferred to the water droplets). This effect reduces drywell ambient temperature and pressure until equilibrium conditions are established.	
	g. The convective cooling process occurs at a rate much slower than the evaporative cooling process. An operator can effectively control the magnitude of a containment temperature/pressure reduction from convective cooling by terminating operation of drywell sprays.	
	h. Considering the pressure drop concerns described above, engineering calculations have determined that primary containment integrity is assured when drywell sprays are operated in the safe area of Drywell Spray Initiation Limit Curve (Curve 5).	See OPL171.201 for Curve 5 basis
	i. If drywell temperature and pressure are within the safe area of Curve 5, the operator continues at Step DW/T-9. If drywell temperature and pressure are not within the safe area of Curve 5, then drywell spray operation is not permitted and the operator is directed to emergency depressurize the RPV at Step DW/T-12.	
12.	Step DW/T-9	
	a. This action step directs the operator to stop operation of recirculation pumps and drywell blowers. Because this step is prioritized with the miniature before decision step DW/T-6 symbol, this action must be performed before drywell temperature reaches 280°F, drywell design temperature.	

	INSTRUCTOR NOTES
• Drywell pressure dropping below the high drywell pressure scram setpoint (2.45 psig).	
a. The purpose of the Drywell Spray Initiation Limit is to preclude containment failure due to excessive differential pressures between the suppression chamber and the drywell, and to prevent inadvertently de-inerting the containment.	
b. Drywell spray operation effects a drywell pressure and temperature reduction through the combined effects of evaporative cooling and convective cooling.	
(1) Evaporative cooling occurs when water is sprayed into a superheated atmosphere. The water at the surface of each droplet is heated and flashed to steam until the surrounding atmosphere saturates, absorbing heat energy from the atmosphere.	
(2) In the drywell, this cooling process results in an immediate, rapid, large reduction in pressure which occurs at a rate much faster than can be compensated for by the suppression chamber to drywell and reactor building to suppression chamber vacuum breakers. Unrestricted operation of drywell sprays could cause an excessive negative differential pressure to occur between the drywell and the suppression chamber, large enough to possibly cause a loss of primary containment integrity.	
(3) Convective cooling occurs when water is sprayed into a saturated atmosphere. The sprayed water droplets absorb heat from the surrounding atmosphere through convective heat transfer, reducing drywell ambient temperature and pressure until equilibrium conditions are established.	

Examination Outline Cross-reference:

295030 Low Suppression Pool Wtr Lvl / 5

EK1.02 (10CFR 55.41.8)

Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL:

- Pump NPSH

Level	RO	SRO
Tier #	1	------
Group #	1	------
K/A #	295030EK1.02	
Importance Rating	3.5	------

Proposed Question: **# 15**

Given the following Unit 2 conditions:

- Drywell Pressure is stable at 6.2 psig
- Suppression Chamber Pressure is stable at 5 psig
- RHR Pump 2A is currently operating at the Net Positive Suction Head (NPSH) limit
- A large, unisolable leak occurs just upstream of RHR PUMP 2B SUPPR SUCT VLV, 2-FCV-74-24

Which ONE of the following completes the statement?

As a result of the leak, the EOI-related NPSH limit for RHR Pump 2A becomes __**(1)**__; making cavitation __**(2)**__ likely for a given Suppression Pool Temperature and RHR Pump flow rate.

A. **(1)** more restrictive;
(2) less

B. **(1)** more restrictive;
(2) more

C. **(1)** less restrictive;
(2) less

D. **(1)** less restrictive;
(2) more

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part correct (see ‘B’ below). Second part incorrect because cavitation becomes *more* likely for a lower torus level (pressure). Plausible if the applicant assumes that the more restrictive limit somehow prevents cavitation.

B **CORRECT:** (See attached excerpts) To accommodate suppression pool water levels above the minimum LCO water level, suppression chamber airspace pressure is expressed as “overpressure” in the NPSH Limit. “Overpressure” is the sum of suppression chamber pressure and the hydrostatic head of water above the minimum LCO water level and must be determined by the operator when using the NPSH Limit. As “pressure” lowers, cavitation becomes more likely.

C INCORRECT: First part incorrect because the limit becomes *more* restrictive as torus level (pressure) lowers. Also incorrect because cavitation becomes more likely as the limit becomes more restrictive. Plausible if applicant thinks that a less restrictive curve moves the pump further away from cavitation.

D INCORRECT: First part incorrect because the limit becomes *more* restrictive as torus level (pressure) lowers. Second part correct. Plausible if applicant thinks that the RHR pump operating point changes when torus pressure changes, i.e., flow changes.

RO Level Justification: Tests the candidate's knowledge of NPSH theory and operational implications associated with a lowering Suppression Pool Level; which equates to pressure at the suction of the pump. "Overpressure" is the sum of suppression chamber pressure and the hydrostatic head of water above the minimum LCO water level and must be determined by the operator when using the NPSH Limit. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.201, Rev. 7 (Attach if not previously provided)
EOI Program Man. Sect. 2-IV, Rev. 10
EOI Program Man. Sect. 2-VI-N, Rev. 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _V.B.11 / 12__________ (As available)

Question Source: Bank # Hatch 07 NRC Q#56
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Hatch 2007-2008

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BANK QUESTION FROM HATCH NRC EXAM…

56. 295026EK2.03 015

An RHR pump is currently operating at the net positive suction head (NPSH) limit.

Which ONE of the following describes how torus pressure and torus temperature impact RHR pump operation?

As torus pressure decreases, the EOP NPSH limit becomes __________ making cavitation ______ likely for a given torus temperature and pump flow rate.

A. less restrictive/ less

B. less restrictive / more

C. more restrictive / less

D✓ more restrictive / more

A. Incorrect because as the limit becomes more restrictive as torus pressure lowers. Also incorrect because cavitation becomes more likely as the limit becomes more restrictive. Plausible if applicant thinks that the RHR pump operating point changes when torus pressure changes, i.e., flow changes.

B. Incorrect because as the limit becomes more restrictive as torus pressure lowers. Plausible if applicant thinks that a less restrictive curve moves the pump further away from cavitation.

C. Incorrect because cavitation becomes more likely for a lower torus pressure. Plausible if the applicant assumes that the more restrictive limit somehow prevents cavitation.

D. Correct.

CURVE 2

RHR NPSH LIMITS

Refer to EOI Program Manual, Section 2-VI-N, NPSH Limit Worksheet 15.

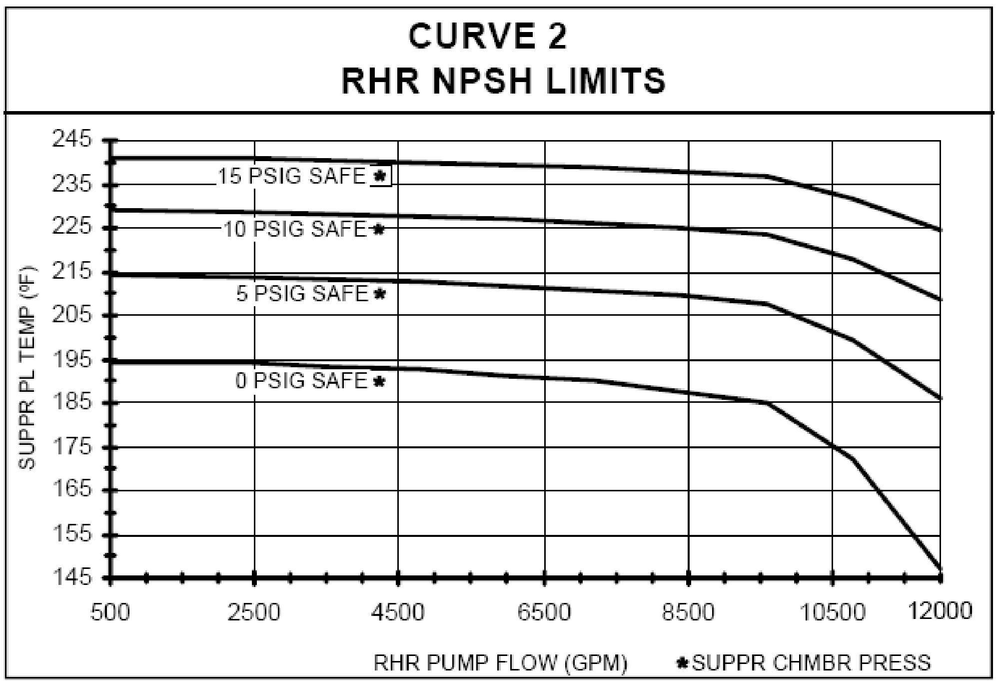
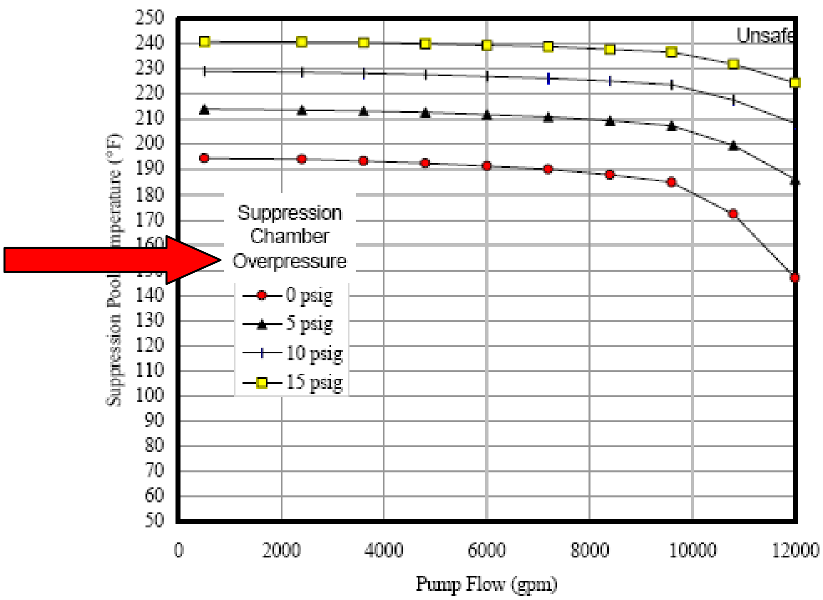


Figure 2 – RHR NPSH Limit



			INSTRUCTOR NOTES
	m.	There are two items to note concerning this table: 1) Except for the Shutdown Floodup instrument, all drywell temperatures are not applicable, because there is very little vertical pipe run in the drywell. This means very little error can be caused by elevated drywell temperatures (until boiling occurs), and 2) The "MAX SC RUN TEMP" is the highest temperature reading which can be obtained from Table 6, Secondary Containment Instrument Runs.	
6.	**Caution #2**		See EOI flow
	"Operation of RHR or CS with suction from the Suppr pl may result in equipment damage if:		charts
	• Pump flow is above the NPSH limit (curve 1 or 2)		TP-17/18
	OR		
	• Suppr Pl lvl is below the vortex limit (10 ft.)		
	a.	The NPSH Limit is reached when available NPSH (NPSHa) equals the NPSH required by the pump vendor (NPSHreq). For use in the EOIs, it is helpful to express the NPSH Limit in terms that are recognizable and measurable by the control room operator. Therefore, the NPSH Limit is calculated as a function of pump flow and suppression pool temperature for selected suppression chamber airspace pressures. To accommodate suppression pool water levels above the minimum LCO water level, suppression chamber airspace pressure is expressed as "overpressure" in the NPSH Limit. Overpressure is the sum of suppression chamber pressure and the hydrostatic head of water above the minimum LCO water level and must be determined by the operator when using the NPSH Limit.	SER 03-05

Examination Outline Cross-reference:

295031 Reactor Low Water Level / 2

EK1.03 (10CFR 55.41.10)

Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL

- Water level effects on reactor power

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295031EK1.03	
Importance Rating	3.7	------

Proposed Question: **# 16**

Which ONE of the following completes the statements?

In EOI C-5, "Level / Power Control," Reactor Water Level is lowered to less than (–) 50 inches to **__(1)__** thereby reducing reactor power. After Hot Shutdown Boron Weight is injected into the Reactor, level is raised in order to **__(2)__**.

A. **(1)** MAXIMIZE core inlet subcooling
(2) promote mixing of the Boron throughout the core.

B. **(1)** reduce natural circulation driving head **AND** core flow
(2) promote mixing of the Boron throughout the core.

C. **(1)** MAXIMIZE core inlet subcooling
(2) allow MSIVs to be reopened to commence a normal Cooldown.

D. **(1)** reduce natural circulation driving head **AND** core flow
(2) allow MSIVs to be reopened to commence a normal Cooldown.

Proposed Answer: **B**

Explanation (Optional):

A CORRECT: Part 1 incorrect – Plausible in that core inlet subcooling does play into the ATWS level control strategy. If the reactor is not shutdown, RPV water level must be controlled to minimize core inlet subcooling, thereby preventing or mitigating the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities. Part 2 correct – See Explanation B.

B **CORRECT**: Part 1 correct - 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. Part 2 correct - The Hot Shutdown Boron Weight (HSBW) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions. When an amount of boron sufficient to shutdown the reactor has been injected into the RPV, mixing is accomplished by raising RPV water level.

C INCORRECT: Part 1 incorrect – See Explanation A. Part 2 incorrect – Plausible in that after Cold Shutdown Boron has been injected, EOI-1 directs commencing plant cooldown. If the condenser is available, it would be preferred to cooldown to the condenser.

D INCORRECT: Part 1 correct – See Explanation B. Part 2 incorrect – See Explanation C.

Justification: Requires Knowledge of the operational implications of Water level effects on reactor power as it applies to REACTOR LOW WATER LEVEL to correctly answer.

Technical Reference(s): OPL171.205 Rev. 8 (Attach if not previously provided)

EOI Program Manual Sect. 0-V-K Rev. 0

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

Question Source: Bank #

Modified Bank # LaSalle 08 #41 (Note changes or attach parent)

New

Question History: Last NRC Exam LaSalle 2008

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

DISCUSSION: STEP C5-11

This action step directs the operator to deliberately lower RPV water level to effect a reduction in reactor power. EOI Appendix 4 provides step-by-step guidance for the most direct actions that will stop and prevent injection flow into the RPV. The injection system status table allows the operator to track the status of those significant systems used to control RPV water level.

Lowering RPV water level is accomplished by stopping and preventing all injection into the RPV, except from SLC and CRD, since these two systems are low flow, and may be needed to establish and maintain reactor subcriticality conditions. With essentially no makeup of reactor coolant, RPV water level then decreases by boil-off.

Power oscillations may occur when RPV water level is lowered significantly below the normal operating range with the reactor still at power. These oscillations have been analyzed, and determined to result in thermal transients well within the design capabilities of the fuel. The oscillations are discussed here to indicate to the operator that they are to be expected, and were considered in developing the steps that require deliberately lowering RPV water level with the reactor at power.

This step is reached only when previous attempts have so far been ineffective in stopping the suppression pool heatup. While this procedure is being utilized for RPV water level control, concurrent actions in EOI-1, RPV Control, have already directed the operator to reject as much heat as possible to the main condenser in RC/P Section, and insert control rods by alternate means and inject SLC in the RC/Q Section. Additionally, the concurrent action in EOI-2, Primary Containment Control, has directed the operator to maximize suppression pool cooling. Once these concurrent actions have been performed, the only remaining available mechanism for reactor power control is through the control of RPV water level.

The only additional action that remains available to mitigate the consequence of a failure-to-scram condition is deliberately lowering RPV water level to effect a reduction in reactor power. Lowering RPV water level reduces natural circulation driving head and core flow, resulting in a reduction of reactor power and the subsequent decrease in the rate of heat addition to the suppression pool. This process occurs as follows:

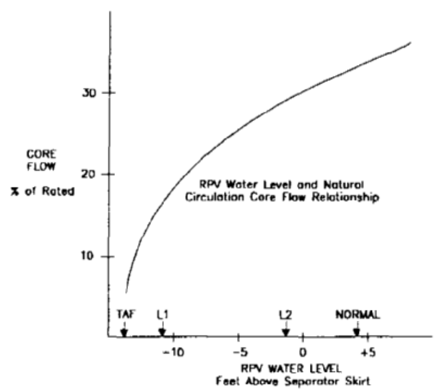
1. Following recirculation pump trips in the RC/Q section of EOI-1, RPV Control, the reactor is in a natural circulation mode. The natural circulation driving head is a function of the height of the fluid columns (RPV water level) and the fluid density differences between the regions inside and outside of the core shroud (void fraction directly affects the fluid density inside the core shroud).

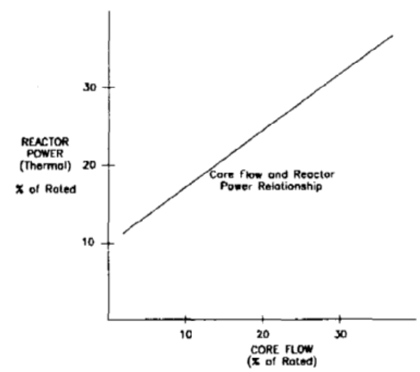
DISCUSSION: STEP C5-11 (Continued)

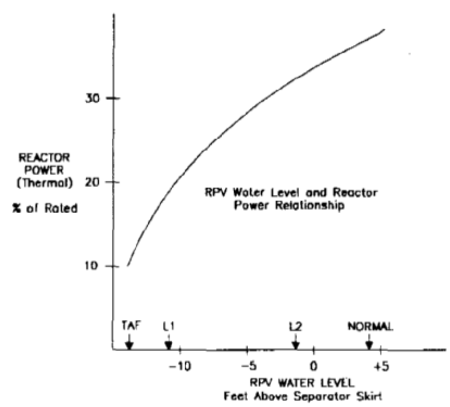
2. As RPV water level is lowered, the height of the fluid columns is reduced, thereby, reducing the natural circulation driving head.
3. As the natural circulation driving head is reduced, the natural circulation flow through the core is also reduced.
4. The reduced core flow results in a reduced rate 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 reactor power begins to decrease.
8. The reduced power results in a reduced steam generation rate.
9. The reduced steam generation rate results in a reduced void fraction.
10. When the void fraction drops to its original value (with some slight adjustment to account for reduced doppler reactivity), the reactor returns to criticality at a lower power level.

DISCUSSION: STEP C5-11 (Continued)

These interrelationships between RPV water level, natural circulation core flow, and reactor power have been observed in BWRs with RPV water level in or near the normal operating band. Computer analysis and scale model tests have confirmed the continued validity of these fundamental thermal hydraulics and reactor physics principles for RPV water levels at and below the elevation of the steam separators. The interrelationships between RPV water level, natural circulation core flow, and reactor power are graphically illustrated in the following figures:







DISCUSSION: STEP C5-23

This decision step has the operator evaluate SLC tank level to determine if the amount of boron injected into the RPV is sufficient to achieve reactor subcriticality.

Engineering calculations have determined that when contents of the SLC tank have been injected into the RPV to a SLC tank level of <A.72>, the reactor will be subcritical irrespective of control rod position, when RPV water level is raised to uniformly mix injected boron. An SLC tank level of <A.72> corresponds to Hot Shutdown Boron Weight. Hot Shutdown Boron Weight is defined to be least weight of soluble boron which, if injected into the RPV and uniformly mixed, will maintain the reactor shutdown under hot standby conditions.

While the RC/Q Section in EOI-1, RPV Control, provided direction to inject boron, this procedure has directed the operator to maintain RPV water level below the normal operating range to suppress reactor power by reducing natural circulation core flow. If RPV water level has been lowered to and maintained near <A.71>, Minimum Steam Cooling RPV Water Level, little if any natural circulation flow exists within the RPV. Three-dimensional scale model tests have been conducted that confirm that little boron mixing occurs under these conditions. Injected boron concentrates in the lower plenum region of the RPV, and does not contribute to reactor shutdown until in-core mixing is achieved. When an amount of boron sufficient to achieve reactor shutdown has been injected into the RPV, mixing is accomplished by raising RPV water level, and thereby increasing natural circulation flow through the vessel.

When contents of the SLC tank have been injected into the RPV to a SLC tank level of <A.72>, the operator is directed to continue at Step C5-24. If Hot Shutdown Boron Weight of boron has not yet been injected into the RPV, the operator returns to Step C5-14, effectively remaining in a loop until the proper amount of boron has been injected.

b. Unless mitigated, these conditions ultimately result in loss of NPSH for ECCS pumps taking suction on the suppression pool, containment over pressurization, and (ultimately) loss of primary containment integrity—which in turn could lead to a loss of adequate core cooling and uncontrolled release of radioactivity to the environment.

c. These conditions, combined with the inability to shut down the reactor through control rod insertion, dictate a requirement to promptly reduce reactor power since, as long as these conditions exist, suppression pool heat up will continue.

d. Reactor power must be reduced so that injection of the hot shutdown weight of boron can be completed before suppression pool temperature exceeds the Heat Capacity Temperature Limit.

7. Step C5-7

a. If the conditions of Step C5-6 exist, the operator is or has been directed to reject as much heat as possible from the RPV to the main condenser (RC/P-7 second and third overrides)

b. To place all available suppression pool cooling into operation (Step SP/T-3)

c. To trip the recirculation pumps (Step RC/Q-7), and to concurrently inject boron and manually insert control rods (Steps RC/Q-10 and RC/Q-21).

d. 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.

e. 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.

(2) Depressurizing the RPV is preferred over restoring RPV water level through the use of systems which inject inside the shroud because of the large reactor power excursions which may result from the in-shroud injection of large volumes of relatively cold and unborated water.

Up to this point, only outside shroud systems were listed

15. Step C5-17

Obj.V.B.6c

a. The Hot Shutdown Boron Weight (HSBW) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions.

67% SLC tank level

b. Since boron injects into the lower plenum, little boron mixing occurs under these conditions. 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.

c. When an amount of boron sufficient to shutdown the reactor has been injected into the RPV, mixing is accomplished by raising RPV water level, thereby raising natural circulation flow through the vessel.

16. Step C5-18

a. Maintaining RPV water level below the normal operating range suppresses reactor power by reducing natural circulation core flow. If RPV water level is lowered to and maintained near -180”, little if any natural circulation flow exists within the RPV.

b. Restoration of RPV water level to the normal control band is therefore delayed until the SLC Tank level reaches 67% which corresponds to the Hot Shutdown Boron Weight.

D. C5, Level/Power Control

1. Purpose of C5, Level/Power Control

a. The actions specified in C5 Level/Power Control effect control of RPV water level under conditions when it cannot be determined that reactor will remain subcritical under all conditions without boron.

b. The actions to control RPV water level in C5 differ from those in RC/L of EOI-1 to address four basic concerns:

(1) When boron is injected into the RPV, the systems used for control of RPV water level must be operated so as to minimize boron dilution and cold water injection, and to promote boron mixing.

(2) If the reactor is not shutdown, RPV water level must be controlled to minimize core inlet subcooling, thereby preventing or mitigating the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities.

(3) If the reactor is not shutdown and suppression pool temperature continues to rise, RPV water level must be controlled not only to adequately cool the core and minimize inlet subcooling, but also to minimize suppression pool heatup.

(4) Even if boron has not been injected into the RPV and the reactor is shutdown on control rods under hot conditions, injection of cold water could cause criticality, with no negative reactivity feedback occurring until reactor power rises to the heating range.

1

WHEN THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS (SEE NOTE)

OR

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

OR

THE RX IS SUBCRITICAL AND NO BORON HAS BEEN INJECTED INTO THE RPV,

THEN **CONTINUE**

RC/P-13

CAUTION

#3 ELEVATED SUPPR CHMBR PRESS MAY TRIP RCIC

#6 HPCI OR RCIC SUCTION TEMP ABOVE 140 ºF

DEPRESSURIZE THE RPV BELOW THE SHUTDOWN COOLING RPV PRESS INTERLOCK WITH ONE OR MORE OF THE FOLLOWING DEPRESSURIZATION SYSTEMS. **MAINTAIN** COOLDOWN RATE BELOW 100 ºF/HR.

DEPRESSURIZATION SYSTEM	APPX
MAIN TURB BYPASS VLVs IF THE MAIN CONDENSER IS AVAILABLE (USE APPX TO OPEN MSIVs)	8B
MSRVs ONLY WHEN SUPPR LVL IS ABOVE 5.5 FT IF .... "MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW" ANNUNCIATOR (XA-55-3D-18) IS IN ALARM, THEN ... **MINIMIZE** MSRV CYCLING BY USING SUSTAINED OPENING FOR DEPRESSURIZATION	11A
HPCI, WITH CST SUCTION IF POSSIBLE	11C
RCIC, WITH CST SUCTION IF POSSIBLE	11B
RFPTs ON MIN FLOW	11F
MAIN STEAM SYSTEM DRAINS	11D
STEAM SEALS	11G
SJAEs	11G
OFF GAS	11G
RWCU IF NO BORON HAS BEEN INJECTED	11E

RC/P-14

1-EOI-1	PAGE 1 OF 1
RPV CONTROL	
UNIT 1 BROWNS FERRY NUCLEAR PLANT	
REV: 0	

Question 41 LaSalle 07-01 NRC ILT Exam

Safety Function: 3.2 Reactor Water Inventory Control

System: EPE 295031 Reactor Low Water Level

K/A: EK1.03 - Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL : Water level effects on reactor power

I.R.: 3.7 / 4.1 10CFR 55 Content Part 41.10

License Level: RO Cognitive Level: Memory

Objective: 433.00.04 PRA: No

Question Exposure: 07-01 NRC ILT Exam

Source: New Question developed for the 07-01 NRC Exam

During the actions in the LGA-010, “Failure to Scram”, Level leg, reactor water level is lowered to –60 to –100 inches in order to ____(1)_____ and after hot shutdown boron weight is added level is raised in order to ______(2)______.

A. (1) void the core to minimize power generation and reduce heat rate to the suppression pool
(2) promote mixing of the boron throughout the core

B. (1) trip Recirculation Pumps to minimize power generation and reduce heat rate to the suppression pool
(2) allow MSIV’s to be reopened to commence a normal Cooldown

C. (1) promote concentrating boron in the high power areas of the core
(2) promote mixing of the boron throughout the core

D. (1) widen the pressure band for using SRV’s and RCIC to maintain reactor pressure band
(2) allow MSIV’s to be reopened to commence a normal Cooldown

---

Correct Answer A.: (1) void the core to minimize power generation and reduce heat rate to the suppression pool
(2) promote mixing of the boron throughout the core

Explanation: This answer summarizes information presented in the LGA-010 Lesson Plan when explaining the actions taken in the ‘Level’ leg of LGA-010, Failure to Scram.

Distracter B: (1) trip Recirculation Pumps to minimize power generation and reduce heat rate to the suppression pool
(2) allow MSIV’s to be reopened to commence a normal Cooldown

Distracter B is incorrect as RR pumps are tripped in the power leg in a controlled manner. Pressure is maintained steady in an ATWS condition

Distracter C: (1) promote concentrating boron in the high power areas of the core
(2) promote mixing of the boron throughout the core

Distracter C is incorrect as the LGA-010 lesson plan explains that injected boron concentrates in the lower plenum and does not contribute to reactor shutdown until mixing is accomplished by raising level.

Distracter D: (1) widen the pressure band for using SRV's and RCIC to maintain reactor pressure band
(2) allow MSIV's to be reopened to commence a normal Cooldown

Distracter D is incorrect as the level band is adjusted in LGA-001 when SRVs are being used for pressure control, and pressure is maintained steady in an ATWS condition.

References:
LGA-010 Lesson Plan, pages 1, 25, 29, and 35.

Examination Outline Cross-reference:	Level	RO	SRO
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	Tier #	1	------
EK2.13 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following:	K/A #	295037EK2.13	
• Alternate boron injection methods: Plant-Specific	Importance Rating	3.4	------

Proposed Question: **# 17**

Which ONE of the following completes the statements?

During Anticipated Transient Without Scram (ATWS) conditions on Unit 2, 2-EOI-1, “RPV Control” requires Standby Liquid Control (SLC) initiation **__(1)__**.

If the Squib Valves fail to actuate (fire) during execution of 2-EOI Appendix-3A, “SLC Injection,” Sodium Pentaborate can be injected with CRD Pump **__(2)__** as directed by 2-EOI Appendix-3B, “Alternate SLC Injection.”

A. **(1) BEFORE** Suppression Pool Temperature rises to 110 ^{o}F **ONLY**.
   (2) 1B

B. **(1) BEFORE** Suppression Pool Temperature rises to 110 ^{o}F **ONLY**.
   (2) 2A

C. **(1) BEFORE** Suppression Pool Temperature rises to 110 ^{o}F **OR WHEN** APRM Peak-to-Peak Oscillations persist above 25%.
   (2) 1B

D. **(1) BEFORE** Suppression Pool Temperature rises to 110 ^{o}F **OR WHEN** APRM Peak-to-Peak Oscillations persist above 25%.
   (2) 2A

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part incorrect – (See attached excerpts) Although SLC initiation is required for this condition, it is NOT the *ONLY* condition where SLC initiation is required. Second part is correct – as detailed in ‘C’ below.

B INCORRECT: First part incorrect – as detailed in ‘A’ above. Second part is incorrect – (See attached excerpts) Unit 2 does not use it’s respective CRD Pump for Alternate Boron Injection, unlike Units 1 and 3. This is a unit difference specific to BFN.

C **CORRECT:** First part correct – (See attached excerpts) – 2-EOI-1, RC/Q, Steps 9 and 10 both result, when conditions are met, in the requirement to inject Boron. One criterion is based on power oscillations and the other is based upon Suppression Pool Temperature. They could happen concurrently, or otherwise. Second part correct – (See attached excerpts) 2-EOI Appendix 3B specifies, for Unit 2, that Unit 1’s CRD Pump 1B is used when executing Alternate SLC Injection on Unit 2.

D INCORRECT: First part correct – as detailed in 'C' above. Second part incorrect – as detailed in 'B' above.

RO Level Justification: Tests the candidate's knowledge of ATWS conditions relative to EOIs requiring SLC initiation. Additionally, tests the candidate's knowledge of sources for Alternate Boron Injection that are unit specific; wherein a unit difference is tested. Compounding level of difficulty is the fact that Units 1 and 3 utilize Unit 2's SLC Tank with their installed respective pumps. Unit 2 utilizes Unit 1's SLC Tank and Unit 1's CRD Pump 1B for Alternate Boron Injection.

Technical Reference(s): 2-EOI-1, Rev. 12 (Attach if not previously provided)
OPL171.039, Rev. 16
2-EOI Appendix-3B, Rev. 5
3-EOI Appendix-3B, Rev. 4

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.039, V.B.6 (As available)
OPL171.202, V.B.15

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

Question History: Last NRC Exam

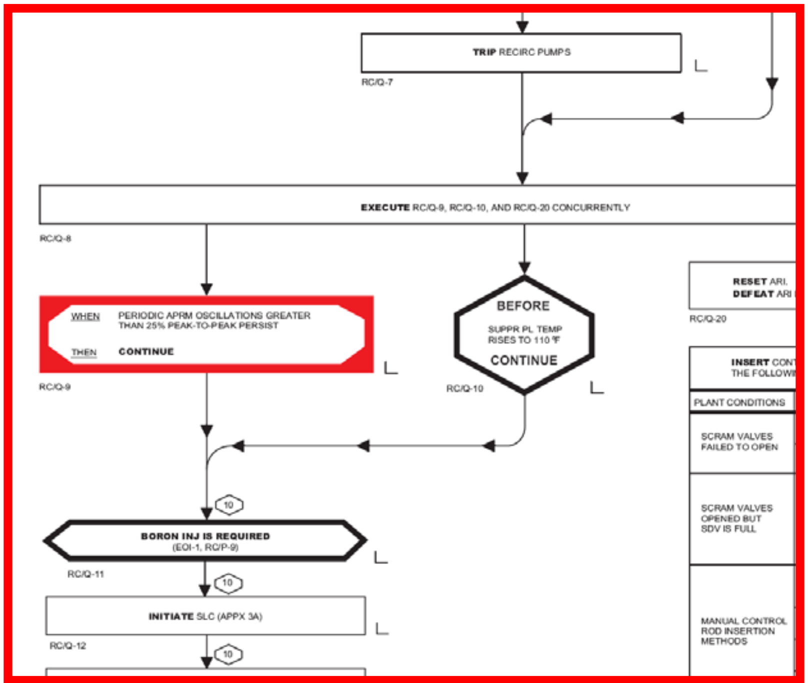
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

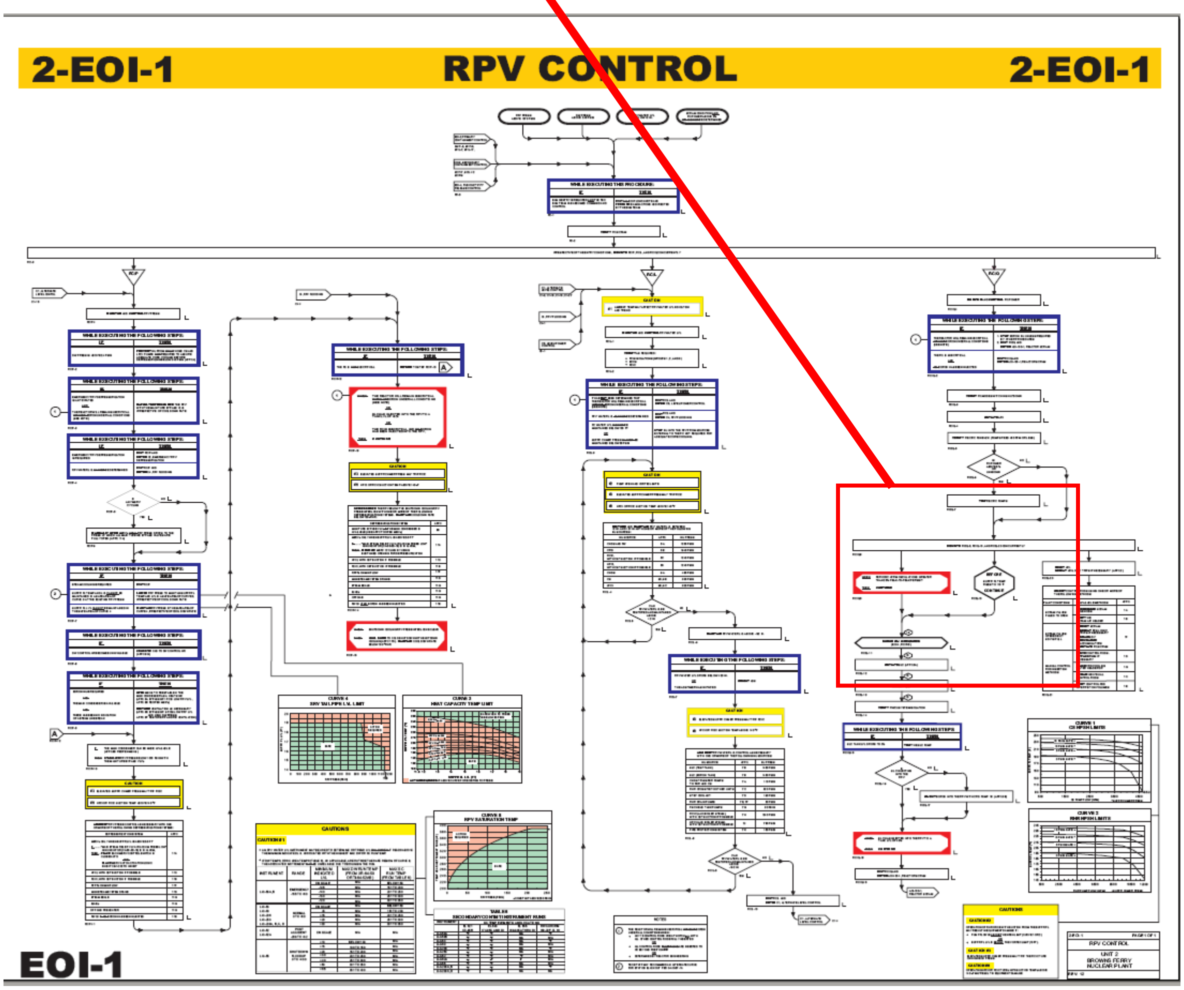
Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

Excerpt from 2-EOI-1, "RPV Control," RC/Q leg





	INSTRUCTOR NOTES
c) The SLC injection should take place no later than two hours after accident initiation.	
d) Initial assumed pH of the Suppression Pool is 5.3. Injection of SLC should maintain pH at or above this value.	
4. Alternate Boron Injection Line-up, Unit 2	
a) If U2 SLC System is unable to inject boron into the reactor, then U1 SLC tank aligned to the 1B CRD pump can be used to inject boron into the U2 reactor.	Obj. V.B.6 Obj. V.C.6 Obj. V.D.8 Obj. V.E.8 TP-5 TP-6
b) This line-up utilizes U1 SLC storage tank as a source of borated water and the 1B CRD pump as the motive force for injection.	
c) The Unit 1 SLC storage tank drain line is connected by hose to the 1B CRD pump suction filter, via the EOI strainer cover.	
d) 1B CRD pump is started and aligned to the U2 CRD System. This injects SLC to the under core region.	
5. Alternate Boron Injection Line-up, Unit 3	
a) This line-up utilizes U2 SLC storage tank as a source of borated water, and one of the U3 CRD pumps as the motive force for injection.	
b) The U2 SLC storage tank drain is connected by hose to the 3A or 3B CRD pump suction filter, via the EOI strainer cover.	
c) The CRD pump is then started, this injects SLC to the under core region.	
6. Abnormal System Operation	
a) A break in the SLC line where it penetrates the reactor vessel will cause the following:	Obj. V.B.5 Obj. V.C.4
b) Lower indicated core differential pressure.	
c) Indicated differential pressure on non-fully instrumented jet pumps will lower.	
d) Indicated jet pump total developed head will lower.	

2-EOI APPENDIX-3B

ALTERNATE SLC INJECTION

LOCATION: Units 1 and 2 Reactor Building

ATTACHMENTS:
1. Tools and Equipment
2. Unit 1 el 565 Reactor Building and Hose Routing
3. Unit 1 el 621 Reactor Building and Hose Routing

(√)

1. **NOTIFY** Unit 1 and Unit 2 Operators and **CONTINUE** in this procedure. ____

2. **VERIFY** CRD pump 1B is shut down but available for use on Unit 2. ____

NOTE: Steps 3 through 6 may be performed concurrently with steps 7 through 12.

3. **REFER TO** Attachment 1 and **REMOVE** 1½-in. Red Rubber Hose from EOI Equipment Storage Boxes on El 565 ft, and El 621 ft. ____

4. **REFER TO** Attachment 3 and **ROUTE** BOTH sections of hose in El 621 ft box from Unit 1 SLC Tank Drain (El 621 ft NW) across West side of El 621 ft Unit 1, down SW stairwell to El 565 ft. ____

5. **REFER TO** Attachment 2, **OBTAIN** ALL three sections of hose from the El 565 ft EOI box, and **ROUTE** from SW stairwell across Unit 1 RB South, around to NE corner, and down stairwell to Unit 1 CRD pumps on El 541 ft. ____

6. **COUPLE** ALL sections of the hose and **TIGHTEN** couplings. ____

7. **ISOLATE** CRD pump 1B as follows:

   a. **RACK OUT** 1-BKR-085-0002, CONTROL ROD HYDRAULIC FEED PUMP 1B (4KV Shutdown Board A, Compartment 13). ____

   b. **NOTIFY** Unit 1 Operator to close 1-FCV-85-8, CRD PUMP B DISCHARGE VALVE, to Unit 1. ____

   c. **NOTIFY** Unit 2 Operator to close 2-FCV-85-8, CRD PUMP 1B DISCH TO U2. ____

Added to support plausibility…

3-EOI APPENDIX-3B

ALTERNATE SLC INJECTION

LOCATION: Units 2 and 3 Reactor Building

ATTACHMENTS:
1. Tools and Equipment
2. Unit 3 el 565 Reactor Building and Hose Routing
3. Unit 2 el 621 Reactor Building and Hose Routing

(√)

1. **NOTIFY** Unit 2 and Unit 3 Operators and **CONTINUE** in this procedure. ____

2. **VERIFY** CRD pump 3B is shut down and available for use. ____

NOTE: Steps 3 through 6 may be performed concurrently with steps 7 through 12.

3. **REFER TO** Attachment 1 and **REMOVE** 1½-in. rubber hoses from EOI Equipment Storage Boxes on El 565 ft, and El 621 ft. ____

4. **REFER TO** Attachment 3 and **ROUTE** the sections of hose in El 621 ft box from Unit 2 SLC Tank Drain (El 621 ft NE) across East side of El 621 ft Unit 2, down U-2/U-3 stairwell to El 565 ft. ____

5. **REFER TO** Attachment 2, and **ROUTE** the sections of hose from the El 565 ft EOI box from Unit 2/Unit 3 stairwell across Unit 3 RB South, around to NE corner, and down stairwell to Unit 3 CRD pumps on El 541 ft. ____

6. **COUPLE** ALL sections of the hose and **TIGHTEN** couplings. ____

7. **RACK OUT** 3-BKR-085-0002, CONTROL ROD DRIVE HYDRAULIC FEED PUMP 3B (4KV Shutdown Board 3EA, Compartment 11). ____

Examination Outline Cross-reference:	Level	RO	SRO
295038 High Off-site Release Rate / 9	Tier #	1	------
G2.4.9 (10CFR 55.41.10)	Group #	1	------
Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	K/A #	295038G2.4.9	
	Importance Rating	3.8	------

Proposed Question: **# 18**

Unit 1 has been operating for one week with increasing amounts of fuel bundle leaks. Suppression efforts have been unsuccessful and the trigger point for shutting down the reactor on excessive Stack release rates is rapidly approaching when the following alarms are received :

- MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C, (1-9-3A, Window 27)
- OG AVG ANNUAL RELEASE RATE EXCEEDED 1-RA-90-157C (1-9-4C, Window 27)

Which ONE of the following completes the statements for this condition?

The requirement to insert a manual scram is driven by a valid __**(1)**__ alarm.

Immediately following the scram, the ARP-specific **IF / THEN** directive to **CLOSE** MSIVs is based upon a determination that __**(2)**__.

A. **(1)** MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C
(2) releases are still in excess of Offsite Dose Calculation Manual limits.

B. **(1)** OG AVG ANNUAL RELEASE RATE EXCEEDED 1-RA-90-157C
(2) the reactor will remain subcritical without boron under all conditions.

C. **(1)** MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C
(2) the reactor will remain subcritical without boron under all conditions.

D. **(1)** OG AVG ANNUAL RELEASE RATE EXCEEDED 1-RA-90-157C
(2) releases are still in excess of Offsite Dose Calculation Manual limits.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part correct – as detailed in 'C' below. Second part is incorrect in that action(s) will be required if ODCM limits are being exceeded; but closing of the MSIVs is not specified. The ARP for Main Steam Line Rad Hi-Hi is very specific on closing MSIVs if not in Level /Power Control Contingency C-5 though.

B INCORRECT: First part incorrect – (See attached excerpts) The ARP for Offgas Average Annual Release Rate Exceeded is very specific on specifying a power reduction, but does not drive the scram directly; wherein the MS Line Rad Hi-Hi does. Second part correct – as detailed in 'C' below.

C **CORRECT:** (See attached excerpts) The Main Steam Line Rad Hi-Hi alarm, once validated, requires a core flow runback followed by a manual scram. Additionally, ARP specifies that if *not* in C-5 that MSIVs must be closed. If the reactor is shutdown under all conditions without boron, EOI Contingency C-5 will not be executed. Candidate must understand strategies associated with EOI/Contingency implementation.

D INCORRECT: Both parts incorrect – as detailed in 'A' and 'B' above.

RO Level Justification: Tests candidate's knowledge of shutdown (ALL RODS IN) implications as they relate to excessive fuel failures inside the reactor core and the resultant high offsite release rates. As the ARP only specifies whether or not you are in "C-5,"additionally tests the candidate's knowledge of strategies associated with EOI and EOI Contingency implementation. Candidate must determine whether or not C-5 requires execution for these conditions. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-ARP-9-4C, Rev. 18 (Attach if not previously provided)
1-ARP-9-3A, Rev. 38

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009 V.B.14.a (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0018 Page 34 of 43

OG AVG ANNUAL RELEASE LIMIT EXCEEDED

1-RA-90-157C

27

(Page 1 of 2)

Sensor/Trip Point:

1-RE-090-0157 2.5 R/hr (Alarm from recorder) (2500 mR/hr)

Sensor Location: Elevation 565
Turbine Building
Column B-T3
Recorder is on Panel 1-9-2.

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. **DETERMINE** if the Off Gas Annual Release Rate Limit is exceeded, **THEN** **PERFORM** the following:
   1. **VERIFY** alarm condition on the following:
      a. OFFGAS RADIATION, 1-RR-90-266 on Panel 1-9-2. ☐
      b. OG PRETREATMENT RADIATION Recorder, 1-RR-90-266, Panel 1-9-2.
      c. OG PRETREATMENT RAD MON RTMR, 1-RM-90-157 on Panel 1-9-10. ☐

B. **NOTIFY** Radiation Protection. ☐

NOTE

High Off-Gas flow can sweep settled particulates into flow stream and cause momentary rise in monitor reading. Low Off-Gas flow can result in improper dilution and cause monitor reading to rise.

C. **VERIFY** Off-Gas flow normal and proper sample flow to the monitor. ☐

Continued on Next Page

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0018 Page 35 of 43

OG AVG ANNUAL RELEASE LIMIT EXCEEDED, Window 27
(Page 2 of 2)

Operator Action: (Continued)

NOTE
Load reduction may be required to keep Off-Gas within ODCM limits.

D. **REQUEST** Chemistry perform radiochemical analysis to determine source. ☐

E. **WITH** OPS MGT and Shift Manager's permission, **PLACE** charcoal beds in parallel with another unit. **REFER TO** 2-OI-66. ☐

F. **IF** fuel damage is suspected, **THEN** **REFER TO** 1-SR-3.4.6.1 for dose equivalent iodine-131 determination. ☐

G. **REFER TO** 0-SI-4.8.B.1.a.1 and SR-3.4.6.1-a for ODCM compliance and to determine if power level reduction is required. ☐

H. **IF** directed by Shift Manager or Unit Supervisor, **THEN** **REDUCE** reactor power to maintain off-gas radiation within ODCM limits. ☐

I. **REFER TO** EPIP-1. ☐

References: GE 729E814 Series 1-47E610-90-1 ODCM 5.5.1
FSAR Sections 1.6.4.4.6, 7.12.2.2, and 13.6.2
Technical Specifications 4.6.B.6 and 4.8.B.1.a.1
Technical Requirements Manual 3.3.9.1, 3.3.5.1, 3.7.2.1

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0038 Page 14 of 53

MAIN STEAM LINE RADIATION HIGH
1-RA-90-135A
7

(Page 1 of 2)

Sensor/Trip Point:

1-RM-90-136 Channel A
1-RM-90-137 Channel C

[NRC/C] Setpoint is 1.5 X normal full power background including N-16 contribution and HWC System injection[nco 940247001]

Sensor Location: Panel 1-9-10

Probable Cause:

A. SI/SR in progress.
B. Air injection from placing standby cond demin in service.
C. Resin trap failure (RWCU or Cond Demin).
D. Fuel damage.
E. Sensor malfunction.
F. RCIC in service.
G. Placing HWC in service.

Automatic Action: None

Operator Action:

A. **CHECK** following radiation recorders on Panel 1-9-2:
   1. MAIN STEAM LINE RADIATION monitor, 1-RR-90-135. ☐
   2. OFFGAS PRETREATMENT RADIATION, 1-RR-90-157. ☐
   3. OFFGAS POST-TREATMENT RADIATION, 1-RR-90-265. ☐
   4. STACK GAS/CONT RM RADIATION, 0-RR-90-147. ☐

B. **NOTIFY** RAD PRO. ☐
C. [NRC/C] **REQUEST** Chemistry to perform radiochemical analysis of primary coolant. [NCO 940247001] ☐
D. **IF** off-gas PRETREATMENT RADIATION, 1-RR-90-157, has risen significantly (30% above previous hour average), **THEN REQUEST** Chemistry to perform analysis of pretreatment off-gas. ☐
E. **SHUTDOWN** Hydrogen Water Chemistry. **REFER TO** 1-OI-4. ☐
F. **REFER TO** 0-SI-4.8.B.1.A.1 for ODCM compliance and to determine if power level reduction is required. ☐

Continued on Next Page

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0038 Page 15 of 53

MAIN STEAM LINE RADIATION HIGH, 2-XA-55-135A, Window 7
(Page 2 of 2)

Operator Action: (Continued)

G. [NRC/C] **LOWER** reactor power to maintain off-gas radiation within ODCM limits as directed by Unit Supervisor. [NCO 940247001] ☐

H. IF ODCM limits are exceeded, **THEN** **REFER TO** EPIP-1. ☐

References: 1-47E610-90-1 47W600-11 1-45E620-3 1-729E814-1

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0038 Page 41 of 53

MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C	27

(Page 1 of 1)

Sensor/Trip Point:

1-RM-90-136 1-RM-90-137	3.0 x normal full power background including N-16 contribution and HWC System injection.

Sensor Location: Radiation monitor drawers are on Panel 1-9-10 in the control room.

Probable Cause:
A. Radiation is three times the normal full power background.
B. Sensor malfunctions.
C. SI (SR) in progress.

Automatic Action:
A. Mechanical vacuum pumps trip.
B. Vacuum pump suction valves 1-FCV-066-0036 and 1-FCV-066-0040 close.

Operator Action:
A. **VERIFY** the alarm on 1-RM-90-136 and 1-RM-90-137 on Panel 1-9-10. ☐
B. **CONFIRM** main steam line radiation level on recorder 1-RR-90-135, Panel 1-9-2. ☐
C. **IF** alarm is valid and Reactor Scram has not occurred, **THEN PERFORM** the following:
   1. **IF** core flow is above 60%, **THEN LOWER** core flow to between 50-60%. ☐
   2. **MANUALLY SCRAM** the Reactor. ☐
   3. **REFER TO** 1-AOI-100-1. ☐

D. **IF** plant conditions **DO NOT** require execution of 1-C-5, **THEN VERIFY** the MSIVs closed. ☐
E. **NOTIFY** RAD PRO. ☐
F. **VERIFY** actions of 1-ARP-9-3A Window 7 have been completed. ☐
G. **IF** Technical Specifications limits are exceeded, **THEN REFER TO** EPIP-1. ☐

References: 1-47E610-90-1 | 730E915-9, 10 | 1-45E620-5

Examination Outline Cross-reference:	Level	RO	SRO
600000 Plant Fire On Site / 8	Tier #	1	------
AK1.02 (10CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to PLANT FIRE ON SITE:	K/A #	600000AK1.02	
• Fire fighting	Importance Rating	2.9	------

Proposed Question: **# 19**

The following plant conditions currently exist on Unit 3:

- A Seal Oil Skid fire has erupted **AND** the sprinkler system has been manually initiated
- Fire header pressure has been 115 psig for 25 seconds after actuation of the sprinkler system
- A small leak in the Drywell resulted in Drywell Pressure of 3 psig and steady

Which ONE of the following completes the statement?

Based on these conditions, the Diesel Fire Pump ______.

A. **AND ALL THREE** Electric Fire Pumps are operating.

B. **AND ALL THREE** Electric Fire Pumps are in standby.

C. is in standby **AND ONE** Electric Fire Pump is operating.

D. is in standby **AND TWO** Electric Fire Pumps are operating.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Two Electric Fire Pumps are running but Diesel Fire Pump is not. If system header pressure is not >120 psig 45 seconds after the initiation signal, the Diesel Fire Pump automatically starts. (ONLY 25 sec has elapsed)

B INCORRECT: Automatic start of the Fire Pumps is locked out on High Drywell Pressure (2.45 psig) in conjunction with Low Reactor Pressure (450 psig) or Low Low Low Reactor Water Level (-122 in.) AND Diesel Generators are supplying power to the 4kV Shutdown Boards. With no conditions resulting in D/G's supplying the 4 kV Shutdown boards, the Fire Pumps are not locked out.

C INCORRECT: Two Electric Fire Pumps are running. Plausible in that candidate may believe that manual initiation of the sprinkler system would not result in automatic start of one Electric Fire Pump.

D **CORRECT:** The first fire pump will automatically start on any spray, sprinkler or fog system manual or automatic actuation. This initial fire pump auto-start is based on a sensed fire condition; NOT on low pressure. If system header pressure is not >120 psig 15 seconds after the initiation signal, a second Fire Pump automatically starts.

RO Level Justification: Tests the candidate's knowledge of interlocks associated with fire pumps in response to a plant fire on site. Additionally, the question tests the candidates knowledge of the impact of high drywell pressure on the fire pumps' start logic.

Technical Reference(s): OPL171.049 Rev 15 (Attach if not previously provided)

0-OI-26 Rev 90

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _V.B.4_____________ (As available)

Question Source:	Bank #		
	Modified Bank #	OPL171.049 #23	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

OPL171.049
Revision 15
Page 35 of 52

INSTRUCTOR NOTES

6. When one or more of the following initiation signals are present the fire pump will automatically start:

   a. Transformers, Shunt Reactor, or Unit 2 HPCI room temperature equal to or greater than 225°F

   b. Turbine Building protected areas or Unit 1/3 HPCI room temperature rise of 12°F/min

   c. Transformer of Shunt Reactor differential, overcurrent or sudden pressure relays actuated

   d. Any spray, sprinkler or fog system manual or automatic actuation

   NOTE: Typical system pressure demand is such that one Fire Pump will operate in case of a fire in the buildings, and two pumps will operate in case of a transformer fire.

Obj. V.D.10
Obj. V.E.11
0-45E644-1
0-45W643
This initial fire pump auto-start is based on a sensed fire condition; NOT on low pressure.

7. Starting any Fire Pump automatically closes the RSW Head Tank isolation valves (FCV-25-70 and FCV-25-32).

8. Stopping all operating Fire Pumps automatically opens RSW Head Tank isolation valve FCV-25-32. The operator must reopen FCV-25-70 at Panel 1-9-20 or locally.

9. Fire Pump Selector switch, XS-26-43 on Control Room Panel 1-9-20, should NOT be left in the OFF position because this would prevent the automatic start of a Fire Pump.

Obj.V.B.5/V.C.4
Obj.V.D.9/V.E.10

10. Due to the anti-pumping feature of the electric Fire Pump circuit breakers, the first pump in the selected sequence will not restart if it is manually tripped with an automatic start signal present.

2-45E765-7
Obj.V.B.5/V.C.4
Obj.V.D.9/V.E.10

11. If system header pressure is not >120 psig 15 seconds after the initiation signal, a second Fire Pump automatically starts.

12. If system header pressure is not >120 psig 30 seconds after the initiation signal, the third Fire Pump automatically starts.

BFN Unit 0	High Pressure Fire Protection System	0-OI-26 Rev. 0090 Page 15 of 63

Date ________

5.0 STARTUP

5.1 Automatic Start of a Fire Pump

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

- Transformers, Shunt Reactor, or Unit 2 HPCI Room, temperature equal to or greater than 225°F. ☐
- Turbine Building protected areas or U1&3 HPCI Rooms temperature rise of 12°F/min. ☐
- Transformer or Shunt Reactor differential, overcurrent or sudden pressure relays actuated. ☐
- Any spray, sprinkler or fog system manual or automatic actuation. ☐

NOTE

Automatic start of the fire pumps is locked out on the following:

- High Drywell Pressure (2.45 psig) in conjunction with Low Reactor Pressure (450 psig) or Low Low Low Reactor Water Level (-122 in.).

  AND

- Diesel Generators are supplying power to the 4160V Shutdown Boards.

NOTES

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

Examination Outline Cross-reference:	Level	RO	SRO
700000 Generator Voltage and Electric Grid Disturbances / 6	Tier #	1	------
AA1.03 (10CFR 55.41.10)	Group #	1	------
Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: • Voltage regulator controls	K/A #	700000AA1.03	
	Importance Rating	3.8	------

Proposed Question: **# 20**

Unit 3 is operating at 80% Reactor Power **AND** the crew has entered 0-AOI-57-1E, "Grid Instability," due to the 500 kV system voltage being at 541 kV. The crew reaches the following step in the procedure:

"**LOWER** reactive power until system voltage returns to 530 kV"

Which ONE of the following identifies how to lower reactive power **AND** the 161 kV Capacitor Bank Status that will restore the system voltage in accordance with 0-AOI-57-1E?

A. Depress the EHC Load Set LOWER pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are **IN** service.

B. Depress the EHC Load Set LOWER pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are **OUT** of service

C. Place the Generator Field Voltage Auto Adjust (90P), 3-HS-57-26, to the LOWER position; check the 161 kV Capacitor Banks are **IN** service.

D. Place the Generator Field Voltage Auto Adjust (90P), 3-HS-57-26, to the LOWER position; check the 161 kV Capacitor Banks are **OUT** of service.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 and 2 incorrect – Depressing the EHC load set LOWER pushbutton would have an effect on load / voltage at current power levels; but, is not a procedurally allowed action. Additionally, the 161 kV Capacitor Banks are not in service and will not aid in restoring high system voltage.

B INCORRECT: Part 1 incorrect (See above) - Part 2 is correct as required by 0-AOI-57-1E (See below).

C INCORRECT: Part 1 is correct and Part 2 is incorrect.

D **CORRECT:** (See attached excerpt) Part 1 correct – Per 0-AOI-57-1E, LOWER reactive power to system voltage returns to 530KV OR UNTIL Generator Reactive Power reaches -150 MVAR, Per 3-OI-47, To adjust GENERATOR MVAR, 3-EI-57-51, in the positive or lagging direction, PLACE GENERATOR FIELD VOLTAGE AUTO ADJUST (90P), 3-HS-57-26, in RAISE UNTIL desired MVAR is indicated. Part 2 correct – Per 0-AOI-57-1E, CHECK 161KV Cap Banks are OUT of service.

RO Level Justification: Tests the candidate's ability to recognize the need and method to adjust the voltage regulator controls to lower reactive power during grid instability conditions. Additionally, knowledge of the role the Capacitor Banks provide in this condition is necessary. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-AOI-57-1E Rev 7 (Attach if not previously provided)
3-OI-47 Rev 89
OPL171.036 Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.13 / 14 (As available)

Question Source:	Bank #		
	Modified Bank #	HLT 0801 Q#20	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

BANK QUESTION THAT WAS MODIFIED…

Exam Development Input 1

Record	K/A Number	Pedigree	SRO-Only	Currently Chosen For
1869	700000AK1.01	BFN 0801 NRC Question #20	NO	

Question Stem

Unit 3 is operating at 80% Reactor Power and the crew has entered 0-AOI-57-1E, "Grid Instability," due to the 530 kV system voltage being at 513 kV. The crew reaches the following step in the procedure:

"RAISE reactive power until voltage returns to 520 kV"

Which ONE of the following identifies how to raise reactive power AND the 161 kV Capacitor Bank Status that will restore the system voltage in accordance with 0-AOI-57-1E?

Answer

Place the Generator Field Voltage Auto Adjust (90P), 3-HS-57-26, to the RAISE position; check the 161 kV Capacitor Banks are IN service.

Distractor 1

Depress the EHC load set RAISE pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are IN service.

Distractor 2

Depress the EHC load set RAISE pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are OUT of service

Distractor 3

Place the Generator Field Voltage Auto Adjust (90P), 3-HS-57-26, to the RAISE □position; check the 161 kV Capacitor Banks are OUT of service.

Record: 1858 of 1939

BFN Unit 3	Turbine-Generator System	3-OI-47 Rev. 0089 Page 87 of 240

6.1 Normal Operation (continued)

[10] **MAINTAIN** GENERATOR MVAR, 3-EI-57-51, ≤200MVAR outgoing, and those of Illustration 6, Generator KVAR Limitations, (Capability Curve), the above note and as directed by the Transmission Operator as follows:

[10.1] To adjust GENERATOR MVAR, 3-EI-57-51, in the positive or lagging direction, **PLACE** GENERATOR FIELD VOLTAGE AUTO ADJUST (90P), 3-HS-57-26, in RAISE **UNTIL** desired MVAR is indicated. ☐

[10.2] To adjust GENERATOR MVAR, 3-EI-57-51, in the negative or leading direction, **PLACE** GENERATOR FIELD VOLTAGE AUTO ADJUST (90P), 3-HS-57-26, in LOWER **UNTIL** desired MVAR is indicated. ☐

[10.3] **PERFORM** the following to minimize generator heat load or check GENERATOR MVAR, 3-EI-57-51, accuracy:

[10.3.1] **ADJUST** GENERATOR MVAR, 3-EI-57-51, per Steps 6.1[10.1] or 6.1[10.2] for zero MVAR **AND MONITOR** GENERATOR PHASE A(B)(C) amps, 3-EI-57-47(48)(49). ☐

[10.3.2] **WHEN** minimum amps are indicated on GENERATOR PHASE A(B)(C) amps, 3-EI-57-47(48)(49), **THEN**

ZERO MVAR has been obtained. ☐

[10.4] **ADJUST** GENERATOR FIELD VOLTAGE MANUAL ADJUST (70P), 3-HS-57-25, **UNTIL** GEN TRANSFER VOLTS, 3-EI-57-41, indicates zero. ☐

[11] **PERFORM** Illustration 7, Turbine-Generator Bearing Metal Temperature, daily. ☐

BFN Unit 0	Grid Instability	0-AOI-57-1E Rev. 0007 Page 7 of 18

4.2 Subsequent Action (continued)

[6] **IF** grid instability is characterized by system voltage being maintained outside the normal limits of 525 + 5 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 515KV, **THEN**

PERFORM the following:

[6.3] **RAISE** reactive power to system voltage returns to 520KV **OR UNTIL** Generator Reactive Power reaches +200 MVAR, ☐

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

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

3. Indication of amperes (each phase), megawatts, megavars, and voltage for each 500kV line is available on panel 9-23.

B. 161 kV System

Safety: Extreme High Voltage

1. Purpose

This system provides adequate off-site power to start the units, carry common station auxiliary loads, and if necessary to carry the emergency loads for a unit in a design basis accident. It also provides power for operation of the cooling tower equipment.

TP-2

2. The arrangement consists of two buses with cross-tie Oil Circuit Breakers (OCBs) and capacitor banks. Each of the two 161kV lines, Athens and Trinity, connects to a common station transformer (CSST A or B) through a MOD, and to a cooling tower transformer (CTT 1 or 2) through a disconnect.

Obj. V.B.3
Obj. V.D.3

3. The 161KV lines are normally crosstied.

4. The Capacitor banks are normally in automatic and in standby. The Capacitor banks will automatically close into the 161kV Sys to raise the voltage if start bus voltage reaches 4050V (lowering).

Obj V.B.14
Obj V.D.11

Capacitor Banks are connected to 161 kV Bus 2 in order to regulate voltage to CSSTs. This is done to maintain Start Buses 1A and 1B voltage between 4035 V and 4285 V which assures that an adequate source of power is available to operate plant safety equipment and safely shutdown the plant after an accident.

SER 03-05 Capacitor Banks are energy storage devices used to supply capacitive reactive power to the distribution system and offset the inductive reactive power demand from lines, transformers, and inductive loads.

45E761 sh 1 & 2

Examination Outline Cross-reference:	Level	RO	SRO
295002 Loss of Main Condenser Vac / 3	Tier #	1	------
AK2.02 (10CFR 55.41.10)	Group #	2	------
Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following:	K/A #	295002AK2.02	
• Main turbine	Importance Rating	3.1	------

Proposed Question: **# 21**

Given that Main Condenser Vacuum is degrading, which ONE of the following completes the statements?

A **RISING** Off-Gas flow on OG FLOW TO 6-HOUR HOLDUP VOLUME, 3-FR-66-20, would be indicative of **__(1)__**.

To ensure a Manual Scram is inserted prior to automatic action occurring, a Control Room Panel 3-9-6 "Hotwell Pressure" reading of **__(2)__** inches Hg Vacuum would be a valid Trigger Value in accordance with OPDP-1, "Conduct of Operations."

A. **(1)** air in-leakage to the Main Condenser.
(2) 22.5

B. **(1)** First Stage Steam Jet Air Ejector stalling.
(2) 22.5

C. **(1)** air in-leakage to the Main Condenser.
(2) 25

D. **(1)** First Stage Steam Jet Air Ejector stalling.
(2) 25

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part is correct – see 'C' below. Second part is incorrect - see 'C' below. So, in this case, the unintended consequence (AUTOMATIC Rx Scram) would have already occurred. If the candidate were to set the Trigger Value off of the actual setpoint (-21.8 in.), this value would be chosen. The value of -25 inches would be too far away to be a valid Trigger Value in this case, wherein they may still be attempting to lower power.

B INCORRECT: (See attached excerpts). First part incorrect in that according to 3-AOI-47-3 specifically, First Stage SJAE stalling will cause REDUCED Off-Gas flows, but will still result in the same consequences. Second part is incorrect – see 'A' above.

C **CORRECT:** (See attached excerpts). According to 3-AOI-47-3 specifically, a RISING Off-Gas flow rate is indicative of Condenser Air In-leakage. It goes on to discuss the OE where the procedurally-specified, actual trip setpoint for the instruments of -21.8 in. Hg Vacuum does not correspond to the same observable value in the Control Room. The Control Room indication value of approximately -24.3 (but well before -22.5 in.) is where the Main Turbine actually trips on Low Condenser Vacuum. OPDP-1 sets forth requirements on establishing Trigger Values for Manual Insertion of a Reactor Scram in cases such as this.

D INCORRECT: First part is incorrect – see 'B' above. Second part is correct.

RO Level Justification: Tests the candidate's knowledge of potential transient conditions that can impact condenser vacuum, along with the subsequent impact to the Main Turbine (and Reactor) itself. An element of "Conduct of Operations" is tested with setting a Trigger Value. Additionally, plant Operating Experience is also tested in the relationship of the trip setpoint to where the turbine actually trips based upon Control Room indications. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-AOI-47-3, Rev. 11 (Attach if not previously provided)
3-OI-47, Rev. 89
OPDP-1, Rev. 15
OPL171.010, Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.12 / 23 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

BFN Unit 3	Loss of Condenser Vacuum	3-AOI-47-3 Rev. 0011 Page 5 of 11

NOTES

1) Rising Off-Gas flow would indicate condenser inleakage if the Off-Gas System is functioning properly. Low Off-Gas flow in conjunction with low condenser vacuum could be indicative of an Off-Gas problem.

2) During operations with valid CONDENSER A, B, OR C VACUUM LOW 3-PA-47-125 alarm, and condensate temperature of 136 F or greater at the inlet of the SJAE(ICS point 2-28), reduced SJAE First Stage performance (stalling) may occur. This condition will cause reduced Off Gas flow and a loss of vacuum/turbine trip.
[BFPER 02-016091-000]

3.0 AUTOMATIC ACTIONS

NOTE

Turbine trip is expected around 24.3 inches Hg as indicated on 3-XR-2-2 due to differences between instrument taps for turbine trip and indicated vacuum. (PER 89506)

A. Any of the following will cause a turbine trip:

1. Condenser A, both 3-PS-047-072A and 72B at 21.8" Hg vacuum.

2. Condenser B, both 3-PS-047-073A and 73B at 21.8" Hg vacuum.

3. Condenser C, both 3-PS-047-074A and 74B at 21.8" Hg vacuum.

B. RFP turbines trip and main turbine bypass valves closure occurs at -7" Hg hotwell pressure.

3.4 Turning Gear Operation (continued)

D. For relatively short outages where restart is expected, the Turbine must be maintained on turning gear as long as any shell temperature is above 500°F.

E. If it is necessary to discontinue turning gear operation while the rotors are still hot as indicated by shell metal temperature ≥ 500°F, oil flow to the bearings should be maintained to prevent bearing damage due to overheating.

F. If lube oil flow must be stopped with a shell metal temperature greater than 500°F, bearing temperatures should be monitored on THRUST/JOURNAL BRG TEMPERATURE,3-TR-47-23, to ensure Main Turbine bearing metal temperatures do **NOT** exceed 300°F.

G. Following any shutdown, turning gear operation may be discontinued indefinitely after shell metal temperatures are less than 500°F [GEK-92566C]

H. When the turbine is to be removed from turning gear operation for greater than 24 hours, a WO should be completed for Electrical Maintenance to remove the main generator and exciter brushes.

3.5 Turbine Trips

A. High Reactor Water Level Trip logic for the Main Turbine at +55 inches is taken from Narrow Range level instruments 3-LI-3-208A, 3-LI-3-208B, 3-LI-3-208C, and 3-LI-3-208D. The logic is arranged in two channels; Channel A is fed from 3-LI-3-208A and 3-LI-3-208C. Channel B is fed from 3-LI-3-208B and 3-LI-3-208D. A trip of the Main Turbine and the RFPTs will occur if both instruments in Channel A, or Channel B sense reactor water level at ≥ +55 inches.

B. Turbine trip on low main condenser vacuum is expected around an indicated 24.3 inches Hg, instead of the 21.8 inches currently stated in this procedure, due to differences between instrument taps for turbine trip and indicated vacuum.

This condition was discovered during maintenance activities on Unit 3 when condenser vacuum was being monitored by operations using 3-XR-2 on Panel 3-9-6 that is fed by 3-PT-2-1. The instrument tap for 3-PT-2-1 is located just above the condenser tubes, which is the point of highest vacuum. The instrument taps for the sensing lines feeding the turbine trip switches are located just below the LP turbines. Because of the Volumetric differences between the two locations of the taps, and the steam flow direction from top to bottom, the sensed vacuum is greater at the lower tap than at the higher tap. (See PER 89506)

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3.3 Conservative Decision Making (continued)

D. When the control room team identifies or is made aware of a slowly degrading trend the principles of operational decision making are applied. Shift Management will request an Operational Decision-Making Issue (ODMI) in accordance with OPDP-11 and the corrective action program. The issue is tracked by station management until resolved or a plan to correct the issue is in place. The crew is provided guidance, in a timely manner consistent with the degrading condition rate of change. The plan to cope with the potential consequences of the issue should take into account the rate of degradation. Guidance should be provided on actions to take should the rate of degradation change or predefined limits be reached including when to, remove the component or system from service, maneuver the plant or shut the reactor down.

3.4 Expectations for Inserting a Manual Scram or Manual Reactor Trip

Licensed Operators shall take no manual action that will cause an automatic scram. Operators shall without hesitation insert a manual scram/manual reactor trip whenever any of the following conditions occurs:

- Safety of the reactor is in jeopardy.
- Operating parameters exceed any of the reactor protection setpoints and an automatic shutdown does not occur.
- Core thermal hydraulic instability is observed and mitigating actions are ineffective (BWR).
- As directed by plant procedures.
- When a pre-determined trigger value is reached.

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. (e.g. Establishment of manual level control following automatic FCV failure to control level or manual start of an EDG that failed to auto start.)

B. When operating in manual mode, the Unit Supervisor will specify the frequency of monitoring, control bands and trigger values as appropriate.

C. When manual operation is no longer required or the automatic function is restored, return systems to automatic or standby mode.

D. When practical, before placing controls in manual for activities which require manual control, review system response and actions to be taken during potential off normal events.

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9.0 DEFINITIONS

Control Board Walk Down - A detailed review of each control board by the oncoming and off-going Operator. Items checked should include annunciators, switch alignment and other light indications.

Controls - Apparatus and mechanisms, which directly affect the reactivity or power level of the reactor when manipulated.

Critical Parameter - An operating parameter that bears close monitoring because it alerts an Operator that plant or component conditions are significantly degrading and operator actions to mitigate the condition will be required. Standard critical parameters are Rx power, Rx pressure, and level (SG Level for PWR and Rx Level for BWR).

Flagging - The practice of placing a flag (typically a border for annunciator windows or post-it flag, colored tape or other similar item for other devices) on an annunciator, switch, drawer, etc., to help an Operator recognize/focus on the correct component/annunciator or perform the "R" in STAR. These are placed and removed in a timely manner so as not to cause confusion later.

Key Performance Indicator (KPI) - An indicator used to measure and track department or plant performance over time.

Narrative Log - A chronological sequence of events as related to unit/plant activities. The Narrative Log will normally be maintained using the eSOMS computer application. If eSOMS is not available, then either hard bound ledger(s), or duplicate type books with numbered pages (kept in the various operating locations at the plant as determined by Operations management) may be used.

Safe Operating Envelope - Those system operating parameters defined by operating procedures and bounded by technical specifications and other licensing basis documents that ensure system and component design are not unduly challenged.

Shot-on-Goal - A term used to describe a situation where an activity is identified at the authorization phase to have an unplanned plant impact, or that has been presented for authorization without the following rules or process:

- An activity that if allowed would have resulted in an unplanned LCO, ODCM or TRM or unplanned equipment reliability issue or
- An activity presented for authorization not on the approved work schedule and presented without the appropriate documentation in accordance with SPP-7.1.

Trigger Value- A predetermined **critical parameter** value that is established where an operator/crew will take actions to address a significantly degrading condition.

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Turbine Trip	Setpoint	Warning	First-Out Alarm
Overspeed Electrical	107% (1926 rpm) Backup elec. 109% (1962 rpm)		TURB TRIPPED TRIP OVERSPEED XA-55-1-1
Generator and Transformer Faults	86 devices		TURB TRIPPED ELECTRICAL TROUBLE XA-55-1-2
Main Condenser Vacuum Low	Trip 21.8 inches Hg vacuum (indicated will be ~ 24.3 in) Alarm @ 24.3 inches, actual	CONDENSER A,B, OR C VACUUM LOW XA-55-7B-17	TURB TRIPPED COND VAC LOW XA-55-1-3
Moisture Separator Drain Tank Level High	11 ft above El 586 floor level	MOIST SEP LC RES LEVEL HIGH XA-55-7C-2,3,4,16,17,18	TURB TRIPPED MOIS SEP LEVEL HIGH XA-55-1-4
Stator Coolant Failures	85 deg° C (81° C U-1/3), or 468 gpm (542 gpm U-1/3) >7726 stator amps (70 sec TD)	GEN STATOR COOL SYS ABNORMAL XA-55-7A-22 TURBINE TRIP TIMER INITIATED XA-55-8A	TURB TRIPPED STAT COOLANT SYS FAILURE XA-55-1-5
MSOP Discharge Pressure Low	105 psig >1300 rpm 2/3 logic		TURB TRIPPED MN SHAFT OIL PUMP INOP XA-55-1-6

Turbine Trip Annunciators (Cont)

Examination Outline Cross-reference:	Level	RO	SRO
295009 Low Reactor Water Level / 2	Tier #	1	------
AK1.05 (10CFR 55.41.10)	Group #	2	------
Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL: • Natural circulation	K/A #	295009AK1.05	
	Importance Rating	3.3	------

Proposed Question: **# 22**

Unit 2 is in Mode 4 with the following conditions:

- Reactor Level is (+) 45 inches
- **ALL** Shutdown Cooling has been lost

Which ONE of the following identifies the reason that 2-AOI-74-1, "Loss of Shutdown Cooling," requires a HIGHER Reactor Water Level established **AND** maintained?

The HIGHER Reactor Water Level ______.

A. provides additional mass of water in the Reactor Vessel to delay boiling.

B. floods the Moisture Separators which provides a path for natural circulation.

C. allows the Main Steam Line Drains to provide a drain path for feed **AND** bleed.

D. provides greater Net Positive Suction Head for the Reactor Water Cleanup Pumps.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Although additional mass of water would delay boiling, the reason level is raised to provide path for Natural Circulation.

B **CORRECT**: If forced circulation is lost level is raised to (+) 80 inches per 2-AOI-74-1 to flood the Moisture Separators and provide a path for natural circulation.

C INCORRECT: Feed and Bleed through MSL Drains would mitigate a loss of Shutdown Cooling but level is not raised to the MSL level of (+) 117 inches RPV Water Level.

D INCORRECT: It is true that raising level would provide greater Net Positive Suction Head for RWCU Pump; however, strategy to raise level would not be used if any forced circulation is available. Therefore, RWCU would be out of service.

Justification: Candidate must recognize the implications of low reactor water level, i.e. < (+) 80 inches with a loss of Shutdown Cooling and the reason for raising level.

Technical Reference(s): 2-OI-74 Rev. 150 (Attach if not previously provided)

2-AOI-74-1 Rev. 33

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.074 V.B.7 (As available)

Question Source:

Bank #	DAEC 07 #49	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam Duane Arnold 07

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 2	Loss of Shutdown Cooling	2-AOI-74-1 Rev. 0033 Page 15 of 31

4.2 Subsequent Actions (continued)

[12.10] **WHEN** time permits after RHR pump is started, **THEN**

VERIFY RHR Pump Breaker charging spring recharged by observing amber breaker spring charged light is on and closing spring target indicates charged. ☐

[12.11] **SLOWLY THROTTLE** RHR HX 2A(2C)(2B)(2D) RHRSW OUTLET VALVE, 2-FCV-23-34(40)(46)(52), to obtain desired cooldown rate. ☐

[13] **IF** necessary, **RAISE** RWCU flow rate to maximum AND maximize RWCU blowdown as required to maintain reactor coolant temperatures less than 200°F on all indications. **REFER TO** 2-OI-69. ☐

CAUTION

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

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

PERFORM the following: ☐

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

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

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

[15] **IF** the affected loop of RHR cannot be placed back in Shutdown Cooling, **THEN**

RESTORE power to affected breakers per 2-POI-74-2 if applicable (Otherwise N/A)

<u>AND</u>

PLACE the alternate loop of RHR in Shutdown Cooling. **REFER TO** 2-OI-74. (Otherwise N/A) ☐

3.4 Shutdown Cooling (continued)

F. When in Shutdown Cooling, reactor temperature should be maintained ≥72°F and only be controlled by throttling RHRSW flow. This is to assure adequate mixing of reactor water.

1. [NER/C] Reactor vessel water temperatures < 68°F exceed the temperature reactivity assumed in the criticality analysis. [INPO SER 90-017]

2. [NER/C] Maintaining water temperature below 100°F minimizes the release of soluble activity. [GE SIL 541]

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.

H. [NER/C] Placing Shutdown Cooling (SDC) in service shortly after the SDC isolation is reset minimizes the release of soluble activity. [GE SIL 541]

I. During SDC operation, the ICS screen (RHRSHUT) if available or Appendix O for RHR/SDC should be monitored frequently to ensure thermal stratification does not occur. If thermal stratification does occur, as indicated by ≥ 50°F ΔT between bottom head drain and feedwater nozzle temperatures, SDC flow is required to be raised and/or a recirculation pump placed in service to provide adequate mixing. If neither of the above can be performed, then reactor water level should be verified >70" to maximize natural circulation within the vessel.

J. Care should be exercised when changing the operating mode or any system parameter while SFSP or reactor cavity operations are in progress. This precludes the possible introduction of sediment/dirt into the SFSP or reactor cavity, thereby reducing water clarity. Notify the refuel floor SRO, if applicable, for permission to alter RHR/SDC System alignment and/or parameters.

K. [PER 141380] The preferred method to initiate Shutdown Cooling is for the RPV steam dome pressure to be at 15 psig or less.

Duane Arnold NRC 2007

RO 47	K/A Number 295021	Statement AK3.05	IR 3.6	Origin N	Source Question NA
LOK F	10CFR55.41(b)5	LOD (1-5)	Reference Documents AOP 149 Rev 25		
Loss of Shutdown Cooling Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : Establishing alternate heat removal flow paths.					

What is the reason that AOP 149, LOSS OF SHUTDOWN COOLING requires a HIGHER RPV Water Level established and maintained?

The HIGHER RPV Water Level:

a. floods the Moisture Separators which provides a path for natural circulation.

b. provides additional mass of water in the Reactor Vessel which will delay boiling.

c. allows the Main Steam Line Drains to provide a drain path for feed and bleed.

d. provides greater Net Positive Suction Head for the Reactor Water Cleanup Pumps.

Correct Answer: A when the Moisture Separators are flooded, a path is provided for natural circulation.
Plausible Distractors: B is plausible: true, but not the stated reason. C is plausible: Steam Lines are not flooded 214 inches RPV Water Level. D is plausible: true, but this is not a limitation.
Objective Link: None

Examination Outline Cross-reference:	Level	RO	SRO
295010 High Drywell Pressure / 5	Tier #	1	------
AA1.06 (10CFR 55.41.7)	Group #	2	------
Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE:	K/A #	295010AA1.06	
• Leakage detection systems	Importance Rating	3.3	------

Proposed Question: **# 23**

Unit 2 is at 75% Reactor Power when DRYWELL NORM OPERATING PRESS HIGH, (2-9-3B, Window 19) is received. Drywell Pressure is 1.6 psig **AND** slowly trending up.

Which ONE of the following completes the statements?

In attempting to determine whether the Technical Specification **UNIDENTIFIED** leakage rate was rising, the operator would evaluate pump-out **AND** fill-rates for the Drywell **__(1)__** Drain Sump.

If the sump fills faster than the preset allowable time, then the Fill-Rate Timer logic will feed directly into the respective alarm for **__(2)__**.

A. **(1)** Floor
(2) Excessive Sump Pump Operation

B. **(1)** Equipment
(2) Excessive Sump Pump Operation

C. **(1)** Floor
(2) Drain Sump Level Abnormal

D. **(1)** Equipment
(2) Drain Sump Level Abnormal

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (See attached excerpts) Per TS 3.4.4, RCS Operational Leakage, Operational leakage shall be limited to ≤ 5 gpm Unidentified leakage. **BOTH** Floor and Equipment Drains are utilized in calculating operational leakage. In the case of UNIDENTIFIED, this leakage is collected in the Floor Drain Sump; versus IDENTIFIED leakage being collected in the Equipment Drain Sump. Part 2 – correct in that, although counterintuitive in comparison, the fill rate timer does not feed into the Abnormal Sump Level Alarm(s); it feeds into the Excessive Sump Pump Operation Alarm(s), per ARP-9-4C.

B INCORRECT: First part incorrect – as detailed in ‘A’ above. Second part correct – as detailed in ‘A’ above.

C INCORRECT: First part correct – as detailed in ‘A’ above. Second part correct – as detailed in ‘A’ above.

D INCORRECT: First part incorrect – as detailed in ‘A’ above. Second part correct – as detailed in ‘A’ above.

RO Level Justification: To correctly answer, candidate must demonstrate ability to monitor Leakage detection systems in response to HIGH DRYWELL PRESSURE and use the data to determine Tech Spec Operational Leakage. This question is rated as MEM due to the requirement to recall facts.

Technical Reference(s): OPL171.016, Rev. 17 (Attach if not previously provided)
U2 TS 3.4.4 Amm. 253
2-ARP-9-4C, Rev. 30

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.087 V.B.8 (As available)
OPL171.016 V.B.14/16

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; and

c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce LEAKAGE increase to within limits. OR	4 hours

(continued)

		INSTRUCTOR NOTES
b.	2/ 3-TS-77-14(Alarm) setpoint is ≤200ºF. 1-TS-77-14 setpoint is 160° F.	
c.	The design temp for the Sump is 200ºF for Unit 2 and Unit 0, which required drawing 2-47E852-2 & 2-47E852-1 to be revised.	
4.	Flow integrators and pump fill rate timers are used to determine leakage in the drywell	
a.	The integrators indicate drain flow in gallons. The decimal point is disregarded and only the last five digits are used when calculating leakage.	DWFD & ED integrators & timers are tech. Spec. instr. TRM 3.3.10
b.	The equipment drain integrator 2-FQ-77-16 is used to determine identifiable reactor coolant leakage.	Obj. V.B.14 Obj. V.D.17
c.	The floor drain integrator 2-FQ-77-6 is used to determine unidentifiable reactor coolant leakage.	These timers activate alarms on Pnl 9-4. If the alarm will reset, then the fill rate timer is the source of the alarm. If not, then the pump-out rate timer has initiated the alarm & it will not reset until the pump(s) stop.
d.	The equipment drain sump fill rate timer is set for 21 minutes.	
e.	The equipment drain sump pump out rate timer is set for 13 minutes.	
f.	The floor drain sump fill rate timer is set for 82 minutes.	
g.	The floor drain sump pump out timer is set for 8.5 minutes.	
5.	The pump out rate timers and the fill rate timers are set to alarm if the pump runs longer than or restarts before the preset time interval.	Discuss: 2-SR-3.4.4.1A
a.	If the integrated flow readings cannot be obtained from control room instrumentation, the leakage rates may be determined by manually timing the interval between the auto stopping of the sump pump and the auto start of the pump, and dividing the time into 402 gals., the volume of leakage corresponding to the difference of the two levels.	

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0030 Page 6 of 44

DRYWELL FD SUMP LEVEL ABN 2-LA-77-1A	2

(Page 1 of 1)

<u>Sensor/Trip Point</u>:

LIS-77-1A	Alarm 65% (36 inches)
	Alarm 2.5% (11 inches)
LIS-77-1B	Alarm 65% (36 inches)
	Alarm 2.5% (11 inches)

Sensor Location: Panel 25-177
Elevation 565
Column R-12 P-LINE

Probable Cause: High or low level in the drywell floor drain sump.

Automatic Action:

A. Floor Drain Sump Pump selected by Automatic Pump Selector, starts at 30 inch sump level. Second sump pump starts at High Level alarm, 36 inch sump level.
B. Sump level of 15 inches stops all running pumps.

Operator Action:

A. **IF** the level is high, **THEN**
CHECK the sump pumps running, discharge valve open, and flow indicated. ☐
B. **IF** the sump level is low, **THEN**
CHECK the sump pumps off. ☐
C. **IF** level is unknown, **THEN**
DISPATCH personnel to Panel 25-177 to determine sump level. ☐
D. **IF** level is normal, **THEN**
VERIFY bulbs for photo-electric switches are good. ☐

References: 45N620-4 45N779-15
FSAR Sections 4.10.3.3, 4.10.5, 10.16.6, and 13.6.2

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0030 Page 16 of 44

DRYWELL EQPT DR SUMP LEVEL ABN 2-LA-77-14A 9	Sensor/Trip Point:	
	2-LIS-77-14A	Alarm 65% (36 inches)
		Alarm 2.5% (11 inches)
	2-LIS-77-14B	Alarm 65% (36 inches)
		Alarm 2.5% (11 inches)

(Page 1 of 1)

Sensor Location: Panel 25-177
Elevation 565
Column R-12 P-LINE

Probable Cause: High-high or low-low level in the sump.

Automatic Action:
A. Floor Drain Sump Pump selected by Automatic Pump Selector, starts at 30 inch sump level. Second sump pump starts at High Level alarm, 36 inch sump level.
B. Sump level of 15 inches stops all running pumps.

Operator Action:
A. **IF** the level is high, **THEN**
CHECK sump pumps running, discharge valve open and flow indicated on 2-FR-77-16. **MONITOR** for excessive leakage. ☐
B. **IF** the sump level is low, **THEN**
CHECK the sump pumps OFF. ☐
C. **IF** level is unknown, **THEN**
DISPATCH personnel to Panel 25-177 to determine sump level. ☐
D. **IF** level is normal, **THEN**
VERIFY bulbs for photo-electric switches are good. ☐

References: 45N620-4 45N779-14 0-47E610-77-1
FSAR Sections 4.10.3.3, 4.10.5, 10.16.6, and 13.6.2
Technical Specification Section 3.4.4

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0030 Page 18 of 44

DRYWELL FD SUMP PUMP EXCESSIVE OPRN 2-IA-77-1 (11)	Sensor/Trip Point:	
	2-IS-77-1A	8.5 Minutes
	2-IS-77-1B	82 Minutes

(Page 1 of 1)

Sensor Location: Panel 9-19
Auxiliary Instrument Room

Probable Cause:
A. Sensor malfunction.
B. Excess leakage.

Automatic Action: The pump-out rate timers and the fill rate timers are set to alarm if the pumps run longer than or restart before a preset time interval.

Operator Action:
A. **CHECK** the sump pumps are operating with proper flow indicated on recorder DRYWELL EFFLUENT FLOW, 2-FR-77-6 on Panel 2-9-4. ☐
B. **IF** proper flow is **NOT** observed, **THEN** **PERFORM** 2-SI-4.2.E-6, Drywell Equipment/Floor Drain Sump Flow Rate Adjustment. ☐
C. **IF** the sump pump(s) are running, **THEN** **ATTEMPT** to reset fill rate timer with 2-HS-77-1C (2-HS-77-1C resets the fill rate timer only). ☐

NOTES
1) If the alarm will reset, then the fill rate timer is the source of the alarm.
2) If the alarm will **NOT** reset, then the pump-out rate timer has initiated the alarm and the alarm will **NOT** reset until the pump(s) stop.

D. **DETERMINE** the leak rate if possible and that continued operation is within Technical Specification Section 3.4.4. [2-SR-2] ☐
E. **NOTIFY** Site Engineering to evaluate using 2-TI-275E, Drywell Leak Investigation Analysis. ☐
F. **REFER TO** EPIP-1. ☐

References:

0-47E610-77-1	GE730E934 Series	45N620-4
FSAR Sections 4.10.3.3, 4.10.5, 10.16.6, and 13.6.2		
Technical Specifications Section 3.4.4		2-TI-275E

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0030 Page 25 of 44

DRYWELL EQPT DR SUMP PUMP EXCESSIVE OPRN 2-IA-77-14 18	Sensor/Trip Point:	
	2-IS-77-14A	13 Minutes
	2-IS-77-14B	21 Minutes

(Page 1 of 1)

Sensor Location: Panel 9-19
Auxiliary Instrument Room

Probable Cause:
A. Excessive leakage.
B. Sensor malfunction.

Automatic Action: The pump-out rate timers and the fill rate timers are set to alarm if the pump runs longer than, or restarts before a preset time interval.

Operator Action:

A. **CHECK** the sump pumps are operating with proper flow indicated on recorder DRYWELL EFFLUENT FLOW, 2-FR-77-6 on Panel 2-9-4. ☐

B. **IF** proper flow is **NOT** observed, **THEN** **PERFORM** 2-SI-4.2.E-6, Drywell Equipment/Floor Drain Sump Flow Rate Adjustment. ☐

C. **IF** the sump pump(s) are running, **THEN** **ATTEMPT** to reset fill rate timer using 2-HS-77-14D (2-HS-77-14D resets the fill rate timer only). ☐

NOTE

If the alarm will reset, then the fill rate timer is the source of the alarm.

If the alarm will **NOT** reset, then the pump-out rate timer has initiated the alarm and the alarm will **NOT** reset until the pump(s) stop.

D. **DETERMINE** the leak rate if possible and that continued operation is within Technical Specification Section 3.4.4. [2-SR-2] ☐

E. **REFER TO** EPIP-1. ☐

References: 45N620-4 0-47E610-77-1 FSAR Sections 4.10.3.3, 4.10.5, 10.16.6, and 13.6.2

Technical Specifications Section 3.4.4

Examination Outline Cross-reference:

295022 Loss of CRD Pumps / 1

G2.2.38 (10CFR 55.41.10)

Knowledge of conditions and limitations in the facility license.

Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295022G2.2.38	
Importance Rating	3.6	------

Proposed Question: **# 24**

Unit 2 is operating at 100% Reactor Power when the running CRD pump trips **AND** actions to restore a CRD pump have **NOT** been successful. The following alarm is subsequently received during efforts to restore one CRD Pump:

- CONTROL ROD DRIVE UNIT HIGH TEMP, (2-9-5A, Window 17), is in alarm

Which ONE of the following identifies the setpoint for this alarm **AND** the required action in accordance with 2-ARP-9-5A, "Panel 9-5 2-XA-55-5A?"

A. 250 °F; declare the affected Control Rod(s) "SLOW"

B. 250 °F; declare the affected Control Rod(s) "INOPERABLE"

C. 350 °F; declare the affected Control Rod(s) "SLOW"

D. 350 °F; declare the affected Control Rod(s) "INOPERABLE"

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: (See attached excerpts) First part incorrect - Per Tech Spec 3.1.4 Bases, temperatures are permitted to rise to 350 °F prior to affecting Scram times. Then, they would be declared SLOW per note 1 of T.S. 3.1.4 - 1 table; making the second part correct.

B INCORRECT: First part incorrect – as detailed in 'A' above. Second part incorrect in that no procedural guidance is established at BFN for this possibility. All procedures distinctly delineate a declaration of "SLOW."

C **CORRECT:** The candidate must determine that no CRD pump is operating based on indications given. (See attached excerpts) Per T.S. 3.1.4 Bases and 2-TI-393, temperatures are permitted to rise to 350 °F prior to affecting Scram times. Then, they would be declared SLOW per note 1 of T.S. 3.1.4 - 1 table in accordance with 2-TI-393 and ARP-9-5A.

D INCORRECT: First part correct as detailed in 'C' above. Second part is incorrect as detailed in 'B' above.

RO Level Justification: Tests the candidate's knowledge of the implications of a loss of CRD Pumps and the subsequent impact on CRD Mechanism temperatures as they relate to Browns Ferry Technical Specifications. This question is rated as C/A due to the requirement to assemble, sort, and integrate two distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): U2 Tech Spec 3.1.4, Am.253 (Attach if not previously provided)

U2 Tech Spec Bases 3.1.4, Rev. 9

2 -ARP-9-5A, Rev. 45

2-TI-393, Rev. 2

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _OPL171.006 V.B.18 / 22_ (As available)

Question Source: Bank # **X** 0801 #25

Modified Bank # (Note changes or attach parent)

New

Question History: HLT 0801

Last NRC Exam (07 / 2009)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 2	Panel 9-5 2-XA-55-5A	2-ARP-9-5A Rev. 0045 Page 23 of 46

CONTROL ROD
DRIVE UNIT
TEMP HIGH
2-TA-85-7

17

(Page 1 of 2)

Sensor/Trip Point:

TE-85-7 (1 thru 185) 350°F Alarm comes from recorders, 2-TR-85-7A, & 2-TR-85-7B

Sensor Location: Located on each control rod drive.

Probable Cause:
A. Insufficient cooling water flow.
B. Malfunction of sensor.
C. Leaking scram discharge valve.
D. Plugged CRD cooling water orifice.

Automatic Action: None

Operator Action:

A. **VALIDATE** high temp of CRD on recorder 2-TR-85-7A, & 2-TR-85-7B (Panel 2-9-47) or on ICS. ☐

B. **IF** alarm is valid, **THEN** perform the following as directed by the Unit Supervisor: ☐
- **CHECK** cooling water pressure and flow normal on Panel 2-9-5. ☐
- **DISPATCH** personnel to check for HCU scram discharge valve leaking as indicated by elevated discharge piping temperatures for associated CRD. ☐
- **PERFORM** 2-TI-393 for control rods with high temperatures or failed thermocouples. ☐
- **REFER TO** 0-OI-55, 2-OI-85, 2-AOI-85-3. ☐
- **FLUSH** CRD to unblock restricted cooling water flow. **REFER** to 2-OI-85. ☐
- **DECLARE** the control rod, which is in alarm, "SLOW" as directed by 2-TI-393 per Tech Spec. Table 3.1.4-1 Note 1. ☐
- **RAISE** CRD Flow, as directed by Unit Supervisor, if required to keep the drives cool per "CRD Pump Operation At Elevated Flow" section of 2-OI-85. ☐

C. **IF** alarm is invalid, **THEN PERFORM** the following as directed by the Unit Supervisor: ☐
- **REFER TO** 0-OI-55. ☐
- **INITIATE** WO to determine cause of invalid alarm. ☐

Continued on Next Page

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

-----NOTES-----

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds) REACTOR STEAM DOME PRESSURE ≥ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

BASES (continued)

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6).

To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 185 x 7% ≈ 13) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations.

Table 3.1.4-1 is modified by two Notes, which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

Scram times can be adversely affected by high control rod drive temperatures. Temperatures over 350°F may result in a measurable delay in scram time response times for an otherwise normally performing CRD due to the potential for flashing of the hot water in the drive when the scram valves are opened. As a conservative measure, CRDs which have a

(continued)

BASES

LCO (continued)

temperature of greater than 350°F will either be classified as "slow" rods or an engineering evaluation can be performed.

This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods can be conservatively declared inoperable and not accounted for as "slow" control rods.

APPLICABILITY

In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS

A.1

When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analysis. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BFN UNIT 2	EVALUATION OF CRD TEMPERATURE ALARMS	2-TI-393 REV 0002

2.0 REFERENCES, Continued

2.3 Other Documents

2.3.1 GE SIL 173

2.3.2 Letter from C E Jurgens (GE) to F E Hartwig (TVA) dated 6/16/94

2.3.3 Technical Specifications, Section 3.1.4.

2.3.4 TVAN Writer's Guide

3.0 DEFINITIONS

Adjacent HCU - One immediately next to or behind an affected HCU.

4.0 PRECAUTIONS AND LIMITATIONS

4.1 GE SIL 173 discusses the potential effects on scram times for CRDs whose operating temperature is above 350 °F. Scram times can be increased by 0.150 to 0.500 seconds for a CRD operating between 350 °F and 525 °F. CRDs with temperatures above 350 °F are declared slow in accordance with Technical Specification Section 3.1.4 and are added to the population of rods scram time tested per 2-SR-3.1.4.1.

4.2 CRDs with open thermocouples will result in an upscale trip of 2-TA-85-7 and unknown temperature data displayed on ICS. CRDs with unknown temperatures and leaking scram outlet valves are declared slow in accordance with Technical Specification Section 3.1.4 and are added to the population of rods scram time tested per 2-SR-3.1.4.1.

4.3 CRDs which operate above the ICS high temperature alarm point (≥ 240 °F & ≤ 350 °F) should be evaluated by the system engineer. Possible causes of CRD high temperature should be investigated (reference GE SIL 173) and a work order initiated to correct the problem.

4.4 CRDs with high temperatures or open thermocouples shall have their alarms disabled in accordance with OPDP-4 and 0-OI-55 to prevent masking the annunciation of other valid high temperature alarms.

5.0 PREREQUISITES

None

6.0 EQUIPMENT

Infrared Temperature Gun or Equivalent

BFN UNIT 2	EVALUATION OF CRD TEMPERATURE ALARMS	2-TI-393 REV 0002

7.0 PROCEDURE

Section 7.1 of this procedure shall be performed by Operations to evaluate new CRD high temperature alarms as directed by 2-ARP-9-5A.

NOTE:

Appendix E summarizes actions to be taken based on CRD temperature conditions.

7.1 Evaluation of New Alarms

7.1.1 Record the CRD number on Appendix B, Evaluation of New CRD Temperature Alarms. Record the CRD temperature or indicate that the alarm is due to an open thermocouple input (upscale trip of 2-TA-85-7 and unknown temperature data displayed on ICS).

7.1.2 Determine if the scram outlet valve is leaking by comparing the scram outlet piping temperature for the affected CRD to the scram outlet piping temperatures of adjacent HCUs.

7.1.2.1 Measure the temperature of the scram outlet piping for the affected CRD using a infrared temperature gun or equivalent and record on Appendix B.

7.1.2.2 Measure the temperature of the scram outlet piping for two adjacent HCUs and record on Appendix B.

7.1.2.3 If the scram outlet piping temperature for the affected CRD is more than 4 °F greater than the scram outlet piping temperatures of adjacent HCUs,

7.1.2.3.1 Initiate a WO indicating that the scram outlet valve is leaking by and record WO number on Appendix B.

7.1.2.3.2 Declare the affected rod SLOW per Note 1 of Tech Spec Table 3.1.4-1, and contact Reactor Engineering to update the scram time data base to reflect that the rod is slow.

7.1.3 If the alarm is due to an open, or intermittent, thermocouple, disable the associated alarm on the temperature recorder per OPDP-4 and 0-OI-55.

7.1.4 If the CRD temperature is greater than 350 °F and it has been determined that the scram outlet valve is not leaking, perform Section 8.28 of 2-OI-85 to flush the CRD as directed by the System Engineer. If the CRD temperature returns to normal after flushing, document in the Comments field of Appendix B.

7.1.5 If the CRD temperature remains greater than 350 °F,

7.1.5.1 Declare the affected rod SLOW per Note 1 of Tech Spec Table 3.1.4-1, and contact Reactor Engineering to update the scram time data base.

7.1.5.2 Disable the associated alarm on the temperature recorder in accordance with OPDP-4 and 0-OI-55.

7.1.6 Forward this instruction to the CRD System Engineer.

Examination Outline Cross-reference:	Level	RO	SRO
295029 High Suppression Pool Wtr Lvl / 5	Tier #	1	------
EK3.03 (10CFR 55.41.10)	Group #	2	------
Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL:	K/A #	295029EK3.03	
• Reactor SCRAM	Importance Rating	3.4	------

Proposed Question: **# 25**

In accordance with the EOI Program Manual, which ONE of the following completes the statements?

With the reactor operating at 100% power, a Suppression Pool Water Level that is continuously **__(1)__** will result in entry into the **Action Required** area of Curve 4, "SRV Tail Pipe Limit."

Eventually, a Manual Scram is directed from the Suppression Pool Level Control (SP/L) leg of EOI-2, "Primary Containment Control," with the purpose being to attempt to lower **__(2)__** relative to the curve.

A. **(1)** rising **ONLY**
(2) Reactor Pressure

B. **(1)** rising **ONLY**
(2) Suppression Chamber Pressure

C. **(1)** rising **OR** lowering
(2) Reactor Pressure

D. **(1)** rising **OR** lowering
(2) Suppression Chamber Pressure

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (See attached excerpts) A continuously *rising* water level in the Suppression Pool will eventually exceed Curve 4 (approx 16.75 feet @ 100% power). EOI-1 entry in this case would be because level cannot be maintained in the safe area of Curve 4 for the SRV Tail Pipe limit. Bases documents provide the supporting insight on the necessity to attempt to lower reactor pressure relative to the curve.

B INCORRECT: First part correct, as detailed in 'A' above. Second part plausible, but incorrect, based upon the fact that the candidate is not given the curve itself and Suppression Chamber Pressure is a notable parameter associated with other EOI-related curves.

C INCORRECT: First part incorrect, in that rising will exceed the curve, but lowering will not exceed the curve. In the absence of the curve, the candidate could incorrectly determine that uncovering of the SRV tail pipe is also a factor in the Level Limit. Other related EOI curves (HCTL and PSP) can be impacted by a rising or lowering level. Thus, without a reference, this is very plausible. Second part correct, as detailed in 'A' above.

D INCORRECT: Both parts incorrect, but plausible, as detailed in 'B' and 'C' above.

RO Level Justification: Tests the candidate's knowledge of EOI Program Bases associated with entering EOI-1 from EOI-2 on a high Suppression Pool Water Level; and why the Scram, in particular, is directed. Because the candidate is not provided with a reference in this case or told the exact point at which EOI-1 entry is required, this question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-EOI-2, Rev. 0 (Attach if not previously provided)

EOI Program Man., Sect. 0-V-D, Rev. 0

OPL171.203, Rev. 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.11 (As available)

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

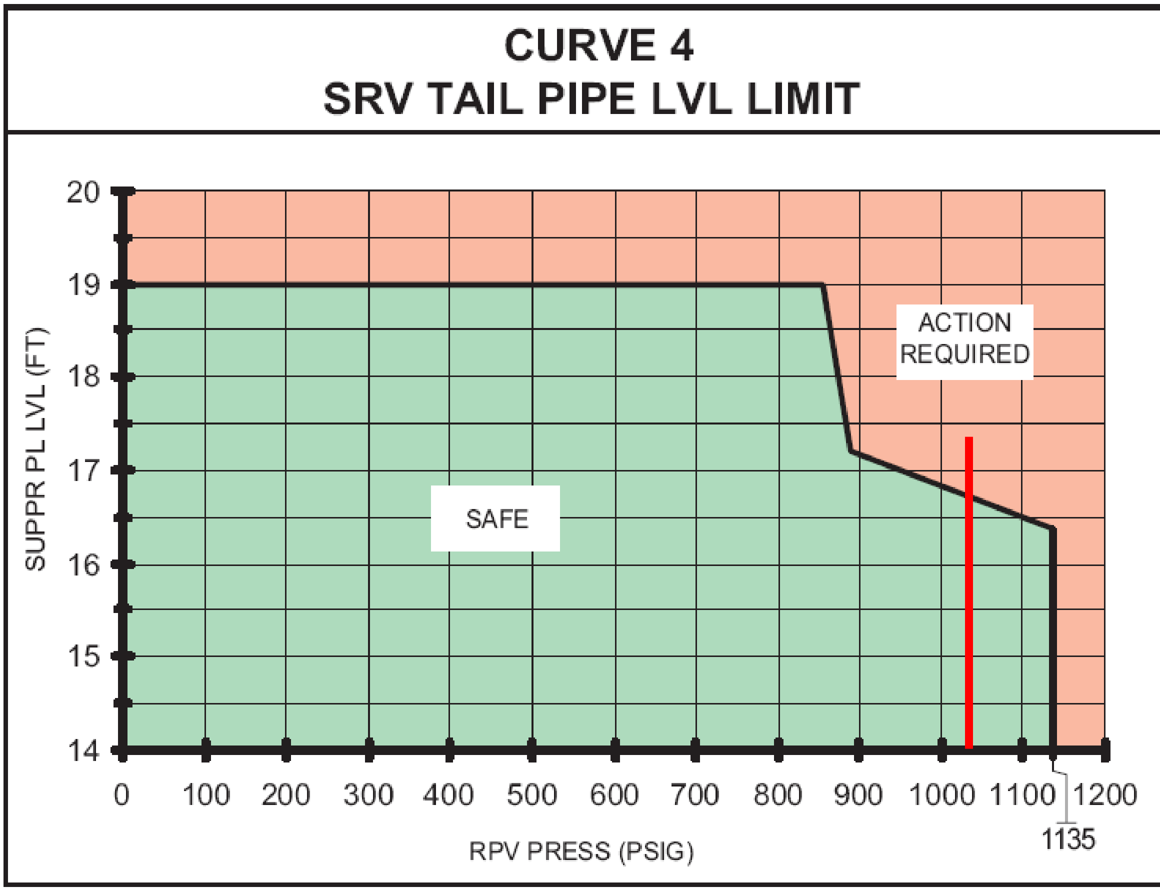
Question Cognitive Level: Memory or Fundamental Knowledge

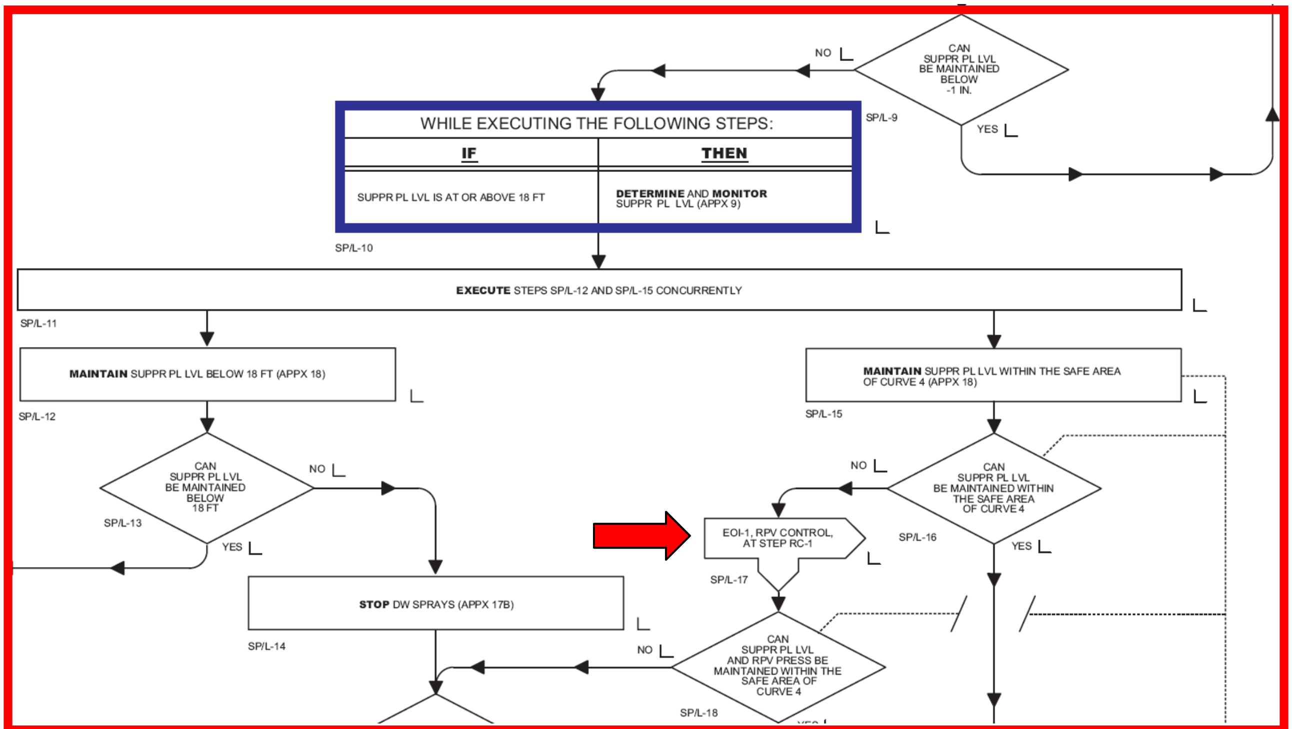
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

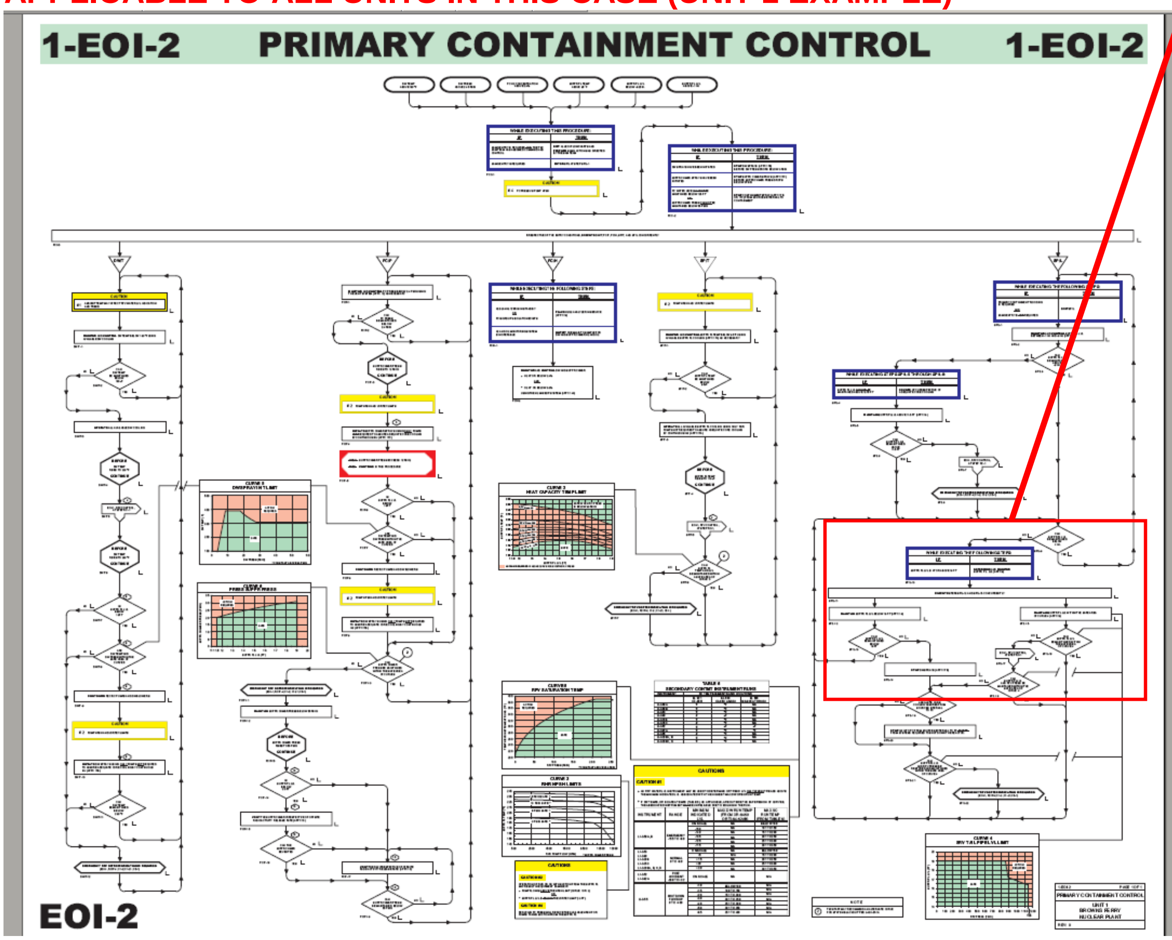
55.43

Comments:


CURVE 4
SRV TAIL PIPE LVL LIMIT
ACTION REQUIRED
SAFE
SUPPR PL LVL (FT)
20
19
18
17
16
15
14
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1135
RPV PRESS (PSIG)



APPLICABLE TO ALL UNITS IN THIS CASE (UNIT 1 EXAMPLE)



DISCUSSION: STEP SP/L-16, 17

This decision step has the operator evaluate both current and future efforts to control suppression pool level, in relation to the current value and trend of suppression pool level, to determine if level can be maintained in the safe area of MSRV Tail Pipe Level Limit Curve (Curve 4).

Engineering calculations have determined that MSRV system damage and containment failure will be prevented when suppression pool level and RPV pressure are maintained in the safe area of MSRV Tail Pipe Level Limit Curve. MSRV Tail Pipe Level Limit is defined to be the highest suppression pool water level at which opening an MSRV will not result in exceeding the capability of MSRV tail pipe, tail pipe supports, quencher, or quencher supports.

If suppression pool level can be maintained in the safe area of Curve 4, the operator returns to Step SP/L-9, continuing with efforts to restore level to normal.

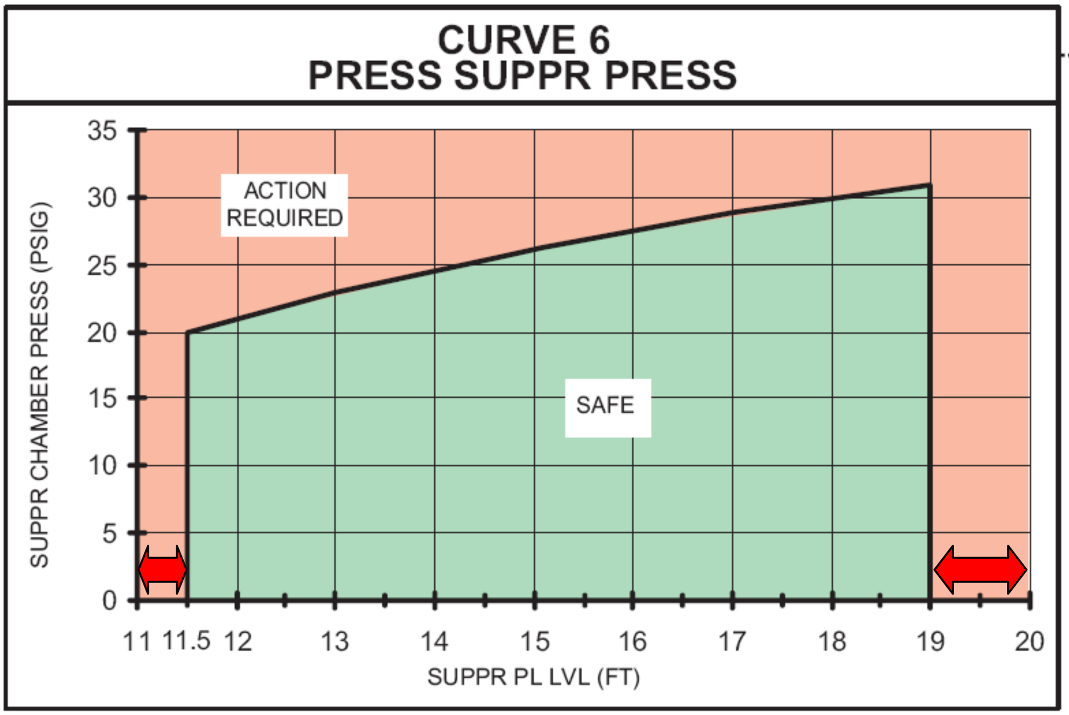
If the operator is unable to maintain suppression pool water level below MSRV Tail Pipe Level Limit, then since MSRV Tail Pipe Level Limit is a function of RPV pressure, an effort is made to control RPV pressure so that MSRV Tail Pipe Level Limit is not exceeded. The operator continues at Step SP/L-17.

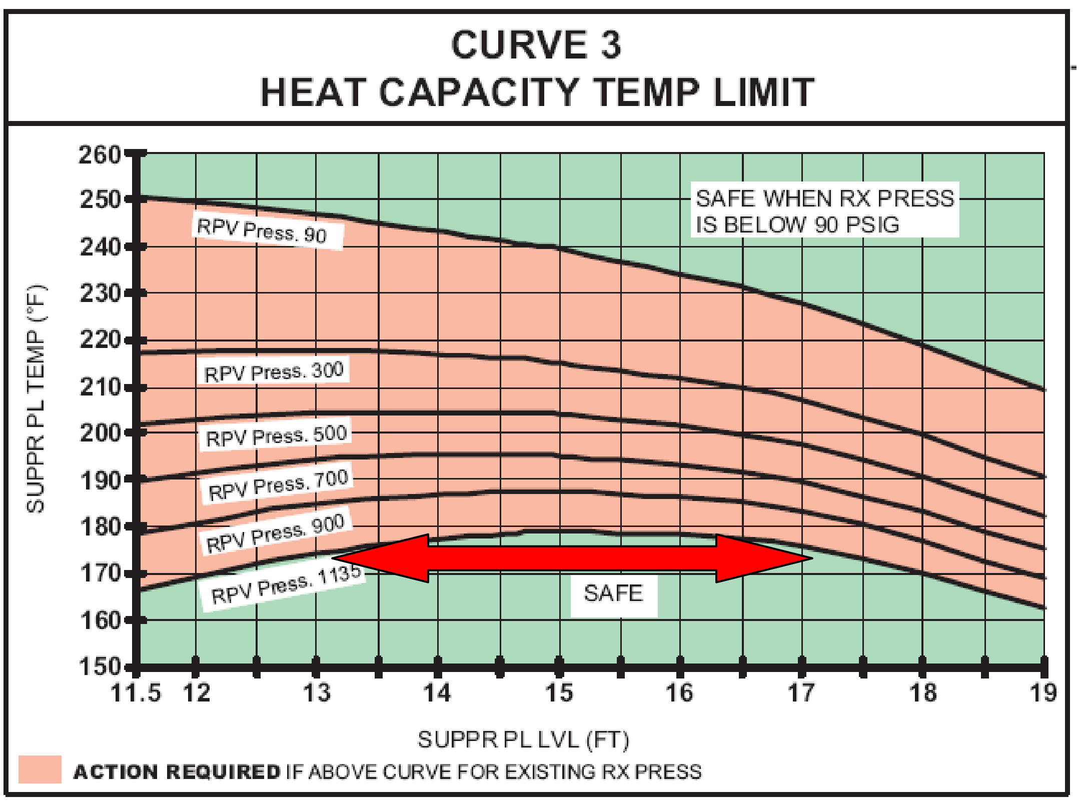
Step SP/L-17 is an execute concurrently step that requires the operator to enter EOI-1, RPV Control, at Step RC-1, and execute actions concurrently with those actions specified in EOI-2.

Entry into EOI-1, RPV Control, is directed so as to control RPV pressure relative to MSRV Tail Pipe Level Limit. EOI-1, RPV Control, requires initiation of a reactor scram if one has not yet been initiated. A scram will reduce core heat and steam generation rate in the RPV to decay heat levels (assuming the scram is successful), and will assist in maintaining RPV pressure below MSRV Tail Pipe Level Limit. Conditions requiring entry into EOI-2 do not necessarily require entry into EOI-1, RPV Control. Therefore, a scram may not yet have been initiated.

Direction to control RPV pressure within the safe area of MSRV Tail Pipe Level Limit Curve is provided in the RC/P Section of EOI-1, RPV Control. In addition, entry into EOI-1, RPV Control, must be made because it is through the RC/P Section that transfer is made to C2, Emergency RPV Depressurization, should it be necessary.

INFO SUPPORTING DISTRACTORS ‘C’ AND ‘D’





Examination Outline Cross-reference:	Level	RO	SRO
295034 Secondary Containment Ventilation High Radiation / 9	Tier #	1	------
EA1.02 (10CFR 55.41.7)	Group #	2	------
Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION:	K/A #	295034EA1.02	
• Process radiation monitoring system	Importance Rating	3.9	------

Proposed Question: **# 26**

On Unit 3, a Refueling Accident has occurred resulting in the following conditions:

- **ALL** Refuel Zone Radiation Monitor Channels are reading 65 mr/hr
- Control Room Ventilation Rad Monitors 90-259A/B are reading 155 cpm
- **ALL** Reactor Zone Radiation Monitor Channels are reading 68 mr/hr

Based on these conditions, which ONE of the following identifies the status of plant systems?

A. **NO** Standby Gas Treatment Systems are in service; **NO** CREV are in service.

B. **ALL** Standby Gas Treatment Systems are in service; **NO** CREV are in service.

C. **NO** Standby Gas Treatment Systems are in service; **ONLY** the selected CREV is in service.

D. **ALL** Standby Gas Treatment Systems are in service; **ONLY** the selected CREV is in service.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Radiation Levels at the Refuel Zone Radiation Monitors and Reactor Zone Radiation Monitor are below the automatic initiation set point of 72 mr/hr for both CREV and SGTS Initiation. Control Room Ventilation Rad Monitors 90-259A/B are reading 155 cpm which is above the ALERT alarm set point but is less than the initiation set point of 221 cpm with 2 minute time delay.

B INCORRECT: Standby Gas Treatment Systems will not be in service. Second part is correct.

C INCORRECT: First part is correct. Second part is incorrect, no CREV will be in service.

D INCORRECT: Both parts are incorrect as explained above.

RO Level Justification: Candidate must recognize automatic initiation conditions to demonstrate ability to monitor and operate process radiation monitoring as it applies to Secondary CTMT Ventilation High Radiation. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.067 Rev. 16 (Attach if not previously provided)
OPL171.018 Rev. 10
3-OI-90 Rev. 55

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.067 V.B.2 (As available)

Question Source: Bank #
Modified Bank # 0801 #10 (Note changes or attach parent)
New

Question History: Last NRC Exam BFN 0801

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments: Conditions in the stem were modified such that one of the distracters became the correct answer making this a significantly modified question.

OPL171.067
Revision 16

a. Automatic start signals are:

(1) High radiation of 221 cpm above background + 2 Min TD (270 cpm Tech Specs) in air inlet ducts to Control Room from (Radiation monitor RM 90-259A Units 1 & 2, Radiation monitor RM 90-259B Unit 3). Either monitor starts selected CREV unit.

(2) Reactor zone ventilation systems radiation high $\geq$72 MR/hr

(3) Refuel zone ventilation systems radiation high $\geq$72 MR/hr

(1) Low reactor water level at +2 inches above instrument zero

(2) High primary containment pressure $\geq$2.45 psig

OPL171.018
Revision 10

2. Initiation Signals — Obj. V.E.3, Obj. V.C.6

   a. System automatically starts with one or more of the following signals.

   These signals on any unit, will start all three SGT trains when the control switch is in AUTO
   .

      (1) High drywell pressure (2.45 psi)

      (2) Low reactor water level: +2.0"

      (3) High radiation, Reactor Zone Ventilation System (72 mRem/hr) 2 monitors in the same channel or 2 monitors downscale (RE-142, -143) one in each channel.

      (4) High radiation, Refueling Zone (72 mRem/hr) 2 monitors in the same channel or 2 monitors downscale (RE-140, -141) one in each channel.

   b. All 3 SGT trains auto-start on initiation and run until manually stopped. — TP-3 and 4

BFN Unit 3	Radiation Monitoring System	3-OI-90 Rev. 0055 Page 8 of 50

3.0 PRECAUTIONS AND LIMITATIONS

A. The following Radiation Monitoring subsystems initiate the listed automatic actions and isolations on high radiation trip signals:

1. Main Steam Line (3 times normal full-load background radiation).

   a. Mechanical Vacuum Pump trip and suction valve isolation.

2. Off-Gas Post-Treatment

   a. High or High-High - opens Adsorber Inlet Valve, 3-FCV-66-113A, and closes Adsorber Bypass Valve, 3-FCV-66-113B, if 3-XS-66-113 is in AUTO.

   b. High-High-High - closes Off-Gas System Isolation Valve, 3-FCV-66-28 (5-second time delay).

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.)

5. Control Room Ventilation Monitoring (221 cpm above background high activity <u>or</u> two channels downscale/inop)

   a. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)

B. Abnormal or significant rises in radiation levels are required to be reported to the Unit Supervisor.

BFN Unit 3	Radiation Monitoring System	3-OI-90 Rev. 0055 Page 43 of 50

Illustration 4
(Page 4 of 9)

Control Room Radiation Monitors 0-90-RM-259A and 259B

2.0 COMPONENTS (REFER TO FIGURE 1)

A. **DISPLAY** SCREEN (shown in detail in Figure 2)

1. Contains parameters and information on the first line with a parameter driven bar graph on the second line. The parameter and proper units are displayed on the first line. The information updates every couple of seconds.

2. The Alarm Status Indicator in the top right corner of the display screen will show alarm conditions with a flashing letter. Listed below are the possible flashing letters that can be received:

   a. "H" - indicates high alarm

   b. "A" - indicates alert alarm

   c. "?" - indicates alarm on channel other than the one currently displayed.

   d. "D" - dose accumulation exceeds the dose setpoint.

   e. "S" - count sum accumulation exceeds the setpoint.

B. **HIGH** red light

1. Normally extinguished. Illuminates when high radiation is detected (approximately 221 cpm above background) in the free air supply duct to the Control Room.

2. When this condition is reached this light will illuminate, the NORM green light will extinguish, a flashing "H" will appear in the top right corner of the display, and an audible alarm will sound.

C. **ALERT** amber light

Normally extinguished. Illuminates when high radiation is detected in the free air supply duct to the Control Room (approximately 150 cpm depending on background). When this condition is reached this light will illuminate, the NORM green light will extinguish, a flashing "A" will appear in the top right corner of the display, and an audible alarm will sound.

D. **NORM** green light

Normally illuminated, indicating normal operation.

BFN 0801 NRC #10

On Unit 3, a Refueling Accident has occurred resulting in the following conditions:

- **ALL** Refuel Zone Radiation Monitor Channels are reading 65 mr/hr
- Control Room Ventilation Rad Monitors 90-259A/B are reading 100 cpm
- **ALL** Reactor Zone Radiation Monitor Channels are reading 75 mr/hr

Based on these conditions, which ONE of the following identifies the status of plant systems?

A. **NO** Standby Gas Treatment Systems are in service; **NO** CREV is in service.

B. **ALL** Standby Gas Treatment Systems are in service; **NO** CREV is in service.

C. **NO** Standby Gas Treatment Systems are in service; **ONLY** the selected CREV is in service.

D. **ALL** Standby Gas Treatment Systems are in service; **ONLY** the selected CREV is in service.

ANSWER: **D**

Examination Outline Cross-reference:	Level	RO	SRO
295036 Secondary Containment High Sump/Area Water Level / 5	Tier #	1	------
EA2.01 (10CFR 55.41.10)	Group #	2	------
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:	K/A #	295036EA2.01	
• Operability of components within the affected area	Importance Rating	3.0	------

Proposed Question: **# 27**

A Condensate Transfer System leak spraying on Unit 1 Loop II Core Spray Room Cooler has resulted in the following:

- Loop II Core Spray Room Cooler has tripped **AND** will **NOT** reset
- CORE SPRAY LOOP II PUMP ROOM FLOOD LEVEL HIGH, (1-9-4C, Window 31), is in alarm

Which ONE of the following completes the statements?

Loop II Core Spray __**(1)**__operable.

Entry into 1-EOI-3, "Secondary Containment Control," __**(2)**__ required.

A. **(1)** is
(2) is

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

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

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

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – See explanation B. Part 2 correct - See explanation C.

B INCORRECT: Part 1 incorrect - Plausible in that the Core Spray System will function with the CS Room Cooler out of service. Part 2 incorrect – Plausible in that not all alarms associated with degrading conditions occur at the EOI Entry level.

C **CORRECT**: Part 1 correct - Per TR 3.5.3, the equipment area cooler associated with each Core Spray Pump must be operable when that Core Spray System is considered to be operable. With the area cooler inoperable, Condition A requires the pumps served by the cooler be declared inoperable immediately. Part 2 correct - The Reactor Building floor drain sump High-High level (66") is an EOI-3 entry condition.

D INCORRECT: Part 1 correct and Part 2 incorrect as explained above.

Justification: Candidate must recognize the impact of the leakage in the area and associated loss of Loop I Core Spray Room Cooler on the associated Core Spray System operability. Since question can be answered knowing ≤ 1 hour Tech Spec Actions, this is an RO question. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL 171.204 Rev. 7 (Attach if not previously provided)

U1 TRM, Section 3.5.3 Rev. 0

1-ARP-9-4C Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.204 V.B.2 (As available)

OPL171.045 V.B.4 / 6

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

A secondary containment floor drain sump water level above maximum normal operating level is an indication that steam or water may be discharging into secondary containment. Maximum normal operating floor drain sump water level is defined to be the highest value of secondary containment floor drain sump water level expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

f. Area water levels above 2 inches

Secondary containment area water level above maximum normal operating level is an indication that steam or water may be discharging into secondary containment. Maximum normal operating secondary containment water level is defined to be the highest value of secondary containment area water level expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

g. Area radiation level above the maximum normal operating value of Table 4.

Secondary containment area radiation level above the maximum normal operating value of Table 4 is an indication that water from a primary system, or from a primary to secondary system leak, may be discharging into secondary containment. Maximum normal operating secondary containment area radiation level is defined to be the highest value of secondary containment areas radiation expected to occur during normal plant operating conditions with all directly associated support and control system functioning properly.

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0018 Page 39 of 43

CORE SPRAY LOOP II PUMP ROOM FLOOD LEVEL HIGH 1-LA-77-25B 31

(Page 1 of 1)

Sensor/Trip Point:

1-LS-77-25B ≥ 2 Inches of Water on the Floor

Sensor Location: Sensor is located near the floor of the northeast core spray pump room.

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

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to VISUALLY CHECK the northeast core spray pump room. ☐

B. **IF** alarm is valid, **THEN** **PERFORM** the following:

- **VERIFY** the floor drain sump pumps running. ☐
- **CHECK** the floor drains for proper drainage. ☐
- **IF** possible, **THEN** **DETERMINE** the source of the leak and the leak rate. ☐
- **ENTER** 1-EOI-3 Flowchart. ☐

NOTE
The floor drain and equipment drain sump pumps may need to be locked out to prevent Radwaste flooding.

- **NOTIFY** Radwaste operator to monitor floor drain collector tank and waste collector tank levels. ☐
- **NOTIFY** Radiation Protection. ☐

References: 0-47E610-77-1 FSAR Sections 13.6.2 and F.7.15

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Equipment Area Cooler inoperable.	A.1 Declare the pump(s) served by that cooler inoperable. (Refer to applicable TS and TRM LCOs)	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.5.3.1	Verify each Equipment Area Cooler automatically starts when the associated Core Spray or RHR pump is started.	92 days

Example of elevated parameter alarm to support distractors
Drywell Pressure Abnormal alarms before EOI Entry of 2.45 psig

BFN Unit 1	Panel 9-5 1-XA-55-5B	1-ARP-9-5B Rev. 0016 Page 35 of 42

DRYWELL PRESSURE ABNORMAL 1-PA-64-56	31

(Page 1 of 1)

Sensor/Trip Point:

1-PS-064-0056E	1.65 psig rising
1-PS-064-0056F	0.22 psig lowering

Sensor Location: 1-LPNL-925-0005B
Elevation 593
Column No. S-R3

Probable Cause:

A. Drywell Δ P air compressor failure.
B. Loss of RBCCW.
C. Breach of primary containment.
   1. Drywell vent valves open or leaking.
   2. Drywell vacuum breaker open or leaking.

D. LOCA.
E. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** the alarm using multiple indications. ☐
B. **IF** RBCCW has been lost, **THEN** **REFER TO** 1-AOI-70-1. ☐
C. **REFER TO** 1-AOI-64-1. ☐

References: 1-45E620-6-2 1-47E610-64-1 1-730E915-17

Examination Outline Cross-reference:	Level	RO	SRO
203000 RHR/LPCI: Injection Mode	Tier #	2	------
A1.01 (10CFR 55.41.5)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including:	K/A #	203000A1.01	
• Reactor water level	Importance Rating	4.2	------

Proposed Question: **# 28**

Given the following conditions:

- Unit 2 has experienced a Loss of Coolant Accident (LOCA)
- Drywell Sprays are required in accordance with "Primary Containment Control," 2-EOI-2

Which ONE of the following plant conditions must exist prior to opening **BOTH** the Residual Heat Removal (RHR) SYS I Inboard **AND** Outboard Drywell Spray Valves?

A. Reactor Level must be greater than (-) 155 inches (Emergency Range) with **ONLY** the CONT SPRAY VLV SEL SWITCH in SELECT.

B. Reactor Level must be greater than (-) 162 inches (Post Accident Range) with **ONLY** the CONT SPRAY VLV SEL SWITCH in SELECT.

C. Reactor Level is greater than (-) 183 inches (Post Accident Range) with **ONLY** the CONT SPRAY VLV SEL SWITCH in SELECT.

D. Reactor Level is less than (-) 200 inches (Post Accident Range) with **ONLY** the 2/3 CORE HEIGHT KEYLOCK BYPASS SWITCH in BYPASS.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: With RPV level > -155 inches adequate core cooling is assured by submergence and therefore spraying containment is allowed. If candidate determines that top of active fuel verses 2/3 core height is necessary, it becomes plausible. Also level must be above -183" not -155"

B INCORRECT: With RPV level > -162 inches (top of active fuel) adequate core cooling is assured. If candidate determines that top of active fuel versus 2/3's core height is necessary, it becomes plausible. Also level must be above -183" not -162"

C **CORRECT**: OPL171.044, TP 33 shows that under the conditions only the CONT SPRAY VL V SEL SWITCH in SELECT is required to allow spraying containment. Contact K 14A1B is closed if level is above 2/3's core height (-183") as referenced on the Post Accident range (-268" to +60"). Contact K61A1B is closed due to an accident signal (level <-122" on the Wide range level instruments). Therefore to complete the logic only the SELECT switch is required to be closed which energizes relays K58A1B (Seal In) and K50A/B which allows spraying containment.

D INCORRECT**:** With RPV level < -200 inches, you are below 2/3's core height which requires both the CONT SPRAY VL V SEL SWITCH in SELECT and 2/3 CORE HEIGHT KEYLOCK BYPASS switch in BYPASS.

Justification: This question satisfies the KIA statement by requiring the candidate to use specific values of reactor water level to determine the conditions which allow diverting RHR from a LPCI Injection lineup to containment control. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-74 Rev. 150 (Attach if not previously provided)

OPL171.044 Rev. 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 V.B.10 (As available)

Question Source: Bank # BFN 0610 #1

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam BFN 2008

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

INSTRUCTOR NOTES

e. Reactor vessel water level -183 inches indicated on Post Accident Flooding Range

(1) LIS used to provide a level permissive signal to Containment Cooling valves based on level inside the shroud. Permits containment spray only if level is >2/3 core height with LOCA signal present.

LIS-3-52 & 3-62A

(2) A keylock bypass switch can bypass the 2/3 core height interlock.

f. Drywell pressure 1.96 psig

(1) Pressure switches used to provide pressure permissive signal to containment spray valves. Permits opening of containment spray only if pressure is significant in containment after accident.

(2) DWP decreasing to less than 1.96 psig will automatically close the spray valves if an accident signal is present.

Monitoring plant systems is critical during a transient when conditions change rapidly. SER 03-05

g. Reactor vessel low water level +2 inches (Lvl 3)

Level switches used to initiate: PCIS Group 2 isolation of Shutdown Cooling, CS System, RHR SDC isolation valves, RHR inbd isolation valves, RWCU isolation and a reactor scram.

h. RHR pressure high permissive > 100 psig

TP-1 and 2

Pressure switches on discharge of each RHR pump indicate pump running to ADS.

i. Reactor low pressure permissive < 230 psig

(1) Pressure switch used to automatically isolate Reactor Recirculation Sys discharge valve with LPCI initiation signal present.

FCV-68-3/79

(2) Same RPV pressure logic setup as with LPCI logic

- Only Respective Divisional logic will close the valve.

2. Valve Interlocks

Obj. V.B.10

a. RHR pump suction from Suppression Pool

Obj. V.C.5
(74-1, 74-12, 74-

(1) No automatic closing or opening interlocks

INSTRUCTOR NOTES

(2) Interlocks prevent normal opening if in-line Suppression Pool Spray valve (74- 58; 74-72) not full closed

Obj. V.C.5

(3) Automatically closes on a LPCI signal

(4) The in-line valve interlock and/or the automatic closure signal can be bypassed if the following exist:

(a) Reactor level >-183 inches and LPCI initiation signal present and Select-Reset switch to SELECT position.

(b) Reactor level interlock and LPCI initiation signal may be bypassed by use of keylock bypass switch (XS 74-122/130)

Amber light above keylock switch indicates switch in B/P position

(5) Loop I/II respective valve can not be opened if either pump shutdown cooling suction valve not full closed.

(6) Emergency position at breaker bypasses in-line valve interlock and LPCI closure signal. Valves can only be operated at the breaker.

Obj. V.D.8

U2 & U3-74-57

U1-74-71

(7) EMERGENCY position does not bypass the interlock with the SDC valves.

(8) Separate bypass switch allows bypassing interlock from Valves 74-2/13 (74-25/36)

Local switch for flushing.

(9) Control Circuit

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0150 Page 23 of 449

3.6 Interlocks (continued)

8. To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton is required to be depressed after either of following occur:

    a. Isolation signal has been reset **OR**

    b. 2-FCV-74-47 **OR** 2-FCV-74-48 is fully closed.

9. If after a GROUP II Isolation, RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67) is given an OPEN signal prior to depressing the RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET 2-XS-74-126(132), then the valve will trave full open and full close unless given a close signal prior to traveling full open.

10. The RHR spray/cooling valves, 2-FCV-74-57(71), receive an auto closure signal in the presence of a LPCI initiation signal and they are interlocked to prevent opening if the in-line torus spray valve, 2-FCV-74-58(72), is not fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

    a. Reactor level is >2/3 core height **AND**

    b. LPCI initiation signal is present **AND**

    c. The select reset switch is in the SELECT position.

    The requirements for >2/3 core height and a LPCI initiation signal may be BYPASSED using the keylock bypass switch, 2-XS-74-122/30.

11. If primary containment cooling is desired with reactor level at <2/3 core height, the keylock bypass switch is required to be placed in BYPASS <u>before</u> the select reset switch is placed in SELECT to ensure relay logic is made up.\

12. The RHR torus spray valves, 2-FCV-74-58(72), have the same in-line valve interlocks as those outlined in 3.6A 10 for the torus spray/cooling valves. Additionally these valves have an interlock preventing opening unless drywell pressure is ≥1.96 psig which cannot be bypassed.

13. The RHR torus cooling/test valves, 2-FCV-74-59(73), receive an auto closure signal in the presence of a LPCI initiation signal. Auto closure may be bypassed by the same conditions/actions outlined in Step 3.6A.10

1. RO 203000A1.01 001/C/A/T2G1/RHR/DWSP/1/203000A1.01/4.2/4.3/RO/MODIFIED 11/17/07

Given the following conditions:

- Unit 2 has experienced a Loss of Coolant Accident (LOCA).
- Drywell sprays are required in accordance with 2-EOI-2 flowchart.

Which ONE of the following plant conditions must exist prior to opening BOTH the Residual Heat Removal (RHR) SYS I Inboard AND Outboard Drywell Spray Valves?

A. RPV level must be greater than (-)155 inches (Emergency Range) with ONLY the the CONT SPRAY VLV SEL SWITCH in SELECT.

B. RPV level must be greater than (-)162 inches (Post Accident Range) with ONLY the CONT SPRAY VLV SEL SWITCH in SELECT.

C.✓ RPV level is greater than (-)183 inches (Post Accident Range) with ONLY the CONT SPRAY VLV SEL SWITCH in SELECT.

D. RPV level is less than (-)200 inches (Post Accident Range) with ONLY the 2/3 CORE HEIGHT KEYLOCK BYPASS SWITCH in BYPASS.

K/A Statement:

203000 RHR/LPCI: Injection Mode
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Reactor water level

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific values of reactor water level to determine the conditions which allow diverting RHR from a LPCI Injection lineup to containment control.

Level of Knowledge Justification: This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

References: 2-OI-74, OPL171.044, TP-33

0610 NRC RO Exam
modified from OPL171.044 #22
REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

Examination Outline Cross-reference:	Level	RO	SRO
203000 RHR/LPCI: Injection Mode	Tier #	2	------
A1.04 (10CFR 55.41.5)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including:	K/A #	203000A1.04	
• System pressure	Importance Rating	3.6	------

Proposed Question: **# 29**

Unit 1 was operating at 100% Reactor Power when a LOCA occurred resulting in the following:

- Reactor Pressure is 405 psig
- Reactor Level is (-) 140 inches

Which ONE of the following completes the statement?

The RHR SYS I LPCI OUTBD INJECT VALVE, 1-FCV-74-52, is **__(1)__** **AND** the RHR SYS I MIN FLOW VALVE, 1-FCV-74-7, is **__(2)__**.

A. **(1)** OPEN
(2) CLOSED

B. **(1)** CLOSED
(2) CLOSED

C. **(1)** OPEN
(2) OPEN

D. **(1)** CLOSED
(2) OPEN

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: Part 1 incorrect – Plausible in that the Reactor Pressure associated with the LPCI Injection Valve interlock could be confused with the RHR Shutoff Head Pressure of 320 psig. Part 2 incorrect – Plausible in that Reactor Pressure is less than Shutoff Head for Condensate Booster Pumps which have similar pump design criteria and could therefore be confused with RHR Shutoff Head Pressure.

C **CORRECT**: Part 1 correct – Reactor Water Level is less than RHR LPCI Initiation Level of (-) 122 inches and Reactor Pressure is less than LPCI Injection Valve interlock of 450 psig. Therefore, the LPCI injection valves are open. Part 2 correct – Reactor Pressure is greater than the RHR Pumps Shutoff Head of 320 psig. With system flow less than 5800 gpm, the LPCI Minimum Flow Valves will be open.

D INCORRECT: Part 1 incorrect – See Explanation B. Part 2 correct – See Explanation C

Justification: To successfully answer, candidate must have knowledge of RHR System response in LPCI Mode to demonstrate ability to predict and/or monitor changes in parameters associated with system pressure.

Technical Reference(s): OPL171.044 Rev. 17 (Attach if not previously provided)
0-TI-394 Rev. 4 / 1-OI-74 Rev. 69

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044_V.B.18 (As available)

Question Source: Bank # La Salle 08 #14
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam La Salle 2008

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments: Question identified as a Bank question although it has been modified from original question associated with Core Spray System at La Salle. Adapted to RHR LPCI for BFN. Does not meet the criteria of a significantly modified question since no distracters were significantly changed.

BFN Unit 1	Residual Heat Removal System	1-OI-74 Rev. 0069 Page 13 of 332

3.3 LPCI

A. LPCI will initiate on any of the following signals:

1. Reactor Vessel low-low-low water level (-122 inches)(Level 1).

2. High Drywell Pressure (2.45 psig) with low Reactor Vessel Pressure (450 psig).

B. Manually stopping an RHR pump after LPCI initiation disables automatic restart of that pump until the initiation signal is reset. The affected RHR pump can still be started manually.

C. Upon an automatic LPCI initiation with normal power available, RHR Pump 1A will start immediately, THEN 1B, 1C, 1D sequentially start at 7 second intervals. Otherwise, all RHR pumps start immediately once diesel power is available (and normal power unavailable).

D. As soon as practicable after an RHR pump(s) auto start, the corresponding control room handswitch should be placed in normal-after-start position to ensure the handswitch disagreement light(s) and pump tripped annunciator(s) function as designed.

E. If RECIRC PUMP 1A(1B) DISCHARGE VALVE, 1-FCV-068-0003(0079) is declared inoperable while the valve is OPEN, the associated LPCI subsystem is required to be declared INOPERABLE. (Tech Spec Bases SR 3.5.1.5)

3.4 Shutdown Cooling

A. Prior to initiating Shutdown Cooling, RHR should be flushed to Radwaste until conductivity is less than 2.0 micromho/cm with < 0.1 ppm chlorides (unless directed otherwise by 1-AOI-74-1, Loss of Shutdown Cooling). If CS&S has been aligned as the keep fill source, a chemistry sample should be requested and results analyzed to determine if flushing is required.

B. Reactor water quality requirements of Technical Requirements Manual Section 3.4.1 apply to the RHR Loop to be placed in Shutdown Cooling. In order to ensure optimum coolant chemistry, analysis should meet CI-13.1 Cold Shutdown Limits which are more conservative based on EPRI Guidelines.

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3.6 Interlocks

A. The RHR system is equipped with pump and valve interlocks to assure the following:

1. All RHR pump flow is directed to the LPCI injection path during ECCS initiation.

2. Protection of low pressure piping from high reactor pressures.

3. A pump suction path is fully open prior to pump start.

4. Suction Path Interlocks:

   a. An RHR pump will not start or will trip, if running, unless its corresponding torus suction valve is open or the SDC suction valve and the SDC suction supply valves, 1-FCV-74-47 and 48, are open.

   b. The torus suction valves cannot be opened unless the corresponding pumps SDC suction valve is fully closed.

   c. The SDC suction valves cannot be opened unless the corresponding pumps Torus suction valve is fully closed.

5. RHR Minimum Flow Valve Interlocks:

   a. The RHR minimum flow valves auto close if both pumps in the corresponding loop are off and either pump's SDC suction valve is open.

   b. The minimum flow valves open and close on a low flow of 5800 gpm after a 10 second TD. The tolerance of the flow switch may allow the setpoint of the min flow valve to be from 4500 gpm to 7000 gpm. Operation outside of this expanded range should be investigated. The analytical limit as listed in design criteria BFN-50-7074 is 11000 gpm for min flow valve closure. (MD-Q0074-960020 Rev 01)

   c. Placing the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), in INHIBIT, will simulate a high flow and the minimum flow valve will remain closed regardless of flow.

3.6 Interlocks (continued)

d. Opening RHR SYSTEM I(II)MIN FLOW VALVE, 1-HS-74-7A(30A), from 1-PNL-9-3, with the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), in INHIBIT, will cause the minimum flow valves to travel full open and full close unless the RHR SYSTEM I(II) MIN FLOW VALVE, 1-HS-74-7A(30A) is placed in closed position to break the OPEN seal in contacts.

e. [II/C] Local operation of the RHR minimum flow valves will bypass the intended function of the Minimum Flow Inhibit switch and can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFPER941099]

f. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), with the respective RHR loop in standby readiness, can cause inadvertent damage to that loop RHR pump(s) should RHR pump(s) auto start. [BFA-890790003P]

g. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), with the respective RHR loop in Shutdown Cooling, can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFA-890790003P]

6. The RHR outboard LPCI injection valves, 1-FCV-74-52(66), have throttling capability. They receive an auto open signal in the presence of a LPCI initiation signal when Reactor pressure is ≤450 psig and are interlocked open under these conditions until the appropriate LPCI SYS I(II) OUTBD INJ VLV BYPASS SEL keylock switch, 1-HS-74-155A(155B) is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is >450 psig if its in-line companion valve 1-FCV-74-53(67) is not fully closed.

7. If Unit 1 reactor pressure exceeds 100 psig or a Group II isolation occurs on Unit 1 while Shutdown Cooling is in operation, the following will occur for the given condition:

(100 psig) RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 1-FCV-74-47 and 1-FCV-74-48, close, thus tripping operating Unit 1 RHR Pumps.

(Group II) RHR SYS I and II LPCI INBD INJECT VALVEs, 1-FCV-74-53 and 1-FCV-74-67, close and RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 1-FCV-74-47 and 1-FCV-74-48, close, thus tripping operating Unit 1 RHR Pumps.

4. Test Mode — TP-4
   a. Manual operation - can be performed during normal plant operation.
   b. Test valve sized for maximum flow (same valve as Suppression Pool Cooling).
   c. Isolates on LPCI injection signal.
5. LPCI Mode — TP-3
   a. Automatic, accident mode of RHR — Obj. V.B.18
   b. Initiated by a "one-out-of-two-twice" logic.
      (1) Low vessel water level, Level 1(-122 inches) , OR
      (2) High drywell pressure ($\geq$2.45 psig) and reactor pressure permissive ($\leq$450 psig)
   c. This signal starts the RHR pumps.
   d. Reactor Recirculation pumps are tripped at -45 inches. (Level 2)
   e. All valves not needed for LPCI injection automatically isolate and are interlocked shut as previously described.
   f. Sends permissive to ADS when RHR pump pressure is sensed >100 psig.
   g. Minimum flow valves open if injection flow in loop is <5800. Automatically shut as injection valves open and injection flow increases to >5800 gpm. — IE Bulletin 86-01
   h. Reactor pressure decreases through the break in conjunction with either HPCI or ADS.
      (1) As reactor pressure decreases to <450 psig the LPCI outboard injection valves open and are interlocked full open. They can then be throttled. Interlocking the injection path isolated until less than 450 psig protects the RHR piping from over-pressurization.
      (2) As pressure reaches <230 psig both Reactor Recirculation pump discharge valves shut and interlock shut.

(3) As pressure reaches ~300 psig the RHR System injects into both Reactor Recirculation System loops. Flow will increase until at ~30 psid Suppression Pool pressure to reactor pressure flow will reach ~10,000 gpm per pump.

(4) Even with a failure of the most limiting component, reflood time on design basis LOCA is sufficient to keep Peak Clad Temperature <2200°F when combined with CS pumps. — Note the RHR and CS system redundancy.

(a) Most limiting failure is LPCI injection valve failing closed, which eliminates 2 RHR pumps

(b) With the most limiting failure there are still four CS pumps and two RHR pumps available for injection.

6. Miscellaneous

a. Refueling — TP-9 and 14

Supplements the fuel pool cooling capability.

b. Unit cross-ties — TP-1 and 2

MOVs permit Unit 1, System 2 to take suction from Unit 2, System 1 or vice-versa. Also U2, System 2 can be aligned with U3, System 1.

c. Condensate Storage Tank (CST) supply — TP-1 and 2

Each RHR pump has a normally locked closed supply from the CST for flushing, testing, and suction supply (EOI Appendices 10C and 10D).

G. Normal RHR Standby Condition — TP-1 and 2, Obj. V.D.5

Valve Lineup:

1. Suppression pool suction – open
2. RHR pumps off
3. Minimum flow valve open
4. Test valve closed
5. Heat exchanger manual outlet open
6. Service water to heat exchanger (if in use)

RPV AND CONTAINMENT EOI/SAMG SUPPORT INFORMATION

INJECTION SOURCES				
System	Pumps	Capacity (gpm)	Shutoff Head (psig)	Motive Force
HPCI	1	5,000 (150-1150 psig)	1240	Steam
RCIC	1	600 (150-1150 psig)	1240	Steam
CRD	2	98 each	1640	Motor
Feedwater	3	11,200 each	1210	Steam
Condensate Booster	3	10,800 each (300 psig)	410	Offsite Power
Core Spray	2 loops	6250 per loop (105 psig)	330	Motor
RHR (LPCI)	4	10,000 each (0 psig)	320	Motor
Condensate	3	10,830 each (103 psig)	130	Offsite Power

Severe Accident Management Guidelines	
Technical Support Guidelines	
0-TI-394 Illustration 1	Rev 4

Question 14 LaSalle 07-01 NRC ILT Exam

Safety Function: 3.2 Reactor Water Inventory Control

System: 209001 LPCS

K/A: A1.04 - Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Reactor pressure.

I.R.: 3.7 / 3.7 10CFR 55 Content Part 41.5 / 45.5

License Level: RO Cognitive Level: High

Objective: 063.00.05 PRA: No

Question Exposure: 07-01 NRC ILT Exam

Source: New Question developed for the 07-01 NRC ILT Exam

Stem:

Unit-1 was operating at 100% reactor power when a LOCA occurred resulting in the following:
- The reactor scrammed
- RPV pressure has dropped to 480 psig
- RPV level is -140 inches and steady
- HPCS and RCIC are injecting to the RPV

Based on these conditions, the LPCS Injection Valve is (1)__________ and the LPCS Min Flow Valve is (2)__________.

A. (1) OPEN and
(2) OPEN

B. (1) OPEN and
(2) CLOSED

C. (1) CLOSED and
(2) OPEN

D. (1) CLOSED and
(2) CLOSED

Correct Answer A: (1) OPEN and (2) OPEN

Explanation: LPCS will initiate and the pump will start when RPV level is <-129 inches. The LPCs Injection Valve will open when reactor pressure drops below the 500 pound interlock with an initiation signal present. The LPCS Min Flow Valve will be OPEN until RPV pressure falls below pump shutoff head (440 psig), and injection into the vessel can be established.

Distracter B: (1) OPEN and (2) CLOSED

Distracter C: (1) CLOSED and (2) OPEN

Distracter D: (1) CLOSED and (2) CLOSED

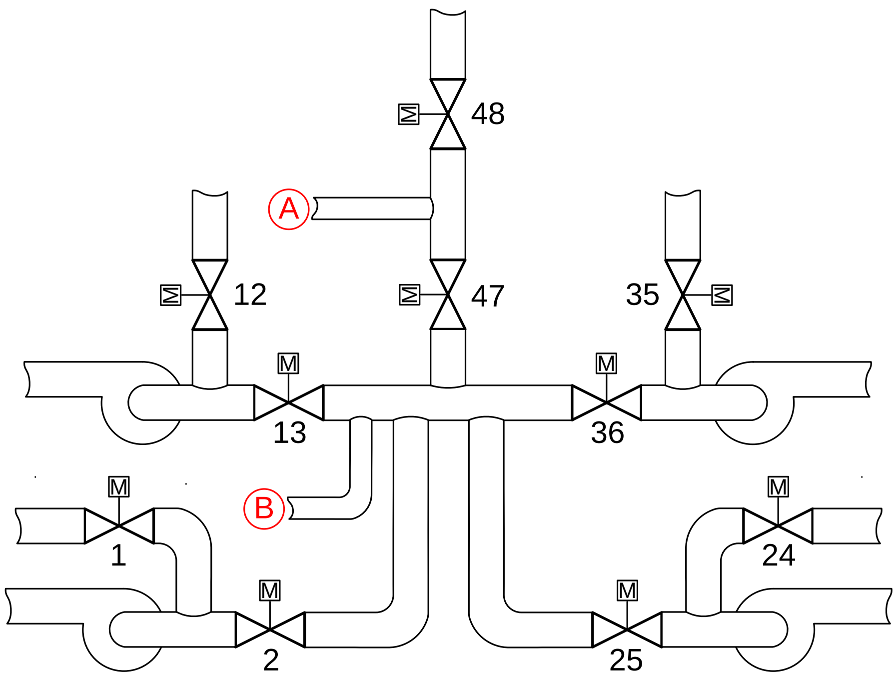
Reason Distracters B, C, and D are incorrect: The answers are combinations of the open and closed positions that are incorrect.

References:
Lesson Plan 63, LPCS

Examination Outline Cross-reference:	Level	RO	SRO
205000 Shutdown Cooling	Tier #	2	------
K3.05 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on the following:	K/A #	205000K3.05	
• Fuel pool cooling assist: Plant-Specific	Importance Rating	2.6	------

Proposed Question: **# 30**

On Unit 3, RHR Pump 3C has been placed in Supplemental Fuel Pool Cooling.



3-FCV-74-1 / 12 / 24 / 35 – SUPPR POOL SUCT VLVs
**ALL other valves associated with Shutdown Cooling

Which ONE of the following completes the statements?

Suction from Fuel Pool Cooling ties into RHR at point __**(1)**__. A Shutdown Cooling Isolation __**(2)**__ result in **LOSS** of Supplemental Fuel Pool Cooling.

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

B. **(1)** B
(2) will

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

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

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: First part incorrect - (See attached) – the tap for Supplemental Fuel Pool Cooling suction ties in just before the 13 valve, which is also in the Shutdown Cooling suction path, **BUT NOT** between the 47 and 48 valves. Second part incorrect – in that the 47 and 48 valves do, in fact, receive a close signal as a function of a Shutdown Cooling Isolation signal (logic attached FCV-74-47 AND 48); and if this were the proper configuration, the potential closing of these valves would isolate the path.

B INCORRECT: First part correct - (See attached) – the tap for Supplemental Fuel Pool Cooling suction ties in just before the 13 valve, which is also in the Shutdown Cooling suction path,. Second part incorrect – in that to lose Sup. Fuel Pool Cooling from this lineup, the 13 valve (associated w/ RHR Pump 'C') would have to travel closed on an isolation signal from Shutdown Cooling; and it does **NOT** receive a close signal as a function of a Shutdown Cooling Isolation signal (similar logic attached FCV-74-2).

C INCORRECT: Part 1 incorrect – as detailed in 'A' above. Part 2 correct – but, for a different configuration. (See attached procedural excerpt) This is plausible in that leads are lifted to relays in this procedural sequence and valve control circuits are manipulated to off-normal conditions. This could just as easily have been performed to prevent closure of one of the Shutdown Cooling Isolation Valves; the 47 valve in this case, to keep the flowpath available (as the loop will be declared inoperable for LPCI anyway). Additionally, due to the availability of the ADHR System, this is a very infrequently performed procedure sequence; adding to the level of difficulty.

D **CORRECT:** (See attached) – the tap for Supplemental Fuel Pool Cooling suction ties in just before the 13. Part 2 correct – (see attached) because the tap is downstream of the 47 & 48 valve, a Shutdown Cooling Isolation signal would **NOT** cause a loss of suction path while in Supplemental Fuel Pool Cooling.

Justification: Requires Knowledge of the effect that a SHUTDOWN COOLING SYSTEM Isolation will have on Supplemental Fuel Pool Cooling. Adding to the level of difficulty, the availability of a primary cooling system (ADHR) yields the fact that this lineup is rarely used at BFN. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.044 Rev. 17 (Attach if not previously provided)

3-OI-74 Rev. 95

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.044 V.B.3 / 10 / 19 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

VALVE	1-74-1	1-74-12	1-74-24	1-74-35	2-74-1	2-74-12	2-74-24	2-74-35
Interlock Valve(s)	74-2	74-13	74-25	74-36	74-2	74-13	74-25	74-36
Normal/Emerg SW			X	X	X	X		
Local Controls	X	X	X	X	X	X	X	X
Controls at Bkr			X	X	X	X		
Panel 9-3 controls	X	X	X	X	X	X	X	X
Local Indication lgts	X	X	X	X	X	X	X	X
Lights on Bkr			X	X	X	X		
Lights on Pnl 9-3	X	X	X	X	X	X	X	X
S/D-NORMAL SW	X	X	X	X	X	X	X	X

VALVE	3-74-1	3-74-12	3-74-24	3-74-35
Interlock Valve(s)	74-2	74-13	74-25	74-36
Normal/Emerg Sw	X	X		
Local Controls	X	X	X	X
Controls at Bkr	X	X		
Panel 9-3 controls	X	X	X	X
Local Indication lgts	X	X	X	X
Lights on Bkr	X	X		
Lights on Pnl 9-3	X	X	X	X
S/D-NORMAL SW			X	

b. RHR pump Shutdown Cooling suction interlocks — (74-2. 74-13; 74-25, 74-36) Obj. V.C.5

(1) No automatic closing or opening signals — TP-17, 18, 19, 20

(2) Valve cannot be opened unless the corresponding RHR pump Suppression Pool suction valve is fully closed. — Interlock cannot be Bypassed Obj. V.D.8

(3) "EMERGENCY" position allows operation at breaker only. — "Emergency" switches on all valves on all three units

(4) Loop I or Loop II valves cannot be opened with S/P spray cooling/test valve not full closed on loop. — DCN 41208/51222 added key lock to bypass interlock *Interlock bypassed by procedure for SDC flush*

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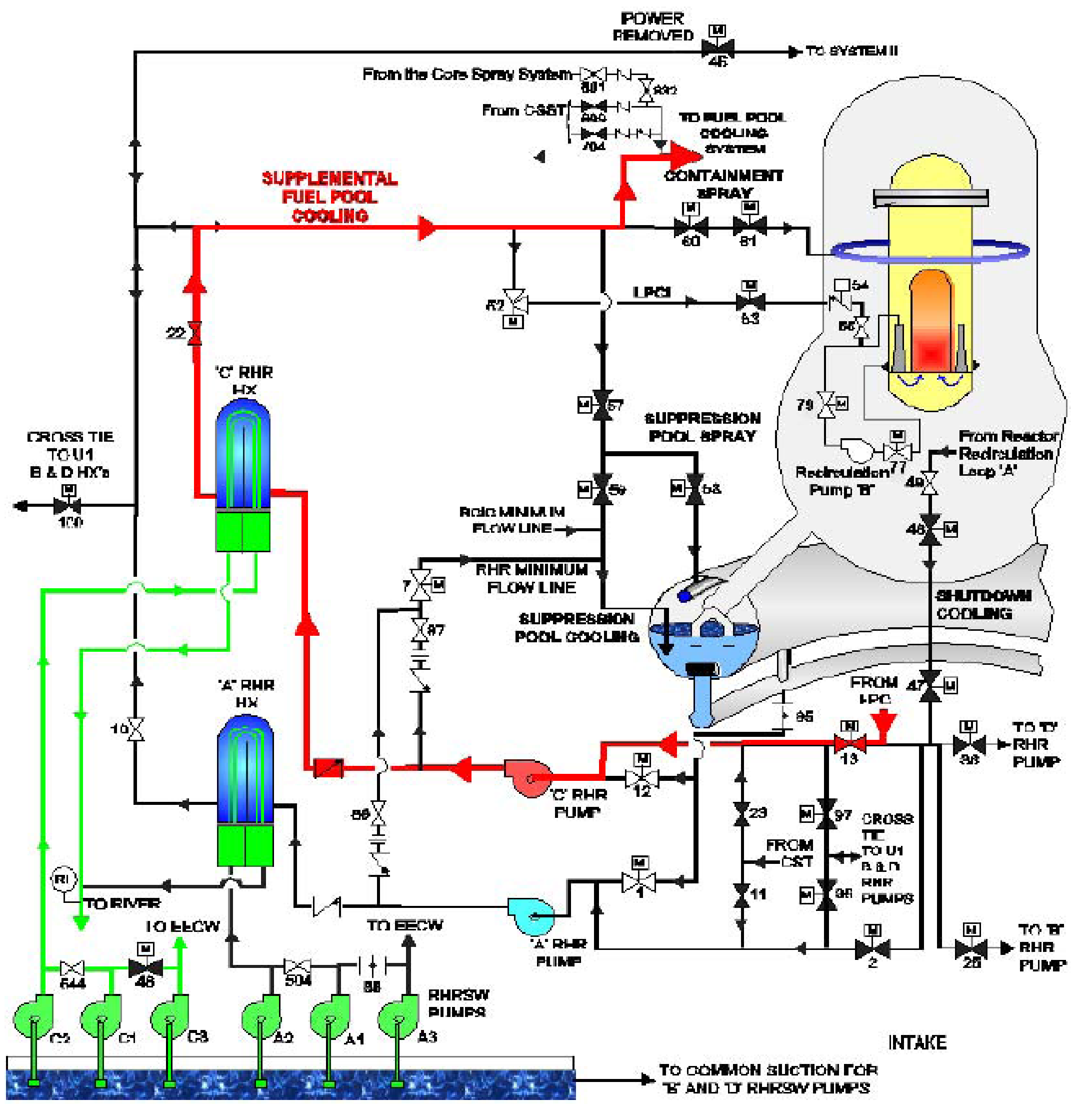

TP 9 RHR SYSTEM SUPPLEMENTAL FUEL POOL COOLING FLOW DIAGRAM UNIT 2 SYSTEM I

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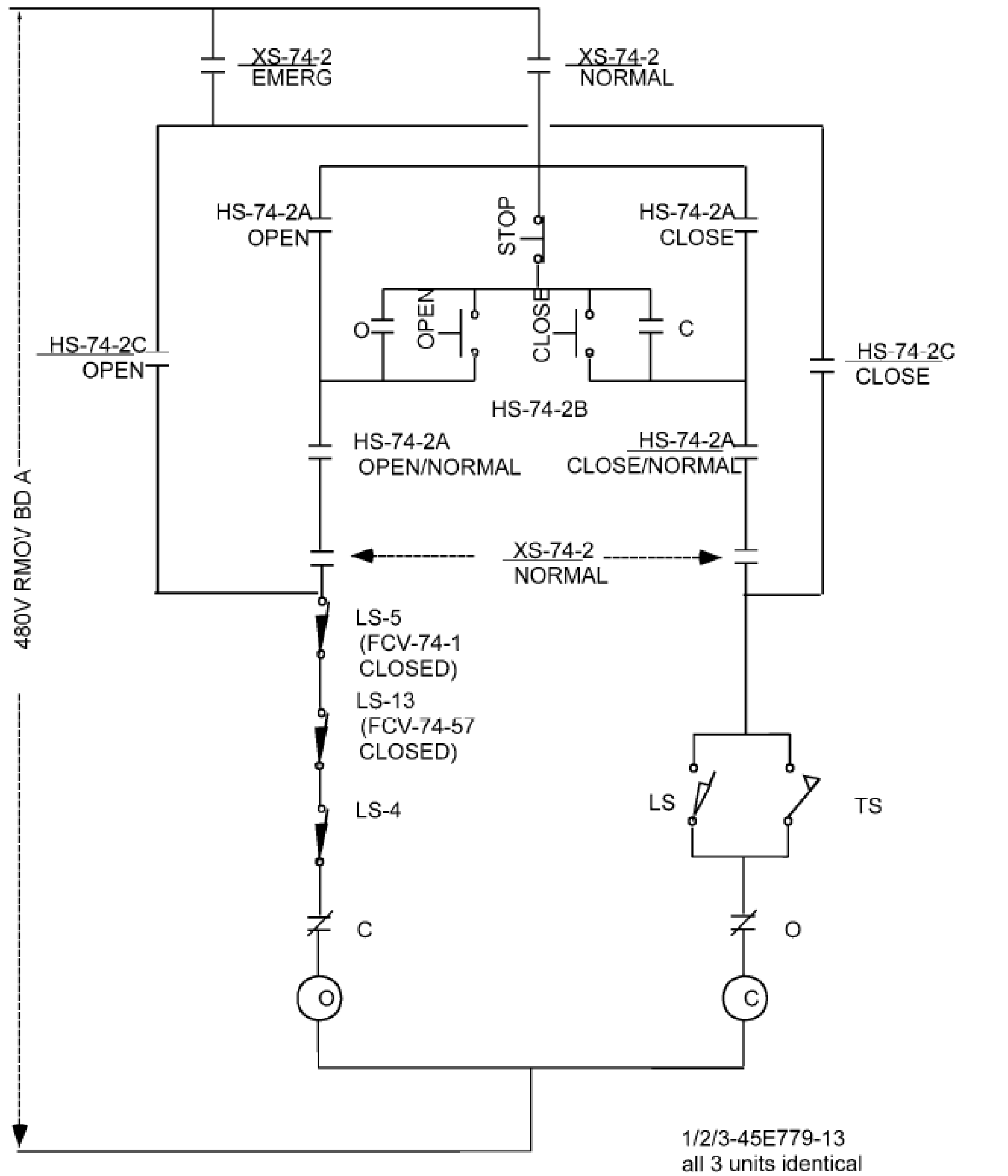

FCV-74-2 shown - see reference drawing table for FCV-74-13, 25, 36 interfaces

TP 17 'A' RHR PUMP SHUTDOWN COOLING SUCTION VALVE FCV-74-2 CONTROL CIRCUIT

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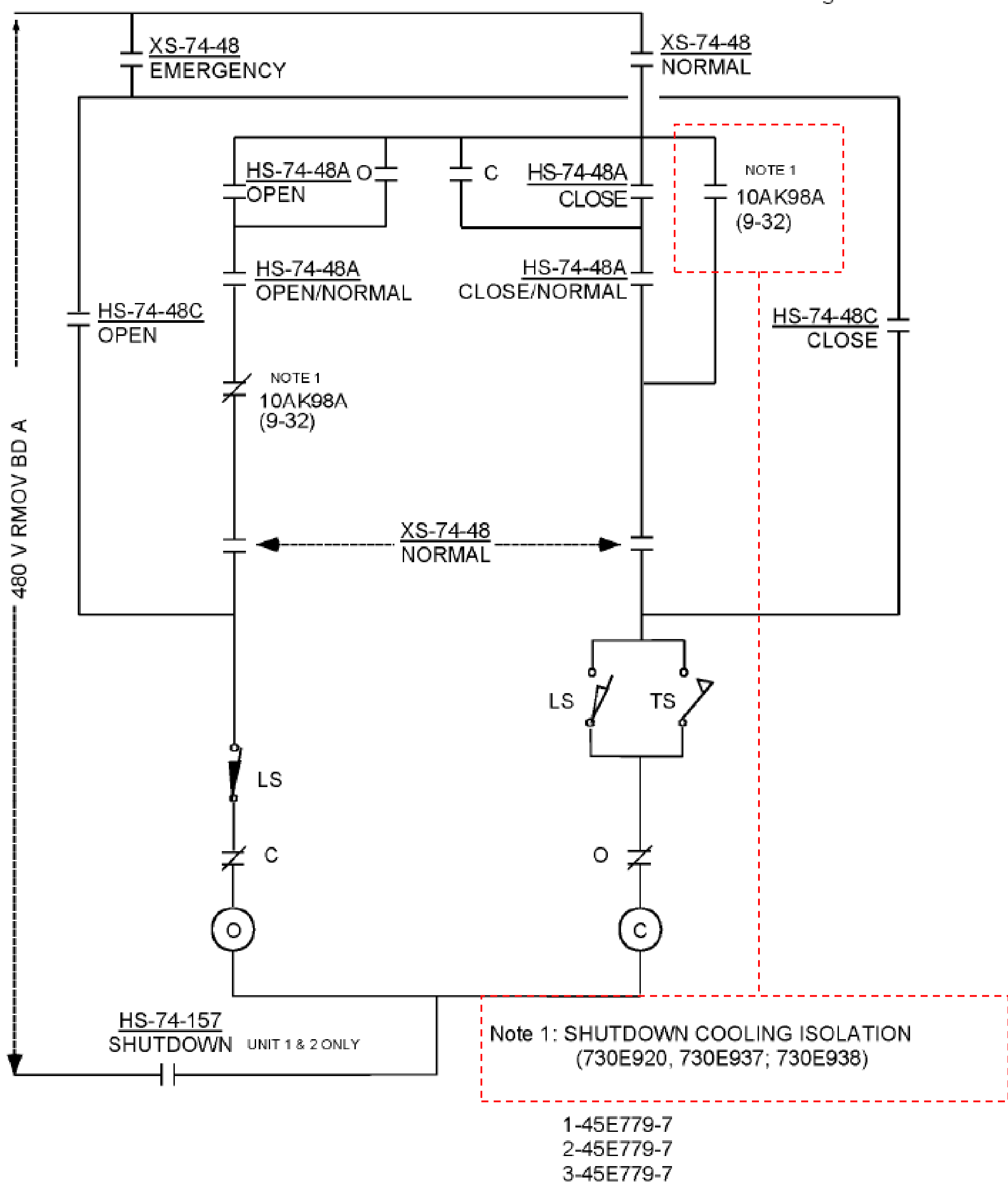

TP 20 RHR SHUTDOWN COOLING SUCTION INBOARD VALVE FCV-74-48 CONTROL CIRCUIT

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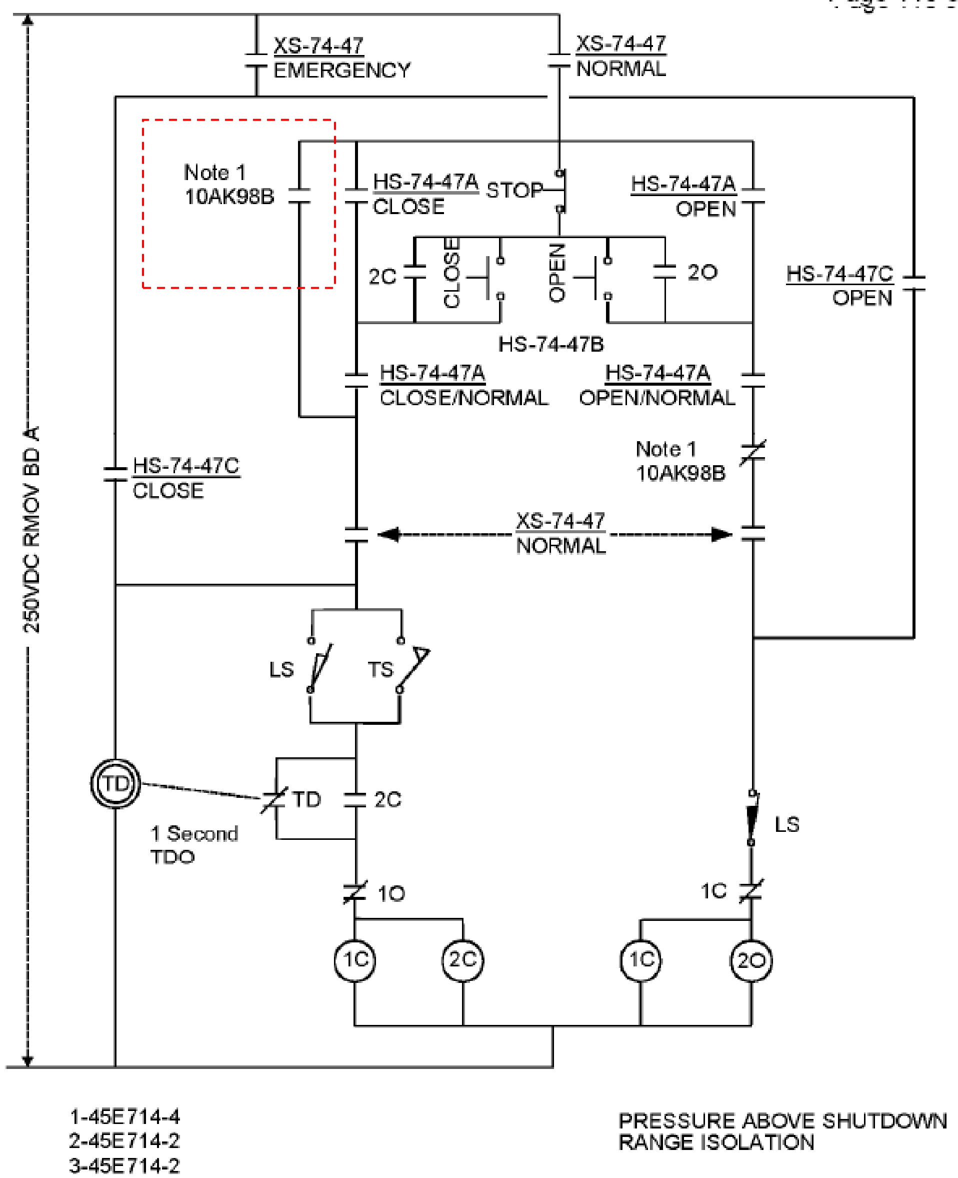

TP 19 RHR SHUTDOWN COOLING SUCTION OUTBOARD VALVE FCV-74-47 CONTROL CIRCUIT

BFN Unit 3	Residual Heat Removal System	3-OI-74 Rev. 0095 Page 175 of 374

8.12 Initiation of Supplemental Fuel Pool Cooling with RHR Pumps A or C(B or D) (continued)

[3] **IF** RHR Loop II is to be used for Supplemental Fuel Pool Cooling, **THEN**

VERIFY the following conditions:

A. RHR **NOT** required to be operable. ☐

B. RHR Loops I and II are in Standby Readiness. **REFER TO** Section 4.0. ☐

[4] **VERIFY** Fuel Pool Cooling System lined up for normal operation. **REFER TO** 3-OI-78. ☐

[5] **NOTIFY** Chemistry that RHRSW is to be placed in service and RHR is to be used for Supplemental Fuel Pool Cooling. ☐

[6] **NOTIFY** other units of placing Supplemental Fuel Pool Cooling with RHR Pumps A or C(B or D) in service, the subsequent start of common equipment (i.e., RHRSW pumps) and associated alarms are to be expected. ☐

[7] **ALIGN** RHR pump to be placed in Supplemental Fuel Pool Cooling. **REFER TO** the following steps and **USE** the following table: ☐

RHR Pump to be Used	3A	3B	3C	3D
Suction Valve Interlock - Panel - Relay	3-9-32 10A-K19A	3-9-33 10A-K19B	3-9-32 10A-K22A	3-9-33 10A-K22B
RHR PUMP ( ) SUPPR POOL SUCT VLV	3-FCV-74-1	3-FCV-74-24	3-FCV-74-12	3-FCV-74-35
RHR PUMP ( ) SD SD COOLING SUCT VLV	3-FCV-74-2	3-FCV-74-25	3-FCV-74-13	3-FCV-74-36
RHR PUMP ( ) SEAL WTR VENT	3-VTV-074-0548A 3-VTV-074-0549A	3-VTV-074-0548B 3-VTV-074-0549B	3-VTV-074-0548C 3-VTV-074-0549C	3-VTV-074-0548D 3-VTV-074-0549D
TELL-TALE VENT SHUTOFF VLV HIGH POINT TELLTALE VENT RHR HTX ( )	3-SHV-074-0749A 3-FSV-074-0142	3-SHV-074-0749B 3-FSV-074-0144	3-SHV-074-0749C 3-FSV-074-0413	3-SHV-074-0749D 3-FSV-074-0145

8.12 Initiation of Supplemental Fuel Pool Cooling with RHR Pumps A or C(B or D) (continued)

NOTE
The following step allows the RHR Pump to start with its suction valves closed.

[7.1] **REQUEST** Electrical Maintenance lift the lead at Terminal 1 or 2, or boot Contact 1 of the associated relay (above table) and document on Appendix A. ☐

[7.2] **PLACE** RHR VLV 3-FCV-74-24 APP R CONT CIRCUIT DISABLE HS, 3-HS-074-0024C, on 480V RMOV Board 3B Compartment 9B1, in SHUTDOWN to allow closing of 3-FCV-74-24 for flushing RHR Loop II. ☐

[7.3] **CLOSE** RHR PUMP SUPPR POOL SUCT VLV for the RHR pump to be used (above table). ☐

[7.4] **OPEN** RHR PUMP SD COOLING SUCT VLV for the RHR pump to be used (above table). ☐

NOTE
Closed loop vents should be vented for 1 minute.

[7.5] **OPEN** the vent valves on the RHR pump and heat exchanger to be used until a solid stream of water is observed, **THEN CLOSE** (above table) (local). ☐

[8] **IF** time permits, **THEN**

FLUSH RHR System. **REFER TO** Section 8.14. ☐

[9] **UNLOCK** and **OPEN** FUEL POOL CLG TO RHR XCONN VLV, 3-SHV-074-0091, (Rx Bldg, El 541', SW Corner Room). ☐

[10] **OPEN** FPC TO RHR SUCT, 3-SHV-078-0534, (Rx Bldg, El 621'). ☐

Examination Outline Cross-reference:	Level	RO	SRO
206000 HPCI	Tier #	2	------
K2.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following: • System pumps: BWR-2,3,4	K/A #	206000K2.02	
	Importance Rating	2.8	------

Proposed Question: **# 31**

Which ONE of the following completes the statement?

The power supply to the Unit 2 HPCI Aux Oil Pump is ________.

A. 250 VDC RMOV BD 2A.

B. 250 VDC RMOV BD 2B.

C. 480 VAC RMOV BD 2A.

D. 480 VAC RMOV BD 2B.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** 250 VDC RMOV BD 2A is the power supply to the HPCI Aux Oil Pump. See Attached Electrical Lineup Checklist.

B INCORRECT: This is, in fact a power supply to HPCI components; just not the Aux Oil Pump. Refer to attached PRESTARTUP REQUIREMENTS. This is also a primary power supply to RCIC components.

C INCORRECT: This is, in fact a power supply to HPCI components; just not the Aux Oil Pump. Refer to attached PRESTARTUP REQUIREMENTS. This is also a primary power supply to RCIC components.

D INCORRECT: This is, in fact a power supply to HPCI components; just not the Aux Oil Pump. Refer to attached PRESTARTUP REQUIREMENTS. This is also a primary power supply to RCIC components.

RO Level Justification: Tests whether the candidate has knowledge of power supplies to HPCI components, including the HPCI Auxiliary Oil Pump. Level of difficulty is compounded by the similarities of HPCI and RCIC in conjunction with the complex electrical distribution system at BFN. HPCI is a Div II System with ‘B’ Logic as the primary logic; but it comes from an ‘A’ Board. RCIC is the opposite – ‘A’ Logic from a ‘B’ Board. This often creates confusion between the power supplies for the two systems.

Technical Reference(s): 2-OI-73, Rev. 83 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.1/7 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:
55.41 X
55.43

Comments:

BFN Unit 2	**High Pressure Coolant Injection System**	**2-OI-73 Rev. 0083 Page 18 of 83**

4.0 PRESTARTUP/STANDBY READINESS REQUIREMENTS

NOTE

When Section 4.0 is required to be verified by subsequent Sections, Section 4.0 is required to be performed.

[1] **VERIFY** the following related system requirements are satisfied:

[1.1] HPCI turbine oil reservoir oil level is within the requirements of ILLUSTRATION 3. ☐

[1.2] HPCI booster pump pedestal bearings oil level is visible in sight glasses. ☐

[1.3] The following panels are energized. REFER TO 0-OI-57B, 0-OI-57C, and 0-OI-57D:

A. 480V Reactor MOV Board 2A. ☐

B. 480V Reactor MOV Board 2B. ☐

C. 250VDC Reactor MOV Board 2A. ☐

D. 250VDC Reactor MOV Board 2B. ☐

E. 240V Lighting Board 2A. ☐

F. Panel 2-9-9, Cabinet 5. ☐

[1.4] The following system lineup checklists are current:

A. Attachment 1, HPCI System Valve Lineup Checklist, Unit 2. ☐

B. Attachment 2, HPCI System Panel Lineup Checklist, Unit 2. ☐

C. Attachment 3, HPCI System Electrical Lineup Checklist, Unit 2. ☐

D. Attachment 4, HPCI System Instrument Inspection Checklist, Unit 2. ☐

BFN Unit 2	Attachment 3 ELECTRICAL LINEUP CHECKLIST	2-OI-73/ATT-3 Rev. 0082 Page 5 of 7

4.0 ATTACHMENT DATA

Performed On: ________________

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
Reactor Bldg. - 240V Lighting Board 2A - El 621'			
3D3	HPCI AUX MOTOR HTR	ON	____ ____
Control Bay 480V Reactor MOV Board 2A - El 621'			
17E	HPCI STEAM SUPPLY LINE ISOLATION VALVE FCV-73-2	ON	____ ____
Control Bay - 250V Reactor MOV Board 2A Div II - El 621'			
1D2	HPCI PUMP SUCTION FROM CONDENSATE STORAGE TANK VALVE FCV-73-40 (MO 23-17)	ON	____ ____
3D	HPCI STM SUPPLY VALVE TO TURB FCV-73-16 (MO 23-14)	ON	____ ____
4A	HPCI TURBINE AUX. OIL PUMP	ON	____ ____
4D	HPCI PUMP SUCTION FROM SUPPRESSION CHAMBER VALVE FCV-73-26 (MO 23-58)	ON	____ ____
5A	HPCI PUMP DISCHARGE VALVE FCV-73-34 (MO 23-20)	ON	____ ____
6A	HPCI TEST BYPASS TO CONDENSATE STORAGE TANK VALVE FCV-73-35 (MO 23-21)	ON	

Examination Outline Cross-reference:	Level	RO	SRO
211000 SLC	Tier #	2	------
A3.01 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including: • Pump discharge pressure: Plant-Specific	K/A #	211000A3.01	
	Importance Rating	3.5	------

Proposed Question: **# 33**

Unit 3 is executing 3-EOI-1, "RPV Control," due to a Scram **AND** an ATWS. The Unit Operator (UO) initiates Standby Liquid Control (SLC) per 3-EOI Appendix-3A, "SLC Injection." The following is observed **TEN** minutes later:

- SLC Storage Tank Level is 79%
- SLC Pump Discharge Pressure is 1100 psig and steady
- Reactor Pressure is 1000 psig

Based upon the **ABOVE** indications, which ONE of the following completes the statement?

The running SLC Pump is discharging __**(1)**__ **AND** the blue SQUIB VALVE A and B CONTINUITY lights, on Panel 3-9-5, are expected to be __**(2)**__ for this condition.

A. **(1)** to the Reactor Vessel
(2) illuminated

B. **(1)** to the Reactor Vessel
(2) extinguished

C. **(1)** through a Relief Valve
(2) illuminated

D. **(1)** through a Relief Valve
(2) extinguished

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part correct – see 'B' below. Second part incorrect in that if the Squib Valves fire as intended, the blue continuity lights will be extinguished. In reality, for the blue continuity lights, either state is plausible.

B **CORRECT:** (See attached excerpts) Being a positive displacement pump, SLC Pump discharge pressure should be at or slightly higher than RPV pressure. Additionally, the tank level lowering at approximately 1% per minute supports injection to the RPV. If the Squib Valves fire as intended, the blue continuity lights will be extinguished. This fact is verified in 3-EOI Appendix 3A as a function of proper system operation.

C INCORRECT: First part incorrect in that the SLC Relief Valve setpoint is 1425 psig $\pm$ 75 psig. Thus, indicated pressure would be significantly higher if the Relief Valve was the discharge path. Also, indicated tank level would not lower if the Relief Valve was open; as the relief discharges back to the tank. The second part is also incorrect for same reason detailed in 'A' above.

D INCORRECT: First part incorrect – see 'C' above. Second part is correct, as detailed in 'B' above.

RO Level Justification: Tests the candidate's ability to monitor SLC System operations in reference to SLC Pump Discharge Pressure and relationship to injection along with tank level indications. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.039, Rev. 16 (Attach if not previously provided)

3-EOI Appendix 3A, Rev. 1

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.3.e/f (As available)

V.B.5 / 6

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

OPL171.039
Revision 16
Page 3 of 7

a) The total tank capacity is 4,901 gallons; however, the overflow is connected at the tank level corresponding to 4,850 gallons.

b) Normal storage tank level is 85.6% to 95.4%, which provides 4150 to 4626 gallons of solution for injection into the reactor vessel. *

c) Storage tank level is measured by an air bubbler system, (supplied by control air) which senses the amount of backpressure applied by the head of water in the tank. Obj. V.B.3.g Obj. V.C.2.g

d) The storage tank air sparger is:

(1) Supplied by service air.

(2) Used to mix the sodium pentaborate solution during initial storage tank filling and prior to routine sampling.

e) Storage tank heater:

(1) Heats storage tank water to 80° F for initial sodium pentaborate solution mixing. Obj. V.D.3.b Obj. V.E.3.b

(2) Maintains the sodium pentaborate solution ≥10° F above the solution saturation temperature. Obj. V.B.3.c Obj. V.C.2.c

(3) Controlled with the heater control in ON or AUTOMATIC position.

(a) In ON, the heater will stay on continuously.

(b) In AUTOMATIC, the heater will energize to maintain the solution at the temperature setpoint on the controller.

(c) If the tank heater is being controlled in ON or AUTOMATIC and the tank solution level lowers to 830 gallons, the heater will automatically cut off, to prevent possible damage from overheating.

(4) Powered from 480V Reactor Building Vent Board. Obj. V.B.5.c Obj. V.C.4.d

(5) Installed in a dry well to allow heater removal without draining the tank.

f) Storage tank outlet strainer:

	INSTRUCTOR NOTES
4. SLC Pumps	
a) Two 100% capacity, triplex, positive displacement piston pumps are installed in parallel.	Obj. V.B.5.c Obj. V.C.4.d Obj. V.D.4 Obj. V.E.4 Obj. V.B.5.c Obj. V.C.4.d Obj. V.B.3.f Obj. V.C.2.f
b) 'A' pump is powered from 480V Shutdown Board A.	
c) 'B' pump is powered from 480V Shutdown Board B.	
d) Electrically interlocked so that only one pump will run at a time. This prevents system overpressurization.	
e) The pumps are manually started from the main control room using the key-lock switch on panel 9-5, or locally, using the Test Permissive Transfer Switch at Panel 25-19.	
f) A control room start signal will fire the explosive valves. A local start will <u>not</u> fire the explosive valves.	
g) Either pump is capable of supplying a system flow of approximately 50 gpm at a system pressure of 1275 psig.	Obj. V.D.3.d Obj. V.E.3.d
h) Each pump discharge has a relief valve, set at 1425 $\pm$ 75 psig, to protect the pump and the system from overpressurization.	Obj. V.B.3.f Obj. V.C.2.f Obj. V.D.3.e Obj. V.E.3.e
i) Each pump contains internal suction and discharge check valves, which open at approximately 5 psid, allowing only forward flow through an idle pump. (INPO O&MR 341).	
j) Pump motors are protected by an undervoltage trip.	
5. Accumulators	
a) An accumulator is installed between each pump and its discharge check valve.	
b) Dampens the pressure pulsations that are inherent with piston-type, positive-displacement pumps.	Obj. V.D.3.d Obj. V.E.3.d
c) A steel vessel accumulator, containing a synthetic bladder, with one side charged to ~450 psig nitrogen gas and SLC solution on the other side.	

3-EOI APPENDIX-3A

SLC INJECTION

LOCATION: Unit 3 Control Room

ATTACHMENTS: None

(√)

1. **UNLOCK** and **PLACE** 3-HS-63-6A, SLC PUMP 3A/3B, control switch in START PUMP 3A or START PUMP 3B position. ____

2. **CHECK** SLC System for injection by observing the following:

   - Selected pump starts, as indicated by red light illuminated above pump control switch. ____
   - Squib valves fire, as indicated by SQUIB VALVE A and B CONTINUITY blue lights extinguished, ____
   - SLC SQUIB VALVE CONTINUITY LOST Annunciator in alarm on Panel 3-9-5 (3-XA-55-5B, Window 20). ____
   - 3-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. ____
   - System flow, as indicated by 3-IL-63-11, SLC FLOW, red light illuminated on Panel 3-9-5, ____
   - SLC INJECTION FLOW TO REACTOR Annunciator in alarm on Panel 3-9-5 (3-XA-55-5B, Window 14). ____

3. IF .....Proper system operation CANNOT be verified, THEN ...**RETURN** to Step 1 and **START** other SLC pump. ____

4. **VERIFY** RWCU isolation by observing the following:

   - RWCU Pumps 3A and 3B tripped ____
   - 3-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed ____
   - 3-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed ____
   - 3-FCV-69-12, RWCU RETURN ISOLATION VALVE closed. ____

5. **VERIFY** ADS inhibited. ____

6. **MONITOR** reactor power for downward trend. ____

7. **MONITOR** 3-LI-63-1A, SLC STORAGE TANK LEVEL, and **CHECK** that level is dropping approximately 1% per minute. ____

8. WHEN ...EITHER of the following exists:

   - SLC tank level drops to 0%,

   OR

   - As directed by SRO,

   THEN ...**STOP** SLC Pump 3A or 3B. ____

9. **NOTIFY** Chemistry to mix additional solution to compensate for dilution as directed by the SRO. ____

10. WHEN ...Directed by SRO to perform system flush,
    THEN ...**REFER** to 3-OI-63, Section 8.1, for system flush. ____

LAST PAGE

PICTURE FROM UNIT 3 SIMULATOR FOR COMPARISON

Examination Outline Cross-reference:	Level	RO	SRO
209001 LPCS	Tier #	2	------
K1.08 (10CFR 55.41.7)	Group #	1	------
Knowledge of the physical connections and/or cause-effect relationships between the LOW PRESSURE CORE SPRAY (LPCS) SYSTEM and the following:	K/A #	209001K1.08	
• A.C. electrical power	Importance Rating	3.2	------

Proposed Question: **# 32**

Unit 2 has experienced a LOCA with the following plant conditions:

- Drywell Pressure is 3.5 psig and rising
- Reactor Water Level is (-) 120 inches and lowering
- Reactor Pressure is 105 psig and lowering
- 4kV Shutdown Board C is locked out

Which ONE of the following predicts the total injection flowrate for the Loop 2 Core Spray Pumps? (**Assume no operator actions**)

A. 0 gpm.

B. 2400 gpm.

C. 6250 gpm.

D. 9100 gpm.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: (See attached excerpts) Although RPV Level initiation of (-) 122 inches has not been reached, Core Spray should have auto-started on Drywell Pressure > 2.5 psig with Reactor Pressure < 450 psig. Each loop of Core Spray is designed to supply 6250 gpm against head of 105 psig. With 4 kV SD Board C locked out, CS Pump 2B is de-energized and CS Pump 2D is interlocked to not auto-start. System 2 CS flow would be 0 gpm under these conditions.

B INCORRECT: Value of 2400 gpm based on expected Min flow for Core Spray. If power loss had affected System 1 Inboard Injection Valve, Pumps would be running on Min Flow between 2200 gpm and 2600 gpm.

C INCORRECT: Plausibility based on impact of loss of 4kV Shutdown Board C. The power loss does not affect ‘A’ and ‘C’ Core Spray. Pump ‘A’ is supplied from SD Board ‘A’ and Pump ‘C’ is supplied from SD Board ‘B’. System 1 Inboard Injection Valve is power from 480 RMOV Shutdown Board 2A which is still energized by SD Board ‘A’

D INCORRECT: Value of 9100 gpm is plausible and corresponds to expected flow for pump run out. For the identified plant conditions, Core Spray Pumps are not in a run out condition.

RO Level Justification: Tests the candidate's knowledge of Core Spray AC power supplies and the cause-effect relationship of losing an AC power board when initiation conditions are present. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.045 Rev 15 (Attach if not previously provided)

2-OI-75, Rev. 95

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.2/3/4 (As available)

Question Source:	Bank #		
	Modified Bank #	HLT 0801 Q#16	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BANK QUESTION THAT WAS MODIFIED…

Exam Development Input 1

Record	K/A Number	Pedigree	SRO-Only	Currently Chosen For
1865	295031EA2.03	BFN 0801 NRC Question #16	NO	

Question Stem

Unit 2 has experienced a LOCA with the following plant conditions:

•Drywell Pressure is 3.5 psig and rising
•Reactor Water Level is (-) 120 inches and lowering
•Reactor Pressure is 105 psig and lowering
•4kV Shutdown Board C is locked out

Which ONE of the following predicts the total injection flowrate for the Loop I Core Spray Pumps?

Answer

6250 gpm.

Distractor 1

0 gpm.

Distractor 2

2400 gpm.

Distractor 3

9100 gpm.

Record: 1854 of 1939

INSTRUCTOR NOTES

X. Lesson Body

A. General Description

Employ .ppt's as appropriate

1. The purpose of the Core Spray System is to protect against over-heating the fuel in the event of a LOCA. This is accomplished by spraying water directly on the fuel from spray spargers located within the shroud.

Obj. V.B.1
Obj. V.E.1

2. There are two independent, redundant, 100% capacity Core Spray Systems (System I - A and C pumps; System II - B and D pumps).

3. System is initiated on either:

Obj. V.B.2.h
Obj. V.C.1.g
Obj. V.D.4
Obj. V.E.6

a. Low-low-low water level -122", Level 1

OR

b. High drywell pressure (2.45 psig) in conjunction with low reactor pressure (450 psig).

4. Upon initiation, water is pumped from the suppression pool (torus) to the reactor.

TP-1
Obj. V.D.4
Obj. V.E.6

c. Two 360° spray spargers within the shroud disperse the water.

d. Each system contains two pumps in parallel.

e. Emergency power is received from diesel generators through the 4kV shutdown boards.

Obj. V.B.3.a
Obj. V.C.2.a
Obj. V.D.10.a
Obj. V.E.12.a

INSTRUCTOR NOTES

b. Pump motors have an oil-lubricated thrust bearing at the top.

c. Each pump motor has a heater which prevents condensation in the motor windings. This heater is normally on and is automatically de-energized when the pump breaker closes.

d. Each loop delivers at least 6,250 gpm against a system head corresponding to a 105 psi differential pressure between the reactor vessel and the primary containment.

e. Power supplies for all Core Spray System pump motors are shown on the following chart for Units 1, 2, and 3:

Obj. V.B.3.a
Obj. V.C.2.a
Obj. V.D.10.a
Obj. V.E.12.a

PUMP	1A	2A	3A	1B	2B	3B	1C	2C	3C	1D	2D	3D
LOOP (SYS)	I	I	I	II	II	II	I	I	I	II	II	II
ELECT. DIV.	I	I	I	II	II	II	I	I	I	II	II	II
S/D BD.	A	A	3EA	C	C	3EC	B	B	3EB	D	D	3ED
D/G	A	A	3A	C	C	3C	B	B	3B	D	D	3D

f. Pumps are located in the basement of the Reactor Building (NE, NW Quads-Elev. 519). The head of water from torus provides NPSH requirements.

Obj. V.B.2.f
Obj. V.C.1.e
Obj. V.E.13.a

3. Spargers

TP-2
Obj. V.B.1
Obj. V.D.1
Obj. V.E.3
Obj. V.E.13.b

a. Two 360° spray spargers located within core shroud.

b. Each sparger is split into two 180° segments.

c. Spargers are separate, each receiving flow from one of the two loops provided.

3.8 Power Supplies

A. Core Spray Breaker closure with the breaker racked to the test position will result in a auto start of the EECW pumps if the NVA or DGVA relay is allowed to time-out prior to opening of the breaker.

Core Spray Pump	EECW Pump (if aligned for service)
2A	C1, B3
2B	A1, D3
2C	A1, D3
2D	C1, B3

B. If one pump in a Core Spray System has its 4kV Shutdown Board de-energized, NEITHER pump in that loop will auto start AND may NOT be considered operable. REFER TO TS 3.5.1/3.5.2.

C. After operation of 4160V breakers, the charging spring shall be verified to have recharged by Verifying locally, the breaker closing spring target indicates charged and the amber breaker spring charged light is on to insure future breaker operation.

D. Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161kV source and large pumps are started. Unit Supervisors should evaluate placing the Capacitor Banks in Manual prior to starting RHR, CS, CBP, CCW, or COND pumps as referenced in 0-OI-57A.

3.9 Venting requirements

A. The fill and vent section in this OI is intended to be used following maintenance when any portion of the system has been drained. It is not to be used as a substitute for the monthly venting surveillance 2-SR-3.5.1.1 (CSI) (CSII)

B. The associated Core Spray System should be vented before starting any Core Spray pump, unless in an emergency situation.

C. Failure to ensure system is filled and vented prior to racking pump breakers into the connect position following restoration after maintenance may result in equipment damage caused by an inadvertent pump start.

D. Whenever any portion of Core Spray Loop I (II) piping from the suction line to the discharge valve has been drained for any reason, both Core Spray pumps in Loop I(II) are required to be operated in the torus to torus recirc flowpath mode for a minimum of 15 minutes to verify that the suction piping and portions of the discharge piping have been dynamically vented and are free of voids. Section 8.12 of this procedure or the applicable sections of 2-SR-3.5.1.6 (CS I) (CSII) flow rate test or 2-SR-3.5.1.6 (CS I) (CSII) comprehensive pump test procedures, may be used.

INSTRUCTOR NOTES

d. Nozzles direct the Core Spray water toward the vertical centerline of the fuel.

4. Valves and Piping — TP-1

a. Motor-operated valves are all AC and located outside the drywell. System I valves powered from 480V Rx MOV BD A; System II valves from B. — Obj. V.B.3.b, Obj. V.C.2.b

b. Controlled from Panel 9-3 in the Control Room and from local pushbutton stations in the Reactor Building.

c. Thermal expansion relief valves set at 500 psi to protect low pressure discharge piping rated at 520 psi and 150 psi to protect the low pressure suction piping. — Obj. V.C.1, Obj. V.B.2

d. Suction valves FCV 75-2 and 11 (75-30 and 39) are normally open - alarm in Control Room if either suction valve is not fully open. Manual suction HCV-75-3 and 12 (75-31 and 40) physically connect between the MOV suctions and the pump, to permit CST supply to the Core Spray System as necessary. Closed position indication is provided in the control room. — Obj. V.B.5.d

e. Minimum flow valve FCV 75-9 (FCV-75-37), which is normally open. Provides minimum flow protection for each loop and discharges to the torus below water level. These are controlled by FS 75-21 (49) which closes at > 2600 GPM increasing and opens at 2200 GPM decreasing. — Obj. V.B.2.e,j, Obj. V.C.1.i

Restricting orifice in min-flow line restricts flow to 20% of loop flow

Examination Outline Cross-reference:	Level	RO	SRO
212000 RPS	Tier #	2	------
A2.09 (10CFR 55.41.5)	Group #	1	------
Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM (RPS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	212000A2.09	
• High containment/drywell pressure	Importance Rating	4.1	------

Proposed Question: **# 34**

Unit 2 has inserted a manual Reactor Scram. Several control rods failed to insert on the scram. Plant conditions are as follows:

- Reactor Power is 3%
- Reactor Pressure is 960 psig **AND** controlled by EHC
- Drywell Pressure is 2.5 psig
- Mode Switch is in Shutdown
- SCRAM DISCH VOLUME HI LEVEL BYPASS Switch is in NORMAL
- Reactor Water Level is (-) 55 inches

NOTE: 2-EOI Appendix 1F – Manual Scram
2-EOI Appendix 2 – Defeating ARI Logic Trips

Which ONE of the following is required to reset the Scram?

A. Install jumpers per 2-EOI Appendix 1F **AND** Reset ARI.

B. Install jumpers per 2-EOI Appendix 1F **AND** Defeat ARI per 2-EOI Appendix 2.

C. Manually insert control rods using the RMCS until APRM DOWNSCALE / OPRM INOP, (2-9-5A, Window 4) clears.

D. Place the SDV Hi Hi Wtr Trip Bypass Keylock Switch to BYPASS until EAST / WEST CRD DISCH VOL WTR LVL HIGH HALF SCRAM, (2-9-4A, Windows 1/29) clear.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: is incorrect because ARI must be inhibited or scram air pressure will not recover to close the Scram valves, also the SDV vent and Drain valves must be closed.

B **CORRECT**: In accordance with EOI Appendix 1F, "Manual Scram," Jumpers must be installed to defeat RPS Scrams, ARI Defeated per EOI Appendix 2, RPS Reset, then SDV Vent and Drain Valves verified closed.

C INCORRECT: is incorrect because clearing the APRM downscale annunciator will have no effect on the RPS system, the RPS trip in SHUTDOWN is set at *15%*.

D INCORRECT: is incorrect because bypassing the SDV and draining the SDV will not allow RPS to be reset with a high drywell pressure and low reactor water level scram in.

Justification: Requires candidate to demonstrate ability to predict the impacts of High drywell pressure on the REACTOR PROTECTION SYSTEM (RPS); and based on those predictions, use EOI Appendix to correct the consequences of those abnormal conditions. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-EOI Appendix 1F Rev. 4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # Brunswick 07 #10

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Brunswick 2007

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

2-EOI APPENDIX-1F

MANUAL SCRAM

LOCATION: Unit 2 Control Room

ATTACHMENTS:
1. Tools and Equipment
2. Panel 2-9-15, Rear
3. Panel 2-9-17, Rear

(√)

1. **VERIFY** Reactor Scram and ARI reset. ____

   a. IF ..... ARI CANNOT be reset,
   THEN ... **EXECUTE** EOI Appendix 2 concurrently with Step 1.b of this procedure. ____

   b. IF ..... Reactor Scram CANNOT be reset,
   THEN ... **DISPATCH** personnel to Unit 2 Auxiliary Instrument Room to defeat ALL RPS logic trips as follows:

      1) **REFER** to Attachment 1 and **OBTAIN** four 3-ft banana jack jumpers from EOI Equipment Storage Box. ____

      2) **REFER** to Attachment 2 and **JUMPER** the following relay terminals in Panel 2-9-15, Rear:

         a) Relay 5A-K10A (DQ) Terminal 2 to Relay 5A-K12E (ED) Terminal 4, Bay 1. ____

         b) Relay 5A-K10C (AT) Terminal 2 to Relay 5A-K12G (BH) Terminal 4, Bay 3. ____

      3) **REFER** to Attachment 3 and **JUMPER** the following relay terminals in Panel 2-9-17, Rear:

         a) Relay 5A-K10B (DQ) Terminal 2 to Relay 5A-K12F (ED) Terminal 4, Bay 1. ____

         b) Relay 5A-K10D (AT) Terminal 2 to Relay 5A-K12H (BH) Terminal 4, Bay 3. ____

2. WHEN ...RPS Logic has been defeated,
   THEN ...**RESET** Reactor Scram. ____

3. **VERIFY OPEN** Scram Discharge Volume vent and drain valves. ____

ES-401 Sample Written Examination Question Worksheet Form ES-401-5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	212000 A2.09	
Importance Rating	3.6	

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High containment/drywell pressure.

Proposed Question: Common 10

Unit Two (2) has inserted a manual reactor scram due to a loss of RBCCW. Several control rods failed to insert on the scram. Plant conditions:

APRM indicated power	3%
Reactor pressure	960 psig, controlled by EHC
Drywell pressure	2.1 psig
Mode Switch	Shutdown
SDV Hi Hi Wtr Trip Bypass	Normal

Which ONE of the following is required to reset RPS?

A. Install jumpers per LEP-02, Section 3, Reset ARI and Verify the SDV Vent and Drain Valves are OPEN.

B. Install jumpers per LEP-02, Section 3, Inhibit ARI and Verify the SDV Vent and Drain Valves are CLOSED.

C. Manually insert control rods using the RMCS until A-06 Annunciator APRM DOWNSCALE clears.

D. Place the SDV Hi Hi Wtr Trip Bypass Keylock Switch to BYPASS then wait until A-05 Annunciator SDV HI-HI LEVEL RPS TRIP clears.

Proposed Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
215003 IRM	Tier #	2	------
A4.04 (10CFR 55.41.7)	Group #	1	------
Ability to manually operate and/or monitor in the control room: • IRM back panel switches, meters, and indicating lights	K/A #	215003A4.04	
	Importance Rating	3.1	------

Proposed Question: **# 35**

Given the following plant conditions:

- Unit 3 Reactor startup preparations are in progress with NO rods withdrawn
- Instrument Mechanics (IM) are performing the Intermediate Range Monitor (IRM) Functional Surveillance
- NO IRMs are currently bypassed
- The IM has placed the "INOP / INHIBIT" toggle switch for the 'H' Channel IRM in the "INHIBIT" position

Which ONE of the following describes the IRM trip function that is bypassed as a result of this action?

A. IRM "High Voltage Low" INOP TRIP.

B. IRM "Loss of ± 24 VDC" INOP TRIP.

C. IRM "Module Unplugged" INOP TRIP.

D. IRM "Mode Switch Out of Operate" INOP TRIP.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This INOP trip will still function. Each of the possible answers will typically initiate an INOP trip of its associated IRM channel, therefore each distractor is plausible.

B INCORRECT: See explanation A

C INCORRECT: See explanation A

D **CORRECT:** INOP/INHIBIT Pushbuttons (toggle sw for Unit 3) are pushed/flipped to bypass the INOP trip that results from taking mode switch S-1 out of "OPERATE." They are used to allow testing of other scram or rod block signals from the IRM drawer into RPS/RMCS without them being masked by the INOP trip.

Justification: This question satisfies the K/A statement by requiring the candidate to use specific component manipulations to correctly determine the response of the IRM system. Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

Technical Reference(s): OPL171.020 Rev. 11 (Attach if not previously provided)
3-OI-92A Rev. 15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.020 V.B.5 / 7 (As available)

Question Source:
Bank # BFN 0610 #7
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam BFN 0610

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

BFN Unit 3	Intermediate Range Monitors	3-OI-92A Rev. 0015 Page 7 of 15

3.0 PRECAUTIONS AND LIMITATIONS (continued)

H. The IRMs produce the following trip outputs to the Reactor Protection System auto-scram circuitry:

1. High-High (> 116.4 on 125 scale).

2. Inop (module unplugged, mode switch not in OPERATE, HV power supply low voltage, loss of 24VDC power supply to IRM drawer).

3. In addition, by removing the blue shorting links (2 total links), the IRMs are placed in the non-coincident trip logic where any <u>one</u> channel, if tripped, will produce a full reactor scram. The 2/4 Voters are also in this logic such that a trip output from any one Voter yields a full Reactor Scram.

I. The time required to drive a detector from full out to full in is approximately 3 minutes.

J. The INOP TRIP BY-PASS switches located on the IRM drawers on Panel 9-12 by-pass the IRM switch position out-of-operate trip. These switches are to be used only during testing of the IRM channels.

K. [NRC/C] Upon return to service of 24-VDC Neutron Monitoring Battery A or B, Instrument Maintenance is required to perform functional tests on SRMs and IRMs that are powered from the affected battery board. [NRC IE Inspector Follow-up Item 86-40-03]

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INSTRUCTOR NOTES

2. Panel 9-12

TP-9

a. Reset switch - Allows resetting of seal-in trip lights on front of drawer.

Obj.V.B.5/V.B.7

b. Function switch positions

(1) 'Operate' - IRM channel functions as described.

(2) 'Standby' - same as operate, except gives Inop trip to yield maximum design protection before channel is removed from service.

Fundamentals: How is this feature a conservative design feature?
Ans: inserts trip to ensure that trip signal is not bypassed when testing IRM instrumentation.

(3) 'Zero 1' - Removes signal from output amplifier so that output amplifier, local meter, and recorder can be zeroed.

(4) 'Zero 2' - Removes voltage from range switch. This deselects all ranges. This, in turn, causes no input to be sent to attenuator and allows setting the zero adjust on output amplifier.

(5) '125 '- Input is removed from attenuator same as Zero 2 position. A calibration signal is substituted which will yield 125 on the 125 scale. Used to set gain of output amplifier.

(6) '40' - Produces a 40 reading on the 125 scale.

c. INOP/INHIBIT Pushbuttons

Obj. V.B.6
Obj. V.C.4

(1) Pushed to bypass the INOP trip that results from taking mode switch S-1 out of "operate."

(2) Used to allow testing of other scram or rod block signals from the IRM drawer into RPS/RMCS without them being masked by the INOP trip.

DCN W18726A replaced the INOP/INHIBIT Pushbutton with a toggle switch for the U-3 IRM drawers. (UNIT DIFFERENCE)

7. RO 215003A4.04 001/MEM/SYS/IRM/B6/215003A4.04//RO/SRO/MODIFIED 11/17/07

Given the following plant conditions:

- Unit 3 reactor startup preparations are in progress with NO rods withdrawn.
- Instrument Mechanics are performing the Intermediate Range Monitor (IRM) functional surveillance.
- No IRMs are currently bypassed.
- The Instrument Mechanic technician has placed the "INOP INHIBIT" toggle switch for the 'H' Channel IRM in the "INHIBIT" position.

Which ONE of the following describes the IRM trip function that is bypassed as a result of this action?

A. IRM "High Voltage Low" INOP TRIP is bypassed.

B. IRM "Loss of ± 24 VDC" INOP TRIP is bypassed.

C. IRM "Module Unplugged" INOP TRIP is bypassed.

D.✓ IRM "Mode Switch Out of Operate" INOP TRIP is bypassed.

K/A Statement:

215003 IRM
A4.04 - Ability to manually operate and/or monitor in the control room: IRM back panel switches, meters, and indicating lights

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific component manipulations to correctly determine the response of the IRM system.

References: OPL171.020 rev. 10, pg 22, c.(1) & (2) and 3-OI-92A rev. 14, pg 7 of 15 P&L 3.0.J

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

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MODIFIED FROM OPL171.020 #28

REFERENCE PROVIDED: None

Plausibility Analysis:

In order to answer this question correctly the candidate must determine the following:

1. The function switch is bypassed by the "INOP INHIBIT" pushbutton.

NOTE: Each of the possible answers below will typically initiate an INOP trip of it's associated IRM channel, therefore each distractor is plausible.

D - correct: INOP/INHIBIT Pushbuttons (toggle sw for Unit 3) are pushed/flipped to bypass the INOP trip that results from taking mode switch S-1 out of "OPERATE." They are used to allow testing of other scram or rod block signals from the IRM drawer into RPS/RMCS without them being masked by the INOP trip.

A - incorrect: This INOP trip will still function.

B - incorrect: This INOP trip will still function.

C - incorrect: This INOP trip will still function.

Examination Outline Cross-reference:

215003 IRM

A4.05 (10CFR 55.41.7)

Ability to manually operate and/or monitor in the control room:

- Trip bypasses

Level	RO	SRO
Tier #	2	------
Group #	1	------
K/A #	215003A4.05	
Importance Rating	3.4	------

Proposed Question: **# 36**

Unit 1 is in Mode 2 with the following conditions:

- Source Range Monitor (SRM) 'A' is reading 6.2 x 10^4 cps
- SRM 'D' mode switch (S-1) is in the STANDBY position
- Intermediate Range Monitor (IRM) 'D' is downscale on Range 1 (output has been lost)
- IRM 'C' is reading 85 of 125 scale on Range 8
- **ALL** other IRMs are reading mid scale on Range 8 **OR** 9

Based on the above indications, which ONE of the following has caused a Rod Block signal to be generated?

A. IRM High.

B. SRM High.

C. IRM Downscale.

D. SRM Inoperable.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: IRM High Flux Rod Block Trip occurs at >104.6 on 125 SCALE. Plausible in that reading is above level which IRM is required to be ranged up in accordance with 1-GOI-100-1A, but less than the Rod Block signal.

B INCORRECT: SRM A is elevated but below its High Flux Rod Block set point of 6.8 x 10^4 cps.

C INCORRECT: Loss of IRM detector output would result in a rod block based on IRM downscale. However, this trip is bypassed with Detector on Range 1.

D **CORRECT**: SRM B Mode Switch being placed in STANDBY will result in a SRM Inop alarm with a concurrent rod block. However, this function is bypassed with IRMs on Range 8. Since IRM D is on Range 1 due to the operability issue, the bypass logic is not made up.

Justification: To successfully answer this question, candidate must recognize those conditions which will cause IRM / SRM Rod Block signals and when those trips are bypassed. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.020 Rev 11 / 1-OI-92A Rev 8 (Attach if not previously provided)

OPL171.019 Rev 13 / 1-OI-92 Rev 6

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.020 V.B.5 (As available)

OPL171.019 V.B.5/V.B.8

Question Source:

Bank #	
Modified Bank #	
New	X

(Note changes or attach parent)

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

Instructor Notes

(2) Mode switch S-1

(a) This switch changes the mode of SRM channel operation to allow for maintenance or calibration.

(b) The switch has six positions: — Self Check

(i) Operate

(ii) Standby

(iii) Zero

(iv) Period

(v) 10^5

(vi) 10

(c) Operate position - The SRM operates normally (as described in this lesson).

(d) Standby position - Same as the Operate position, except that it causes an Inop Trip signal. This trip signal will initiate a control rod block unless the mode switch is in RUN OR the associated IRM channels are on range 8 or above.

	INSTRUCTOR NOTES
(2) Active coating	Obj. V.B.1.
(a) 1.25 mg U_3O_8 (1/5 of that in SRM Detector)	
(b) Uranium is 90 percent enriched in U235 (same as SRM)	
(c) Inner surface of outer electrode only (same as SRM)	
(3) Fill-gas is Argon-filled to 1.2 atmospheres (17.7 psia)	
(4) Operating voltage is 100V DC (350V DC for SRM)	
(5) Operates as an "ion chamber"	
b. Reason for differences between SRM & IRM detectors	Obj. V.B.2.
(1) Less sensitivity needed by IRM since it operates at higher fluxes.	
(2) Internal heat dissipation	
(3) IRMs may remain in core until reactor is at significant power levels (10-20 percent).	GOI-100-1A has IRMs withdrawn after recorders are switched to APRM and RBM and mode SW in RUN
(4) Detector gets quite hot due to fissions occurring internal to detector. Quantity of U235 limited (1/5 of SRM) to keep from burning up detector.	
c. Loss of IRM detector output would result in a rod block based on IRM downscale, which is bypassed with the Reactor Mode Switch in RUN or IRM range switch selected to position 1)	Obj V.B.11
d. It is extremely important to minimize exposure to IRM detectors by removing them from core as soon as they are no longer needed.	<u>Procedure use</u> Fundamentals - Why? Ans: preserves the life span of IRM detectors.

BFN Unit 1	Intermediate Range Monitors	1-OI-92A Rev. 0008 Page 15 of 15

Illustration 1
(Page 1 of 1)

IRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
IRM High	>104.6 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 or 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

BFN Unit 1	Source Range Monitors	1-OI-92 Rev. 0006 Page 14 of 14

Illustration 1
(Page 1 of 1)

SRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
SRM High	6.8×10^4 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	2×10^5 counts per second	Scram if shorting links removed

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0020 Page 82 of 165

5.0 INSTRUCTION STEPS (continued)

NOTE
1) Completing paper closure of 1-SR-3.3.1.1.5 is not required prior to performing Step 5.0[32]. However, all AC steps must be VERIFIED COMPLETED SATISFACTORILY prior to withdrawing SRMs.
2) Tech Spec Bases state that overlap between SRMs and IRMs exists when IRM downscale indications have cleared and IRM readings are on-scale and trending higher prior to SRMs reaching 10^5 cps.

[32] **VERIFY** SRM/IRM overlap by obtaining data and completing 1-SR-3.3.1.1.5 SRM and IRMs Overlap Verification.

(R) ________ ________ ________
Initials Date Time
Reactor Engineer

NOTES
1) SRMs are fully withdrawn when IRMs are on Range 3 or above and indicating above their downscale trip point.
2) If a shutdown margin test has been performed using a different rod sequence, 1-SR-3.1.3.5(A) will provide required actions to insert all control rods, establish normal sequence and perform the subsequent start up with re-entry at Step 5.0[23].

[33] **WITHDRAW** SRMs as necessary to maintain them on scale between 10^2 cps and 10^5 cps.

________ ________ ________
Initials Date Time

[34] **MAINTAIN** IRMs on scale between approximately 25 and 75 using IRM range switches.

________ ________ ________
Initials Date Time

[35] **ENSURE** 1-SI-4.6.B.1-4 has been satisfactorily completed prior to pressurizing Reactor.

(R) ________ ________ ________
Initials Date Time
Chem Shift Supv

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor	Tier #	2	------
G2.2.2 (10CFR 55.41.7)	Group #	1	------
Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	K/A #	215004G2.2.2	
	Importance Rating	4.6	------

Proposed Question: **# 37**

A plant start up on Unit 3 is in progress. A control rod block has occurred. The following nuclear instrument indications are noted:

	SRM A	SRM B	SRM C	SRM D
Position	Full in	Mid-position	Mid-position	Full in
Counts (CPS)	9.5×10^3	95	80	8.0×10^3

IRM A	IRM B	IRM C	IRM D	IRM E	IRM F	IRM G	IRM H
25/125	15/125	35/125	55/125	75/125	75/125	30/125	25/125
Range 3	Range 2	Range 3	Range 3	Range 2	Range 2	Range 3	Range 3

Which ONE of the following identifies the MINIMUM action needed to clear the ROD WITHDRAWAL BLOCK?

A. Insert SRM B **ONLY**

B. Insert SRM B **AND** SRM C

C. Range up on IRM B **AND** IRM F to range 3

D. Range up on IRM E **AND** IRM F to range 3

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausible in that with IRM C on range 3, candidate may believe SRM C Detector Not Full In Rod Block is bypassed. However, with any associated IRM (A, C, E or G) not on range 3, the trip remains in force.

B **CORRECT**: SRM RETRACT NOT PERMITTED will alarm and cause a rod block with SRM counts <145cps with associated IRMs ≤ Range 2 and the Detector not Full In.

C INCORRECT: Plausible in that it would clear the Control Rod Block from SRM B. However, it would result in IRM B causing a rod block due to IRM downscale.

D INCORRECT: Plausible in that ranging up IRM E and F would not result in an IRM downscale rod block. However, a rod block would remain with IRM B still on range 2.

Justification: Candidate must demonstrate ability to manipulate the console controls for SRMs as required to operate the facility between shutdown and designated power levels.

Technical Reference(s): 3-OI-92 Rev. 13 (Attach if not previously provided)

OPL171.019 Rev. 13

Proposed references to be provided to applicants during examination: ________

Learning Objective: OPL171.019 V.B.8 (As available)

Question Source: Bank #

Modified Bank # Perry 09 #37 (Note changes or attach parent)

New

Question History: Last NRC Exam Perry 2009

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 3	Source Range Monitors	3-OI-92 Rev. 0013 Page 13 of 13

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 105 counts per second	Scram if shorting links removed

Instructor Notes

b. Alarms, Interlocks, Trips and Annunciators — Obj.V.B.8 Obj.V.C.2/V.D.5

Annunciator/Function	Setpoint	Bypassed	Instructor Notes
SRM Hi(Alarm and Rod Block) (Panel 9-5A, Window13)	6.8×10^4	IRM range 8 or above OR in Run Mode	
INOP(Alarm and Rod Block) (Panel 9-5A, Window 13)		IRM range 8 or above, OR in Run Mode	
(1) module unplugged;			Obj. V.B.5
(2) switch not in Operate			Obj. V.C.1
(3) HV Power supply voltage Low			Obj. V.D.4
(4) Loss of +/- 24 VDC power supply			Loss of power gives Rod Block
SRM DOWNSCALE (Alarm and Rod Block) (Panel 9-5A, Window 6)	<5cps	IRM range 3 or in RUN Mode	
SRM SHORT PERIOD (Alarm only) (Panel 9-5A, Window 20)	30 seconds	Never	
SRM RETRACT NOT PERMITTED (Alarm and Rod Block)	<145cps	IRM range 3 OR in RUN Mode OR Detector Full-in.	Obj.V.B.7

c. Alarms other than annunciators on panel 9-5 — Obj.V.B.8 Obj.V.C.2

(1) Each SRM channel has a set of four lights on the apron section:

(a) Hi Hi (red) — Refer to OI-92 for current setpoints (2×10^5 cps)

(b) High/INOP (amber)

(c) Downscale (white)

(d) Bypassed (white)

(2) Each SRM channel has a white RETRACT PERMISSIVE light above its respective LCR meter. ($\geq$ minimum cps energizes the light.) — Obj. V.B.6 Set points in OI-92

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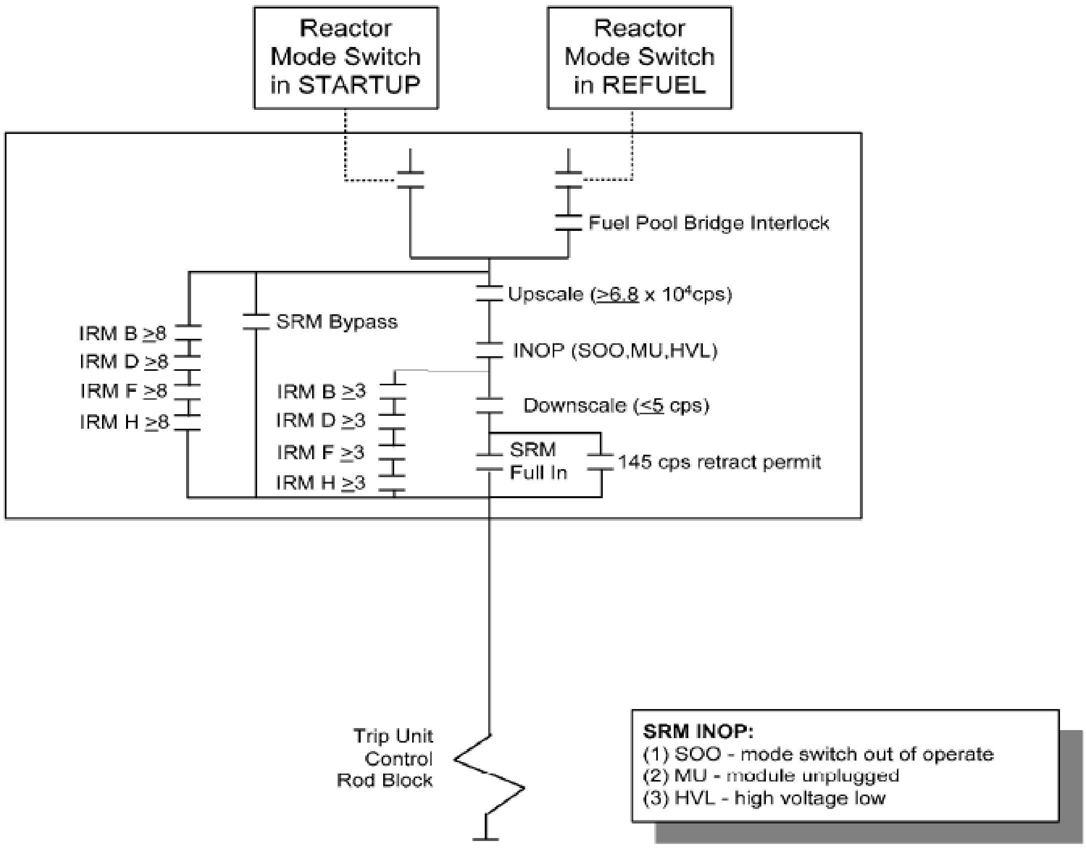

2-730E321 (Partial)

TP-10: SRM ROD BLOCK DIAGRAM

BFN Unit 3	Intermediate Range Monitors	3-OI-92A Rev. 0015 Page 15 of 15

Illustration 1
(Page 1 of 1)

IRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
IRM High	>104.6 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

PERRY
NRC EXAM - 2009

QUESTION RO 49

A plant start up is in progress. A control rod block has occurred. The following nuclear instrument indications are noted:

	SRM A	SRM B	SRM C	SRM D
Position	Full in	Mid-position	Mid-position	Full in
Counts (CPS)	$9.5x10^4$	95	80	$8.0x10^4$

IRM A	IRM B	IRM C	IRM D	IRM E	IRM F	IRM G	IRM H
25/125	15/125	35/125	55/125	75/125	75/125	30/125	25/125
Range 3	Range 2	Range 3	Range 3	Range 2	Range 2	Range 3	Range 3

What is the minimum action needed to clear the ROD WITHDRAWAL BLOCK?

A. Only Insert SRM B

B. Insert SRM B and SRM C

C. Range up on IRM B & IRM F to range 3

D. Range up on IRM E & IRM F to range 3

Examination Outline Cross-reference:	Level	RO	SRO
215005 APRM / LPRM	Tier #	2	------
K4.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of AVERAGE POWER RANGE MONITOR / LOCAL POWER RANGE MONITOR (APRM / LPRM) SYSTEM design feature(s) and/or interlocks which provide for the following:	K/A #	215005K4.01	
• Rod withdrawal blocks	Importance Rating	3.7	------

Proposed Question: **# 38**

Unit 2 APRMs have the following indications:

- APRM 1 - 106%
- APRM 2 - 104%
- APRM 3 - 104%
- APRM 4 - 105%
- Recirc Loop A flow 60%
- Recirc Loop B flow 64%

Which ONE of the following identifies the expected plant response to these conditions?

A. Control Rod Withdrawal Block **ONLY**

B. Half Scram **AND** Control Rod Withdrawal Block

C. Full Scram **AND** Control Rod Withdrawal Block

D. Flow Compare Inverse Video Alarm on ODA **ONLY**

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: w = (60 + 64)/2 = 62%
rod block setpoint = (.66(62) + 59) = 99.92%
scram setpoint = (.66(62) + 65) = 105.92%
All APRM channels are >the rod block set point.

B INCORRECT: ONLY APRM 1 is above the STP calculated upscale Scram set point of 105.92. PRNM system only provides half scrams during voter testing. Plausible in that typical RPS logic results in a half scram with one channel above the trip set point.

C INCORRECT: Only 1 channel is above the upscale set point. Requires 2 APRM inputs above setpoint (.66w + 65) to initiate a Full Reactor Scram.

D INCORRECT**:** Flow compare function looks for 5% mismatch between flows for same loop. Additionally, a Control Rod Withdrawal Block will occur with current conditions.

Justification: Question requires knowledge of APRM design features and interlocks which provide Control Rod Withdrawal Blocks to successfully answer. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.148 Rev 12 (Attach if not previously provided)

2-OI-92C Rev. 34

2-OI-92B Rev. 38

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.148_V.B.31 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam BFN 0606

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

	INSTRUCTOR NOTES
i. 2/4 Logic Module	TP-7, 8, 9, 10
(1) The main functions of the voters are to perform:	Obj. V.B.15.c
(a) Voting of the APRM trip outputs for RPS	
(b) Monitor APRM channel bypass conditions	
(c) Provide isolation (interface) between APRM and external equipment.	
(d) The 2/4 Logic module monitors the trip signals from the four APRMs and initiates Reactor Protection System (RPS) safety functions in the event of specific trip conditions.	Full reactor scram vs. 1/2 scram
(e) A reactor scram output is provided when at least two of the same type of trip inputs are in a tripped state.	TOP row of lights 2nd row of lights
• Either from APRM HIGH HIGH/INOP OR • any OPRM algorithm set point exceeded	
(f) Each voter gets 4 input signals (one from each APRM / OPRM channel) for APRM trip functions and it gets another 4 input signals for OPRM trip functions.	TP-11
(g) The OPRM automatic trip functions are generated by any of the cells within any of the APRM/OPRM channels as calculated by any of the three algorithms.	

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INSTRUCTOR NOTES

(2) The STP signal is used by the APRM for flow biased rod blocks and scram set points.

(3) Flow Biased Scram and Rod Block generation

(a) The APRM calculates a flow-biased setpoint by comparing reactor power and reactor recirculation flow

(b) At 100% power, both recirculation pumps are running and the SLO value in the flow biased calculation is zero (0)

w = flow as calculated by the APRM instrument.

(c) With one recirculation pump tripped or secured, a 10% bias is added to the flow biased calculation to add a conservatism to the calculation:

Δw =Δ flow = 0% for 2 loop operation and 10% for single loop operation. This is the conservative bias added to the calculation during single loop operation.

(d) Flow biased setpoint for reactor scram is

0.66(w-Δw) + 65%

(e) Flow biased setpoint for Control Rod Block is

0.66(w-Δw) +59%

Obj V.D.7.b
Obj V.D.7.c

(f) Examples

(i) Given that Neutron Flux indicates 85% power and Reactor Recirculation Flow indicates 40% flow, calculate the setpoint for the alarm and rod block. (Assume both reactor recirculation pumps are running.)

.66(40-0) +65 = scram setpoint = 91.4% STP

.66 (40-0) + 59 = rod block setpoint = 85.4% STP

(ii) How is this different if in single loop operation?

SLO
.66(40-10)+65= 84.8% STP
.66(40-10)+59= 78.8% STP

BFN Unit 2	Rod Block Monitor	2-OI-92C Rev. 0034 Page 14 of 16

Illustration 1
(Page 1 of 1)

RBM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
RBM Downscale	92%	Rod Block
RBM Inop	1. Local RBM Chassis Mode Switch NOT in OPERATE 2. LOSS of Input Power (Module Unplugged) 3. RBM fails to null 4. Less than 50% of LPRM inputs operable for rod selected 5. Null sequence in progress 6. Self-Test Detected Critical Fault 7. Communication link to the reference APRM is lost or invalid. 8. More than one rod selected	Rod Block
RBM Upscale 1. Low 2. Intermediate 3. High	 1. 25% to 60% STP Alarms at 121.8% 2. > 60% to 80 % STP alarms at 117.0% 3. > 80% STP Alarms at 112.0%	Rod Block
1. Recirc Flow Compare	1. >5% mismatch between APRMs	Flow Compare Inverse Video Alarm

BFN Unit 2	Average Power Range Monitoring	2-OI-92B Rev. 0038 Page 22 of 30

Illustration 1
(Page 1 of 5)
APRM Trip Outputs

APRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
APRM Downscale	≥5%	1. Rod Block if REACTOR MODE SWITCH in RUN.
APRM Inop	1. APRM Chassis Mode not in OPERATE (keylock to INOP). 2. Loss of Input Power to APRM. 3. Self Test detected Critical Fault in the APRM instrument. 4. Firmware Watchdog timer has timed out	1. One Channel detected, no alarm or RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).
APRM Inop Condition	1. < 20 LPRMs in OPERATE, or < 3 per level.	1. <20 LPRMs total or <3 per level results in a Rod Block and a trouble alarm on the display panel. This does not yield an automatic APRM trip, but does, however, make the associated APRM INOP.
APRM High	1. DLO ≤ (0.66W + 59%) SLO ≤ (0.66W(W-10%) + 59%) [W = Total Recirc Drive Flow in % rated]. 2. Neutron Flux Clamp Rod Block ≥ 113% 3. ≤ 10% APRM Flux.	1. Rod Block if REACTOR MODE SWITCH in RUN. 2. Rod Block in all REACTOR MODE SWITCH positions except RUN.
APRM High High	1. DLO ≤ (0.66W + 65%) 2. SLO ≤(0.66(W-10%) + 65%) [W = Total Recirc Drive Flow in % rated]. 2. ≤ 119% APRM Flux. 3. ≤ 14% APRM Flux.	1. Scram. 2. Scram in all REACTOR MODE SWITCH positions except RUN.
APRM Flow Converter	1. ≤ 5% mismatch between APRM Channels. 2. 107% Flow monitor upscale.	1. Flow compare inverse video alarm. 2. Rod Block.
OPRM Inop	< 23 Operable Cells - A cell is inop when it has < 2 operable LPRM's	Annunciation Only
OPRM Pre-Trip Condition	Any one of three algorithms, period, growth, or amplitude exceeds its pre-trip alarm setpoint for an operable OPRM cell.	Rod Block
OPRM Trip	Any one of the three algorithms, period, growth, or amplitude for an operable OPRM cell has exceeded its trip value:	1. One Channel detected, no RPS output signal. 2. Two Channels detected, RPS output signal to all four Voters (Full Reactor Scram).

All OPRM setpoints are bypassed when the Reactor Mode Switch is not in RUN or the Reactor is not operating in the Power/Flow region where instabilities can occur (≥25% Power & <60% Recirc Drive Flow).

Examination Outline Cross-reference:	Level	RO	SRO
217000 RCIC	Tier #	2	------
K6.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC):	K/A #	217000K6.04	
• Condensate storage and transfer system	Importance Rating	3.5	------

Proposed Question: **# 39**

After a Reactor Scram on Unit 2, the following plant conditions exist:

- Main Turbine Bypass Valves failed closed
- HPCI **AND** RCIC have been MANUALLY started in CST to CST pressure control mode
- Subsequently, Condensate Storage Tank (CST) level dropped to 6500 gallons

Assuming **NO** operator action has been taken, which ONE of the following completes the statement?

RCIC is __**(1)**__ with suction from the __**(2)**__.

A. **(1)** operating at shutoff head
(2) CST.

B. **(1)** operating in pressure control
(2) CST.

C. **(1)** operating at shutoff head
(2) Suppression Pool.

D. **(1)** pumping to the CST
(2) Suppression Pool.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Part 1 correct - At ~7000 gallons in the CST, HPCI auto swaps from CST suction to Suppression Pool (Torus) suction. When this occurs the CST Test Return Isolation valve, 2-FCV-73-36, receives a close signal from the Torus suction valves opening; to prevent pumping the Torus to the CST. RCIC uses the HPCI Test Return line for flow path to the CST with RCIC connecting upstream of 2-FCV-73-36. RCIC min flow valve will not open with the absence of an initiation signal. Part 2 correct – Normal standby lineup has RCIC suction aligned to the CST. RCIC does not have an automatic suction swap from CST to Suppression Pool.

B INCORRECT: Part 1 incorrect – plausible in that the isolation of the CST test return valves is in response to HPCI low CST suction swap. With RCIC not having a low CST suction swap, candidate may fail to recognize RCIC flow path is also affected. Part 2 correct as explained above.

C INCORRECT: Part 1 correct as detailed in explanation A. Part 2 is incorrect but plausible in that HPCI does have automatic suction swap to the Suppression Pool with CST level < 7000 gallons.

D INCORRECT: Parts 1 and 2 incorrect but plausible as explained above.

Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions to determine the effect of low CST level on RCIC operation. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.040 Rev. 23 (Attach if not previously provided)
OPL171.042 Rev. 20
2-OI-71 Rev. 59

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.040 V.B.11 (As available)

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

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INSTRUCTOR NOTES

1. If both of the suppression pool suction line isolation valves are fully open, HPCI CST suction valve automatically closes

Obj. V.B.4
Obj. V.C.4
SER 3-05

2. If during HPCI operation, suppression pool water level increases to 5.25" above zero or if CST level drops to 552'6" above sea level (7000 gallons), then HPCI pump suction valves from the suppression pool (73-26 and 73-27) open. (This will then cause the CST suction valve to close once the SP suction valves get full open).

NOTE: There are normally 300,000 gallons available in the CST for HPCI and RCIC use.

3. A flow switch tapped in parallel with the HPCI system flow controller closes the minimum flow bypass valve to suppression pool (73-30) at 1255 gpm increasing; and opens it at 900 gpm decreasing, only if an auto start signal is present. Minimum flow valve closes on a Turbine Trip signal.

4. If either of the suppression pool suction line isolation valves (73-26 or 73-27) are full open then the HPCI test line to the CST valves (73-35 and 73-36) will close.

Obj. V.B.4
Obj. V.C.4

5. If the HPCI turbine isolation valve (73-16) is fully closed, then gland seal condenser condensate pump discharge valves to clean radwaste (73-17A and 73-17B) will open if the gland seal condenser hotwell has high level.

6. If the HPCI turbine isolation valve (73-16) is fully closed, then HPCI turbine steam line drain pot discharge isolation valves to the main condenser (73-6A and 73-6B) will open.

7. If 73-16 is full closed, the auxiliary oil pump will not start from the control room. When 73-16 opens 10% and the control switch is in the start position, the auxiliary oil pump will run.

BFN Unit 2	Reactor Core Isolation Cooling	2-OI-71 Rev. 0059 Page 9 of 75

3.0 PRECAUTIONS AND LIMITATIONS

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

1. High RPV water level (+51 in.); 579 in. above vessel zero. The RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, will close at +51 in. and will **RE-OPEN** when RCIC re-initiates at -45 in. RPV water level.
2. Turbine overspeed (Mechanical, 122.3% of rated speed).
3. Pump low suction pressure (10 inches Hg vacuum).
4. Turbine high exhaust pressure (50 psig).
5. Any isolation signal.
6. Remote manual trip (RCIC TURBINE TRIP push-button, 2-HS-71-9A, depressed).

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

1. RCIC steamline space temperature at ≤180°F Torus Area or ≤180°F RCIC Pump Room.
2. RCIC turbine high steam flow (150% flow, 3-second time delay.)
3. RCIC turbine steam line low pressure (73 psig).
4. RCIC turbine exhaust diaphragms ruptured (10 psig).
5. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION push-button, 2-HS-71-54, depressed, only if RCIC initiation signal is present).

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

D. In the presence of a RCIC initiation signal, the RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve will NOT auto open on low flow if an initiation signal is NOT present.

E. RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, will open on receipt of an initiation signal even with RCIC turbine manually tripped resulting in slowly draining CST to Suppression Chamber.

(3) Unit 3 power supply to the EGM Control Box is Div I ECCS Inverter.

(4) If Bus B fails, B channel trip logic and B channel isolation logic will be inoperative.

e. Steam line break — Obj. V.B.11.a. Obj. V.C.7.a

RCIC is provided with two independent flow to detect high steam flow. High steam flow of ≥150% for 3 seconds measured on either one or both flow elements will isolate FCV 71-2 and 71-3.

f. Low CST level — Obj. V.B.11.b. Obj. V.C.7.b Obj. V.E.13 Obj. V.E.14

RCIC has no automatic transfer from CST to torus. OI-71 directs transfer when HPCI auto transfers on low CST level or high torus level and if RCIC trips on low suction pressure 10" Hg vacuum.

g. High suppression pool temperature — Obj. V.B.11.b Obj. V.C.7.b Obj. V.B.11.c

(1) RCIC is normally aligned to the CST for pump suction cooling water. Suppression pool temperature will adversely affect the pool's capacity as a heat sink. While performing RCIC surveillance, pool temperature is monitored and pool cooling is directed at 95°F bulk temperature. Calculations have shown a 1°F rise torus temperature for every 16 minutes of operation.

(2) When RCIC is operating on suppression pool suction and the following alarms are received: — Obj. V.D.10 Obj. V.D.11 Obj. V.B.12

RCIC OIL CLR OUTLET DISCH OIL HI TEMP	(120°F)
RCIC GOVERNOR END BEARING HIGH TEMP	(160°F)
RCIC COUPL END BEARING TEMP HIGH	(160°F)

RCIC STEAM LINE LEAK DETECTION TEMP HI	TS-71-41A/B/C/D	138° F	3D-10

5. Valve Interlocks — TP-12

a. Steam valves — Obj. V.B.6.

(1) Isolation valves FCV-71-2 and 3 auto close on RCIC isolation signal.

(2) Steam line drain pot to main condenser hotwell valves FCV-71-6A and 6B automatically shut if steam supply to turbine valve FCV-71-8 is not full closed.

(3) Steam supply valve FCV-71-8 auto opens on an initiation signal. Auto closes on high reactor water level of +51".

b. Condensate suction valve FCV-71-19

(1) Automatically opens on initiation signal provided at least one of the torus suction valves is not fully open

(2) Automatically closes when both torus suction valves are fully open (FCV 71-17, 71-18)

c. Minimum flow valve FCV-71-34

(1) Automatically closes on turbine trip signal I (except overspeed with an initiation signal)

(2) Opens at system flows ≤60 gpm if a RCIC automatic start signal has been initiated. Closes at system flow ≥120 gpm

(3) On Manual Trip a closure signal applied as long as trip pushbutton is depressed

d. Test valve FCV-71-38

(1) Normally closed. Automatically closes (if open) on initiation signal

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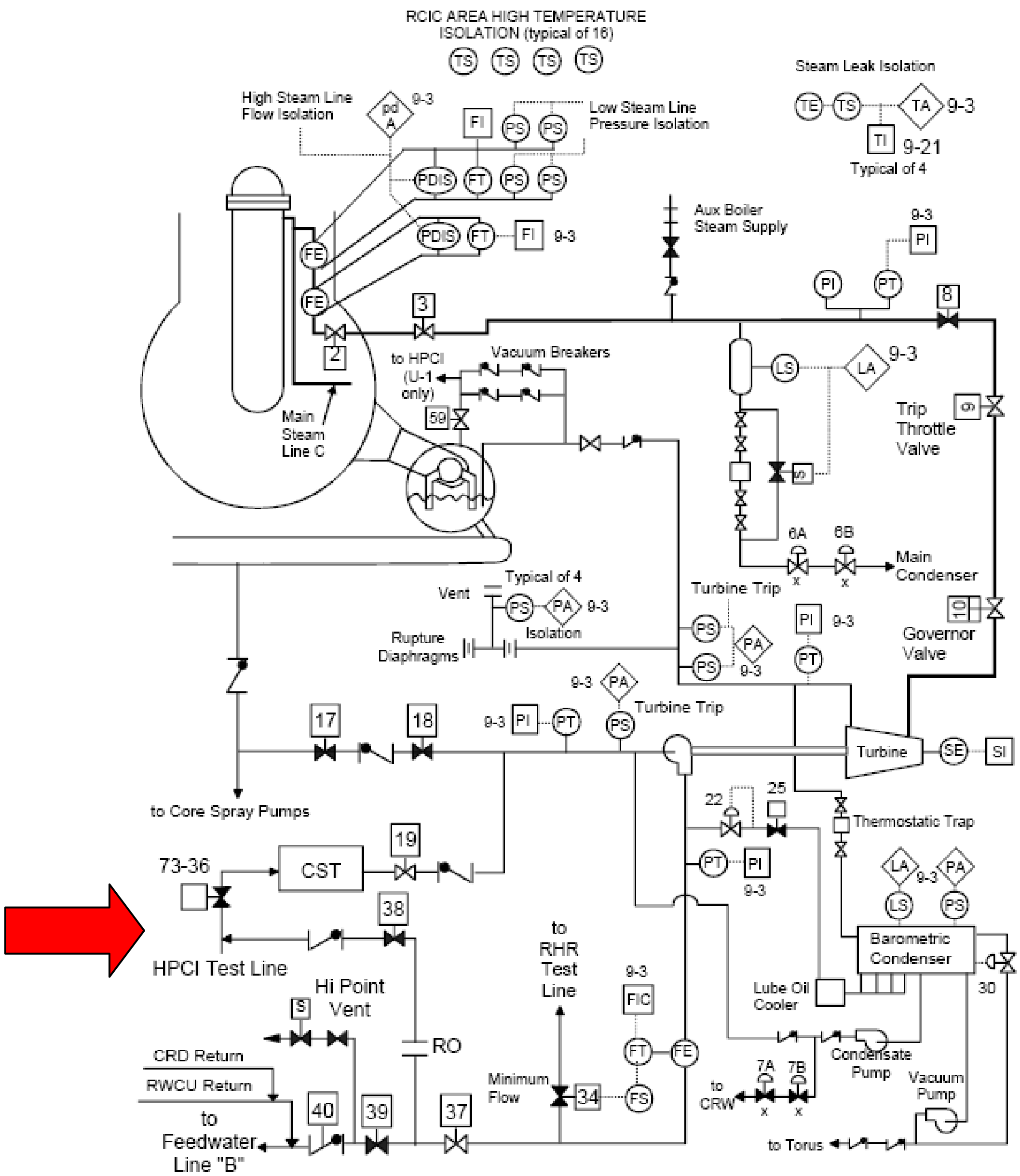

TP-1: RCIC System Flow Path

Examination Outline Cross-reference:	Level	RO	SRO
218000 ADS	Tier #	2	------
K4.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) design feature(s) and/or interlocks which provide for the following:	K/A #	218000K4.02	
• Allows manual initiation of ADS logic	Importance Rating	3.8	------

Proposed Question: **# 40**

A Recirculation Loop leak results in a Unit 2 Drywell Pressure of 2.5 psig.

Six minutes later, plant conditions are as follows:

- Reactor Water Level is (-) 110 inches
- Drywell Pressure is 5.1 psig
- Core Spray Pumps 2A **AND** 2D are being manually started
- **NO** other ECCS Pumps are available

Which ONE of the following identifies the status of ADS?

A. ADS Valves will **NOT** Automatically actuate **BUT** can be opened **MANUALLY**.

B. ADS Valves will open **IMMEDIATELY** if Reactor Water Level reaches Level 1.

C. ADS Valves will open 95 seconds after the 2A **AND** 2D Core Spray Pumps started.

D. ADS Valves will open **IMMEDIATELY** after the 2A **AND** 2D Core Spray Pumps started.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Following conditions must be met before ADS Initiation will occur: Two coincidental signals of High Drywell Pressure 2.45 psig and Reactor Level 1 (-) 122 inches or Reactor Level 1 (-) 122 inches for 265 seconds plus confirmatory low Reactor Level (+) 2 inches and any RHR Pump or either A or B and C or D Core Spray Pumps running. This answer is correct because the Level 1 initiation signal has not been satisfied.

B INCORRECT: This is incorrect because when level 1 is reached, the 95 second timer has to run out before ADS will initiate. Plausible if candidate thinks conditions already met for 95 second timer to start

C INCORRECT: The 95 second time delay does not start until three indications are provided (reactor vessel ≤ (+) 2 inches, and reactor vessel ≤ (-) 122 inches, and Drywell pressure ≥ 2.45 psig, or at or ≤ (-) 122 inches for 265 seconds. Plausible in that candidate may confuse Level 1 for 265 seconds with High Drywell Pressure for more than 265 seconds.

D INCORRECT: This is incorrect because although 1 RHR or 2 Core Spray pumps must be running, ADS will not initiate until RPV level reaches Level 1. Plausible in that candidate may confuse Level 1 for 265 seconds with High Drywell Pressure for more than 265 seconds and that conditions already met to start the 95 second timer.

Justification: To correctly answer this question, candidate must recognize condition not met for automatic initiation of ADS requiring manual operation of valves to initiate ADS. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.043 Rev 13 (Attach if not previously provided)

2-OI-1 Rev. 46

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.043 V.B.4 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam BFN 0801 #40

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

INSTRUCTOR NOTES

d. EOI Appendix 8G crossties CAD to DWCA

PROCEDURE USE & ADHERENCE

4. ADS systems controls

TP-2

a. Consists of pressure and water level sensors arranged in the trip systems that control a solenoid-operated pilot air valve

b. The solenoid-operated valve controls the pneumatic pressure applied to a diaphragm actuator which controls the SRV directly

DCN 51106 Cable & Switch configuration / modifications

c. Cables from sensors lead to the Control Room where logic arrangements are formed in cabinets

d. Control channels are separated to limit the effects of electrical failures

e. A two-position control switch is provided in the Control Room for control of the ADS valves

1) Two positions are OPEN and AUTO

HP Use SELF-CHECKING

2) In OPEN, the switch energizes a DC solenoid which allows pneumatic pressure to be applied to the diaphragm actuator of the relief valve

Pressure relief consists of actuation of reactor pressure on internal pilot or by electro-pneumatic operation via pressure switches.

NOTE:
The relief valves can be manually opened to provide a controlled nuclear system cooldown under conditions where the normal heat sink is not available

3) In AUTO, the valves are controlled by the ADS logic and pressure relief logic

f. Four of the six ADS valves may also be controlled from a backup control board which is provided to facilitate plant shutdown and cooldown from outside the Control Room

UNIT DIFFERENCE, DCN 51106 adds new panel “25-658” to Unit 1

5. Automatic Depressurization Initiation Logic

a. The following conditions must be met before automatic depressurization will occur

Obj. V.B.4
Obj. V.C.3
Obj. V.D.3
Obj. V.E.4

1) Two coincident signals of high drywell pressure (+2.45 psig) and low low low reactor vessel

water level (-122")
OR
-122" for 265 sec. — LT-3-58A-D

2) A confirmatory low reactor vessel water level signal (+2") (Tech Spec Value 0") — LT-3-184, LT-3-185

3) Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running — Obj. V.C.4, Obj. V.D.4

NOTE:
This signal comes from pressure switches on the discharge of the pumps which give permissives in the logic above a set pressure of 100 psig for RHR pumps and 185 psig for the Core Spray pumps.

RHR	CS
PS-74-8A and 8B (Pump A)	PS-75-7 (Pump A)
PS-74-31A and 31B (Pump B)	PS-75-35 (Pump B)
PS-74-19A and 19B (Pump C)	PS-75-16 (Pump C)
PS-74-42A and 42B (Pump D)	PS-75-44 (Pump D)

Associated shutdown boards must be energized for the respective pumps.

4) A 95-second timer must be timed out

b. The high drywell pressure signal seals in immediately upon receipt of the signal — Obj. V.C.4, Obj. V.D.4
1) Must be manually reset after the signal has cleared — PS-64-57A-D
2) Indicative of a breach in the process system barrier inside the drywell — HP Procedure Use and Adherence

c. The reactor vessel low water level signals (-122" and +2") indicate that fuel is in danger of becoming overheated — Obj. V.B.4, Obj. V.C.3, Obj. V.D.3, Obj. V.E.4
1) The -122" water level signal would not normally occur unless the HPCI System had failed — K 28, 29, & 30, Obj. V.C.4
2) These signals do not seal — Obj. V.D.4
3) The -122" water level initiation setpoint is selected to open the SRVs and depressurize the reactor vessel in time to allow fuel cooling by the Core Spray and LPCI Systems following a LOCA, in the event that the other makeup systems (Feedwater, CRD Hydraulic, RCIC, — TP-3, Obj. V.C.4, Obj. V.D.4

BFN Unit 2	Main Steam System	2-OI-1 Rev. 0046 Page 13 of 64

3.4 Main Steam Relief Valve (MSRV / ADS)

A. Whenever both the acoustic monitor and the temperature indication on a relief valve fail to indicate in the Control Room, the Technical Specifications Section 3.3.3.1 should be consulted to determine what limiting conditions for operation apply.

B. In the event that a relief valve fails to function as designed and the cause of the malfunction is not clearly determined and then corrected, the valve should be considered inoperable and Technical Specifications Section 3.5.1 and 3.4.3 should be consulted to determine what limiting conditions for operation apply.

C. ADS will initiate when ALL of the following conditions are met:

1. A confirmatory Low reactor water level signals (+2.0 inches), REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE, 2-9-3C Window 3

2. Two coincident signals for each of the following parameters:

   a. high drywell pressure (+2.45 psig) in conjunction with low low low reactor water level (-122 inches), ADS BLOWDOWN HIGH DRYWELL PRESS SEAL-IN, 2-XA-55-9-3C Window 33 and RX WTR LVL LOW LOW LOW ECCS/ESF INIT 2-LA-3-58A, 2-XA-55-9-3C Window 28

   OR

   b. low low low reactor water level (-122 inches), RX WTR LVL LOW LOW LOW ECCS/ESF INIT 2-LA-3-58A, 2-XA-55-9-3C Window 28, for 265 seconds (High drywell pressure bypass)

3. One RHR pump OR two Core Spray pumps (A or B and C or D) running, RHR OR CS PUMPS RUNNING ADS BLOWDOWN PERMISSIVE, 2-XA-55-9-3C Window 10.

4. When ALL of the above logic is satisfied, then a 95 second timer starts (ADS BLOWDOWN TIMERS INITIATED, 2-XA-55-9-3C, Window 11) and the timer must be timed out to initiate ADS blowdown.

D. Depressing 2-XS-1-159 and -161 on Panel 2-9-3 will reset the ADS Blowdown Timers. They also reset an ADS initiation, if the timers have timed out. ADS will re-initiate upon subsequent timing out of the timer provided the low level and pump logic signals still exist. The timer setpoint is 95 seconds, however setpoint tolerance allows it to be as low as 77 seconds.

BFN 0801 NRC #40

A Recirculation Loop leak results in Unit 2 Drywell Pressure of 2.5 psig.

Six minutes later, plant conditions are as follows:

- Reactor water level is (-) 110 inches and lowering slowly
- Drywell pressure is 5.1 psig

A Unit Operator manually starts Core Spray Pumps 2A **AND** 2C.

Which ONE of the following identifies the status of ADS?

A. ADS 95-second timer is still reset (i.e., has **NOT** yet started).

B. ADS Valves will open **IMMEDIATELY** after the Core Spray Pumps starts.

C. ADS Valves are closed but will open 95 seconds after the Core Spray Pumps start.

D. ADS Valves are closed but will open **IMMEDIATELY** when Reactor Level reaches Level 1.

Examination Outline Cross-reference:	Level	RO	SRO
218000 ADS	Tier #	2	------
K4.03 (10CFR 55.41.7)	Group #	1	------
Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) design feature(s) and/or interlocks which provide for the following: • ADS logic control	K/A #	218000K4.03	
	Importance Rating	3.8	------

Proposed Question: **# 41**

A small break LOCA has occurred on Unit 2 with a failure of **ALL** high pressure injection. Conditions have deteriorated to the point of automatic initiation of ADS.

Which ONE of the following describes when the ADS Valves will close?

A. When Reactor Pressure drops to 150 psig.

B. When **ALL** Low Pressure ECCS Pumps are secured.

C. When Reactor Water Level rises above (-) 122 inches.

D. When Reactor Pressure lowers to 20 psig above Suppression Chamber Pressure.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Reactor Pressure of 150 psig has no affect on ADS Logic. Plausible in that it is a recognizable value associated with ADS. ADS valves are not required to be OPERABLE with reactor steam dome pressure less than 150 psig.

B INCORRECT: The Low Pressure ECCS pumps need to be running to initiate ADS, but a seal in maintains initiation logic even if Low Pressure ECCS Pumps are removed from service. Plausible in that candidate may not recognize this portion of the logic is sealed in.

C INCORRECT: Level 1 (-122 inches) is required to initiate ADS, but a seal in maintains initiation logic even after Reactor Level rises above – 122 inches. Plausible in that candidate may not recognize this portion of the logic is sealed in.

D **CORRECT:** ADS SRVs will auto close with a differential pressure < 20 psig.

Justification: Tests candidates’ knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) design feature(s) and interlocks which provide for ADS logic control.

Technical Reference(s):	OPL171.009 Rev.11	(Attach if not previously provided)
	QPL171.043 Rev. 13	

Proposed references to be provided to applicants during examination: NONE

Learning Objective:	OPL171.009 V.B.2 OPL171.043 V.B.3	(As available)

Question Source:	Bank #	BFN 2004 #18	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	BFN 2004	

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

INSTRUCTOR NOTES

d. EOI Appendix 8G crossties CAD to DWCA

PROCEDURE USE & ADHERENCE

4. ADS systems controls

TP-2

a. Consists of pressure and water level sensors arranged in the trip systems that control a solenoid-operated pilot air valve

b. The solenoid-operated valve controls the pneumatic pressure applied to a diaphragm actuator which controls the SRV directly

DCN 51106 Cable & Switch configuration / modifications

c. Cables from sensors lead to the Control Room where logic arrangements are formed in cabinets

d. Control channels are separated to limit the effects of electrical failures

e. A two-position control switch is provided in the Control Room for control of the ADS valves

1) Two positions are OPEN and AUTO

HP Use SELF-CHECKING

2) In OPEN, the switch energizes a DC solenoid which allows pneumatic pressure to be applied to the diaphragm actuator of the relief valve

Pressure relief consists of actuation of reactor pressure on internal pilot or by electro-pneumatic operation via pressure switches.

NOTE:
The relief valves can be manually opened to provide a controlled nuclear system cooldown under conditions where the normal heat sink is not available

3) In AUTO, the valves are controlled by the ADS logic and pressure relief logic

f. Four of the six ADS valves may also be controlled from a backup control board which is provided to facilitate plant shutdown and cooldown from outside the Control Room

UNIT DIFFERENCE, DCN 51106 adds new panel “25-658” to Unit 1

5. Automatic Depressurization Initiation Logic

a. The following conditions must be met before automatic depressurization will occur

Obj. V.B.4
Obj. V.C.3
Obj. V.D.3
Obj. V.E.4

1) Two coincident signals of high drywell pressure (+2.45 psig) and low low low reactor vessel

INSTRUCTOR NOTES

water level (-122")
OR
-122" for 265 sec. — LT-3-58A-D

2) A confirmatory low reactor vessel water level signal (+2") (Tech Spec Value 0") — LT-3-184, LT-3-185

3) Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running — Obj. V.C.4, Obj. V.D.4

NOTE:
This signal comes from pressure switches on the discharge of the pumps which give permissives in the logic above a set pressure of 100 psig for RHR pumps and 185 psig for the Core Spray pumps.

RHR	CS
PS-74-8A and 8B (Pump A)	PS-75-7 (Pump A)
PS-74-31A and 31B (Pump B)	PS-75-35 (Pump B)
PS-74-19A and 19B (Pump C)	PS-75-16 (Pump C)
PS-74-42A and 42B (Pump D)	PS-75-44 (Pump D)

Associated shutdown boards must be energized for the respective pumps.

4) A 95-second timer must be timed out

b. The high drywell pressure signal seals in immediately upon receipt of the signal — Obj. V.C.4, Obj. V.D.4

1) Must be manually reset after the signal has cleared — PS-64-57A-D

2) Indicative of a breach in the process system barrier inside the drywell — HP Procedure Use and Adherence

c. The reactor vessel low water level signals (-122" and +2") indicate that fuel is in danger of becoming overheated — Obj. V.B.4, Obj. V.C.3, Obj. V.D.3, Obj. V.E.4

1) The -122" water level signal would not normally occur unless the HPCI System had failed — K 28, 29, & 30, Obj. V.C.4

2) These signals do not seal — Obj. V.D.4

3) The -122" water level initiation setpoint is selected to open the SRVs and depressurize the reactor vessel in time to allow fuel cooling by the Core Spray and LPCI Systems following a LOCA, in the event that the other makeup systems (Feedwater, CRD Hydraulic, RCIC, — TP-3, Obj. V.C.4, Obj. V.D.4

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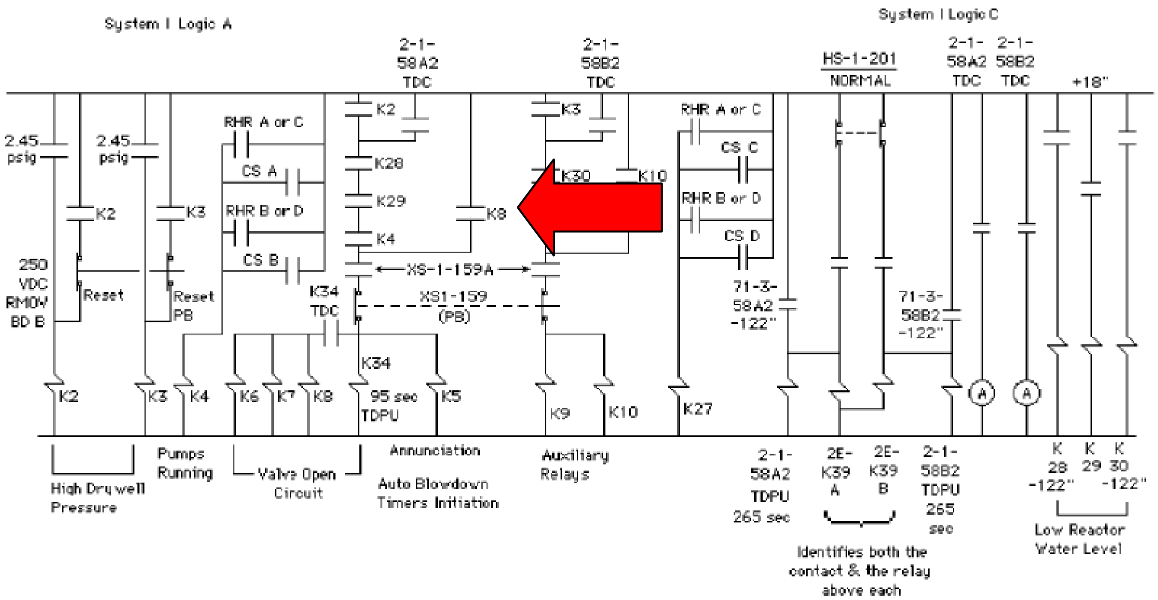

Keylocks placed in INHIBIT position activates annunciators on Panel 9-3 (Window 18C/31C)

XS-1-159A = Keylock Inhibit Selector Switch A and C logic

XS-1-161A = Keylock Inhibit Selector Switch B and D logic

TP-3 ADS Initiation Logic

c. Valve functions

(1) Safety function (the DC solenoid valves do not operate in this mode) protects against nuclear over pressurization (pressure actuated).

Review light indicators - green light only.

(2) Relief functions

(a) To provide automatic depressurization for small breaks in the nuclear steam system so that the LPCI mode of RHR and the Core Spray system can operate to protect the fuel barrier. This function is part of the Automatic Depressurization System (ADS). This function can be overridden with the use of two keylock switches on 9-3 (ADS LOGIC INHIBIT SWITCHES)

6 valves are used for ADS, which is discussed in OPL171.043.

Red light indicates solenoid energized <u>not</u> valve open.

Check redundant indications (SRV tail pipe temp, acoustic monitor, MSL flow lowering, Steam flow/feed flow mismatch and Rx power initially goes down, then goes to a higher indicated value)

(b) Manual operation by the operator

(c) The valves are designed such that 50 psi differential is required (between Reactor Pressure and Drywell Pressure) to open the valves and they will auto close with a differential pressure $\leq$ 20 psi.

TP-3

(d) Pressure switches are utilized to work in the 'relief mode'. The 'relief mode' energizes a solenoid valve to open the SRV. The setpoints are the same as in the safety mode. This is done to compensate for potential setpoint drift in the 'safety relief' mode. This function works in tandem with the 'safety mode' as a backup

TP-4

Browns Ferry Nuclear Plant 2004-301
SRO Inital Exam

18. 218000A1.04 001/T2G1//ADS/MEM 4.1/4.2/B BF04301/R/TCK

A small break LOCA has occurred on Unit 2 with a failure of all high pressure injection. Conditions have deteriorated to the point of auto initiation of ADS.

Which ONE of the following describes when the ADS valves will close assuming all ADS valves remain in Auto?

A. When all low pressure ECCS pumps are secured.

B. When reactor water level rises above -122 inches.

C. When reactor pressure drops below 150 psig.

D.✓ When reactor pressure lowers to 20 psig above suppression chamber pressure.

K/A 228000 A1.04 Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: Reactor pressure. (4.1/4.2)

References: QPL171.043, Rev.10, pg
OPL171.009, Rev.8, pg 17 of 57
Enabling Objective #B3 (OPbl71.043)

A. Incorrect since the pumps need to be running to initiate ADS, not to secure it

B. Incorrect since the ADS valves have already been actuated and remain that way until the reactor is depressurized.

C. Incorrect since this is the pressure at which the ADS valves are required to be operable and has no affect on ABS operation.

B. Correct answer.

Examination Outline Cross-reference:	Level	RO	SRO
223002 PCIS/Nuclear Steam Supply Shutoff	Tier #	2	------
K1.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of the physical connections and/or cause-effect relationships between the PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) / NUCLEAR STEAM SUPPLY SHUT-OFF and the following:	K/A #	223002K1.04	
• High pressure coolant injection: Plant-Specific	Importance Rating	3.5	------

Proposed Question: **# 42**

Which ONE of the following will result in a HPCI Group 4 Isolation on Unit 2?

A. Reactor Pressure of 108 psig.

B. HPCI Pump Room Temperature of 170º F.

C. HPCI Steam Line Flow at 150% of rated for 5 seconds.

D. HPCI Pressure between Exhaust Rupture Discs of 12 psig.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This pressure would cause Group 4 Isolation on Unit 1 ONLY. Unit 1 Reactor Pressure low pressure set point is 110 psig. Unit 2 Group 4 set point is 105 psig.

B INCORRECT: Pump Room Isolation Temperature is 185 º F. If Torus Area Temp > 165 º F a Group 4 Isolation will occur.

C INCORRECT: HPCI isolates on high steam line flow of 200% of rated for 3 seconds. This Steam Line flow rate is above Group 1 isolation of 135% and Group 4 time delay of 3 seconds is exceeded.

D **CORRECT:** HPCI High Pressure Between Rupture Discs of 10 psig results in a Group 4 Isolation.

Justification: Tests candidate's knowledge of parameters and setpoints that will result in HPCI Isolation from the Primary Containment Isolation System.

Technical Reference(s): 2-OI-73, Rev. 83 (Attach if not previously provided)

OPL171.042, Rev.20

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.042 V.B.2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 2	High Pressure Coolant Injection System	2-OI-73 Rev. 0083 Page 12 of 83

3.4 Initiation

A. When any of the following signals are received, the HPCI System automatically initiates:

1. Low RPV water level at -45".

2. High drywell pressure at 2.45 psig.

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.

C. HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, will open on receipt of an initiation signal even with HPCI Auxiliary Oil pump in PULL-TO-LOCK position resulting in slowly draining CST to Suppression Chamber.

3.5 Isolation

A. When any of the following signals are received, the HPCI System automatically isolates: (REFER TO 2-AOI-64-2b, Group 4 HPCI Isolation.)

1. High steamline flow at 85 psid(approximately 200%) of rated (3 sec time delay).

2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room.

3. Low RPV pressure at 105 psig (does not seal-in).

4. High pressure between rupture diaphragms at 10 psig.

5. Remote Manual HPCI (AUTO-INIT) MANUAL ISOLATION pushbutton, 2-HS-73-61, if automatic initiation signal is present.

INSTRUCTOR NOTES

(1) Minimum flow bypass valve (73-30) opens if it was closed.

Obj. V.D.5
Obj. V.E.6

(2) Suction valve from CST (73-40) opens if it was closed. (Provided at least one torus suction valve is closed).

Mgmt Exp 06-003: Once HPCI T&P'd iaw EOI App. 4, must evaluate per CAUTION #5 prior to re-start. Read 06-003.

(3) HPCI pump discharge valves to feedwater system (73-34 and 73-44) receive open signal.

(4) Test line isolation valves (73-35 and 73-36) close if they were open.

EFFECTIVE COMMUNICATION

(5) The steam supply valve to the turbine (73-16) opens.

(6) The auxiliary oil pump and GSC blower are started by initiation signal.

(7) As oil pressure increases, the turbine control and stop valves open when hydraulic pressure is available to admit steam to the turbine.

(8) The minimum flow bypass valve (73-30) will be shut automatically when HPCI flow becomes adequate.

(9) The turbine control system will maintain turbine speed to provide constant flow.

2. HPCI Isolation

TP-10
Obj. V.B.2.c
Obj. V.C.2.c
Obj. V.D.6
Obj. V.E.7

a. Conditions causing HPCI isolation

(1) Low reactor pressure 105 psig for Units 2&3, (one out of two twice). 110 psig for Unit 1. DCN 51237

Only signal that does not seal in
Unit Difference

(2) High HPCI area temperature $\geq$165°F (Torus Area) or $\geq$185°F (HPCI Pump Room) (one out of two twice)

SER 3-05

INSTRUCTOR NOTES

(3) High HPCI steam line flow (200%) (3-second time delay) (one out of two)

(4) Turbine exhaust inboard disc rupture 10 psig (one out of two twice)

(5) Remote manual isolation pushbutton if an auto initiation signal exists.

b. Actions on HPCI isolation — Obj. V.B.3.b, Obj. V.C.3.b, SER 3-05

(1) HPCI steam line isolation valves (73-2, 73-3, and 73-81) close.

(2) Suppression pool suction valves (73-26 and 73-27) close.

(3) HPCI turbine trips. This will close the minimum flow bypass to the suppression pool (73-30.)

c. Resetting an isolation

(1) Low reactor pressure isolation will be automatically reset if reactor pressure is restored. — 73-2, 73-3 & 73-81 valves will not auto open under any condition.

(2) The other isolation signals seal in. When the condition has cleared, push the HPCI auto isolation circuit reset pushbuttons.

d. HPCI Turbine Trip — TP-9

(1) Conditions causing HPCI turbine trip — Obj. V.B.2.b, Obj. V.C.2.b, Obj. V.D.7, Obj. V.E.8, SER 3-05

(a) High (Level 8) reactor vessel water level +51" (2 of 2)

(b) High HPCI turbine exhaust discharge pressure 140 psig (1 of 2)

BFN Unit 1	High Pressure Coolant Injection System	1-OI-73 Rev. 0011 Page 8 of 76

3.0 PRECAUTIONS AND LIMITATIONS

A. The HPCI turbine automatically trips on any of the following:

1. RPV water level high at +51 inches
2. Low pump suction pressure at 19.3" HG Vacuum (4.7 sec time delay)
3. Turbine high exhaust pressure at 140 psig
4. Any isolation signal
5. Remote Manual HPCI TURBINE TRIP pushbutton, 1-HS-73-18A

B. HPCI turbine overspeed at 122% (~5000 rpm) of rated speed (~4100 rpm) results in a hydraulic trip. The hydraulic trip occurs when operating oil is ported from the HPCI TURBINE STOP VALVE, 1-FCV-073-0018, causing the stop valve to close under spring force. Once the stop valve is closed, the piston of the hydraulic trip resets. With the HPCI turbine under load, the field-adjusted reset should occur between 2500 and 3000 rpm, and the startup sequence should commence. Since the overspeed trip condition does not result in any automatic trip signals in the HPCI control circuit, the HPCI PUMP MIN FLOW VALVE, 1-FCV-073-0030 does not close as a direct result of the turbine overspeed.

C. The HPCI System automatically isolates on any of the following: (REFER TO 1-AOI-64-2b, Group 4 HPCI Isolation.)

1. High steamline flow at 85 psid (~200% of rated) (3 sec time delay)
2. Steamline space temperature at 165°F Torus Area or 185°F HPCI Pump Room
3. Low RPV pressure at 110 psig (does not seal-in)
4. High pressure between rupture diaphragms at 10 psig
5. Remote Manual HPCI AUTO-INIT MANUAL ISOLATION pushbutton, 1-HS-73-61, if automatic initiation signal is present

D. HPCI System automatically initiates from one-out-of-two-taken twice logic from either: (**REFER TO** Section 5.1).

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3.2 Main Steam Isolation Valves (MSIV)

3.2.1 MSIV Closure

A. The MSIVs should be fast closed when the reactor is shutdown and no steam flow, unless required to be slow closed by surveillance, test instruction, or an abnormal condition. [BFNPER 164499]

B. When a MSIV is closed at power, the potential exists for an isolation of the Hydrogen Water Chemistry System to occur. This is due to the possibility of a hydrogen bubble becoming entrained in the main steam line drains and subsequently being released when the main steam line drains reposition in response to a MSIV closure. This scenario can result in a small Off Gas System hydrogen spike of sufficient strength to cause a automatic isolation of the Hydrogen Water Chemistry System.

C. Closure of all MSIVs could cause turbine shaft damage if main condenser vacuum is maintained and seal steam supply is not established from the auxiliary boiler.

3.2.2 MSIV Isolation

A. Main steam tunnel temperature should not be allowed to exceed 189°F to prevent MSIV isolation.

B. Whenever reactor pressure is reduced to 852 psig and the reactor mode switch is in RUN position, the MSIVs will close.

C. The MSIVs will close if 250 Vdc and 120 Vac power to the MSIV control logic is de-energized.

D. Reactor power should be ≤ 66% prior to closing an MSIV greater than 15 percent during closure testing. This should prevent a high steam line flow MSIV closure and subsequent reactor scram.

E. Placing all MSIV Handswitches in the Close Position allows the PCIS group one trip logic to be reset. Leaving any Handswitch in the Open Position prevents resetting the group one logic.

F. The PCIS group one trip parameters do not exceed trip setpoints.

1. Reactor water level above -122 in.
2. MSL flow less than 135%.
3. MSL tunnel temperature less than 189°F.
4. MSL pressure greater than 852 psig if in Mode 1.

(d)	MSIV testing problem	GE SIL 568 DCN T27975
i.	GE SIL 568 describes two cases where MSIVs failed to close upon command. LS-5 settings were revised to reflect 0-85% and 85-100% on Units-2 and 3. The poppet may hang up on grooves worn into the guide ribs (due to poppet vibration or rocking).	Cover SEOPR 96-02-001-06 on MSL flow differences due to DCN V36612A DCN 51143 DCN 51173 (Handout 2) All 3 units have had this Limit switch change
ii.	The 90% LS setting may not reveal this problem. Therefore, changing the limit switch settings to 85% will provide the operator with more useful information whether or not the poppet is free to travel.	PJB: Plan for success, but anticipate problems and evaluate contingencies
(6)	Operation from Panel 25-32	Obj. V.B.12 Obj. V.C.6
(a)	All MSIVs have control transfer switches and control switches on panel 25-32	Obj. V.D.4
(b)	When control of the valves is transferred to panel 25-32, automatic isolation signals are disabled. Valve position indication are also lost on panel 9-3 when control is transferred to panel 25-32.	
(7)	Signals which cause automatic closure of MSIVs are:	TP-9 Obj. V.B.10 Obj. V.E.6
(a)	Low-low-low reactor water level (Level 1) -122"	

(b) Steam tunnel high temperature, 189^{O}F

(c) High steam flow in any main steam line $\geq$ 135% .45 sec TD.

(d) Main Steam Line low pressure (852 psig) with Mode SW in "RUN"

(e) An automatic isolation causes: — Obj. V.B.11 Obj. V.E.7

(i) All MSIVs to close
(ii) Drain valves 1-55 and 1-56 to shut.

Briefly discuss other systems affected

(8) Reasons for isolation signals:

(a) -122" Rx Water Level (Level 1)

i. Low enough to prevent spurious initiation
ii. High enough to initiate isolation and ECCS so that:
- No melting of the fuel cladding occurs.
- Post accident cooling may be accomplished.
- The guidelines of 10 CFR 100 are not exceeded.

(b) 189°F Steam Tunnel High Temp

i. Detect small (15 gpm) steam leaks in the steam tunnel.
ii. Provide backup to high steam flow isolation on large breaks outside containment.
iii. Prevent exceeding 10 CFR 100 guidelines

Examination Outline Cross-reference:	Level	RO	SRO
223002 PCIS/Nuclear Steam Supply Shutoff	Tier #	2	------
K1.08 (10CFR 55.41.7)	Group #	1	------
Knowledge of the physical connections and/or cause-effect relationships between the PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) / NUCLEAR STEAM SUPPLY SHUT-OFF and the following: • Shutdown cooling system/RHR	K/A #	223002K1.08	
	Importance Rating	3.4	------

Proposed Question: **# 43**

Preparations are underway to place Unit 2 in Cold Shutdown following a Scram. When the operator started the 2B RHR Pump for Shutdown Cooling (SDC), Reactor Water Level lowered to 0 inches.

Which ONE of the following completes both of the following statements for using RHR Loop 1 LPCI to restore vessel level in accordance with 2-AOI-74-1, “Loss of Shutdown Cooling?”

The RHR SYS 1 SD CLG INBD INJECT ISOL RESET pushbutton, 2-XS-74-126, **__(1)__** to be depressed.

Following the start of Loop 1 RHR Pump, the operator is required to open **__(2)__**.

A. **(1)** is required
(2) RHR SYS I OUTBD INJECT VALVE, 2-FCV-74-52.

B. **(1)** is required
(2) RHR SYS I INBD INJECT VALVE, 2-FCV-74-53.

C. **(1)** is **NOT** required
(2) RHR SYS I OUTBD INJECT VALVE, 2-FCV-74-52.

D. **(1)** is **NOT** required
(2) RHR SYS I INBD INJECT VALVE, 2-FCV-74-53.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: (See attached excerpts) Part 1 correct - A Group II Isolation signal has been initiated at + 2 inches. Per 2-AOI-74-1, The RHR SYS 1 SD CLG INBD INJECT ISOL RESET pushbutton, 2-XS-74-126 is required to be depressed. Part 2 is incorrect - 2-74-52 is unaffected by the isolation signal, but will need to be re-throttled following re-establishment of flow [5.2.4].

B **CORRECT:** (See attached excerpts) A Group II Isolation signal has been initiated at + 2 inches. Per 2-AOI-74-1, The RHR SYS 1 SD CLG INBD INJECT ISOL RESET pushbutton, 2-XS-74-126 is required to be depressed. 2-FCV-74-53 is affected by the isolation signal, and will need to be re-opened, [5.2.3].

C INCORRECT: Part 1 is incorrect as detailed in ‘B’ above. Part 2 is incorrect as detailed in ‘A’ above.

D INCORRECT: Part 1 is incorrect and part 2 is correct.

RO Level Justification: Tests the candidate's knowledge of physical connections (logic) and cause-effect relationships between PCIS and the Shutdown Cooling Mode of RHR. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-AOI-74-1, Rev. 33 (Attach if not previously provided)
2-OI-74, Rev. 150
OPL171.017, Rev. 15
OPL171.044, Rev. 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.017 V.B.2.g / 4 (As available)
OPL171.044 V.B.10 / 19

Question Source: Bank # **X** 0801 # 30
Modified Bank # (Note changes or attach parent)
New

Question History: HLT 0801
Last NRC Exam (7 / 2009)

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

F. Valve Groups

The isolation valves at BFN are categorized into one of seven "groups" (1,2,3,4,5,6,8), based generally upon the type of system(s) isolated and associated isolation signals.

Group 7 was removed from PCIS long ago. These are HPCI/RCIC drain valves which close on system initiation.

Details regarding valves in each group are provided in Table 1.
A basic description of each group is as follows:

1. Group 1

Obj. V.B.2
Obj. V.C.2

This group includes the Main Steam Isolation Vavles (MSIVs), main steam line drains, and reactor water sample line isolation valves.

The signals which will initiate a Group 1 isolation are as follows:

RPV low low low level (-122" or Level 1)
*MSL High Flow .45 sec TD (135%)
*MSL Area High Temperature (189°F)
*MSL Low Pressure (852 psig Mode Switch in RUN)

*(MSIVs and MSL Drains only)

2. Group 2

Obj. V.C.2
Obj. V.B.2

This group includes Residual Heat Removal (RHR), drywell floor/equipment drain sump, and PSC Head Tank Pump suction valves.

The signals which will initiate a Group 2 isolation are:

2-730E927-13

RPV low level (+2"or Level 3)
Drywell High Pressure (+2.45 psig)
Reactor High pressure (105psig) (SDC) only.

TABLE 1
(continued)

Initiation Signals	**Group 2**	Valve Type	Location Ref. to Drywell	Power to Open (3)	Power to Close (4)	Re-Open Manipulation
Level 3; Rx low level: +2"	RHR shutdown cooling supply (FCV 74-47)	MO Gate	Outside	DC	DC	(9)
Hi Drywell Press +2.45 psig	RHR shutdown cooling supply (FCV 74-48)	MO Gate	Inside	AC	AC	(9)
Hi Reactor pressure <105 psig. SDC only.	RHR LPCI to Reactor (FCV 74-53 &67)	MO Gate	Outside	AC	AC	(10)
	Drain pump A isolation valve (FCV 75-57 & -58)	AO Gate	Outside	Air/AC	Spring	(9)
	Drywell equipment drain disch isolation valves (FCV 77-15A &-15B)	AO Gate	Outside	Air/AC	Spring	(9)
	Drywell floor drain disch isolation valves (FCV 77-2A &-2B)	AO Gate	Outside	Air/AC	Spring	(9)

Switch out of STOP initiates RWCU isolation.

H. Isolation Reset and Bypass

1. Isolation Reset

The conditions/operator actions necessary to reset the isolation signals for each group are covered in detail in the associated system lesson plans. General information for each isolation group is provided below:

a. Group 1

Obj. V.B.3
Obj. V.C.3

Resetting a Group 1 isolation requires all MSIV control switches to be in the CLOSE position, and the RESET switches on panel 9-4 (HS-64-16A-S32/S33) to be rotated both to the left (however, operational practice is to rotate to both the left and right). Rotating the switches to the left re-energizes the AC and DC solenoids.

Operations Work Expectation

PCIS light on 9-4 will illuminate when both switches are taken to the left

b. Group 2

Obj. V.B.3
Obj. V.C.3

All Group 2 isolation valves are reset by rotating the RESET switches on Panel 9-4 to the right, except for the LPCI Injection valves (FCV-75-53 & 67), which only requires depression of the FCV-74-53(67) SD CLG ISOL RESET pushbuttons (XS-74-126/132) on Panel 9-3.

Operational practice is to rotate to both the left and right (Operations Work Expectation)

c. Group 3

Obj. V.B.3
Obj. V.C.3

The Group 3 (RWCU) isolation valves are reset by rotating the RESET switches on Panel 9-4 to the right.

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INSTRUCTOR NOTES

(5) Interlock with RHR pump, SDC path suction lineup established or pump trip occurs

Use of 74 for flush prior to SDC.
Obj. V.D.4

(6) Controls Circuits

VALVE	1-74-2	1-74-13	1-74-25	1-74-36	2-74-2	2-74-13	2-74-25	2-74-36
Operate Interlock	74-1/57	74-12/57	74-24/71	74-35/71	74-1/57	74-12/57	74-24/71	74-35/71
Outgoing interlock	74-7	74-7	74-30	74-30	74-7	74-7	74-30	74-30
Normal/Emerg Sw	X	X	X	X	X	X	X	X
Local Controls	X	X	X	X	X	X	X	X
Controls at Bkr	X	X	X	X	X	X	X	X
Panel 9-3 controls	X	X	X	X	X	X	X	X
Local Indication lgts	X	X	X	X	X	X	X	X
Lights on Bkr	X	X	X	X	X	X	X	X
Lights on Pnl 9-3	X	X	X	X	X	X	X	X
VALVE	**3-74-2**	**3-74-13**	**3-74-25**	**3-74-36**				
Operate Interlock	74-1/57	74-12/57	74-24/71	74-35/71				
Outgoing Interlock	74-7	74-7	74-30	74-30				
Normal/Emerg Sw	X	X	X	X				
Local Controls	X	X	X	X				
Controls at Bkr	X	X	X	X				
Panel 9-3 controls	X	X	X	X				
Local Indication lgts	X	X	X	X				
Lights on Bkr	X	X	X	X				
Lights on Pnl 9-3	X	X	X	X				

c. Shutdown Cooling suction supply valves

(Inbd-48; Outbd-47)
TP-21, 22 and 23
Obj. V.C.5

(1) No automatic opening interlocks

(2) Valves automatically close on:

(a) Reactor pressure > 100 psig

(b) Group 2 isolation (drywell pressure > 2.45 psig or level < 2 inches)

Obj. V.D.8

(3) Switch 74-157 (480 RMOV 1/2A) must be in SHUTDOWN to operate 74-48 from any location. (Units 1 & 2 only)

The breaker for 74-47 is normally open for App R concerns

(4) "EMERGENCY" position allows operation at breaker only. Removes PCIS logic closure circuit.

Emergency Switch on both valves on all three units.

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3.6 Interlocks (continued)

d. Opening RHR SYSTEM I(II) MIN FLOW VALVE 2-HS-74-7A(30A) from 2-PNL-9-3, with the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 2-HS-74-148(149), in INHIBIT, causes the minimum flow valves to travel full open and full close unless the RHR SYSTEM I(II) MIN FLOW VALVE, 2-HS-74-7A(30A) is placed in closed position to break the OPEN seal in contacts.

e. [II/C] Local operation of the RHR minimum flow valves bypasses the intended function of the Minimum Flow Inhibit switch and can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFPER941099]

f. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 2-HS-74-148(149), with the respective RHR loop in standby readiness, can cause inadvertent damage to that loop RHR pump(s) should RHR pump(s) auto start. [BFA-890790003P]

g. [PRD/C] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT Switch, 2-HS-74-148(149), with the respective RHR loop in Shutdown Cooling, can cause inadvertent drainage of the Reactor vessel to the Suppression Pool. [BFA-890790003P]

6. The RHR Outboard LPCI injection valves, 2-FCV-74-52(66), have throttling capability. They receive an auto open signal in the presence of a LPCI initiation signal when Reactor pressure is ≤450 psig and are interlocked open under these conditions or until the appropriate LPCI SYS I (SYS II) OUTBD INJ VLV BYPASS SEL keylock Switch, 2-HS-74-155A(155B), is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is >450 psig if its in-line companion valve 2-FCV-74-53(67) is not fully closed.

7. If Unit 2 reactor pressure exceeds 100 psig or a Group II isolation occurs on Unit 2 while Shutdown Cooling is in operation, the following will occur for the given condition:

(100 psig)	RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 <u>and</u> 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.
(Group II)	RHR SYS I and II LPCI INBD INJECT VALVEs, 2-FCV-74-53 and 2-FCV-74-67, close <u>and</u> Unit 2 RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.

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3.6 Interlocks (continued)

8. To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton is required to be depressed after either of following occur:

   a. Isolation signal has been reset **OR**

   b. 2-FCV-74-47 **OR** 2-FCV-74-48 is fully closed.

9. If after a GROUP II Isolation, RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67) is given an OPEN signal prior to depressing the RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET 2-XS-74-126(132), then the valve will trave full open and full close unless given a close signal prior to traveling full open.

10. The RHR spray/cooling valves, 2-FCV-74-57(71), receive an auto closure signal in the presence of a LPCI initiation signal and they are interlocked to prevent opening if the in-line torus spray valve, 2-FCV-74-58(72), is not fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

    a. Reactor level is >2/3 core height **AND**

    b. LPCI initiation signal is present **AND**

    c. The select reset switch is in the SELECT position.

    The requirements for >2/3 core height and a LPCI initiation signal may be BYPASSED using the keylock bypass switch, 2-XS-74-122/30.

11. If primary containment cooling is desired with reactor level at <2/3 core height, the keylock bypass switch is required to be placed in BYPASS before the select reset switch is placed in SELECT to ensure relay logic is made up.\

12. The RHR torus spray valves, 2-FCV-74-58(72), have the same in-line valve interlocks as those outlined in 3.6A 10 for the torus spray/cooling valves. Additionally these valves have an interlock preventing opening unless drywell pressure is ≥1.96 psig which cannot be bypassed.

13. The RHR torus cooling/test valves, 2-FCV-74-59(73), receive an auto closure signal in the presence of a LPCI initiation signal. Auto closure may be bypassed by the same conditions/actions outlined in Step 3.6A.10

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4.2 Subsequent Actions (continued)

[5] **IF** Shutdown Cooling isolates on low RPV water level or high Drywell press (GROUP 2 ISOL) AND RPV water level needs restoring using LPCI, **THEN** (Otherwise N/A)

PERFORM the following before reaching -122 inches RPV water level: ☐

NOTE

The LPCI inboard injection valve that is aligned per 2-POI-74-2 will already be in the required accident position with the breakers open and will **NOT** isolate.

[5.1] **PERFORM** the following on a group 2 isolation: ☐

[5.1.1] **IF** 2-POI-74-2 is in effect, **THEN**

VERIFY CLOSED one of the following valves: (Otherwise N/A) ☐

- RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-47. ☐
- RHR SHUTDOWN COOLING SUCT INBD ISOL VLV, 2-FCV-74-48.

<u>**AND**</u> ☐

- **VERIFY CLOSED** the LPCI inboard injection valve **NOT** aligned for 2-POI-74-2, (RHR SYS I LPCI INBD INJECT VALVE, 2-FCV-74-53 OR RHR SYS II LPCI INBD INJECT VALVE, 2-FCV-74-67) ☐

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4.2 Subsequent Actions (continued)

[5.1.2] **IF** 2-POI-74-2 is **NOT** in effect, **THEN**

VERIFY CLOSED the following valves on a Group 2 isolation: ☐

- RHR SHUTDOWN COOLING SUCT OUTBD ISOL VLV, 2-FCV-74-47. ☐
- RHR SHUTDOWN COOLING SUCT INBD ISOL VLV, 2-FCV-74-48. ☐
- RHR SYS I LPCI INBD INJECT VALVE, 2-FCV-74-53. ☐
- RHR SYS II LPCI INBD INJECT VALVE, 2-FCV-74-67. ☐

[5.2] **DEPRESS** RHR SYS I(II) SD CLG INBD INJECT ISOL RESET, 2-XS-74-126 and 2-XS-74-132. ☐

[5.2.1] **VERIFY** 2-IL-74-126 and 2-IL-74-132 extinguished. ☐

[5.2.2] **VERIFY** RHR Pumps in the RHR Loop **NOT** aligned for Shutdown Cooling, are operating. ☐

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

[5.2.4] **THROTTLE OPEN**, RHR SYS I(II) LPCI OUTBD INJECT VALVE, 2-FCV-74-52(66). ☐

[5.3] **IF** the RHR loop that was in shutdown cooling is needed for RPV water level makeup, **THEN**

PERFORM the following: ☐

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

Examination Outline Cross-reference:	Level	RO	SRO
239002 SRVs	Tier #	2	------
K2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following:	K/A #	239002K2.01	
• SRV solenoids	Importance Rating	2.8	------

Proposed Question: **# 44**

Which ONE of the following completes the statements?

ALTERNATE electrical power for those Unit 3 Safety Relief Valve (SRV) Solenoids, where available, is supplied from 250 VDC **__(1)__**.

Upon experiencing undervoltage conditions on the normal power supply, the transfer to SRV Solenoid alternate power supplies **__(2)__**.

A. **(1)** RMOV Boards **ONLY**.
(2) occurs automatically.

B. **(1)** RMOV Boards **ONLY**.
(2) **MUST** be performed manually.

C. **(1)** RMOV Boards **AND** Battery Boards.
(2) occurs automatically.

D. **(1)** RMOV Boards **AND** Battery Boards.
(2) **MUST** be performed manually.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: First part incorrect. Second part correct– see 'C' below. ALL primary sources of power are from 250 VDC RMOV Boards, lending plausibility to the first part.

B INCORRECT: Both parts incorrect – see 'C' below. During Control Room Abandonment (See attached 3-AOI-100-2 Excerpt), manual transfers associated with power supplies are necessary, lending plausibility.

C **CORRECT:** (See attached Lesson Plan excerpts) – *Primary* power supplies to SRVs are from **ONLY** 250 VDC RMOV Boards. *Alternate* power supplies are from BOTH 250 VDC RMOV **AND** Battery Boards. Undervoltage relays, located in the Remote Shutdown Panels, will automatically transfer power supplies on a loss of normal voltage.

D INCORRECT: First part correct. Second part incorrect– see 'C' above.

RO Level Justification: Tests whether the candidate has knowledge of power supplies to SRV solenoids and how those power supplies interrelate. Adding to the level of difficulty, these valves with alternate power supplies are utilized at the Remote Shutdown Panel; where a MANUAL transfer of power supplies to EMERGENCY is initiated. Additionally, normal power supplies are distributed among the three 250 VDC boards such that candidate may assume there is adequate time to manually transfer power, as the SRV LCO is predicated upon the Safety Mode only.

Technical Reference(s): OPL171.043, Rev. 13 (Attach if not previously provided)

OPL171.009, Rev. 11

3-AOI-100-2, Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.3/5 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

INSTRUCTOR NOTES

5. ADS valves required Drywell Control Air for pilot actuation mode

D. Unit Differences

1. Valves controlled from 25-32

a. Unit 1, 1-5, 1-22, 1-30, 1-34

b. Unit 2, 1-5, 1-22, 1-30, 1- 34

c. Unit 3, 1-5, 1-22, 1-34, 1-41

2. Unit 3 power supplies

a. 1-5, 1-34 250V RMOV Bd 3C, alternate from battery board 2, Panel 7

b. 1-18, 1-19 250V RMOV Bd 3B

c. 1-22, 1-41 250V RMOV Bd 3A, with first alternate 250V RMOV Bd 3C and second alternate Battery Board 2 panel 7

E. Technical Specifications

Obj. V.B.6
Obj. V.C.6

1. Section 3.3.3.2
2. Section 3.3.5.1
3. Section 3.5.1
4. TRM Section 3.3.3.4

F. Industry Events

1. LER # 1999-004-00 – (Limerick Station) On 6/04/99, an engineering review of Tech Specs, UFSAR, and Design Bases Documents determined that the backup instrument gas bottles are required to support ADS valve operability. It was subsequently determined that clearances had previously been applied that removed a portion of the backup instrument gas bottles from service for greater that the TS LCO allowed

Instructor review LER and discuss applicable portions with class

QUESTIONING ATTITUDE

INSTRUCTOR NOTES

f.	The power supply for the LOGIC and the solenoid valves is 250VDC	All ADS valves with alternate power supplies can be manually operated from backup control panel (25-32)
g.	250V RMOV Bd B supplies LOGIC Power for both system I & II	
h.	A Loss of 250V RMOV Bd B would prevent actuation	
i.	250V RMOV Bd A supplies Power for relays in system II of ADS Logic	
j.	A Loss of 250V RMOV Bd A would prevent system II actuation	
k.	PCV 1-22 is powered from 250V RMOV Board 2A with alternate supply from 250V RMOV Board 2B	See section F. Unit Differences for U-3 Power Supplies
l.	PCVs 1-19, 1-31 are powered from 250V RMOV Board 2B. There is no alternate power to these valves	
m.	PCVs 1-5 and 1-34 are normally powered from 250V RMOV Board 2C with alternate power supply from Battery Board 1 panel	
n.	PCV 1-30 is normally powered from 250V RMOV Board 2A with a first alternate to 250V RMOV Board 2C and a second alternate to Battery Board 1 panel 7	
o.	Valves powered from 250V RMOV Bd 2C required alternate sources due to RMOV Board 2C not being environmentally qualified for a line break in secondary containment	
p.	The transfer occurs automatically when undervoltage relays (mounted on panel 2-25-32) sense a loss of power to 250V RMOV Bd 2	DCN 51106

i. Manual demand (hand switch)

ii. Automatic blowdown demand (ADS) for 6 valves which are controlled by ADS.

iii. RPV high pressure

PT-3-204A-D

(e) The operating air is supplied from the drywell control air system.

(f) The SRV solenoids are powered from 250 VDC RMOV Boards or Battery Boards. Some SRV power supplies have relays in the bottom of panel 25-32 that allow them to swap to an alternate supply when the normal supply is lost.

Appendix C: shows SRV's that have alternate power supply (s).

(i) On Unit 1 , SRV's 1-5, 1-22,1-30, and 1-34 have auto transfer capabilities (for power supplies)

(ii) On Unit 2, SRV's 1-5, 1-22, 1-30, and 1-34 have auto transfer capabilities (for power supplies).

(iii) On Unit 3, SRV's 1-5, 1-22, 1-34, and 1-41 have auto transfer capabilities (for power supplies).

(g) Loss of air or power to an SRV would inhibit the relief function but not the safety function. Per TS 3.4.3 MSRV operability is based on the safety function (spring action) and not the 'relief' function

Obj. V.B.4

BFN Unit 3	Control Room Abandonment	3-AOI-100-2 Rev. 0018 Page 9 of 92

4.2 Unit 3 Subsequent Actions (continued)

[3] **PLACE** the following MSRV disconnect switches in DISCT at Panel 3-25-32:

Switch No.	Description	
3-XS-1-4	MAIN STM LINE A RELIEF VALVE DISCT	☐
3-XS-1-42	MAIN STM LINE D RELIEF VALVE DISCT	☐
3-XS-1-23	MAIN STM LINE B RELIEF VALVE DISCT	☐
3-XS-1-30	MAIN STM LINE C RELIEF VALVE DISCT	☐
3-XS-1-180	MAIN STM LINE D RELIEF VALVE DISCT	☐

[4] **PLACE** the following MSRV transfer switches in EMERG at Panel 3-25-32:

Switch No.	Description	(√)
3-XS-1-22	MAIN STM LINE B RELIEF VALVE XFR	☐
3-XS-1-5	MAIN STM LINE A RELIEF VALVE XFR	☐
3-XS-1-41	MAIN STM LINE D RELIEF VALVE XFR	☐
3-XS-1-34	MAIN STM LINE C RELIEF VALVE XFR	☐

NOTE

Use of the following sequence when opening MSRVs should distribute heat evenly in the Suppression Pool.

[5] **MAINTAIN** Reactor Pressure between 800 and 1000 psig using the following sequence at Panel 3-25-32: ☐

A. 3-HS-1-22C, MAIN STM LINE B RELIEF VALVE ☐

B. 3-HS-1-5C, MAIN STM LINE A RELIEF VALVE ☐

C. 3-HS-1-41C, MAIN STM LINE D RELIEF VALVE ☐

Examination Outline Cross-reference:	Level	RO	SRO
259002 Reactor Water Level Control	Tier #	2	------
G2.2.40 (10CFR 55.41.10)	Group #	1	------
Ability to apply technical specifications for a system.	K/A #	259002G2.2.40	
	Importance Rating	3.4	------

Proposed Question: **# 45**

The following conditions exist on Unit 1:

- Reactor Power is 28%
- NORMAL RANGE Level indicator, 1-LI-3-208D, is failed **HIGH** (> 60 inches)

The Unit Operator subsequently observes that NORMAL RANGE Level indicator, 1-LI-3-208A, is drifting upscale.

Which ONE of the following completes the statements?

Tech Spec 3.3.2.2, "Feedwater and Main Turbine High Water Level Trip Instrumentation," **__(1)__** applicable for the current plant conditions.

If 1-LI-3-208A reaches **FULL** scale, the running RFPTs **__(2)__** trip.

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

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

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

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

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: Part 1 is incorrect – Plausible in that Reactor Power is less than 30%. Other Instrumentation TS associated with the Main Turbine are NOT applicable with Power < 30%. Example - Main Turbine Stop Valve Closure / Control Valve Fast Closure, TS 3.3.1.1. Part 2 incorrect – Plausible in that with 2 channels upscale, candidate may believe conditions are met for Level 8 Reactor Feed Pump Trip.

C **CORRECT**: Part 1 is correct - TS 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation is applicable with Reactor Power ≥ 25%. Part 2 is correct either 208A and 208C or 208B and 208D must be picked up to trip the RFPTs. Combination of A and D channels will not cause a trip.

D INCORRECT: Part 1 incorrect – See Explanation B. Part 2 correct – See Explanation C.

Justification: Question requires candidate to demonstrate ability to apply Tech Specs for Reactor Level Instrumentation by recognizing condition which the associated Tech Spec would be applicable. Per Tech Spec 3.3.2.2, feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.026 Rev. 14 (Attach if not previously provided)
OPL171.003 Rev. 19
U1 TS 3.3.2.2 Am 234

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.003 V.B.20 (As available)
OPL171.026 V.B.5

Question Source:	Bank #		
	Modified Bank #	171.026 #3	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

a. The ELEC position of the OVERSPEED TEST TRIP LOCKOUT switch removes electrical overspeed trip for testing. All other trips remain functional.

OI-3 section 8.9

b. 'Amber' light below tachometer on 9-6 and locally will be lit when electrical overspeed condition is reached. (Unit 3 flashes, Unit 2 does not.)

Unit difference

c. Testing of the turbine stop valves is required but the high pressure stop valve can only be tested if the HP control valve is fully closed. Depressing the pushbutton on 9-6 causes the valve to close until it reaches its fully closed position or the pushbutton is released.

d. The Low Pressure Stop Valve can be tested at any time. Depressing the pushbutton on 9-6 causes the valve to travel to the mid (50%) position and remain until the pushbutton is released.

e. High Water Level Trip

1) High water level trip at 55" comes off of LS-3-208A, B, C, D.

These are uncompensated indicators

2) Logic is such that it is 2-out-of-2 taken once. For example, in order for a full turbine trip to occur, either 208A and 208C or 208B and 208D must be picked up.

3) Trip Channel 'A' is 208A & 208C; Trip Channel "B" is 208B & 208D.

4) Two sets of indicating lights (red & green) are installed on panel 9-5 and two reset switches. Normal condition - Green Light on; Trip condition - Red light on;

5) Ready to reset condition - Green & Red lights on

REACTOR VESSEL LEVELS

Level 8	+55 +51	Main Turbine and RFPT trip (LT-3-208) Instruments 208 A→ D Channel A = 208 A & C; or Channel B = 208 B & D = 2-out-of-2 coincidence and HPCI and RCIC turbine trip @ +51" (logic same as above)
Level 7	+39	High water level alarm (REACTOR WATER LEVEL ABNORMAL annunciation 2-9-5A window 8)
Level 6	< +39	Top end of programmed level control range
Level 5	> +27	Low end of programmed level control range
Level 4	+27	Low water level alarm (REACTOR WATER LEVEL ABNORMAL annunciation 2-9-5A window 8)
Level 3	+2	LT-3-184 &18 Reactor scram PCIS Groups 2,3,6,8 isolations Rx Bldg. ventilation isolation SGT trains auto start CREV System initiates SD Cooling suction valves close ADS confirmatory low level EOI-1 entry condition
Level 2	-45	LT-3-58 A→D HPCI initiates RCIC initiates ATWS/ARI/RPT trip
Level 1	-122	PCIS Group 1 isolation Diesel Generators start Core Spray initiates LPCI initiates ADS initiation logic EECW pumps start
Level 0	-183	Containment Spray permissive interlock (2/3 core height)

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels of feedwater and main turbine high water level trip instrumentation per trip system shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more feedwater and main turbine high water level trip channels inoperable, in one trip system.	A.1	Place channel(s) in trip.	7 days
B. One or more feedwater and main turbine high water level trip channels inoperable in each trip system.	B.1	Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

RPS Instrumentation
3.3.1.1

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level - High (continued)					
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 46 gallons
8. Turbine Stop Valve - Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low(d)	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 550 psig
10. Reactor Mode Switch - Shutdown Position	1,2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.14	NA
12. RPS Channel Test Switches	1,2	2	G	SR 3.3.1.1.4	NA
	5(a)	2	H	SR 3.3.1.1.4	NA

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

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

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

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

. **OPL171.026 3**
The following conditions exist on Unit 2:

- 100% Reactor Power
- NORMAL RANGE Level indicator, 2-LI-3-208D, is failed high (>60 inches)

The Unit Operator subsequently observes that NORMAL RANGE Level indicator, 2-LI-3-208A, is drifting upscale.

Which ONE of the following describes what action(s), if any, would occur should 2-LI-3-208A reaches full scale?

A. NO trip action would occur.

B. ONLY the Main Turbine would trip.

C. ONLY the running RFPTs would trip.

D. The running RFPTs AND Main Turbine would trip.

ANSWER: A

Examination Outline Cross-reference:	Level	RO	SRO
261000 SGTS	Tier #	2	------
A1.01 (10CFR 55.41.5)	Group #	1	------
Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM (SGTS) controls including:	K/A #	261000A1.01	
• System flow	Importance Rating	2.9	------

Proposed Question: **# 46**

Which ONE of the following completes the statements?

Following automatic initiation, Standby Gas Treatment System (SGTS) flow to STACK can be monitored in **__(1)__ .**

SGTS is designed such that a **MINIMUM** of**__(2)__** SGTS Subsystem(s) can maintain a negative 1/4-inch of H20 vacuum in Secondary Containment with a **MAXIMUM** inleakage flow of 12,000 scfm.

A. **(1)** Unit 1 Control Room **ONLY**
(2) ONE

B. **(1) ALL** 3 Control Rooms
(2) ONE

C. **(1)** Unit 1 Control Room **ONLY**
(2) TWO

D. **(1) ALL** 3 Control Rooms
(2) TWO

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that some SGTS indications are available in ALL 3 Control Rooms and some are not. Additionally, per 0-OI-65, “Standby Gas Treatment System,” Precautions and Limitations, it is recommended that the trains be started from the Units 1 and 2 Control Rooms due in part to the availability of instrumentation. Part 2 incorrect – SGT System is designed such that two subsystems are required to perform SGT Design Function. Plausible in that EOI App 12 only requires at least one SGTS running for CTMT Venting.

B INCORRECT: Part 1 correct – See Explanation D. Part 1 incorrect – See Explanation A.

C INCORRECT: Part 1 incorrect – See Explanation A. Part 1 correct – See Explanation D.

D **CORRECT:** Part 1 correct - SGTS Total Flow is determined by taking sum of Flow to Stack Indicators 0-FI-65-50B/1(2)(3) and 0-FI-65-71B/1(2)(3). Indicators are located on Panels 1-9-20(Unit 1), 2-9-20(Unit 2), and 3-9-20(Unit 3). Part 2 correct - Upon a secondary containment isolation, the SGT System is designed such that two subsystems maintain a negative 1/4-inch of H20 vacuum in Secondary Containment with an inleakage flow of 12,000 scfm.

Justification: Question tests candidates Ability to monitor changes in system flow associated with operating the STANDBY GAS TREATMENT SYSTEM (SGTS).

Technical Reference(s): O-OI-65 Rev. 52 (Attach if not previously provided)

OPL171.018 Rev. 10

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.018 V.B.1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0052 Page 8 of 41

3.0 PRECAUTIONS AND LIMITATIONS

A. Upon a secondary containment isolation, the SGT System is designed to maintain a negative 1/4-inch of H_20 vacuum in Secondary Containment with an inleakage flow of 12,000 cfm.

B. [NRC/C] All three trains will remain in operation during an accident to satisfy single failure criteria and to minimize the potential release of radioactivity from the Reactor Building into the Control Building air supply intake ducts. [NRC NCO 88 0193 004]

C. [NER/C] Steps should be taken to minimize dust loading and to prevent paint vapors, petroleum fumes, welding smoke, and other airborne contaminants from reaching the HEPA filters and charcoal adsorbers. Normal ventilation should be in operation for a minimum of two (2) hours after painting, fire, smoke, or chemical release has terminated prior to operating SGT System. [CAQR SQP890064]

D. If the SGT System is run within 16 hours of the completion of painting in the areas specified in MAI-5.3 or MAI-5.7, Control of Volatile Organic Compounds section, a determination is to be made using those procedures as to whether additional actions are required to verify SGT System operability. Exceeding MAI-5.7 limits requires performing 0-SR-3.6.4.3.2(A)(B)(C) to verify SGT can perform its intended function.

E. When all SGT Trains are secured and any evolution has the potential to discharge radioactive effluents through the main stack, one Unit 2 and one Unit 3 Stack Dilution Fan should remain in operation. This requirement provides clean air flow through the dilution cross-tie to SGT ducts. This prevents the potential back flow of radioactive effluents through the SGT duct work.

F. The alignment of SBGT trains to perform the PURGING function cannot be used when the average reactor coolant temperature is above 212°F since a postulated LOCA could impact the ability for the SBGT trains to perform their safety function. If the primary containment purge system is inoperable and the average reactor coolant temperature is less than or equal to 212°F, the standby gas treatment system venting path will provide the required filtration. The standby gas treatment system is **NOT** the normal means for PURGING operations since the vent path from containment is a much more restrictive flowpath (slower) than the purge system.

G. In the event that the train charcoal filter temperature rises to 150°F due to iodine adsorption following a LOCA, decay heat removal mode of operation should be initiated when the train is no longer in service.

H. An open decay heat removal damper in a particular train renders that train inoperable for Secondary Containment purposes.

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6.0 SYSTEM OPERATIONS (continued)

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
0-FI-65-50B/1	0-FI-65-50B/2	0-FI-65-50B/3
0-FI-65-71B/1	0-FI-65-71B/2	0-FI-65-71B/3

[2] **IF** an SGT train has been shut down after a LOCA **AND** the train's charcoal filter temperature reaches 150°F, **THEN**

INITIATE decay heat removal. **REFER TO** Section 8.0. ☐

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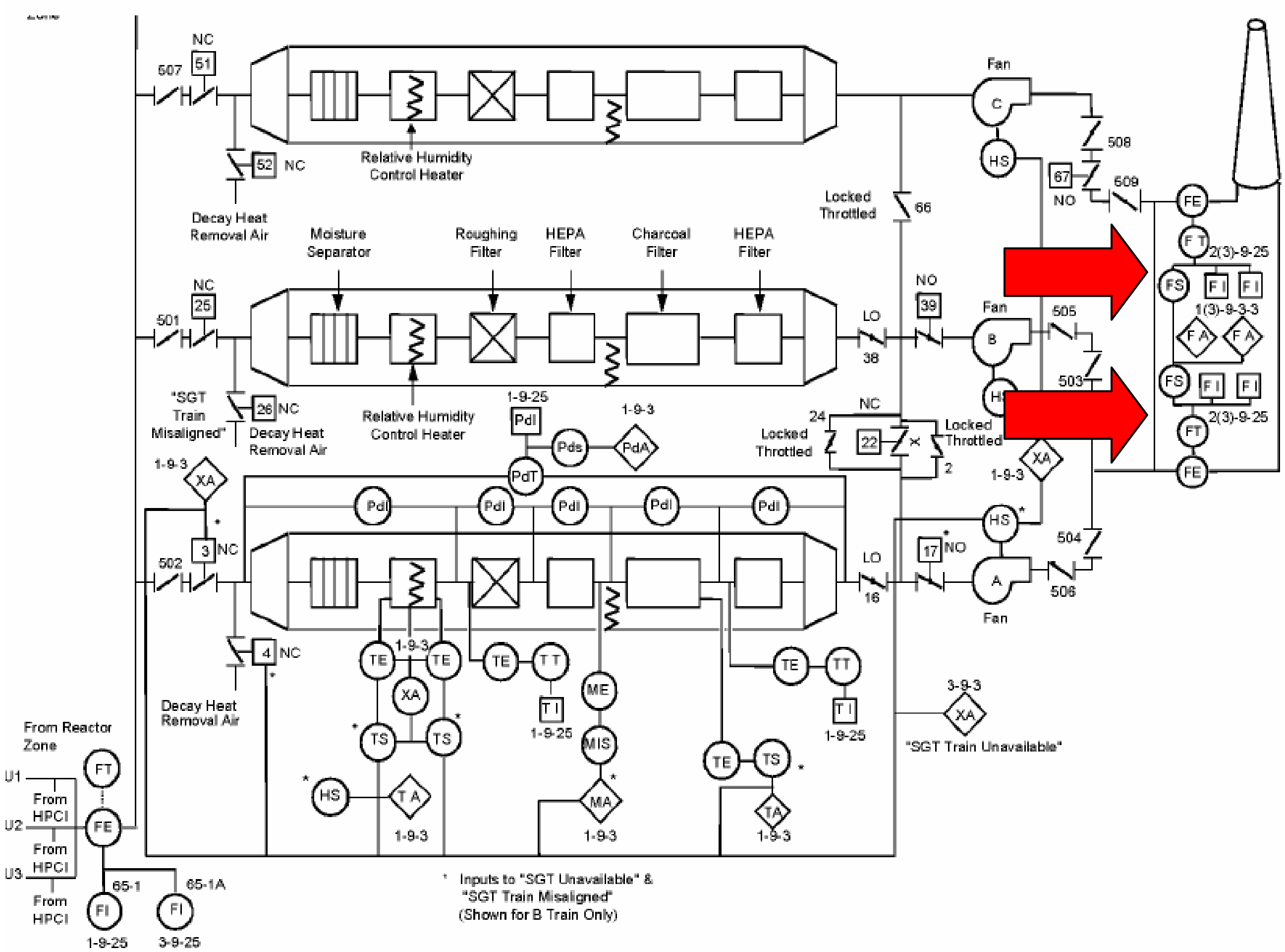

TP-1: STANDBY GAS TREATMENT (SGT) SYSTEM

			INSTRUCTOR NOTES
f. SGT FAN SW A B & C MISALIGNED XA-65-18	Blower HS (18A, 40A, 69A) in PULL-TO-LOCK with no Group 6 isolation present	Annunciation (U-1 only)	
g. SGT TOTAL FLOW LOW FA-65-50	< 8000 scfm with an initiation present	Annunciation (U-1 and U-3)	
h. SGT FILTER BANK A(B,C) DIFF PRESS HI PdA 65-5 (-27, -53)	> 8" H2O	Annunciation (all units)	
i. SGT FILTER BK A (B,C) RH HTR CONT TEMPERATURE TA 65-12A (-34A, -60)	>180˚F	Annunciation (U-1 and U-2 only) Turns off R-H heater control	
j. SGT FILTER BK A (B, C) HEATING ELEMENT POWER LOSS XA-65-12B (-34B, -60)	>180˚F	Annunciation (U-1 and U-2 only) Trips heater element to prevent damage	
k. SGT FILTER BANK A(B,C) TEMP HIGH TA 65-14 (-36, -63)	>150˚F	Annunciation (U-1 and U-2 only)	Obj. V.D.9
l. SGT FILTER BANK A(B, C) RH HIGH MA-65-13, -35 -61	80% relative humidity	Annunciation (U-1 and U-2 only)	Alarm may come in on Low Flow and/or High ambient temps. (SEOPR 95-0-065-002)

Note: For C.2.(h-l) above, each filter-bank (A, B & C) has its own annunciator for each condition; Bank A and B annunciators are located on Pnl 9-3(U1), and C annunciators are on Pnl 9-3(U2).

m. Temperature Controller 65-12, -36, -63	125˚F	Controls charcoal heater to maintain temperature	LER 85-029-01 (Charcoal bed heaters are electrically tagged.)

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0052 Page 11 of 41

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Z. Although all three trains of the SGT System can be started from the Unit 3 Control Room, it is recommended that the trains be started from the Units 1 and 2 Control Rooms due to the availability of instrumentation and shutdown capability.

AA. In the event that both SGT Trains A and B malfunction, these two trains can be manually aligned and controlled to use either filter bank with either fan. One operating fan can draw suction from both filter banks together (50% through each), or full flow through either filter bank alone.

BB. All three SGT Trains are equipped with decay heat removal lines and dampers to allow a small cooling flow of air for filter decay heat removal through a shut down train. When power is lost to a train, the dampers necessary to achieve decay heat cooling are powered by the adjacent train of SGT as follows:

1. TRAIN A DECAY HEAT DMPR, 0-DMP-065-0004, is powered from the same source as Fan B.

2. TRAIN B DECAY HEAT DMPR, 0-DMP-065-0026, is powered from the same source as Fan A.

3. TRAIN C DECAY HEAT DMPR, 0-DMP-065-0052, is powered from the same source as Fan A.

4. FSV-65-24(66)(2) have been replaced with manual valves 0-DMP-065-0024(0066)(0002) which are throttled for decay heat removal.

CC. When train temperature is greater than 180°F, the relative humidity heater turns off. The heater automatically starts when the SGT train starts.

DD. SGT Trains A and B trip on initiation of 480V load-shed logic, but will auto restart in 40 seconds when an initiation signal is present.

EE. SGT FILTER BANK A(B)(C) INLET DAMPERs, 0-DMP-065-0003(0025)(0051) auto open on automatic or manual initiation.

FF. RX ZONE EXH SGT XTIE DMPR OPR, 1-FCO-064-0041, REACTOR ZONE EXH TO SGT CROSSTIE DAMPER, 2(3)-FCO-64-41 and RFF SGT SUCT DMPR OPR, 1-FCO-064-0045 will **NOT** automatically open in the event that SGT Fan A fails to auto start on a PCIS Group 6 isolation signal.

GG. RX ZONE EXH SGT XTIE DMPR OPR, 1- FCO-064-0040, REACTOR ZONE EXH TO SGT CROSSTIE DAMPER, 2(3)-FCO-64-40 and RFF SGT SUCT DMPR OPR, 1-FCO-064-0044 will **NOT** automatically open in the event that SGT Fan B fails to auto start on a PCIS Group 6 isolation signal.

Examination Outline Cross-reference:	Level	RO	SRO
262001 AC Electrical Distribution	Tier #	2	------
K5.02 (10CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION:	K/A #	262001K5.02	
• Breaker control	Importance Rating	2.6	------

Proposed Question: **# 47**

A 4 kV breaker was racked in following maintenance on its associated pump. The breaker was then closed. Thirty minutes later, the breaker experiences a loss of DC Control Power.

Which ONE of the following completes the statement for the loss of DC Control Power?

The breaker **__(1)__ AND __(2)__**.

A. **(1)** will trip **OPEN** on power loss
(2) CAN be closed **LOCALLY** at least once if it becomes necessary.

B. **(1)** will trip **OPEN** on power loss
(2) **CANNOT** be operated **LOCALLY**, even if it becomes necessary.

C. **(1)** **CANNOT** be operated remotely
(2) CAN be closed **LOCALLY** at least once if it becomes necessary.

D. **(1)** **CANNOT** be operated remotely
(2) **CANNOT** be operated **LOCALLY**, even if it becomes necessary.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Breakers do not trip open on loss of control power. Second part is correct.

B INCORRECT: Breakers do not trip open on loss of control power. Local tripping and one closing locally are available.

C **CORRECT:** No remote operation is available when control power is lost, but local tripping is always available. When the breaker was closed prior to the loss of control power, the charging motor energized to charge the springs, so even after control power was lost, the springs were charged and available for one subsequent closure, if attempted.

D INCORRECT: First part correct – See 'C' above. Second part incorrect since breaker can be opened one time locally.

RO Level Justification: Candidate must recognize operational implications of loss of control power on 4 kV breaker operations to successfully answer this question. This question is rated as MEM based upon requiring the recalling of facts.

Technical Reference(s): OPL171.036 Rev. 12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam River Bend 08 #46

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

(3) Diesel Output Breaker (1822) CASA/B accident signal trip removed.

Bkr 1814 3C D/G

Bkr 1826 3D D/G

b. 4KV Shutdown Board C

Alternate Supply Breaker (1624) Shutdown Bus 1 lockout relay 86-1 trip removed.

c. 4KV Shutdown Board D

(1) Diesel Output Breaker (1816) CASA/B accident signal trip removed.

(2) Supply Breaker from Unit 3 (1826) Trips removed: CASA/B accident signal 4KV Shutdown Board 3ED lockout DG 3D overspeed signal DG 3D stop signal.

6. Control Power

a. Normal 250VDC control power for the Shutdown Boards is from either a shutdown battery or a unit battery, depending on the board. All alternate supplies are from unit batteries. If control power is lost indication and control is lost. Breakers must be manually operated

Discuss effects and indications of loss of control power to a breaker.

Reference Final Event Report B-91-135

Shutdown Board	Normal	Alternate
A	SB-A	BB2
B	SB-B	BB2
C	SB-C	BB1
D	SB-D	BB3
3EA	BB1	BB2
3EB	SB-3EB	BB3
3EC	BB3	BB1
3ED	BB2	BB3

b. All control power transfers are manual at the Shutdown Board.

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Initial NRC License Examination
Reactor Operator

QUESTION 46 Rev 1

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	262001 K5.02
	Importance Rating	2.6

Knowledge of the operational implications of breaker control as it applies to the AC electrical distribution system.

Proposed Question:

A 4160 volt breaker was racked in following maintenance on its associated pump. The breaker was then closed. Thirty minutes later, the breaker experienced a loss of DC control power. No operator actions were taken.

Which of the following describes the operational capabilities of this breaker?

A. The breaker will trip open on loss of control power and no further breaker operations are possible.

B. The breaker will trip open on loss of control power and all additional breaker operations must be performed locally.

C. The breaker cannot be remotely operated but can be locally tripped open, then closed and tripped open one more time locally.

D. The breaker cannot be remotely operated but can be locally tripped one time with no further breaker operations possible.

Proposed Answer: C.

Explanation (Optional): Breakers do not trip open on loss of control power. No remote operation is available when control power is loss, but local tripping is always available. When the breaker was closed prior to the loss of control power, the charging motor energized to charge the springs, so even after control power was loss, the springs were charged and available for a subsequent closure.

Technical Reference(s): STM-300, Rev 11

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0300 Obj. 14a

Question Source: Bank # 1105

Question History: Last NRC Exam 1/1997

Examination Outline Cross-reference:	Level	RO	SRO
262002 UPS (AC/DC)	Tier #	2	------
K3.08 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.)will have on the following:	K/A #	262002K3.08	
• Computer operation: Plant-Specific	Importance Rating	2.7	------

Proposed Question: **# 48**

Which ONE of the following completes the statements?

The Unit 2 **AND** Unit 3 Integrated Computer Systems (ICS) are fed from **__(1)__** inverter(s).

If normal power (inverter output) is lost, the Unit 2 **AND** 3 ICS swap to alternate **__(2)__**.

A. **(1)** a common
(2) without interruption

B. **(1)** separate
(2) without interruption

C. **(1)** a common
(2) after a 5 second time delay

D. **(1)** separate
(2) after a 5 second time delay

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Unit 2 AND 3 ICS are supplied from separate inverters. Plausible in that the power supplies for Units 2 & 3 are at the same location - U3 Turbine Bldg Elev. 565 (Room at the east end). Additionally, there are multiple systems, components, and functions shared between Units throughout the site. Part 2 correct - See explanation B.

B **CORRECT**: Part 1 correct – Unit 2 UPS Inverter AC supply is 480V Common Bd 2, Comp 9A and DC supply is Batt Bd 4, Bkr 316. Unit 3 UPS Inverter AC supply is480V Common Bd 3, Compt 8B and DC Supply is Batt Bd 6, Bkr. 316. The UPS consists of an inverter/rectifier and a regulating transformer. The UPS feeds a distribution panel from which the ICS computers are fed. Part 2 correct - Power Supply for Units 1, 2 and 3 ICS Computers is from an uninterruptible power supply (UPS). Upon loss of the normal AC power to the rectifier, the DC source will provide power to the Inverter. A static switch (throw over) is provided such that if power from the inverter is interrupted, it will switch to the regulating transformer.

C INCORRECT: Part 1 incorrect – See explanation A. Part 2 incorrect - The 5 second time delay is plausible in that on a loss of 120 VAC I&C Buses, the transfer switch operates after a 5 second time delay. The time delay transfer allows for momentary outages.

D INCORRECT: Part 1 correct – See explanation B. Part 2 incorrect – See explanation C.

Justification: Requires knowledge of ICS Power Supplies including effect of loss of normal power supply on ICS to correctly answer the question.

Technical Reference(s): OPL171.099 Rev. 9 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.099 V.B.1a (As available)

Question Source:

Bank #		
Modified Bank #	OPL171.099 04	(Note changes or attach parent)
New		

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

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INSTRUCTOR NOTES

b. Power Supply for Units 1, 2 and 3 ICS Computers is from an uninterruptible power supply (UPS). The power supply for Units 1, 2 and 3 is located in Unit 3 Turbine Bldg El. 565 (across from hot tool room). The UPS consists of an inverter/rectifier and a regulating transformer. The UPS feeds a distribution panel from which the ICS computers are fed. — TP-1

(1) Unit 1 Inverter/Rectifier Supplies: — Obj. V.B.1.a

(a) AC supply: 480 VAC Common Bd 1, Comp 9C — Review OI-48 Precautions & Limitations

(b) DC supply: Batt Bd 5, Bkr 316

(c) Dist. Pnls: 1-LPNL-925-532 and 525 — 0-45E704-1,5

(2) Unit 2 Inverter/Rectifier Supplies:

(a) AC supply: 480 VAC Common Bd 2, Compt 9A

(b) DC Supply: Batt Bd 4, Bkr. 316 — 0-45E704-3

(c) Dist Pnls: 2-LPNL-925-532 and 525

INSTRUCTOR NOTES

(3) Unit 3 Inverter/Rectifier Supplies:

(a) AC supply: 480 VAC Common Bd 3, Compt 8B

(b) DC Supply: Batt Bd 6, Bkr. 316

(c) Dist Pnls: 3-LPNL-925-532 and 525 — 0-45E704-2,4

(4) Upon loss of the normal AC power to the rectifier, the DC source will provide power to the Inverter. A static switch (throw over) is provided such that if power from the inverter is interrupted, it will switch to the regulating transformer.

Review OI-48 Precautions & Limitations

Bump less transfer

Procedure Adherence

c. The Unit 1, 2, and 3 ICSs have some hardware and software differences. These are transparent to Operations users with the exception of the following:

DCNs 51082 69172, 69173, OPL171.151

UNIT DIFFERENCES

(1) Unit 3 is running an older version of ICS with DOS based RWM and SPDS (some differences in display appearance). This ICS will be upgraded U3C14 outage.

Modifications to ICS are being implemented in stages. Stages of DCN 69173 will be implemented during the next Unit 3 outage (U3C14) which is scheduled for the spring of 2010.

(2) Unit 2 has an A/B switch in the back of panel 9-5 to toggle the 9-5 monitor between ICS and RWM computers due to the RWM not being upgraded with ICS during the U2C15 outage. The RWM will be upgraded spring 2011 outage.

(3) Unit 1 receives data interface from the Foxboro redundant system for Reactor Water Recirculation, Feedwater Heater Drains, Moisture Separator Drains, and Generator Temperature Monitoring.

I&C Power System:

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Instructor Notes

(a) Each Cabinet has an automatic transfer switch to transfer power supplies on loss of normal power. Should the normal supply voltage drop to 70% (84 volts), the transfer switch operates after a 5 second time delay. The time delay transfer allows for momentary outages. When the normal supply voltage returns to 90% (108 volts) or greater, the switch automatically transfers back to the normal supply instantaneously.

Obj. V.B.1.b; V.C.1 V.D.1.c; V.E.1.c

Note that these transfers are Break-Before-Make

(b) The alternate power supplies are arranged as shown below:

I & C BUS ALTERNATE POWER SUPPLIES

ALT source is a qualified source
Obj. V.B.1.a

I & C Bus	ALTERNATE FEED
1A	Unit 2 9-9 Panel 2 (I &CA)
2A	Unit 3 9-9 Panel 2 (I &CA)
3A	Unit 1 9-9 Panel 2 (I &CA)
1B	Unit 3 9-9 Panel 3 (I &CB)
2B	Unit 1 9-9 Panel 3 (I &CB)
3B	Unit 2 9-9 Panel 3 (I &CB)

Obj.V.C.1.a
Obj.V.D.1.a
Obj.V.E.1.a

Browns Ferry Operations Bank

OPL171.099 04

Which of the following describes the power supplies for the Unit 2 and Unit 3 ICS Computers?

A. Unit 2 and Unit 3 ICS Computers are both powered from one uninterruptible power supply inverter.

B. Unit 2 and Unit 3 ICS Computers are each powered by their own separate uninterruptible power supply inverter.

C. Unit 2 ICS is fed by an uninterruptible power supply inverter, while Unit 3 is fed by Battery Board 1.

D. Unit 3 ICS is fed by an uninterruptible power supply inverter, while Unit 2 is fed by Battery Board 1.

ANSWER: B

Examination Outline Cross-reference:	Level	RO	SRO
263000 DC Electrical Distribution	Tier #	2	------
A2.02 (10CFR 55.41.5)	Group #	1	------
Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	263000A2.02	
• Loss of ventilation during charging	Importance Rating	2.6	------

Proposed Question: **# 49**

Which ONE of the following completes the statements relating to Plant/Station Battery Room HVAC Systems?

If these systems are **NOT** operating properly, the concern is that **__(1)__**.

Because of this, provisions are provided in plant procedures to utilize **__(2)__**.

A. **(1)** lead-calcium batteries tend to release toxic gas into the atmosphere at temperatures above 90 ºF.
(2) an Emergency Exhaust Fan **ONLY**.

B. **(1)** the design limit for hydrogen concentration in the rooms may be reached during battery charging operations.
(2) an Emergency Exhaust Fan **ONLY**.

C. **(1)** lead-calcium batteries tend to release toxic gas into the atmosphere at temperatures above 90 ºF.
(2) an Emergency Exhaust Fan **AND/OR** Portable Temporary Ventilation Equipment.

D. **(1)** the design limit for hydrogen concentration in the rooms may be reached during battery charging operations.
(2) an Emergency Exhaust Fan **AND/OR** Portable Temporary Ventilation Equipment.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Lead-calcium batteries suffer degraded performance at high temperatures but do not release toxic gas as a result. (See Attached Excerpts) An Emergency Exhaust Fan is provided with operating instructions provided in Section 5.12 of 0-OI-31. But, there is another option provided in addition to the Emergency Exhaust Fan; which is portable temporary ventilation provided in Section 8.15 of 0-OI-31.

B INCORRECT: (See Attached Excerpts) First part correct in that hydrogen buildup to explosive levels is the concern. Second part incorrect as detailed in 'A' above.

C INCORRECT: First part incorrect as detailed in 'A' above. (See Attached Excerpts) Second part is correct in that an Emergency Exhaust Fan is provided with operating instructions provided in Section 5.12 of 0-OI-31. AND operation / placement of portable temporary ventilation provided in Section 8.15 of 0-OI-31.

D **CORRECT:** First part correct in that hydrogen buildup to explosive levels is the concern. Second part is correct in that an Emergency Exhaust Fan is provided with operating instructions provided in Section 5.12 of 0-OI-31. AND operation / placement of portable temporary ventilation provided in Section 8.15 of 0-OI-31.

RO Level Justification: Tests the candidate's knowledge of the impacts of a loss / malfunction of ventilation during charging activities on the batteries. In addition, procedurally-directed strategies that are available in the event of that loss/malfunction are tested. This question is rated as MEMORY as it requires a recall of facts.

Technical Reference(s): 0-OI-31, Rev. 136 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source:	Bank #		
	Modified Bank #	HLT 0707 Q # 23	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

7.11 Shutdown of Battery and Board Room Exhaust Fans

CAUTION

Battery Room ventilation is required to prevent buildup of explosive hydrogen concentration.

[1] **REVIEW** all Precautions and Limitations in Section 3.0. ☐

[2] **OBTAIN** Unit Supervisor's approval prior to shutting down the fan(s). ☐

[3] **PERFORM** the following at Panel 25-165 in Unit 1 Mechanical Equipment Room, El 617', to stop the running exhaust fan 1A/1B(3A/3B):

[3.1] **PLACE** BATTERY & BOARD RM EXHAUST FAN 1A/1B(3A/3B), 0-HS-031-0074A(97A), in OFF. ☐

[3.2] **CHECK** that green Off light illuminates on upper left or right section of panel. ☐

- Bat & Bd Rm Exhaust Fan 1A(3A)-upper left section of panel. ☐
- Bat & Board Rm Exhaust Fan 1B(3B)-upper right section of panel. ☐

[4] **REFER TO** Section 8.15 for operation with ventilation out of service. ☐

5.12 Startup of Battery and Board Room Emergency Exhaust Fans 1C(3C)

[1] **VERIFY** that the Battery and Board Room Emergency Exhaust Fans 1C(3C) are in prestartup/standby readiness using Section 4.0. ☐

[2] **REVIEW** all Precautions and Limitations in Section 3.0. ☐

CAUTION

If a normal Battery and Board Room Exhaust Fan is NOT in operation, the emergency exhaust fan should be run. Emergency exhaust fan discharge flow is required to be routed through a normal Battery and Board Room Exhaust Fan and associated suction damper. Thus, a suction damper positioner is required to be repositioned and the damper restrained open prior to emergency exhaust fan operation.

NOTES

1) In the event Battery Room, Communications Battery Room, Battery Board Room, or the RPS MG Set Room temperature rises above normal, the Battery and Board Room emergency exhaust fan should be placed in operation to assist the running exhaust fan.

2) If a Battery and Board Room exhaust fan is NOT in operation, a Battery and Board Room exhaust fan suction damper should be mechanically positioned open or electrically jumpered open in the damper control circuit in accordance with a maintenance work order. These dampers are located in Unit 1 and Unit 3 Ventilation Towers, El 635', with access from the Refuel Floor.

[3] **IF** necessary, **THEN**

START Unit 1-2 fan as follows:

[3.1] At Unit 1 ventilation tower, **VERIFY OPEN** one of the Bat & BD RM EXH FAN 1A(1B) SUCT Dampers, 0-FCO-31-74 or 75. ☐

[3.2] **PLACE** the BATTERY & BD RM EMER EXHAUST FAN 1C Switch, 0-HS-31-113, in ON position (near door 630 at Unit 1 Mechanical Equipment Room, El 617'). ☐

BFN Unit 0	Control Bay and Off-Gas Treatment Building Air Conditioning System	0-OI-31 Rev. 0136 Page 175 of 285

8.15 Temporary Ventilation for Electrical Equipment Rooms

NOTES

1) This instruction contains the temperature requirements and methods for monitoring and supplying temporary ventilation due to the loss or malfunction of HVAC system to one or more of the following rooms:
   - U1/U2 Main Control Room
   - U3 Main Control Room
   - U1, U2, U3 Auxiliary Instrument Rooms
   - U1, U2, U3 125V/250V Battery and Battery Board Rooms
   - U1, U2, U3 Unit Preferred MG Set Rooms
   - U1, U2, U3 RPS MG Set Rooms
   - U3 SHUTDOWN BOARD ROOMs 3EA, 3EB, 3EC, 3ED
   - U1, U2 250V Shutdown Battery Rooms
   - Relay Room
2) MSI-0-000-PRO005, ELECTRICAL EQUIPMENT ROOM EMERGENCY VENTILATION FOLLOWING AN APPENDIX R FIRE EVENT contains additional guidance for compensatory measures to ensure operability of electrical equipment following an Appendix R fire.
3) Temporary portable fans and generators with necessary equipment for placement are located in the T-Warehouse, Row 6

[1] **IF** air conditioning is lost to either 3EA, 3EB, 3EC, or 3ED SHUTDOWN BOARD ROOMs, **THEN**

PERFORM the following:

[1.1] **TURN OFF** all normal (florescent) lighting in 3EA, 3EB, 3EC, 3ED and the Bus Tie Rooms. ☐

[1.2] **VERIFY** Fire Doors 810, 811, and 824, (between the rooms) are OPEN, **THEN**

STATION fire watches as necessary for Appendix R Requirements. ☐

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8.15 **Temporary Ventilation for Electrical Equipment Rooms (continued)**

[2] **PERFORM** Battery Room(s) (explosive gas) monitoring. ☐

[2.1] **MONITOR** for H_2 buildup every 6 days, **THEN**

RECORD on Illustration 2 (detector(s) can be obtained from Fire Operations Personnel). ☐

[2.2] **IF** gas buildup approaches or exceeds the Lower Explosion Limit (LEL), **THEN**

OPEN door(s) and **USE** or **INSTALL** temporary fan(s) as described in Step 8.15[5] to flush room(s) of gases. ☐

[2.2.1] **WHEN** no abnormal gas level exist in room(s), **THEN**

REMOVE fan(s)

AND

RECLOSE door(s). ☐

NOTES

1) A certified degrees Fahrenheit type temperature measuring device is required to be used to monitor the room temperature. The temperature will be measured approximately 52 inches above the room floor in an area where there is the least amount of air recirculation.

2) Portable fans, 110V pedestal or equivalent, are required to be placed inside room(s) as required by the monitoring. Adjust fans vertically (approximately 45°) to force air towards the ceiling area.

[3] **OBTAIN** appropriate temperature measuring device (see Note 1 above), **THEN**

[4] **MONITOR** room temperature ☐

HLT 0707 NRC EXAM Question 23

ES-401 Sample Written Examination Question Worksheet Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
263000K5.01 Knowledge of the operational implications of the following concepts as they apply to the DC Electrical Distribution: Hydrogen Generation during battery charging.	Tier #	2	
	Group #	1	
	K/A #	263000K5.01	
	Importance Rating	2.6	2.9

Proposed Question: **RO # 23**

Which ONE of the following is a concern to plant operation if the Plant/Station Battery Rooms HVAC units are not operating properly?

A. The design limit for hydrogen concentration in the rooms may be reached when the batteries are being charged.

B. Electrical Maintenance will not be able to obtain accurate Cell specific gravity readings if temperature is above 90 ^{o}F.

C. The lead-calcium batteries tend to release toxic gas into the atmosphere above 90 ^{o}F, and access to the room would be limited.

D. The Quarterly Battery SR frequency is lowered to weekly when temperatures are above the 70 ^{o}F to 90 ^{o}F temperature range.

Examination Outline Cross-reference:	Level	RO	SRO
264000 EDGs	Tier #	2	------
A3.02 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL / JET) including:	K/A #	264000A3.02	
• Minimum time for load pickup	Importance Rating	3.1	------

Proposed Question: **# 50**

Which ONE of the following completes the statement in accordance with Tech Spec 3.8.1, "AC Sources - Operating?"

On a Loss of Offsite Power, simultaneous with an ECCS initiation signal on Unit 1, the **MAXIMUM** allowed time for Emergency Diesel Generators to energize their associated Shutdown Boards is ____ seconds.

A. 4

B. 7

C. 10

D. 14

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Recognizable value for D/G Air Start Sequence - If the engine fails to achieve 125 rpm in 4 seconds after start signal, this is a failure to start and the air starting motors shutdown.

B INCORRECT: Recognizable as the time second group of auto-connected loads is sequenced following D/G output breaker closing on DGVA sequencing.

C **CORRECT**: Per TS 3.8.1, On an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal, DG must auto-starts from standby condition and energize permanently connected loads in 10 seconds.

D INCORRECT: Recognizable as the time third group of auto-connected loads is sequenced following D/G output breaker closing on DGVA sequencing.

Justification: To demonstrate the ability to monitor automatic operation of D/G following Loss of Offsite Power in conjunction with an ECCS initiation signal, candidate must know the minimum time required to energize permanently connected loads. Question requires knowledge of bases for TS 3.8.1 LCO statement (above the line) and is therefore an RO question.

Technical Reference(s): OPL171.038, Rev. 17 (Attach if not previously provided)

U1 TS 3.8.1 Am 235

U1 TS BASES 3.8.1 Rev. 52

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.038 V.B.1 (As available)

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BASES

LCO (continued)

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective 4.16 kV shutdown board on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the 4.16 kV shutdown board. The Unit 1 and 2 DGs are provided with a common 480 V load shed logic system with two redundant divisions. The common accident signal logic system, with two redundant divisions, is common to the Unit 1, 2, and 3 DGs. These logic systems must be OPERABLE to ensure the DGs will perform and alignments will occur as assumed during a DBA.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

The AC sources must be separate and independent (to the extent possible) of other AC sources. For the DGs, the separation and independence are complete. For the offsite AC sources, the separation and independence are to the extent practical. A qualified offsite circuit may be connected to more than one division of 4.16 kV shutdown boards and not violate separation criteria. A circuit that is not connected to the Division I or II 4.16 kV shutdown boards is required to have the capability to be connected to at least one division of 4.16 kV shutdown boards to be considered OPERABLE.

(continued)

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INSTRUCTOR NOTES

X. Lesson Body

A. General Description

Safety Objectives

The safety objective of the standby AC power system is to provide a self-contained, highly reliable source of power as required for the engineered safeguard systems so that no single credible event can disable the core standby cooling functions or their supporting auxiliaries.

B. Component Description

1. ** Diesel Generators (all 8)

Obj.V.B.1, Obj.V.D.1 Obj.V.E.1 **Review INPO SOER 83-01 (HO-1). Emphasize importance of operator equipment/ component familiarity & procedural compliance toward preventing Diesel Generator failures.

a. Ratings - 4160 volt, 3 phase, 60 Hz rated for maximum loading with 0.8 power factor lag:
   (1) 2600/2550*KW continuous (>2 hours)
   (2) 2860/2800*KW for 0-2 hours (Short Time Steady State)
   (3) 2850/2815**KW for 0-3 minutes (Cold Engine Instantaneous)
   (4) 3050/3025**KW for >3 min. (Hot Engine Instantaneous)
   (5) 3575 KVA (short time generator only) 0-2 hours
   (6) 3250 KVA (continuous) >2 hours.
   Note: Items (1)(2)(3)(4) & (6) are Engine ratings

See OI-82 P&Ls for ratings

b. Capable of fast starting and being ready to load within 10 seconds.

2. * Reduced rating 1 & 2 (above) apply for engine cooling water outlet temperature exceeding 190°F in conjunction with combustion air exceeding 90°F.

** Reduced rating 3 & 4 (above) apply when combustion air exceeds 90°F regardless of engine cooling water outlet temperature. (For more details see OI 82).

OI-82 P&L 3.2: DG load <550 KW should be avoided to prevent oil/soot accumulation in the exhaust system. Obj.V.C.12

3. Diesel Generator Auxiliaries

a. Lubricating oil system

b. Engine lubricating oil is supplied from the crank case which holds approximately 465 gallons. At the full mark on the dip stick this gives 236.16.gals of usable oil. The engine consumes approximately 0.98 gallons of oil per hour at full load operation.

Obj.V.B.12 Obj.V.C.10 Obj.V.C.7 Level must be within 2" of full mark (197 useful gallons) on diesel crankcase dipstick

INSTRUCTOR NOTES

(12) The redundant start may be canceled before the start circuit locks out by opening the logic breaker and pushing both engine stop push-buttons. Note that pulling the engine driven fuel pump shutoff plunger will not stop the diesel since the electric fuel pump will still be supplying fuel. (OI-82)

3. Accident Operation

a. Accident signal received (**CASx**)

(1) Signals diesel generators to start.

(2) Opens diesel output breakers if shut. Obj.V.B.9

b. If normal voltage is available, load will sequence on as follows: (**NVA**) Obj.V.C.6 Obj.V.D.15 Obj.V.E. 15

Time After Accident	S/D Board A	S/D Board C	S/D Board B	S/D Board D
0	RHR/CS A			
7		RHR/CS B		
14			RHR/CS C	
21				RHR/CS D
28	RHRSW*	RHRSW*	RHRSW*	RHRSW*

*RHRSW pumps assigned for EECW automatic start

c. **If normal voltage is <u>NOT</u> available: (DGVA)** Obj.V.B.9 Obj.V.C.6

(1) After 5-second time delay, all 4kV Shutdown Board loads except 4160/480V transformer breakers are automatically tripped.

(2) Diesel generator output breaker closes when diesel is at speed.

(3) Loads sequence as indicated below

Time After Accident	S/D Board A	S/D Board B	S/D Board C	S/D Board D
0	RHR A	RHR C	RHR B	RHR D
7	CS A	CS C	CS B	CS D
14	RHRSW *	RHRSW *	RHRSW *	RHRSW *

*RHRSW pumps assigned for EECW automatic start

d. Certain 480V loads are shed whenever an accident signal is received in conjunction with the diesel generator tied to the board. (see OPL171.072)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.1.9	---------------------------NOTE-------------------------- All DG starts may be preceded by an engine prelube period. --- Verify, on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ECCS initiation signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through individual timers, 3. achieves steady state voltage ≥ 3940 V and ≤ 4400 V, 4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.	24 months
SR 3.8.1.10	For required Unit 3 DGs, the SRs of Unit 3 Technical Specifications are applicable.	In accordance with applicable SRs

INSTRUCTOR NOTES

(12) The air start sequence is outlined below:

(a) One bank of dual air starting motors (depending upon which start circuit is selected for preferred start at the electrical control cabinet) cranks the engine. If the engine fails to start (<40 rpm in 3 seconds or <125 rpm in 4 seconds after start signal) the air starting motors shutdown.

(b) After a ½ second pause, both start circuits activate so both banks of dual air starting motors crank the engine. If the engine fails to start, the air starting motors shut down and the preferred bank is locked out.

(c) After a ½ second pause, the other start circuit (not selected as preferred) cranks the engine. If the engine fails to start the air starting motors shut down and the second bank is locked out.

(d) Under normal conditions, the air start motors stop when speed reaches 125 RPM or bearing oil pressure reaches 30 psig.

(e) The longest time any of the two start circuits will attempt to start the diesel is 5.5 seconds per start circuit.

(f) Upon activation of a start circuit, the associated governor booster pump starts and runs until speed is above 125 RPM. The electric driven fuel priming pump starts and continues running.

Preferred Start Switch, Start 1 or Start 2 positions

Start 1 selects the 'Left' bank or 'A' bank motors

Start 2 selects the 'Right' bank or 'B' bank motors

Examination Outline Cross-reference:	Level	RO	SRO
264000 EDGs	Tier #	2	------
A3.04 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL / JET) including:	K/A #	264000A3.04	
• Operation of the governor control system on frequency and voltage control	Importance Rating	3.1	------

Proposed Question: **# 51**

Emergency Diesel Generator (DG) 3EA was started for its Monthly Load Test Surveillance.

Which ONE of the following will occur if the DG's output breaker is closed with the DG Mode Selector Switch in the UNITS IN PARALLEL position?

A. The zero droop governor advances the fuel supply to the diesel to raise output frequency to the governor's setpoint. This will cause the DG Output Breaker to trip on overload.

B. The speed regulator lowers the fuel supply to the diesel to lower output voltage to the governor's setpoint. The will cause the DG Output Breaker to trip on undervoltage.

C. The zero droop governor advances the fuel supply to the diesel to raise output frequency to the governor's setpoint. This will cause the DG to trip on overspeed.

D. The speed regulator lowers the fuel supply to the diesel to lower output voltage to the governor's setpoint. This will cause the DG Output Breaker to trip on reverse power.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: The speed regulator senses output frequency, but now the generator output frequency is fixed by the other loads on the grid. If the diesel speed setpoint is higher than grid frequency, the zero droop governor will keep advancing the fuel supply to the diesel in order to try and raise grid/DG output frequency to the governor's setpoint. This will cause the diesel to overload. (495 amps.)

B INCORRECT: First part is incorrect. With DG Mode Selector Switch in the UNITS IN PARALLEL position, DG is in Zero Droop Operation. Second part incorrect.

C INCORRECT: First part is correct but second part is incorrect. Grid will maintain DG speed at Grid frequency.

D INCORRECT: Both parts are incorrect as explained above.

Justification: Requires knowledge of how D/G responds in UNITS IN PARALLEL Mode to demonstrate ability to monitor D/G automatic operation.

Technical Reference(s): OPL171.038 Rev. 17 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.038 V.B.6 (As available)

Question Source: Bank # OPL171.038 #1

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

INSTRUCTOR NOTES

(3) If the generator were to be tied to the grid, when in **Single Unit** or **Units in parallel**, as soon as the output breaker is shut the speed regulator senses output frequency, but now the generator output frequency is fixed by the other machines on the grid. If the diesel speed setpoint is higher than grid frequency, the zero droop governor will keep advancing the fuel supply to the diesel in order to try and raise grid/DG output frequency to the governor's setpoint. This will cause the diesel to overload. (495 amps.)

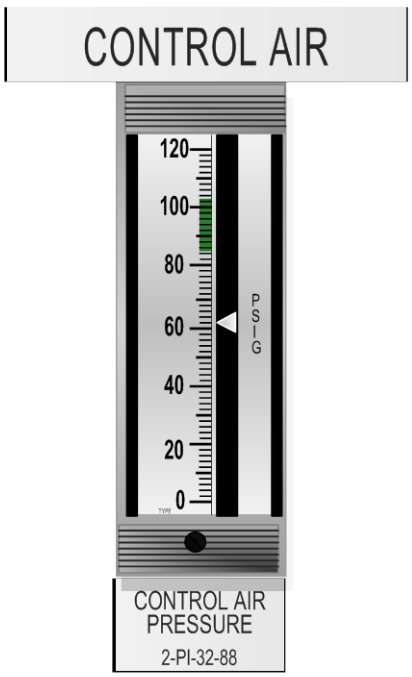
(4) **Droop operation** is in effect for **Parallel with System.** In droop mode the load carried by the diesel is sensed in addition to the output frequency. If the speed setpoint is higher than grid frequency, when the output breaker is shut the governor will see generator output frequency as being too low and start advancing fuel. This will cause the generator load to pick up. As load picks up, it sends a negative speed signal back to the regulator which cancels out the difference between grid frequency and setpoint frequency. When this happens the governor will stop advancing fuel and the engine will steady out at a certain amount of load. To pick up additional load the speed setpoint is adjusted upwards and the load builds up until it has canceled out the additional speed setpoint adjustment. If a droop mode generator was the sole supply to a board, its frequency versus kilowatt load would droop.

(5) Droop mode operation of the governor is controlled by the electronic governor and is in use **only** when the generator mode is "PARALLEL WITH SYSTEM."

Examination Outline Cross-reference:	Level	RO	SRO
300000 Instrument Air	Tier #	2	------
A4.01 (10CFR 55.41.7)	Group #	1	------
Ability to manually operate and/or monitor in the control room: • Pressure gauges	K/A #	300000A4.01	
	Importance Rating	2.6	------

Proposed Question: **# 52**

Unit 2 is at 100% Reactor Power when a Control Air leak results in the following indication:



Based upon the **ABOVE** indications, which ONE of the following is correct?

A. SERVICE AIR XTIE VLV, 0-FCV-33-1, is CLOSED.

B. CONDENSATE DEMIN BYPASS VALVE, 1-FCV-2-130, is OPEN.

C. Unit 2 OUTBOARD MAIN STEAM ISOLATION VALVES are CLOSED.

D. Unit 2 to Unit 3 CONTROL AIR CROSSTIE, 2-PCV-032-3901, is CLOSED.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: SERVICE AIR XTIE VLV 0-FCV-33-1 opens at Control Air System Pressure of 85 psig. Plausible in that this valve has both an automatic opening and automatic closing function associated with degraded Control Air Pressure. Current reading of 62 psig is less than automatic opening set point and greater than the automatic re-closing set point of 30 psig.

B INCORRECT: CONDENSATE DEMIN BYPASS VALVE, 1-FCV-2-130 fails open with Control Air Pressure less than 50 psig. Plausible in that this valve will fail open with degraded Control Air Pressure which has not yet been reached.

C INCORRECT: Main Steam Isolation Valves close with Control Air System Pressure of 45 psig. Plausible in that the MSIVs will fail closed with degraded Control Air Pressure which has not yet been reached.

D **CORRECT**: Unit 2 to Unit 3 Crosstie Valve 2-PCV-032-3901 closes with Control Air Pressure less than 65 psig and will not re-open until Control Air Pressure is 85 psig.

Justification: Tests for required knowledge to demonstrate ability to monitor Control Air Pressure Gauges in the control room. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-32 Rev. 125 (Attach if not previously provided)
OPL171.054 Rev. 14

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.054 V.B.3 (As available)

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

BFN Unit 0	Control Air System	0-OI-32 Rev. 0125 Page 11 of 114

3.0 PRECAUTIONS AND LIMITATIONS (continued)

16. Check Set Points: Controller determines data stored in memory contains unacceptable values

17. Invalid Calibration: Sensor zero value is +/- 10% of its scale

18. 2nd Stage Over Ratio: Machine is not loaded and 2nd stage discharge pressure is >50 psig

L. Control Air Compressors A, B, C, D have the following WARNING signals:

1. Change inlet filter: Inlet vacuum greater than 0.7 psi and the unit is fully loaded

2. Change oil Filter: Filter inlet pressure minus filter outlet pressure is >13 psid and oil manifold temperature is >110°F

3. Sensor failure: oil pressure, intercooler moisture separator temperature or aftercooler outlet temperature sensor recognized as missing or broken

M. Control Air Compressors A, B, C, D must not be operated with housing panels removed as this may cause overheating and high noise levels.

N. Air flow should never be established through an air dryer unless power is supplied to the dryer.

O. Section 6.4, Moisture Trap blowdown, is performed once per shift.

P. [QA/C] Header isolation valves 1-32-586 and 1-32-2378 should be closed during multi-unit operation so that a Control Air failure in Unit 1 will **NOT** result in the possibility of a scram of Unit 2. [CAQR BFP 910093].

Q. 2-PCV-032-3901 is installed on the main control air header between Unit 2 and Unit 3 and 1-PCV-032-3901 is installed on the main control air header between Unit 2 and Unit 1. Each valve automatically closes when Control Air header pressure on either side of the valve drops below 65 psig. This action prevents a control air failure on any one Unit from resulting in a multi-unit scram. Illustration 3 provides more information on this feature.

R. Ingersoll-Rand Air Compressors A, B, C and D use the same 480V breakers as the former Nordberg Air Compressors. However, the Ingersoll-Rand air compressors are automatically started and stopped via contactors integral with the compressor package. A standby compressor must have its supply breaker closed by pushbutton, locally, on 1-LPNL-925-0118. An indication that a supply breaker is closed would be an illuminated POWER ON light next to the Intellisys display or an active Intellisys display on the associated compressor.

OPL171.054
Revision 13 (Typo in Lesson Plan. This is actually in Rev. 14)
Appendix C
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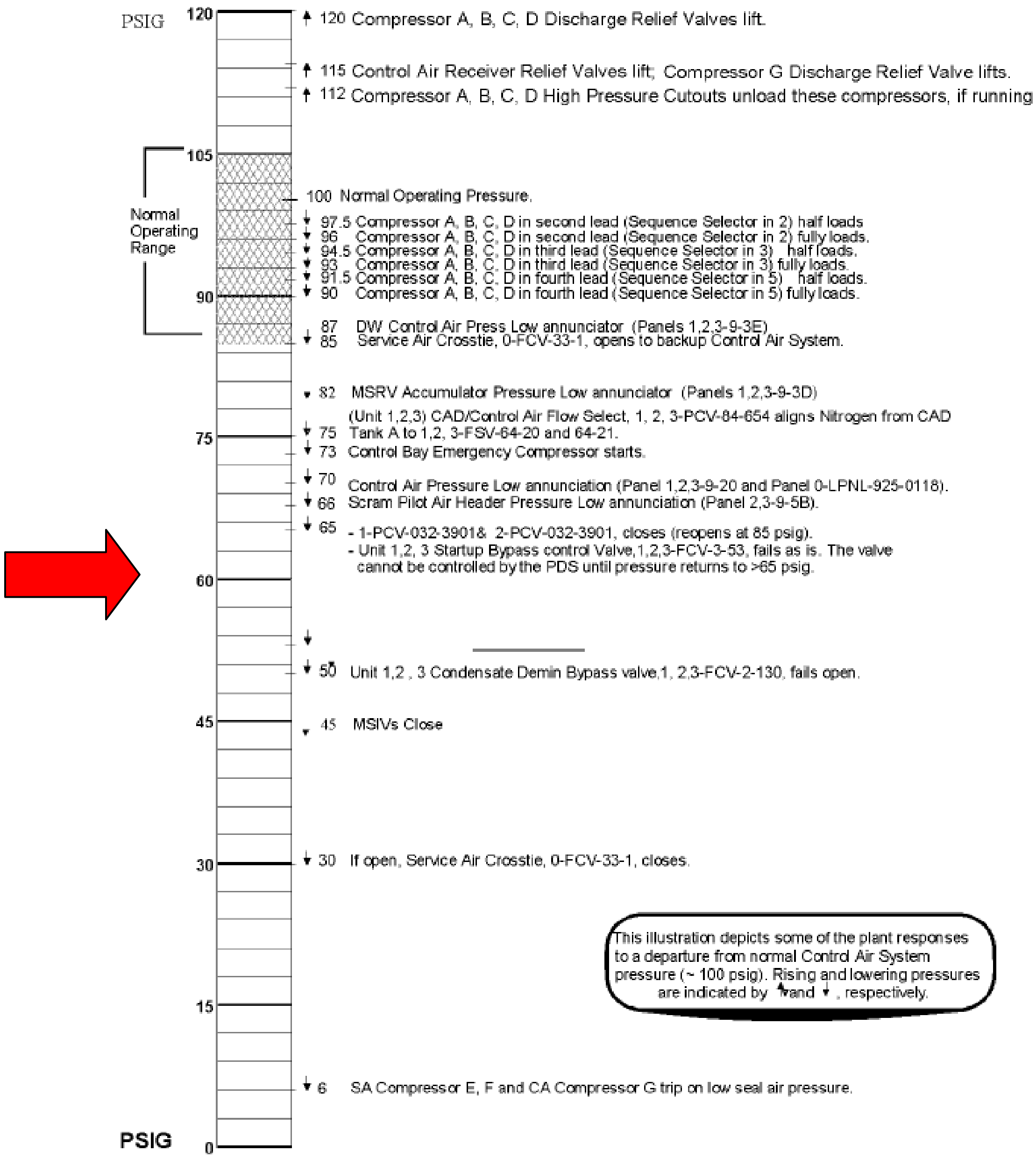


TP-14: CONTROL AIR SYSTEM PRESSURE SPECTRUM

PICTURE FROM SIMULATOR FOR COMPARISON

Examination Outline Cross-reference:	Level	RO	SRO
400000 Component Cooling Water	Tier #	2	------
K6.05 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the CCWS:	K/A #	400000K6.05	
• Pumps	Importance Rating	3.0	------

Proposed Question: **# 53**

The 4KV Shutdown Board A is being fed from its Diesel Generator.

With RHR Service Water (RHRSW) Pump A1 aligned to Emergency Equipment Cooling Water (EECW), Reactor Water Level subsequently drops to (-) 122 inches.

Which ONE of the following completes the statement?

RHRSW Pump A1 will __________.

A. **NOT** trip.

B. trip **AND NOT** restart.

C. trip **AND** then restart after 14 seconds.

D. trip **AND** then restart after 28 seconds.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The shutdown boards will load shed their respective EECW loads when a shutdown board undervoltage condition exists or a LOCA signal in conjunction with a loss of offsite power is received.

B INCORRECT: A1 EECW Pump will trip but second part is incorrect as explained in C.

C **CORRECT**: Under normal conditions, a Common Accident Signal – Low Low Low Reactor water level will auto trip the Diesel Generator output breaker. With the "A" 4KV Shutdown Board being fed from its diesel generator, the breaker trip results in "A" 4KV S/D Bd Undervoltage. The shutdown boards will load shed their respective EECW loads when a shutdown board undervoltage condition exists or a LOCA signal in conjunction with a loss of offsite power is received. When any EECW pump receives an automatic start signal, it will start after a 14 second time delay if its 4kV shutdown board is being fed from its diesel generator.

D INCORRECT**:** When any EECW pump receives an automatic start signal, it will start after a 28 second time delay if normal voltage is available to its 4kV shutdown board. Normal voltage is not available in this case.

Justification: Question tests candidate's knowledge of effect of loss of EECW pump following Common Accident Signal on the EECW system. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.051 Rev. 16 (Attach if not previously provided)
0-OI-67 Rev. 89
OPL171.038 Rev. 17
0-OI-82 Rev. 106

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.051 V.B.9 (As available)

Question Source: Bank # OPL171.051-27
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

OPL.171.051
Revision 16
Page 3 of 10

(1) The signals that start the A3 and C3 pumps and the B3 and D3 pumps also start the B1 and D1 pumps and the A1 and C1 pumps, respectively. One pump should be valved into EECW header service and off RHRSW header service and the pump started at any time that a normally assigned EECW pump is out of service.

Pump Out Of Service	Valve In and Start
A3	A1
B3	B1
C3	C1
D3	D1

(2) RHRSW/EECW pumps A3, B3, C3 and D3 (A1, B1, C1 and D1, if aligned) auto start as follows:

Obj. V.B.8
Obj. V.C.6
Obj. V.D.6
Obj. V.E.11

- common accident signal of 2.45 psig DW pressure with 450 psig Rx pressure, OR
- common accident signal of -122" vessel level (Level 1) is received, OR
- low RCW header pressure is sensed at the Control Air compressors (<30 psig), OR
- low RCW header pressure is sensed at the RBCCW heat exchangers (<15 psig).

(3) RHRSW/EECW pumps B3, D3 (A1, C1 if aligned) will auto start as follows:

Pumps fed from Unit 1 & 2.

Obj. V.B.8/V.B.9
Obj. V.C.6/V.C.7
Obj. V.D.7

- when any Unit 1 or 2 Core Spray pump starts, OR
- any Unit 1 or 2 EDG start ($\geq$ 125 RPM).

(4) RHRSW/EECW pumps A3, C3 (B1, D1 if aligned) will auto start as follows:

Obj. V.E.12/V.D.7

- when any Unit 3 Core Spray pump starts, OR
- any Unit 3 EDG start (≥ 125 RPM).

Pumps fed from Unit 3

(5) The preferred alternate pump for any of the A3 - D3 pumps WILL NOT auto start on the same unit CS pump start or DG start signal as the primary pump therefore an alternate pump must be running to maintain EECW operability.

(6) When any EECW pump receives an automatic start signal, it will start after a 28 second time delay if normal voltage is available to its 4kV shutdown board. If its 4kV shutdown board is being fed from its diesel generator it will start after a 14 second time delay. (These pump start time delays are board specific)

Obj. V.B.9
Obj. V.C.7

E. Chlorination of EECW Piping

1. Hypochlorite addition for EECW system helps prevent the accumulation of Asiatic clams in its piping as well as heat exchangers that it serves. The Calgon Company has been awarded and is assigned chemistry control for all raw water systems at BFN.

Obj.V.B.4.h

2. **NRC Generic Letter 89-13 and SOER 84-1 express the significance of bio-fouling in raw water system heat exchangers for safety systems. Should the chlorination fail to prevent asiatic clam infestations, heat exchanger fouling could cause multiple failures of emergency equipment to meet their design criteria. Operators should be aware that mechanical maintenance and Chemistry will be monitoring the system for biofouling/fouling. If one heat exchanger shows evidence of fouling, the operator should highly suspect fouling in redundant safety system components and take appropriate actions to see that those redundant components are tested. Biofouling has caused inoperable EECW heat exchangers on the Diesel Generators and the RHR seal heat exchangers at BFNP.

**SOER 84-001

Plant/industry experience

[NCO-90-002-005]

F. Chemistry Control

INSTRUCTOR NOTES

(12) The redundant start may be canceled before the start circuit locks out by opening the logic breaker and pushing both engine stop push-buttons. Note that pulling the engine driven fuel pump shutoff plunger will not stop the diesel since the electric fuel pump will still be supplying fuel. (OI-82)

3. Accident Operation

a. Accident signal received (**CASx**)

(1) Signals diesel generators to start.

(2) Opens diesel output breakers if shut. Obj.V.B.9

b. If normal voltage is available, load will sequence on as follows: (**NVA**) Obj.V.C.6 Obj.V.D.15 Obj.V.E. 15

Time After Accident	S/D Board A	S/D Board C	S/D Board B	S/D Board D
0	RHR/CS A			
7		RHR/CS B		
14			RHR/CS C	
21				RHR/CS D
28	RHRSW*	RHRSW*	RHRSW*	RHRSW*

*RHRSW pumps assigned for EECW automatic start

c. **If normal voltage is NOT available: (DGVA)** Obj.V.B.9 Obj.V.C.6

(1) After 5-second time delay, all 4kV Shutdown Board loads except 4160/480V transformer breakers are automatically tripped.

(2) Diesel generator output breaker closes when diesel is at speed.

(3) Loads sequence as indicated below

Time After Accident	S/D Board A	S/D Board B	S/D Board C	S/D Board D
0	RHR A	RHR C	RHR B	RHR D
7	CS A	CS C	CS B	CS D
14	RHRSW *	RHRSW *	RHRSW *	RHRSW *

*RHRSW pumps assigned for EECW automatic start

d. Certain 480V loads are shed whenever an accident signal is received in conjunction with the diesel generator tied to the board. (see OPL171.072)

BFN Unit 0	Standby Diesel Generator System	0-OI-82 Rev. 0106 Page 12 of 179

3.0 PRECAUTIONS AND LIMITATIONS (continued)

M. Personnel working in the D/G rooms should remain aware that the possibility exists of CO_2 discharge into the room. Upon CO_2 initiation, an alarm will sound. Personnel then have 20 seconds to evacuate the area before CO_2 is dispensed. For detection purposes, a wintergreen odor is injected into CO_2 discharge.

N. [NER/C] When the breakers feeding the D/G air dryers (LC-31, bkrs 8, 9, 10 & 11) are opened, the D/G air compressor auto-starts are inhibited. [II-S-91-004]

O. Environmental calculations assume DG battery ambient temperatures are within 40°F to 110°F.

P. When the D/G is the only feed to the shutdown board and in single unit operations, starting an RHR Pump with other 4kV motor loads running on the associated board may result in D/G overload.

Q. After operation of 4160V breakers, the charging spring is required to be verified to have recharged by verifying locally the breaker closing spring target indicates charged and the amber breaker spring charged light is on to ensure future breaker operation.

R. Diesel Generators will automatically start, as follows:

1. Degraded voltage <u>or</u> undervoltage on 4-kV Shutdown Board A, B, C, or D will start its associated Diesel Generator.
2. A Pre-Accident Signal (Reactor Vessel Low Low Low water level <u>OR</u> High Drywell pressure) on Unit 1, Unit 2 <u>or</u> Unit 3 will start all eight Diesel Generators.

S. Under normal conditions, <u>any</u> of the following will auto trip the Diesel Generator output breaker:

1. Differential overcurrent
2. Timed overcurrent
3. Reverse power
4. Loss of field
5. Overspeed
6. Common Accident Signal (Low Low Low Reactor water level <u>OR</u> Low Reactor pressure in conjunction with High Drywell pressure on Unit 1, 2 or Unit 3.)

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0089 Page 8 of 79

3.0 PRECAUTIONS AND LIMITATIONS

A. RHRSW Pumps A3, B3, C3, and D3 are assigned to the EECW System and are referred to in this procedure as RHRSW pumps.

B. RHRSW Pumps A1, B1, C1, and D1 may supply either the RHRSW or EECW Systems. 0-OI-23 should be referred to when using A1, B1, C1, or D1 RHRSW pumps for RHRSW operation.

C. The EECW System is aligned as follows:

1. At least one RHRSW pump, assigned to the EECW System, should be running on each header to maintain the header charged at all times. If no pumps are running on a header and header pressure lowers to ≤ 0 psig, the header shall be declared inoperable and appropriate actions taken, as required by Technical Specifications.
2. Two additional RHRSW pumps, one on each of the north and south headers, are normally lined up to start automatically for EECW system operation, if NOT already running.
3. Only one RHRSW pump in a given RHRSW pump room may be counted toward meeting Technical Specification 3.7.2 requirements for EECW pump operability.

D. If a Number 1 RHRSW Pump is needed to meet minimum Technical Specification operable EECW pump requirements, that pump may be aligned to EECW. To meet EECW requirements, Number 1 RHRSW pumps must be aligned to EECW, the pump started, and should remain running. Number 1 RHRSW pumps do NOT have the same auto start signals as the associated Number 3 RHRSW Pump. When a Number 1 RHRSW Pump is aligned for EECW, its RHRSW function required by the Safe Shutdown Program (Appendix R) is inoperable. Appendix R program equipment operability requirements of FPR-Volume 1 shall be addressed. **REFER TO** Sections 8.1 through 8.4.

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).

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0089 Page 9 of 79

3.0 PRECAUTIONS AND LIMITATIONS (continued)

F. RHRSW Pumps B3 and D3 (14 second delay if diesel supplying board, 28 second delay if normal voltage available), and when lined up for EECW operation, A1 and C1, will auto-start when:

1. Any Unit 1 or 2 Core Spray pump starts.
2. Any Unit 1 or 2 Diesel Generator starts.

G. RHRSW Pumps A3 and C3, and when lined up for EECW operation, B1 and D1, will auto-start when (14 second delay if diesel supplying board, 28 second delay if normal voltage available):

1. Any Unit 3 Core Spray pump starts.
2. Any Unit 3 Diesel Generator starts.

H. EECW System backup water supply valve (FCV-67-53) to the control air compressors will auto open at 30 psig lowering RCW pressure, if EECW pressure is ≥ 106 psig. The valve will auto close on EECW pressure dropping to < 106 psig.

I. EECW System backup water supply valves (FCV-67-50 [North header] and 51 [South header]) to the RBCCW heat exchangers will open at 15 psig lowering RCW pressure if EECW pressure is equal to or greater than the setpoint. These valves will close on EECW pressure dropping below the setpoint. Once closed, the closure seals in until manually reset in accordance with Section 8.7. The north header supply to Unit 1 RBCCW, the north header supply to Unit 2 RBCCW and the South header supply to Unit 3 RBCCW are normally isolated with a manual valve; therefore no flow will occur when either 1-FCV-67-50, 2-FCV-67-50 or 3-FCV-67-51 opens. The EECW pressure setpoints for these valves are listed below in psig:

	Unit 1	Unit 2	Unit 3
FCV-67-50	90	91	92
FCV-67-51	107	109	113

J. The EECW discharge strainer automatically starts its cleaning cycle on pump discharge flow, and the flush valve opens automatically.

BFN Unit 0	Emergency Equipment Cooling Water System	0-OI-67 Rev. 0089 Page 10 of 79

3.0 PRECAUTIONS AND LIMITATIONS (continued)

K. The shutdown boards will load shed their respective EECW loads when a shutdown board undervoltage condition exists or a LOCA signal in conjunction with a loss of offsite power is received.

RHRSW (EECW) PUMP	SHUTDOWN BOARD
A1	A
A3	3EA
B1	3EC
B3	C
C1	B
C3	3EB
D1	3ED
D3	D

L. Because the EECW system is common to all three units, the Unit Operators should contact each other whenever changes to the system are made.

M. [NRC/C] RHRSW Pump Motor nameplate full load current is 53 amps. The maximum allowable continuous running current is 61 amps (based on full load amps multiplied by the motor service factor of 1.15). [SLT 861087005]

N. To prevent possible RHRSW (EECW) pump motor damage, the following start limitations should be observed for the pump motor:

1. Two starts per hour from ambient temperature.
2. One start per hour from rated temperature.
3. When the motor has operated at no load for a minimum of 30 minutes, it may be restarted immediately.
4. The motor should be allowed to stand idle for at least 60 minutes before each additional restart is attempted.

O. The throttle valves that control the flow rate to each equipment cooler in the EECW system are preset and locked. Whenever the valve position is changed, it will have to be reset to the appropriate number of turns from full open in accordance with the valve checklist. Verify the outlet valve is open before setting the inlet valve for the DG engine coolers.

OPL171.051 27

The A 4KV Shutdown Board is being fed from its Diesel Generator and ALL other Shutdown Boards are powered from their NORMAL power supplies.

Which ONE of the following completes the statement?

With the A1 RHRSW Pump aligned to EECW, AND Reactor Water Level subsequently drops to (-)122 inches, A1 RHRSW Pump would __________.

A. NOT trip.

B. trip AND NOT restart.

C. trip AND then restart after 14 seconds.

D. trip AND then restart after 28 seconds.

Examination Outline Cross-reference:	Level	RO	SRO
201002 RMCS	Tier #	2	------
A2.04 (10CFR 55.41.5)	Group #	2	------
Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: • Control rod block	K/A #	201002A2.04	
	Importance Rating	3.2	------

Proposed Question: **# 54**

Unit 3 is at 88% Reactor Power with the "Control Rod Exercise Test for Withdrawn Control Rods," 3-SR-3.1.3.3, in progress when the following indications are received:

- APRM DOWNSCALE / OPRM INOP, (3-9-5A, Window 4) is in alarm
- APRM 1 indicates 0%

Which ONE of the following completes the statement?

This condition will result in a Control Rod __**(1)**__ requiring __**(2)**__ to continue the surveillance.

A. **(1)** withdrawal block **ONLY**
(2) bypassing APRM 1

B. **(1)** withdrawal block **ONLY**
(2) placing APRM 1 Mode Switch to INOP

C. **(1)** withdrawal **AND** insert block
(2) bypassing APRM 1

D. **(1)** withdrawal **AND** insert block
(2) placing APRM 1 Mode Switch to INOP

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 Correct – APRM Downscale will result in a Control Rod withdrawal block ONLY with Mode Switch in RUN. Part 2 Correct – In accordance with 3-ARP-9-5A, IF APRM failed downscale, THEN BYPASS channel. REFER TO 3-OI-92B.

B INCORRECT: Part 1 Correct. Part 2 Incorrect – Removing APRM Mode Switch from OPER position will result in a trip signal. Plausible in that the APRM is Inoperable.

C INCORRECT: Part 1 Incorrect – A Control Rod insert block signal will not be generated from an APRM downscale. Plausible in that various RMCS signals do result in a Control Rod insert block signal. Part 2 Correct.

D INCORRECT: Parts 1 and 2 incorrect as explained above.

Justification: To answer this question, candidate must recognize impact of APRM Downscale on RMCS and identify actions required to correct the abnormal condition to resume Control Rod movement. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 3-ARP-9-5A Rev. 40 (Attach if not previously provided)
OPL171.029 Rev. 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.029 V.B.7 (As available)
OPL171.148 V.B.13

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

Question History: Last NRC Exam La Salle 08 #34

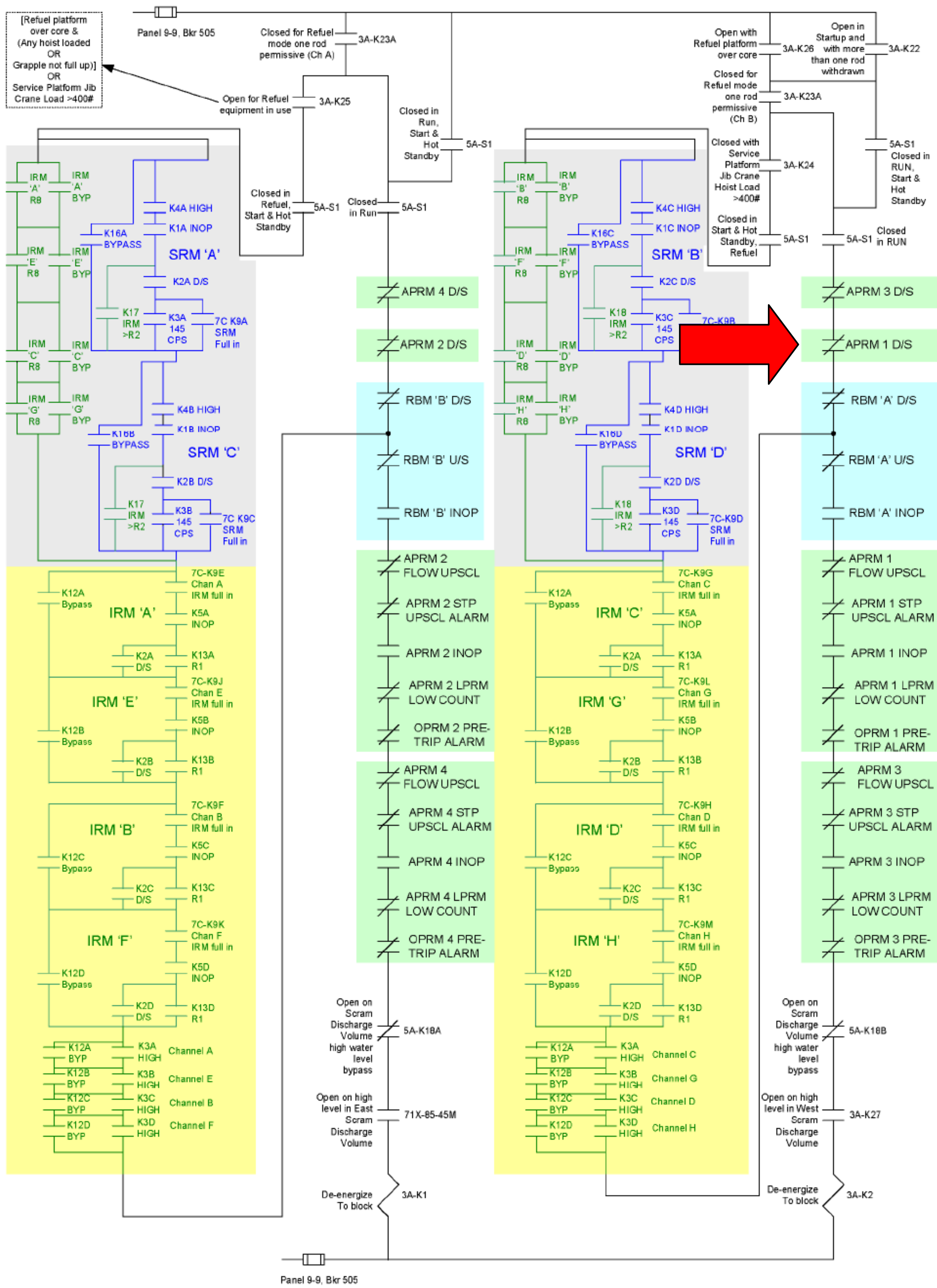
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

OPL171.029
Revision 13
Appendix C
Page 65 of 65



TP-14: ROD WITHDRAWAL BLOCK LOGIC

BFN Unit 3	Panel 9-5 3-XA-55-5A	3-ARP-9-5A Rev. 0040 Page 8 of 46

APRM DOWNSCALE/ OPRM INOP	4

(Page 1 of 1)

Sensor/Trip Point:

APRM Downscale	≤ 5%
OPRM Inop	Less than 23 operable cells

Sensor Location: Control Room Panel 3-9-14.

Probable Cause:
A. Unbypassed APRM channel at or below sensor set point.
B. SI or SR in progress.
C. Unit shutdown (Mode 3 or 4).
D. Failed sensor.

Automatic Action:
A. Rod withdrawal block with Rx Mode Sw. in RUN.(APRM only)

Operator Action:
A. **DETERMINE** which APRM/OPRM channel is downscale/inop. ☐
B. **IF** APRM failed downscale, **THEN** **BYPASS** channel. **REFER TO** 3-OI-92B. ☐
C. **IF** a single OPRM channel is inoperable, **THEN** **BYPASS** channel. **REFER TO** 3-OI-92B. ☐
D. **IF** the OPRM Trip Function is inoperable in MODE 1, **THEN**: **PERFORM** 3-SR-3.3.1.1.I to initiate alternate Thermal-Hydraulic Instability monitoring and required actions. ☐
E. **REFER TO** Tech Spec Table 3.3.1.1-1, TRM Table 3.3.4-1. ☐

References:

3-45E620-6	3-107E5784-20	3-107E5784-03
3-107E5784-03A	3-OI-92B	Technical Specifications
Technical Requirements Manual-TRM		

INSTRUCTOR NOTES

b. Imposed when, with three insert errors existing and an insert block present, a control rod other than one of the insert error control rods is selected.

c. In each case above, the block is applied to force the correction of the error before allowing movement of any other control rods.

d. Withdraw blocks are alarmed on the RWM operator's panel by a WITHDRAW BLOCK indicator light and status indication at all RWM display screens. — Panel 9-5 RWM ROD BLOCK annunciator

9. Insert block — Obj. V.B.8.d Obj. V.C.3.d OI-85 P&L

a. Imposed when a control rod is moved which exceeds the maximum number of allowable insert errors.

(1) The number of allowable insert errors may be varied through use of an off-line RWM system function.

(2) The number of allowable insert errors may be set to values of 0, 1 or 2.

(3) At Browns Ferry, 2 insert errors are allowed; 3 insert errors will cause an insert block.

b. Imposed when a withdraw error has been made, a withdraw block applied, and a control rod other than the withdraw error control rod is selected.

c. In each case above, the block is applied to force correction of the error before allowing further control rod movement.

d. Insert blocks are alarmed on the operator's panel by an INSERT BLOCK indicator light and status indication at all RWM display screens. — Obj. V.B.10

Panel 9-5 RWM ROD BLOCK annunciator

Question 34 LaSalle 07-01 NRC ILT Exam

Safety Function: 3.7 Instrumentation

System: 215002 RBM

K/A: A2.05 - Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RBM high or inoperable: BWR-3,4,5

I.R.: 3.2 / 3.3 10CFR 55 Content Part 41. 5 / 45.6

License Level: RO Cognitive Level: High

Objective: 045.00.05 PRA: No

Question Exposure: 07-01 NRC Exam

New Question developed for the 07-01 NRC ILT Exam

Unit 1 is currently at 90% reactor power with surveillance LOS-RD-SR7 "Channel Interference Monitoring" in progress.
Alarm 1H13-P603, A406, RBM HI/INOP has just illuminated due to the 1A RBM generating a valid "Failure to Null" signal.

This condition will result in a _______(1) _______ requiring _______(2) _______ before continuing with the surveillance.

A. (1) rod insert and withdrawal block
(2) bypassing the 1A RBM at panel 1H13-P603

B. (1) rod withdrawal block ONLY
(2) bypassing the 1A RBM at panel 1H13-P603

C. (1) rod insert and withdrawal block
(2) placing the 1A RBM Mode Switch in "STANDBY" at panel 1H13-P608

D. (1) rod withdrawal bock ONLY
(2) placing the 1A RBM Mode Switch in "STANDBY" at panel 1H13-P608

Correct Answer B:

(1) rod withdrawal block ONLY
(2) bypassing the 1A RBM at panel 1H13-P603

Explanation: Per LOR 1H13-P603-A406, A.1. "BLOCKS control rod withdrawal". B.5 "If a RBM channel is INOP, BYPASS affected channel per LOP-NR-05, "Rod Block Monitor Operation". NOTE: A review of LOP-NR-05 shows that the procedure never provides specific direction to place the RBM in bypass, rather it discusses that only one of the two RBMs may be bypassed, however the operators would be required to bypass the 1A RBM to continue with the surveillance.

Examination Outline Cross-reference:

215002 RBM

A3.04 (10CFR 55.41.7)

Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including:

- Verification of proper functioning / operability: BWR-3,4,5

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	215002A3.04	
Importance Rating	3.6	------

Proposed Question: **# 55**

Unit 2 is at 75% Reactor Power with a Control Rod sequence exchange in progress when the following alarm is received:

- RBM HIGH / INOP, (2-9-5A, Window 24)

Which ONE of the following completes the statement?

The setpoint for this annunciator is __**(1)**__ **AND** the power level displayed on the RBM recorder is determined using the mid-level LPRMs **AND** __**(2)**__level LPRMs.

A. **(1)** 117%
(2) A

B. **(1)** 117%
(2) D

C. **(1)** 121.8%
(2) A

D. **(1)** 121.8%
(2) D

Proposed Answer: **B**

Explanation (Optional):

A CORRECT: Part 1 correct – See explanation B. Part 2 incorrect - The RBM does not use any input from the ‘A’ level LPRMS.

B **CORRECT:** Part 1 correct - The RBM High alarm set point is 117% with STP at 60% to 80%. Part 2 correct - The ‘A’ RBM channel selects two ‘B’ level and two ‘D’ level LPRMS, as well as all 4 ‘C’ level LPRMs. The ‘B’ RBM channel selects the remaining two ‘B’ and ‘D’ level LPRMs and all four ‘C’ level LPRMs.

C INCORRECT: Part 1 incorrect – 121.8 % is the set point for 25% to 60% STP. Part 2 incorrect as explained in A.

D INCORRECT: Part 1 incorrect as explained above. Part 2 correct as explained above.

Justification: To demonstrate ability to monitor automatic operation of the Rod Block Monitor, candidate must know trip set points at various power levels.

Technical Reference(s): OPL171.148 Rev. 12 (Attach if not previously provided)

1-ARP-9-5A Rev. 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.148 V.B.23/26 (As available)

Question Source: Bank # X (Hatch 09 Audit)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

INSTRUCTOR NOTES

c. If a control rod withdrawal block is initiated by the RBM due to exceeding specified values, it applies only to the control rod selected. Movement of other control rods should not be inhibited.

(1) If the RBM is inoperable, none of the selected control rods will be able to be withdrawn.

(2) One channel of the RBM ('A' or 'B') can be manually bypassed using a joystick on Panel 9-5.

(3) Only one channel of the RBM can be bypassed at a time.

3. Basic Operation

a. When a control rod other than a peripheral control rod is selected with STP above 25%, the LPRMs adjacent to the control rod are selected and displayed by the RBM.

b. The RBM initially performs a null sequence, whereby it sets the initial flux level surrounding the control rod at 100%

(1) Null sequence is performed every time following selection of a control rod other than a peripheral control rod.

c. The RBM then determines which setpoint to use based on STP input from the assigned APRM.

Obj. V.B.26.a
Obj. V.B.22.b

(1) Setpoints are determined based on power levels

(a) If STP is between 25% and 60%, the LOW setpoint is used (121.8%)

(b) If STP is between 60% and 80%, the INTERMEDIATE setpoint is used (117.0%).

(c) If STP is above 80%, the HIGH setpoint is used (112.0%)

The setpoints are a percentage of the initial nulled power. The power is initially nulled to 100% based on the LPRM power surrounding the rod. If power then rises to the determined percentage above the initial nulled value, a rod block is imposed.

INSTRUCTOR NOTES

(3) The 'B' RBM channel selects the remaining two 'B' and 'D' level LPRMs and all four 'C' level LPRMs.

(4) The LPRM does not use any input from the 'A' level LPRMS

(5) The following tables show the LPRM selection for the RBM.

RBM Channel 'A'				
Level	A	B	C	D
Upper Right		X	X	
Lower Right			X	X
Upper Left			X	X
Lower Left		X	X	

Obj. V.B.23

RBM Channel 'B'				
Level	A	B	C	D
Upper Right			X	X
Lower Right		X	X	
Upper Left		X	X	
Lower Left			X	X

b. LPRMs values are not used by the RBM if they are from LPRMs that are bypassed in the APRM or auto-bypassed by the RBM. Obj. V.B.23

c. LPRMs that are downscale (<3%) are also bypassed by the RBM. Obj. V.B.25

(1) The LPRM status display will show "AUTO/BYPASS" for the affected LPRM if it the associated LPRM is manually or automatically bypassed. Obj. V.B.24

8. RBM Instrumentation

a. RBM drawer instrumentation is shown in Appendix D.

BFN **Unit 1**	**Panel 9-5** **1-XA-55-5A**	**1-ARP-9-5A** **Rev. 0013** **Page 29 of 44**

RBM
HIGH/INOP
24

(Page 1 of 1)

Sensor/Trip Point:

Relay K1
A10K1 in RBM
Interface module

Relay K3
A10K5 in RBM
Interface module

A. RBM HIGH
   1. LOW SETTINGS 25% to <60% STP Alarms at 121.8%.
   2. INTERMEDIATE SETTING 60% to <80% STP Alarms at 117.0%.
   3. HIGH SETTING ≥80% STP Alarms at 112.0%.

B. RBM INOP
   1. Mode switch **NOT** in operate.
   2. RBM fails to null.
   3. Less than 50% of assigned LPRM inputs
   4. Loss of Input Power (Module unplugged).
   5. More than one rod selected.
   6. Critical self test fault detected
   7. Circuit Board not in circuit.
   8. Self Test Detected Critical Fault.

Sensor Location: Panel 1-9-14, MCR.

Probable Cause:
A. One or more sensor greater than or equal to setpoint.
B. SI (or SR) in progress.
C. Malfunction of sensor.

Automatic Action: Rod withdrawal block of presently selected control rod.

Operator Action:
A. **IF** moving control rods for start-up or power maneuvering, **THEN PERFORM** the following: (otherwise N/A)
   1. **VERIFY** correct control rod selected. ☐
   2. **VERIFY** Rod Out Permit light is not illuminated to ensure selected rod withdrawal is inhibited. ☐
   3. **CHECK** annunciator LPRM HIGH (1-xa-55-5a, Window 12) and matrix light, Panel 1-9-5 to determine if the alarm is due to high flux. ☐
   4. **DESELECT** then **RESELECT** the desired Control Rod to reset the alarm and reinitialize the RBM back to normalized 100%. ☐

Continued on Next Page

HATCH 2009 Audit

21. 215002A1.01 1
Unit 2 is at 75% power with the following conditions:

- o a rod sequence exchange is in progress
- o annunciator RBM UPSCALE OR INOPERATIVE (603-202) alarms

Which ONE of the following choices would complete the following statement?

The setpoint for this annunciator is ______ and the power level displayed on the RBM recorder is determined using the mid-level LPRMs and the ______ level LPRMs.

A. 109.3%
D

B. 109.3%
A

C. 105.5%
D

D. 105.5%
A

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
216000 Nuclear Boiler Inst.	Tier #	2	------
A1.02 (10CFR 55.41.5)	Group #	2	------
Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including:	K/A #	216000A1.02	
• Removing or returning a (sensor) transmitter to service	Importance Rating	2.9	------

Proposed Question: **# 56**

Unit 1 is at 100% Reactor Power. Normal Range Level Transmitter, 1-LT-3-60 is removed from service for maintenance with its input to Feedwater Level Control (FWLC) System bypassed.

During retest of 1-LT-3-60, Instrument Mechanics inadvertently equalize the Normal Range Level Transmitter, 1-LT-3-53.

Which ONE of the following completes the statement?

Indicated Reactor Water Level on Panel 1-9-5 RX WTR LEVEL NORMAL RANGE, 1-LI-3-53 will be __**(1)**__ **AND** the input into the FWLC System from 1-LT-3-53 __**(2)**__ be automatically bypassed.

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

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

C. **(1)** upscale
(2) will

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

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – With variable and reference leg pressures equalized, instrument will indicate upscale not downscale. Part 2 correct – See explanation C

B INCORRECT: Part 1 incorrect as explained above. Part 2 is incorrect – Plausible in that with one channel already bypassed in FWLC, candidate may believe a second channel removed from service will not result in automatically bypassing its input. If two level signals are BAD or invalid, the algorithm will average the remaining two levels and will control on that value. In this instance the two remaining signals are compared to each other. If they deviate by more than 8 inches, a process alarm will be generated, but neither will be declared invalid.

C **CORRECT:** Part 1 correct - The lower the sensed pressure difference the higher the indicated level (Pref - Pvar = △P). With variable and reference leg pressures equalized the pressure difference will be 0 and the instrument will indicate upscale. Part 2 correct – The RFW Control System will use a level signal provided the system determines the signal to be good and valid. A GOOD level signal is one that has NOT failed and is on scale. A VALID level signal is one that does NOT deviate from the average (or median) level by more than 8 inches. When the RFWCS declares a level signal bad or invalid, then the level instrument is automatically bypassed.

D INCORRECT: Part 1 correct as explained in C. Part 2 incorrect as explained in B.

Justification: Required ability to predict affect on Level Indication and FWLC if a Normal Range Level Channel is inadvertently removed from service with another Normal Range Level Channel already out of service.

Technical Reference(s): OPL171.003 Rev. 19 / 1-OI-3 Rev. 22 (Attach if not previously provided)
OPL171.012 Rev. 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.003 V.B.1/V.B.11 (As available)
OPL171.012 V.B.6

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

Question History: Last NRC Exam Susquehanna 08 #34

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

INSTRUCTOR NOTES

(2) A sustained drift in the same direction - detectable over a period of time using calibration data and checking the "as found" vs. "as left" data

(3) A change in process noise

(4) Slow response to or inability to follow planned plant transients or slow response to either an increasing or decreasing test pressure - most likely detectable during calibration procedures

IM's will be monitoring this during SRs

(5) Inability to respond over the entire design range - most likely detectable during calibration procedures

(6) Only the torus wide range water level transmitters on Unit 2 are affected by this bulletin. (additionally these two have not been replaced)

2-LI-64-159A

(0 to 20 feet)

LT-64-159B (Recorder)

C. Reactor vessel instrumentation description

1. Vessel Level Instrumentation

TP-3
Use redundant indications

a. Level instruments are ΔP cells. The level signal is obtained by comparing:

Obj. V.B.3

Fundamentals

(1) The pressure exerted by the actual height of water in the vessel downcomer (variable leg) to

P = Height of water times the density of water.

(2) The pressure exerted by a constant reference column of water (reference leg)

Discuss basic fault effects

(3) The higher the sensed pressure difference the lower the indicated level ($P_{ref} - P_{var} = \Delta P$).

b. A condensing chamber is installed in each reference column pipeline to maintain a constant level in the reference leg piping.

Discuss basic Condensing Chamber operation

h. If two level signals are BAD or invalid, the algorithm will average the remaining two levels and will control on that value. In this instance the two remaining signals are compared to each other. If they deviate by more than 8 inches, a process alarm will be generated, but neither will be declared invalid.

i. If three level signals are BAD or invalid, the algorithm will control on the remaining signal alone.

j. If all four level signals are BAD or invalid, the algorithm will transfer the system to Manual control mode, and generate a process alarm. Should not be able to manually bypass all 4 (using pushbuttons)

2. Main Steam Flow — Obj. V.D.5

a. Four steam flow differential pressure transmitters provide square-rooted signals corresponding to 0 to 5 Mlb/hr flow rates. (actual $\approx$ 4.6 to 4.7Mlb/hr) — Obj. V.B.1

b. The control algorithm checks the input signal quality and discards BAD data signals.

c. Each steam flow signal is adjusted for a flow nozzle adiabatic expansion factor, which is a function of the nozzle geometry and the ratio of the nozzle throat pressure to inlet pressure.

d. The algorithm calculates the average steam line flow and derives a total steam flow by multiplying the average by 4.

e. The total flow is further compensated for density based on the reactor pressure.

Illustration 8
(Page 1 of 7)

RFWCS Instrumentation

1.0 NARROW RANGE REACTOR WATER LEVEL

1.1 Components

LEVEL A, 1-LI-3-53

LEVEL B, 1-LI-3-60

LEVEL C, 1-LI-3-206

LEVEL D, 1-LI-3-253

1.2 Description

The instruments are located on Panel 1-9-5 along with their corresponding bypass pushbuttons. These instruments provide two types of indication and ranges; analog (0 to 60 inches) and digital (-10 to 70 inches). Each instrument has an amber light which illuminates when the signal has been bypassed automatically by the RFW Control System or manually by the Unit Operator.

1.3 System Operation

The RFW Control System will use a level signal provided the system determines the signal to be good and valid. A **GOOD** level signal is one that has **NOT** failed and is on scale. A **VALID** level signal is one that does **NOT** deviate from the average (or median) level by more than 8 inches.

The RFW Control System validates each narrow range level signal by comparing them to the average. A level signal that deviates from the average by more than 8 inches is declared invalid and is bypassed. A level signal that is declared bad by the RFWCS will also be bypassed automatically.

To avoid individual on-scale but faulty level signals from skewing the average, a secondary validation process is used to compare the average level to the median of the valid signals. If the average value differs from the median value by more than 4 inches, then the RFWCS will validate each level signal to the median value instead of the average. In this case, any level signal that varies by more than 8 inches from the median is declared invalid and bypassed by the system.

NRC SUSQUEHANNA RO EXAM 5/16/2008

QUESTION 34

Unit 1 is at 100% power in a normal lineup when I&C inadvertently equalizes the 'A' Narrow Range Level Transmitter, PDT-C32-1N004A (0 psid across DP Cell).

Which of the following will be the indicated level on the 1C652 SIP Panel indicator, LI-C32-1R606A; AND What is the actual RPV level response with Feedwater Level Control in Master Auto? (Assume no operator action)

A. Indicated level at 1C652, LI-C32-1R606A will read upscale.
Actual RPV level will lower to approx. 23 inches and stabilize.

B. Indicated level at 1C652, LI-C32-1R606A will read downscale.
Actual RPV level will rise to approx. 47 inches and stabilize.

C. Indicated level at 1C652, LI-C32-1R606A will read upscale.
Actual RPV level will continue to lower to the scram setpoint.

D. Indicated level at 1C652, LI-C32-1R606A will read downscale.
Actual RPV level will continue to rise to the turbine trip setpoint.

NRC SUSQUEHANNA RO EXAM 5/16/2008

K&A # 216000 Nuclear Boiler Inst.
Importance Rating 2.9

QUESTION 34

K&A Statement: A1.02 - Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including: Removing or returning a sensor (transmitter) to service

Justification:

A. Correct – Indicated level for equalized transmitter will be pegged upscale. FWLC will respond to this level input in Auto and will stabilize at + 23 inches with no immediate operator action required.

B. In-Correct – Indicated level is upscale, not downscale. Actual RPV level will lower and stabilize at + 23 inches. This is the plausible response if the indicated NR level failed downscale with FWLC in Auto.

C. In-Correct – Indicated level is upscale. Actual RPV level will stabilize at + 23 inches with no immediate operator action required. This response is plausible with multiple level instrument failures.

D. In-Correct – Indicated level is upscale, not downscale and actual RPV level will lower and stabilize at + 23 inches with no immediate operator action required. This response is plausible with multiple level instrument failures.

References: ON-145-001 rev 21 page 2
TM-OP-045-ST page 55
TM-OP-080-ST page 56

Student Ref: required No

Learning Objective: TM-OP-045-OB # 10297

Question source: Modified Bank: #63 See original attached

Question History: LOC 21 Cert Exam

Cognitive level: Memory/Fundamental knowledge:
Comprehensive/Analysis: X

10CFR 55.41 X

Comments: Created/Modified by: Morgan
Reviewed by: Michaels

Examination Outline Cross-reference:	Level	RO	SRO
230000 RHR/LPCI: Torus/Pool Spray Mode	Tier #	2	------
A4.04 (10CFR 55.41.7)	Group #	2	------
Ability to manually operate and/or monitor in the control room: • Minimum flow valves	K/A #	230000A4.04	
	Importance Rating	3.1	------

Proposed Question: **# 57**

Unit 1 RHR Loop I is started in Suppression Pool Spray Mode.

Which ONE of the following completes the statement?

RHR SYSTEM I MIN FLOW VALVE, 1-FCV-74-7, will automatically close if flow is _____.

A. 2600 gpm for 10 seconds.

B. 5800 gpm for 10 seconds.

C. 2600 gpm with **NO** time delay.

D. 5800 gpm with **NO** time delay.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Flow is incorrect but time delay is correct. Plausible in that this is the set point for Core Spray Minimum Flow Valve.

B **CORRECT**: The RHR minimum flow valve 1-FCV-74-7 closes on a flow of 5800 gpm after a 10 second TD.

C INCORRECT: Flow and time delay are incorrect.

D INCORRECT: Flow is correct but time delay ins incorrect. Plausible in that the Core Spray Minimum Flow Valve has no time delay.

Justification: Requires knowledge of expected automatic operation of RHR Minimum Flow Valve while in Suppression Pool Spray Mode to correctly answer.

Technical Reference(s): 1-OI-74 Rev. 69 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: OPL171.044 V.B.10 (As available)

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

BFN Unit 1	Residual Heat Removal System	1-OI-74 Rev. 0069 Page 17 of 332

3.6 Interlocks

A. The RHR system is equipped with pump and valve interlocks to assure the following:

1. All RHR pump flow is directed to the LPCI injection path during ECCS initiation.

2. Protection of low pressure piping from high reactor pressures.

3. A pump suction path is fully open prior to pump start.

4. Suction Path Interlocks:

   a. An RHR pump will not start or will trip, if running, unless its corresponding torus suction valve is open or the SDC suction valve and the SDC suction supply valves, 1-FCV-74-47 and 48, are open.

   b. The torus suction valves cannot be opened unless the corresponding pumps SDC suction valve is fully closed.

   c. The SDC suction valves cannot be opened unless the corresponding pumps Torus suction valve is fully closed.

5. RHR Minimum Flow Valve Interlocks:

   a. The RHR minimum flow valves auto close if both pumps in the corresponding loop are off and either pump's SDC suction valve is open.

   b. The minimum flow valves open and close on a low flow of 5800 gpm after a 10 second TD. The tolerance of the flow switch may allow the setpoint of the min flow valve to be from 4500 gpm to 7000 gpm. Operation outside of this expanded range should be investigated. The analytical limit as listed in design criteria BFN-50-7074 is 11000 gpm for min flow valve closure. (MD-Q0074-960020 Rev 01)

   c. Placing the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 1-HS-74-148(149), in INHIBIT, will simulate a high flow and the minimum flow valve will remain closed regardless of flow.

BFN Unit 1	Core Spray System	1-OI-75 Rev. 0015 Page 10 of 102

3.0 PRECAUTIONS AND LIMITATIONS (continued)

J. The CS System will auto initiate from the following signals:

1. RPV water level at or below -122 inches
2. DW pressure at or above 2.45 psig and RPV pressure at or below 450 psig

K. The CS inboard and outboard injection valves have in-line valve interlocks to prevent both valves from being opened with RPV pressure at or above 450 psig. Both receive auto open signals when there is a CS initiation signal and RPV pressure is below 450 psig. The inboard valve may be throttled immediately after initiation.

L. The Core Spray test valve receives an auto closure signal on any CS auto initiation.

M. The Core Spray minimum flow valves receive a closure signal when flow is about 2600 gpm rising and receives an open signal when flow lowers to about 2200 gpm.

N. The preferred method for maintaining the Core Spray system charged is the PSC head tank. If the PSC head tank is NOT available, the CST system may be aligned via 1-SHV-075-0700. The use of 1-SHV-075-0582A or 1-SHV-075-0582B does NOT meet the Tech Spec 3.6.1.3 requirements for Primary Containment integrity UNLESS a dedicated operator is at the controls of the valve who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

O. [NRC/C] The suppression pool water will be highly radioactive after a LOCA. Site Chemistry input/advice should be considered when deciding where to pump the contaminated water. [NRC Inspection Report 89-16]

P. After operation of 4160V breakers the charging spring shall be verified to have recharged by verifying locally, the breaker closing spring target indicates charged and the amber breaker spring charged light is on to ensure future breaker operation.

Q. Leakage of Suppression Pool quality water into the RPV may occur when Core Spray or RHR System pressure is above RPV pressure due to a 1/4 in. hole drilled into the outlet side disc face of CORE SPRAY SYS I(II) INBD INJECT VALVE, 1-FCV-75-25(53) and RHR SYS I(II) LPCI INBD INJECT VALVE, 1-FCV-74-53(67) to eliminate pressure locking concerns of these valves. (DCN W18895A for Core Spray and DCN 51199A for RHR)

R. Failure to ensure system is filled and vented prior to racking pump breakers into the connect position following restoration after maintenance may result in equipment damage caused by an inadvertent pump start.

Examination Outline Cross-reference:	Level	RO	SRO
233000 Fuel Pool Cooling/Cleanup	Tier #	2	------
K2.02 (10CFR 55.41.7)	Group #	2	------
Knowledge of electrical power supplies to the following: • RHR pumps	K/A #	233000K2.02	
	Importance Rating	2.8	------

Proposed Question: **# 58**

Which ONE of the following completes the statement for requirements detailed in 2-OI-74, "Residual Heat Removal System?"

The **NORMAL** power supply to the pump(s) used for the **PREFERRED** method for Supplemental Fuel Pool Cooling is a ________.

A. 4 kV Shutdown Board.

B. 4 kV Common Board.

C. 480 V Shutdown Board.

D. 480 V Reactor Building Vent Board.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The RHR Pumps can be used for Fuel Pool Cooling; but are NOT preferred. If the candidate believes they are, then this is the power supply to the four RHR Pumps generically.

B INCORRECT: This is actually an ALTERNATE upstream power supply board to the board which powers the RHR Drain Pumps; made 4kV to balance out the question. Candidate could potentially believe RHR Pumps are powered from this set of boards (lower probability).

C INCORRECT: Due to the relative importance of the RHR System as a whole, the candidate may believe that this is a safety-related load. If that were the case, it would be powered by this set.

D **CORRECT:** (See attached excerpts) – 2-OI-74 specifies exactly that the RHR Drain Pumps are PREFERRED over the RHR Pumps for Fuel Pool Cooling applications. Lesson Plan OPL171.044 indicates two reasons why – RHR Drain Pump flowrates closely match that of the FPC Pumps and to avoid Min Flow issues on RHR Pumps. Attachment 3 to OI-74 delineates power supplies to both the RHR Drain Pumps and the RHR Pumps, with 480 V Reactor Bldg Vent Boards supplying the RHR Drain Pumps.

RO Level Justification: Tests candidate's knowledge of power supplies of RHR-related pumps that are utilized for Supplemental Fuel Pool Cooling and the relationship of these pumps to the available lineups. This question is rated as C/A due to the requirement to assemble, sort, and integrate two distinct parts of the question to predict an outcome. This requires mentally using specific knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 2-OI-74, Rev. 150 / OPL171.044, Rev. 17 (Attach if not previously provided)

2-OI-74 Att. 3, Rev. 139

Proposed references to be provided to applicants during examination: NONE

Learning Objective: _V.B.3 / 5_ (As available)

Question Source:

Bank #	
Modified Bank #	
New	X

(Note changes or attach parent)

Question History:

Last NRC Exam	

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41	X
55.43	

Comments:

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0150 Page 167 of 449

8.10 Initiation of Supplemental Fuel Pool Cooling with RHR Drain Pump A(B)

CAUTIONS

1) When RHR Drain Pump A is used for Supplemental Fuel Pool Cooling, Loop I of the RHR System is inoperable for LPCI. When RHR Drain Pump B is used for Supplemental Fuel Pool Cooling, Loops I and II of the RHR System are inoperable for LPCI.
2) Care should be exercised when changing the operating mode or any system parameter while SFSP or reactor cavity operations are in progress. This will preclude the possible introduction of sediment/dirt into the SFSP or reactor cavity, thereby reducing water clarity. The refuel floor SRO, if applicable, is required to be contacted for permission to alter RHR/SDC System alignment and/or parameters.

NOTES

1) Supplemental Fuel Pool Cooling should only be used when required to maintain fuel pool temperature below 125°F.
2) An RHR Drain pump is the preferred method for Supplemental Fuel Pool Cooling. If necessary to maintain fuel pool temperature below 125°F, an RHR Pump may be used to raise Supplemental Fuel Pool Cooling in accordance with Section 8.12.
3) All operations are performed at Panel 2-9-3 unless otherwise noted.

[1] **IF** RHR Drain Pump A is to be used for Supplemental Fuel Pool Cooling, **THEN**

VERIFY the following initial conditions are satisfied:

- RHR Loop I is in Standby Readiness **REFER TO** Section 4.0. ☐
- RHR Loop II is in Standby Readiness. (If RHR operability is required by Technical Specification 3.5.1/3.5.2). ☐

d. Standby Coolant Injection Supply (engineered safety feature)

Careful coordination with the other unit and 3 part communications are essential whenever a change is made

RHR Service Water may be injected into the Reactor Vessel as a last means of providing core cooling, once all other sources of cooling have been exhausted.

e. Supplemental Fuel Pool Cooling

(1) Immediately following a total core offload, the decay heat load from the irradiated fuel is high then decreases with time. RHR supplemental cooling may be required to assist the Fuel Pool Cooling System until the heat load is low enough to be maintained by the Fuel Pool Cooling System alone.

(2) Any RHR pump or either the 'A' or 'B' RHR Drain Pumps can be used in this mode. The RHR Drain pumps are the preferred pumps since they most closely match the flow rate of the FPC system.

(3) RHR Drain pumps are preferred to avoid running the RHR pumps below their minimum flow rating.

(4) Suction is from the Fuel Pool Skimmer Surge Tank through piping attached to the RHR System Shutdown Cooling pump suction and a manually operated, normally closed isolation valve.

4. Flow Paths

a. LPCI — TP-3

(1) Suppression pool suction and strainers — Obj. V.B.3, Obj. V.D.1

(2) ECCS ring header — Obj. V.E.2

(3) Two loops (System I and System II) — EOI App. 6B/6C/10C/10D

(4) Suction valves (4, one per pump)

(5) Two pumps per loop

(6) Flow Limiting Orifice — Prevents "runout"

(7) Minimum flow lines, one per loop

(8) Heat exchangers (4, 1 per pump)

BFN Unit 2	Attachment 3 Electrical Lineup Checklist	2-OI-74/ATT-3 Rev. 0139 Page 14 of 15

4.0 ATTACHMENT DATA (continued)

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Control Bay - 480V RMOV 1B - El 593'		
19A	2-BKR-074-0100 RHR HEAT EXCHANGER DISCHARGE CROSSTIE VALVE FCV-74-100 (MO10-171)	OFF[(1)]	____ ____
19A	2-HS-074-0100C RHR HTX A-C DISCH CROSSTIE VLV	OFF	____ ____
19E	2-BKR-074-0096 RHR PUMP 2A SUCTION CROSSTIE VALVE FCV-74-96	OFF[(1)]	____ ____
19E	2-HS-074-0096C RHR PUMP A SUCTION CROSSTIE VLV	OFF	____ ____

[1] Appendix R or the Equipment Qualification Program requires that the valve be closed with its breaker OFF (OPEN) except for testing or until compensatory measures are in place.

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Reactor Bldg - 480V RB Vent Bd 2B - El 565'		
1E	2-BKR-074-0105 RHR SYSTEM DRAIN PUMP 2A	ON	____ ____
9E	2-BKR-074-0107 RHR SYSTEM DRAIN PUMP 2B	ON	____ ____

BFN Unit 2	Attachment 3 Electrical Lineup Checklist	2-OI-74/ATT-3 Rev. 0139 Page 6 of 15

4.0 ATTACHMENT DATA

Performed On: ________________

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Control Bay - 4160V Shutdown Bd A - El 621'		
19	2-BKR-074-0005 RESIDUAL HEAT REMOVAL PUMP 2A	OPEN	____ ____
	Control Bay - 4160V Shutdown Bd C - El 621'		
18	2-BKR-074-0028 RESIDUAL HEAT REMOVAL PUMP 2B	OPEN	____ ____
	Control Bay - 4160V Shutdown Bd B - El 593'		
17	2-BKR-074-0016 RESIDUAL HEAT REMOVAL PUMP 2C	OPEN	____ ____
	Control Bay - 4160V Shutdown Bd D - El 593'		
17	2-BKR-074-0039 RESIDUAL HEAT REMOVAL PUMP 2D	OPEN	____ ____

Examination Outline Cross-reference:	Level	RO	SRO
234000 Fuel Handling Equipment	Tier #	2	------
K5.02 (10CFR 55.41.5)	Group #	2	------
Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT:	K/A #	234000K5.02	
• Fuel handling equipment interlocks	Importance Rating	3.1	------

Proposed Question: **# 59**

On Unit 3, the Mode Switch is in REFUEL **AND ALL** control rods are inserted. The Refueling Bridge operator grappled a fuel bundle, raised the grapple, **AND** commenced moving the bundle towards the core.

Which ONE of the following describes what will result as the Refueling Bridge moves towards the core?

The Refueling Bridge _____.

A. continues over the core **AND** initiates a control rod block.

B. continues over the core **AND** causes **NO** other protective actions.

C. stops before it reaches the core **AND** initiates a control rod block.

D. stops before it reaches the core **AND** causes **NO** other protective actions.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: With a loaded grapple, a rod block is generated when the Refueling Bridge is over the core.

B INCORRECT: Would be true with unloaded grapple.

C INCORRECT: Would be true with one rod withdrawn and second rod selection attempted.

D INCORRECT**:** Would be true with one rod withdrawn.

Justification: Requires knowledge of operational implications of Fuel Handling Equipment and Rod Block interlocks to correctly answer. Question involves the mental process of understanding the material by relating it to its own parts.

Technical Reference(s): 0-GOI-100-3A Rev. 53 (Attach if not previously provided)

OPL171.053 Rev. 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.053 V.B.4 / 5 (As available)

Question Source: Bank # DAEC 07 #35

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Duane Arnold 2007

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 0	Refueling Operations (In-Vessel Operations)	0-GOI-100-3A Rev. 0053 Page 18 of 175

3.3 Refueling Bridge Operation (continued)

C. When operating the refuel bridge in any speed other than JOG, ensure that the grapple or devices being transported have adequate clearance above items stored in the SFSP and Reactor Cavity.

D. Bridge travel toward the core will be stopped if any of the following conditions are met (except when interlocks are jumpered out by instruction in this procedure):

1. Any platform hoist loaded or main grapple **NOT** full up and all rods **NOT** full in with the platform near or over the core.
2. Platform near or over the core with the Mode Switch in other than REFUEL.
3. One rod withdrawn and when withdrawn rod is initially deselected with the Mode Switch in REFUEL. (As long as the rod that is withdrawn is never deselected bridge travel may continue and not be blocked by this interlock.)

E. The Associated Hoist operation will be stopped if any of the following exist.

1. Main Grapple position at full lower (46 ft.). Stops main hoist lower.
2. Main Grapple slack cable signal (< 50 lb. tension on cable) stops main hoist lower.
3. Associated Hoist loaded with all rods **NOT** full in with the platform near or over the core. Stops raise.
4. Associated Hoist overloaded (> 1000 lb.). Stops hoist raise.
5. All rods **NOT** full in with Platform near or over the core. Stops main hoist raise or lower.
6. Associated hoist at full up. Stops raise.

F. A Rod Block will occur if any of the following conditions are met:

1. Any platform hoist loaded or main grapple **NOT** full up with the platform near or over the core with the Mode Switch in REFUEL.
2. Service platform dummy plug not installed.
3. One rod withdrawn and a second rod selected with the Mode Switch in REFUEL.
4. Platform near or over the core with the Mode Switch in STARTUP.

BFN Unit 0	Refueling Operations (In-Vessel Operations)	0-GOI-100-3A Rev. 0053 Page 19 of 175

3.3 Refueling Bridge Operation (continued)

5. One rod withdrawn with the Mode Switch in REFUEL. (As long as all rods are full in, a rod may be selected and withdrawn as long as rod withdrawal signal is present. When the initial withdrawal signal ceases this block will enforce.)

G. [NER/C] Bridge and Trolley speed should be reduced when carrying a double blade guide with the triangular mast fully extended. Operating at a reduced speed which does not result in the 12-inch mast contacting the trolley will assure that no bending takes place. [GE SIL 594] Bridge speed should be limited (as required) in the core with the triangular mast fully extended and loaded to ensure the 12 inch mast does not contact the trolley at full speed or when stopping.

H. [NER/C] During fuel bundle transfers, operators should monitor independent verification of a closed grapple, including load light indication, grapple switch position, slack cable light, and video camera (if available) to verify proper response of the grapple hook to the electrical command to prevent dropping a fuel bundle. This action is required since the grapple switch light can indicate open even though the hooks are still locked closed. [GE SIL 542; GE SIL 618]

I. [NER/C] While using the fuel handling grapple, wedging can occur when a single or double blade guide handle enters into the grapple head if the handle is misaligned during the lead-in. The grapple should be rotated back and forth, as the handle enters the grapple, to ensure the handle is entering in the proper orientation. After the handle enters the main grapple body, handle wedging cannot occur.

When seated properly on the handle, the grapple elevation should read correctly. If the grapple elevation is higher than normal, the grapple should be removed and reoriented, and another attempt made to engage the blade guide handle. [GE SIL 618]

J. When transiting the chute with the mast empty, verify the “Boundary Zone Controller in “NORMAL” or the mast is raised to a level that allows it to clear the top of the chute.

K. If RPIS is unavailable and NO fuel is in the RPV, Attachment 22, Bypassing Bridge Interlocks, may be used to allow in-vessel/fuel pool work activities not associated with Reactor fuel.

L. If Refuel Interlocks are bypassed in accordance with Attachment 22, all operations with control rods **NOT** full in are unaffected.

M. When handling highly irradiated components, a mechanical stop is required to be installed on the hoist cable to prevent raising the component out of the water.

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Appendix C
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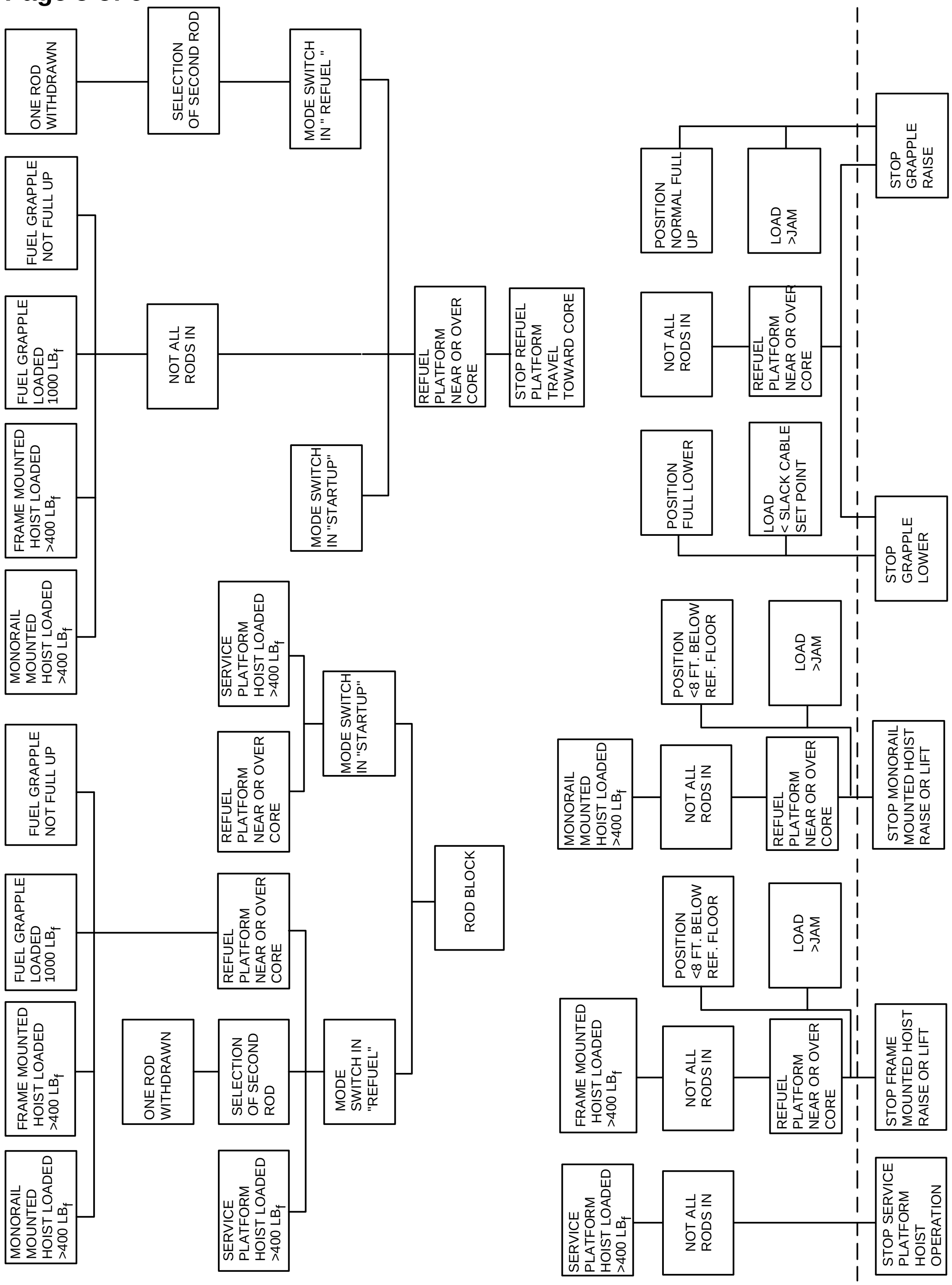

RO	K/A Number	Statement	IR	Origin	Source Question
35	234000	K4.02	3.3	B	2002 Clinton NRC Exam
LOK H	10CFR55.41(b)7	LOD (1-5)	Reference Documents SD 281, Rev 4		
Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and/or interlocks which provide for the following: Prevention of control rod movement during core alterations					

The Mode Switch is in REFUEL and all control rods are inserted. The Refueling Bridge operator grappled a fuel bundle, raised the grapple, and commenced moving the bundle towards the core.

Which ONE of the following describes what will result as the Refueling Bridge moves towards the core?

The Refueling Bridge:

a. continues over the core AND initiates a control rod block.

b. continues over the core AND causes NO other protective actions.

c. stops before it reaches the core AND initiates a control rod block.

d. stops before it reaches the core AND causes NO other protective actions.

Correct Answer: A With a loaded grapple, a rod block is generated when the Refueling Bridge is over the core.
Plausible Distractors: B is plausible: would be true with unloaded grapple. C is plausible: would be true with one rod withdrawn and second rod selection attempted. D is plausible: would be true with one rod withdrawn.
Objective Link: None

Examination Outline Cross-reference:	Level	RO	SRO
241000 Reactor/Turbine Pressure Regulator	Tier #	2	------
K6.06 (10CFR 55.41.7)	Group #	2	------
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR / TURBINE PRESSURE REGULATING SYSTEM: • Reactor pressure	K/A #	241000K6.06	
	Importance Rating	3.8	------

Proposed Question: **# 60**

Unit 3 is at 100% Reactor Power with the EHC system in Header Pressure Control.

Which ONE of the following would be the result if the output of one of the two header pressure transmitters fails **UPSCALE**?

A. The Reactor Scrams on MSIV Closure.

B. The Reactor Scrams on High Reactor Power.

C. The Reactor Scrams on High Reactor Pressure.

D. The other header pressure transmitter maintains Reactor Pressure.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** With EHC Control System in HEADER PRESSURE CONTROL, a single Header pressure input fails high with the reactor mode switch is in the RUN position, a Group 1 isolation will occur when steam line pressure lowers to 852 psig, resulting in a reactor scram

B INCORRECT: Plausible in that if EHC Pressure Control Signal output signal had failed low rather than transmitter output, would result in Turbine Control Valves closing, steam flow to be restricted, and Reactor pressure to increase resulting in high Reactor Power. EHC Pressure Control Signal output failing low would also result in Bypass Valves remaining closed.

C INCORRECT: Plausible in that if EHC Pressure Control Signal output signal had failed low rather than transmitter output, would result in Turbine Control Valves closing, steam flow to be restricted, and Reactor pressure to increase. EHC Pressure Control Signal output failing low would also result in Bypass Valves remaining closed

D INCORRECT: With EHC Control System in HEADER PRESSURE CONTROL, a single Header pressure input fails low. The following may occur: The EHC Control System automatically bypasses the applicable input. Therefore, if the output one of the two header pressure transmitters failed downscale instead of upscale, this would be the correct answer.

Justification: Requires Knowledge of the effect that a malfunction of the Reactor Pressure signal to EHC Logic will have on the REACTOR / TURBINE PRESSURE REGULATING SYSTEM. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.228 Rev 4 (Attach if not previously provided)
3-AOI-47-2 Rev. 6 / 3-OI-47 Rev. 89

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.228 V.B.1 (As available)

Question Source: Bank # OPL171.230 #5
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

BFN Unit 3	Turbine EHC Control System Malfunctions	3-AOI-47-2 Rev. 0006 Page 3 of 8

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, and operator actions for malfunctions of the EHC Control System.

2.0 SYMPTOMS

A. While in REACTOR PRESSURE CONTROL, failed high or low reactor pressure input. The following symptoms may occur:

1. EHC/TSI SYSTEM TROUBLE annunciation, 3-XA-55-7A, Window 6, alarms.
2. On Panel 3-9-7, REACTOR PRESS A(B)(C)(D) BYPASS pushbutton backlight illuminates.

B. While in HEADER PRESSURE CONTROL, a single header pressure input signal fails low. The following symptoms may occur:

1. EHC/TSI SYSTEM TROUBLE annunciation, 3-XA-55-7A, Window 6, alarms.
2. On Panel 3-9-7, HEADER PRESSURE A(B) BYPASS pushbutton backlight illuminates.

C. While in HEADER PRESSURE CONTROL, a single header pressure input signal fails high.

1. The following symptoms may occur:

   a. On Panel 3-9-7, HEADER PRESSURE A(B) BYPASS pushbutton backlight illuminates.

   b. Turbine control valves open to position established by CV POSITION LIMIT setpoint.

   c. Turbine bypass valves open.

   d. Feedwater/Steam flow mismatch.

   e. Reactor pressure lowers.

   f. Generator output rapidly lowers.

BFN Unit 3	Turbine EHC Control System Malfunctions	3-AOI-47-2 Rev. 0006 Page 5 of 8

3.0 AUTOMATIC ACTIONS

A. With EHC Control System in REACTOR PRESSURE CONTROL, a single Reactor Pressure input fails high or low. The following may occur:

The EHC Control System automatically bypasses the applicable input.

B. With EHC Control System in HEADER PRESSURE CONTROL, a single Header pressure input fails low. The following may occur:

The EHC Control System automatically bypasses the applicable input.

C. With EHC Control System in HEADER PRESSURE CONTROL, a single Header pressure input fails high. The following may occur:

If the reactor mode switch is in the RUN position, a Group 1 isolation will occur when steam line pressure lowers to 852 psig, resulting in a reactor scram.

D. Loss of Unit Preferred or Loss of ICS Distribution. The following may occur:

Unit Preferred and ICS are auctioneered power sources to the EHC Control System. Losing one of the power sources should **NOT** affect system operation due to the unaffected power source still supplying power to the system.

BFN Unit 3	Turbine-Generator System	3-OI-47 Rev. 0089 Page 21 of 240

3.9 EHC Controls

A. The EHC Workstation is normally in the VIEW mode. While in this mode, disabled function lettering is grey The only function available to the operator in this mode is the ability to reset an alarm on the Alarm screen.

Performing other functions at the workstation requires that the operator go to the Menu screen, select LOGIN button, type in as user OPS and password OPS. The enabled function lettering is now black. When the extended functions are no longer necessary the operator may return to VIEW mode by typing into user VIEW and password VIEW. Otherwise, the system will automatically revert to VIEW mode after approximately one hour of inactivity.

B. The EHC Control System can be used in either Reactor Pressure control or Header Pressure control. While in Header Pressure control, a single header pressure input failing high could cause the bypass valves to open. While in Reactor Pressure control, a single Reactor Pressure input failing high will not affect the bypass valves. For this reason, Reactor Pressure control is the preferred mode of operation for the EHC Control System.

C. The following pertain to the Max Combined Flow Limit:

1. Maximum combined flow limit setting of 150% (upper limit) precludes exceeding Thermal Limits during a single turbine control valve closure.

2. The max combined flow upper and lower setting limits are 50% and 150%. Normally it is set at 125%.

3. The Maximum Combined Flow Limit setting is adjustable only on the EHC WORK STATION computer (Panels 3-9-7 and 3-9-31).

4. Max Combined Flow Limit setpoint can be found on the following computer screens:

   a. On ICS, EHC TURBINE CONTROL (EHCTC) screen.

   b. On EHC WORK STATION, TURBINE CONTROL screen.

D. [NER/C] Complete failure of the Push Rod-Spring Guide Coupling Bolts on a Control Valve (CV) or a Combined Intermediate Valve (CIV) will give indication of a closed CV or CIV on Panel 3-9-7 even though the valve may actually still be full open. Should this event be believed to have occurred, and the unit appears to be running satisfactorily, DO NOT TRIP the turbine. Notify Operations management and the GE Representative for assistance. (Note: Maintenance will have these bolts completely changed out

BFN Unit 3	Main Steam System	3-OI-1 Rev. 0029 Page 9 of 65

3.2 Main Steam Isolation Valves (MSIV)

3.2.1 MSIV Closure

A. The MSIVs should not be fast closed when the reactor is in the cold shutdown and no steam flow, unless required by surveillance, test instruction, or an abnormal condition. [BFNPER 164499]

B. When an MSIV is closed at power, the potential exists for an isolation of the Hydrogen Water Chemistry System to occur. This is due to the possibility of a hydrogen bubble becoming entrained in the main steam line drains and subsequently being released when the main steam line drains reposition in response to a MSIV closure. This scenario can result in a small Off Gas System hydrogen spike of sufficient strength to cause an automatic isolation of the Hydrogen Water Chemistry System.

C. Closure of all MSIVs could cause turbine shaft damage if main condenser vacuum is maintained and seal steam supply is not established from the auxiliary boiler.

3.2.2 MSIV Isolation

A. Main steam tunnel temperature should not be allowed to exceed 189°F to prevent MSIV isolation.

B. Whenever reactor pressure is reduced to 852 psig and the reactor mode switch is in RUN position, the MSIVs will close.

C. The MSIVs will close if 250 Vdc and 120 Vac power to the MSIV control logic is de-energized.

D. Reactor power should be ≤ 66% prior to closing an MSIV greater than 15 percent during closure testing. This should prevent a high steam line flow MSIV closure and subsequent reactor scram.

E. Placing all MSIV Handswitches in the Close Position allows the PCIS group one trip logic to be reset. Leaving any Handswitch in the Open Position prevents resetting the group one logic.

F. The PCIS group one trip parameters do not exceed trip setpoints.

1. Reactor water level above -122 in.
2. MSL flow less than 135%.
3. MSL tunnel temperature less than 189°F.
4. MSL pressure greater than 852 psig if in Mode 1.

BFN Unit 3	Reactor Protection System	3-OI-99 Rev. 0042 Page 53 of 71

Illustration 2
(Page 1 of 2)

Unit 3 Reactor Scram Initiation Signals

The following conditions initiate a Reactor Scram:

NOTE

If any of the events listed below occur, a Reactor Scram should be automatically initiated. If an automatic scram initiating event occurs and the scram is **NOT** initiated, the operator is required to verify the condition and then manually scram the Reactor and Place Mode Switch in SHUTDOWN.

	Scram	Setpoint	Bypass
A.	Manual	N/A	N/A
B.	Mode Switch in SHUTDOWN	N/A	2-seconds after MODE SWITCH to SHUTDOWN
C.	SRM Hi Hi	2 x 10^5 cps	Shorting links installed
D.	IRM INOP	**NOT** in OPERATE, High Voltage Low Module unplugged, Loss of ± 24 VDC	Mode Switch in RUN (unless companion APRM downscale)
E.	IRM Hi Hi	116.4/125	Mode Switch in RUN
F.	APRM Hi Hi	14%	Mode Switch in RUN
G.	APRM Hi Hi	DLO: (0.66 W + 65) [W = Recirc loop flow in % rated] SLO: ≤ (0.66(W-ΔW) + 65%) [W = Recirc Loop Flow in % rated].	Mode Switch in RUN
H.	APRM Hi Hi	119%	N/A
I.	APRM INOP	• Self-Test Critical Fault. • Loss of Input Power. • Chassis Mode Sw. NOT in OPERATE. • Firmware Watchdog timer timed out.	N/A

BFN **Unit 3**	**Reactor Protection System**	**3-OI-99** **Rev. 0042** **Page 54 of 71**

Illustration 2
(Page 2 of 2)

Unit 3 Reactor Scram Initiation Signals

	Scram	**Setpoint**	**Bypass**
J.	OPRM TRIP	Any one of the three algorithms, period, growth, or amplitude for an operable OPRM cell has exceeded its trip value conditions:	Reactor is NOT operating in the AUTO ENABLE Region of the Power/Flow Map.
K.	Low RPV Water Level (Level 3)	+2.0"	N/A
L.	Hi RPV Pressure	1073 psig	N/A
M.	Hi DW Pressure	2.45 psig	N/A
N.	MSIV closure	90% open (3 Main Steam Lines)	**NOT** in RUN
O.	Scram Discharge Instrument Volume Hi Hi	• Thermal level switches 49 gallons (LS-85-45A,B,G,H) • Float level switches 45 gallons (LS-85-45C,D,E,F)	Mode Switch in SHUTDOWN or REFUEL with keylock switch in BYPASS
P.	TSV Closure	90% open (3 TSVs)	< 30% Rx Power (≤ 148.5 psig 1st stage pressure)
Q.	TCV Fast Closure (load reject)	40% mismatch (amps to cross-under pressure); 850 psig EHC RETS at TCV (1 or 3) & (2 or 4)	< 30% Rx Power (≤ 148.5 psig 1st stage pressure)
R.	Loss of RPS Power	N/A	N/A
S.	Scram Channel Test Switches	Key-locked in AUTO Panels 3-9-15 & 3-9-17	N/A

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

<u>3. Reactor Vessel Steam Dome Pressure - High</u>
(PIS-3-22AA, PIS-3-22BB, PIS-3-22C and PIS-3-22D)

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. The Reactor Vessel Steam Dome Pressure - High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analysis of Reference 4, reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor Fixed Neutron Flux - High signal, not the Reactor Vessel Steam Dome Pressure - High signal), along with the S/RVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

High reactor pressure signals are initiated from four pressure transmitters that sense reactor pressure. The Reactor Vessel Steam Dome Pressure - High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure - High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 when the RCS is pressurized and the potential for pressure increase exists.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.c. Average Power Range Monitor Fixed Neutron Flux - High

The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux - High Function to terminate the CRDA.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux - High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux - High, (Setdown) Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux - High Function is not required in MODE 2.

(continued)

INSTRUCTOR NOTES

2. During turbine start-up and for a brief time following synchronization, the bypass valve control also maintains the reactor steam pressure. Once all the bypass valves are closed, then the turbine control maintains reactor steam pressure either in Header Pressure or Reactor Pressure Control depending on which operating mode is selected. — *Monitor Plant parameters for expected response*

3. Steam pressure control is selectable from either panel 9-7 or the EHC Workstation by selecting HEADER PRESSURE CONTROL or REACTOR PRESSURE CONTROL. — *Obj.V.B.9.a*

4. Header Pressure Control Input Signal

    a. Two redundant pressure transmitters sense header pressure at the main steam throttle just upstream of the main turbine stop valves. — *Powered from within the EHC system*

    b. Both signals are monitored for low, high, difference, and hardware failures.

    c. The higher of the two signals when no failures are detected is selected as the input.

    d. A maximum difference setpoint of 10-PSI is also established to detect a fault and/or transmitter drift from either of the inputs.

    e. In the event a fault is detected, the channel is prohibited from being used in the signal processing and the appropriate BYPASS pushbutton light will illuminate on 9-7 and on the HMI operator interface. — *Obj.V.B.9.c*

    f. Once the failed signal is corrected, depressing the BYPASS pushbutton will reset the BYPASS logic and both input signals will then be processed.

INSTRUCTOR NOTES

g. This mode IS NOT single failure proof - one of the two pressure sensors failing upscale can, and generally will be selected by the logic to control. This will open the TCV's and BPV's to depressurize the header to the MSIV isolation setpoint of 852 psig in RUN Mode.

h. In the unlikely event that both inputs signals are detected as failed, the control logic will automatically switch to reactor pressure control.

i. If header pressure drops below 700-PSI, and reactor pressure control is the controlling mode of operation, the control logic will automatically transfer to header pressure control. If desired, the operator may re-select reactor pressure control after the transfer has been made even though header pressure is below 700-psi. The automatic transfer logic will re-engage if header pressure rises above 725-psi.

5. Reactor Pressure Control Input Signal — TP-3

a. Four (4) redundant pressure transmitters (PT- 204a-d) grouped in pairs with "A" and "B" constituting one pair and "C" and "D" the other pair.

b. A pressure-biasing algorithm determines the lagged high-median value of the four (4) inputs and biases the remaining three (3) input signals to that high median value.

c. The high-median signal is then averaged with the other three signals and is used as "Actual Rx Pressure". — Four biased signals are averaged.

Examination Outline Cross-reference:

268000 Radwaste

K3.04 (10CFR 55.41.5)

Knowledge of the effect that a loss or malfunction of the RADWASTE will have on the following:

- Drain sumps

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	268000K3.04	
Importance Rating	2.7	------

Proposed Question: **# 61**

Unit 2 Turbine Building Floor Drain Sump Pump 'A' has automatically started on high level when the following alarm is received:

- FD COLLECTOR TANK LEVEL HIGH, (0-25-17B, Window 17)

Which ONE of the following completes the statement?

Turbine Building Floor Drain Sump Pump 'A' will ____.

A. trip **IMMEDIATELY**.

B. continue to run with **NO** discharge flow path.

C. continue to pump to the Floor Drain Collector Tank.

D. continue to run with discharge aligned to the Waste Collector Tank.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that it is a common feature for pump to trip if receiving tank level high or low flow. Example – If Cond Head Tank Level High alarms, both condensate transfer pumps secure on high level.

B INCORRECT: Plausible in that it is logical to isolate inputs to tanks on high level and some pumps will not trip even with the low flow condition. Example – If RCIC is operating in CST to CST mode and the Test Return isolates, RCIC will continue to operate dead headed. Min flow valve will not function with absence of initiation signal.

C **CORRECT:** There are no automatic features associated with FD COLLECTOR TANK LEVEL HIGH. Turbine Building Floor Drain Sump Pump 'A' will continue to pump to the Floor Drain Collector tank until high sump level is cleared.

D INCORRECT: Plausible in that it is common feature to re-direct tank inputs on a high tank level. Example – If Laundry Drain Tank Level High is received and FCV-77-179 is open to tank A, it will close to tank A and open to tank B.

Justification: Question tests Candidates knowledge of impact of high level in Floor Drain Collector Tank on Turbine Building Floor Drain Sump Pumps. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): OPL171.084 Rev. 5 (Attach if not previously provided)

0-ARP-25-17B Rev. 5

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New **X**

Question History: Last NRC Exam ______

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 0	0-LPNL-925-0017 XA-55-17B	0-ARP-25-17B Rev. 0005 Page 20 of 50

LA-77-28B
FD COLLECTOR
TANK
LEVEL HIGH
17

(Page 1 of 1)

Sensor/Trip Point:

LRS-77-28	LT-77-28	267" (approx. 98%)

Sensor Location:

LRS-77-28	LT-77-28
0-LPNL-925-0017	Floor Drain Collector Tank
Elevation 565	Elevation 546
Radwaste Control Room	Radwaste Bldg, W-7 D-LINE

Probable Cause:

A. Excessive inleakage.
B. Filter out of service.
C. Valve misalignment.
D. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** tank level with LRS-77-28 on 0-LPNL-925-0017. ☐
B. **IF** floor drain collector tank level is high, **THEN**
   1. **CHECK** STOPPED or **STOP** processing chemical waste tank. ☐
   2. **CHECK** for proper operation of floor drain filter system (flow, ΔP, etc.) and valve alignment. ☐
   3. **CHECK** for source of excessive inleakage (see print 0-47E830-3). ☐
   4. Upon Unit Supervisor approval, **LOCK OUT** all floor drain sump pumps EXCEPT radwaste and drywell. ☐
   5. **BLOW DOWN** floor drain collector tank. **REFER TO** OI-77. ☐

C. **CHECK** for indications of overflow condition:
   1. **CHECK** chemical tank level for rises on LR-77-167. ☐
   2. **CHECK** for excessive floor drain sump pump run time. ☐

D. **NOTIFY** supervisor and WO any failed component/sensor. ☐
E. **PROCESS** the floor drain collector contents. **REFER TO** OI-77. ☐
F. **LOG** valid events and actions taken in NOMS narrative log. ☐

References: 0-47E830-3 0-45E775-3 730E934 Series

OPL 171.084
Revision 5
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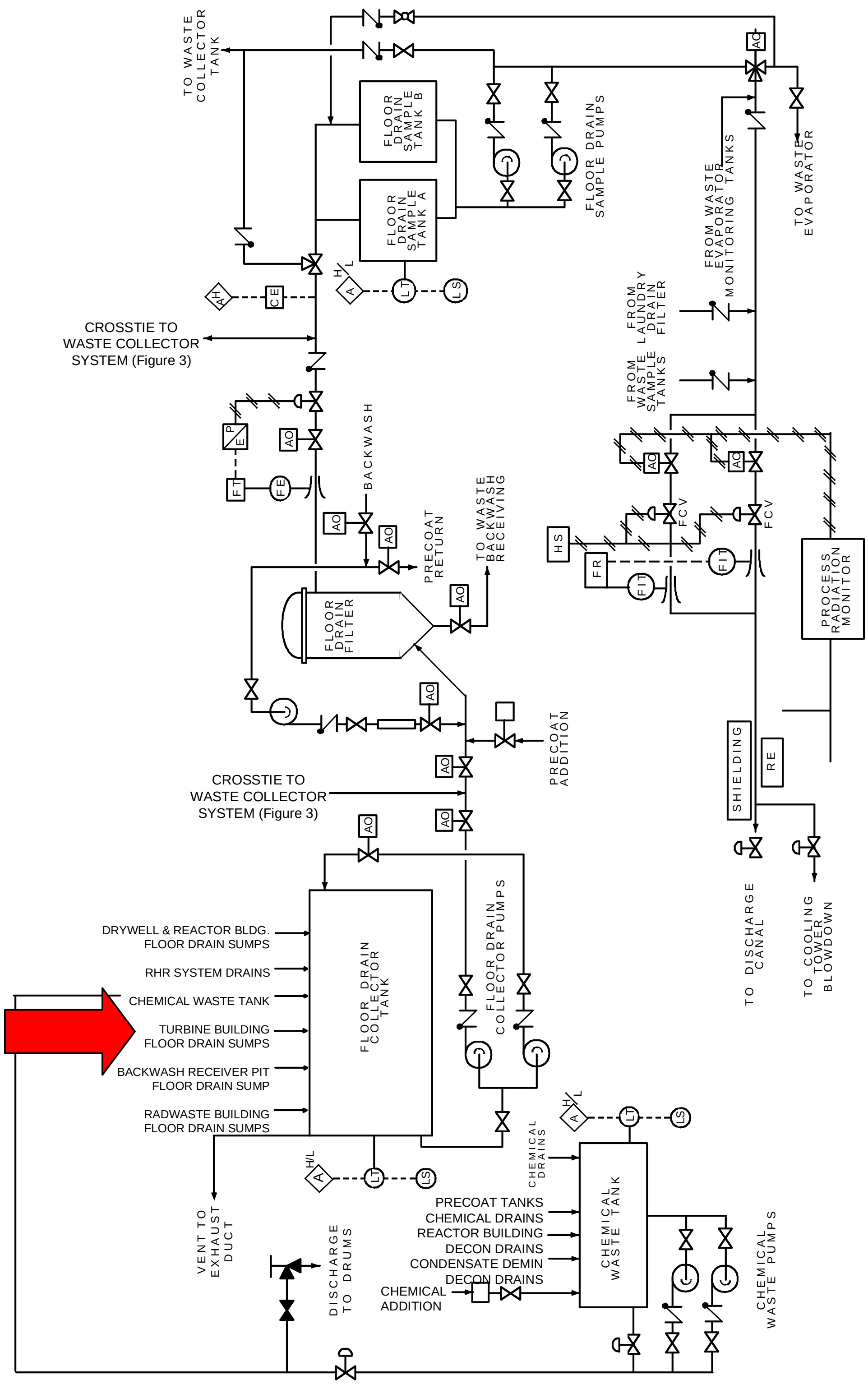


BFN Unit 1	Panel 9-20 1-XA-55-20B	1-ARP-9-20B Rev. 0028 Page 23 of 39

CNDS HEAD TANK
LEVEL
HIGH
0-LA-2-172A

20

(Page 1 of 1)

Sensor/Trip Point:

LS-2-172D 8 ft 7 in (Elevation 726 ft 10 in) rising

Sensor Location: Reactor Building Roof

Probable Cause:
A. Condensate transfer pump(s) malfunction.
B. Level switch malfunction.

Automatic Action: Both condensate transfer pumps secure on high level.

Operator Action:
A. **VERIFY** both Condensate Transfer Pumps OFF, using 0-HS-2-173A and -175A on Panel 1-9-22. ☐
B. **CHECK OPEN** makeup valve LCV-2-177, on Panel 1-9-22. ☐
C. **IF** alarm does **NOT** reset in a reasonable time, **THEN DISPATCH** personnel to **CHECK** level controls at the tank. ☐

References: 0-45E769-4 0-47E610-2-2 1-45E620-12-2

BFN Unit 0	0-LPNL-925-0017 XA-55-17B	0-ARP-25-17B Rev. 0005 Page 21 of 50

LA-77-178A-B LAUNDRY DRAIN TANK LEVEL HIGH 18	Sensor/Trip Point:		
(Page 1 of 1)	LRS-77-178A	LT-77-178A	85" (approx 87%)

Sensor Location:

LRS-77-178	LT-77-178A
0-LPNL-925-0017	Panel 25-78
Elevation 565	Elevation 546
Radwaste Control Room	Radwaste Bldg, W-4 A-LINE

Probable Cause:

A. Transferring tank B contents to tank A.
B. Inleakage - laundry, decont., etc.
C. Valve misalignment.
D. Sensor malfunction.

Automatic Action:

A. **IF** FCV-77-179 is open to tank A, it will close to tank A and open to tank B.
B. **IF** tank B level is **NOT** high, FCV-77-182A closes and FCV-77-182B opens.

Operator Action:

A. **CHECK** laundry tank A level (black pen) and tank B level (red pen) with LRS-77-178 on 0-LPNL-925-0017. ☐
B. **IF** both tank levels are 85%, **THEN** **REFER TO** OI-77 for tanks processing procedure. ☐
C. **IF** unable to determine cause of the level rise, **THEN**
   1. **VERIFY** the correct valve alignment exists for the operation in progress. ☐
   2. **REQUEST** laundry room/decon personnel check for any abnormal condition **AND IF** necessary, **THEN** **HALT** laundry/decon operation until problem is corrected. ☐
   3. **NOTIFY** supervisor. ☐

D. **IF** tank level rises after Chem Lab sample or during release/transfer, **THEN**
   1. **STOP/SECURE** release or transfer operation. ☐
   2. **NOTIFY** supervisor and **DETERMINE** source of inleakage. ☐
   3. **NOTIFY** Chem Lab and **REQUEST** resample. ☐

E. **LOG** valid events and actions taken in NOMS narrative log. ☐

References: 0-47E830-3 0-47E610-77-2 730E934-10

RCIC STEAM LINE LEAK DETECTION TEMP HI	TS-71-41A/B/C/D	138° F	3D-10

5. Valve Interlocks — TP-12

a. Steam valves — Obj. V.B.6.

(1) Isolation valves FCV-71-2 and 3 auto close on RCIC isolation signal.

(2) Steam line drain pot to main condenser hotwell valves FCV-71-6A and 6B automatically shut if steam supply to turbine valve FCV-71-8 is not full closed.

(3) Steam supply valve FCV-71-8 auto opens on an initiation signal. Auto closes on high reactor water level of +51".

b. Condensate suction valve FCV-71-19

(1) Automatically opens on initiation signal provided at least one of the torus suction valves is not fully open

(2) Automatically closes when both torus suction valves are fully open (FCV 71-17, 71-18)

c. Minimum flow valve FCV-71-34

(1) Automatically closes on turbine trip signal l (except overspeed with an initiation signal)

(2) Opens at system flows ≤60 gpm if a RCIC automatic start signal has been initiated. Closes at system flow ≥120 gpm

(3) On Manual Trip a closure signal applied as long as trip pushbutton is depressed

d. Test valve FCV-71-38

(1) Normally closed. Automatically closes (if open) on initiation signal

(3) All isolation signals are sealed in and must be manually reset. Single pushbutton on Panel 9-3 for each logic.

6. Turbine Trips — Obj. V.B.4.c. Obj. V.C.2.c. TP-9 Obj. V.D.4 Obj. V.E.6

a. Signals

(1) Any system isolation signal

(2) Mechanical overspeed (122%) — Obj. V.B.10.a. Obj. V.C.6.a. Obj. V.B.5.c. Obj. V.C.3.c. Obj. V.B.10.d

This trip physically unlatches the Trip Throttle Valve. Must be reset locally

(3) High turbine exhaust pressure (50 psig)

(4) Vessel high level (+51 inches) (Closes FCV 71-8) — "Level 8"

(5) RCIC pump suction pressure low (10" Hg vacuum) — Obj. V.E.10 Obj. V.E.11 Obj. V.E.12

(6) Manual

Remote by energizing solenoid (pushbutton) Local by manual trip lever

b. Action on trip other than high reactor water level — Obj. V.B.5.c. Obj. V.C.3.c.

Trips the 71-9, closes the minimum-flow valve (71-34) except manual trip or overspeed trip with an initiation signal present

Examination Outline Cross-reference:	Level	RO	SRO
272000 Radiation Monitoring System	Tier #	2	------
A3.02 (10CFR 55.41.7)	Group #	2	------
Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including:	K/A #	272000A3.02	
• Offgas system isolation indications	Importance Rating	3.6	------

Proposed Question: **# 62**

Which ONE of the following completes the statement?

MAIN STEAM LINE RAD HIGH-HIGH / INOP, (1-9-3A, Window 27), alarms at __**(1)**__ Normal Full Power Background radiation level **AND ALL** monitors at this radiation level__**(2)**__ result in a trip **AND** isolation of the Mechanical Vacuum Pumps.

A. **(1)** 1.5 times
(2) will

B. **(1)** 1.5 times
(2) will **NOT**

C. **(1)** 3 times
(2) will

D. **(1)** 3 times
(2) will **NOT**

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that this is the set point for MSL RAD HIGH alarm. Part 2 correct - See Explanation C.

B INCORRECT**:** Part 1 incorrect - See Explanation A. Part 2 incorrect – Plausible in that similar radiation monitoring system, Offgas Post Treatment Radiation Monitor does not result in an Isolation of Offgas until Hi-Hi-Hi Rad Level is received. Also, could be confused with MSIVs which do NOT isolate at all on elevated MSL Radiation levels.

C **CORRECT:** Part 1 correct - MAIN STEAM LINE RAD HIGH-HIGH / INOP (55-3A-27) alarm at a radiation level of 3 times the Normal Full Power Background radiation level. Part 2 correct - Mechanical Vacuum Pumps trip and isolate on High-High MSL Radiation 3 X NFP Background.

D INCORRECT: Part 1 correct - See Explanation C. Part 2 incorrect - See Explanation B.

Justification: Candidate must know set point and resulting actions with MSL Rad High-High to demonstrate ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including trip and isolation of the Mechanical Vacuum Pumps which are part of the Offgas System.

Technical Reference(s):	OPL171.033 Rev. 13	(Attach if not previously provided)
	1-OI-66 Rev. 22 / 1-9-3A Rev. 38	

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033_V.B.4 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0038 Page 41 of 53

MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C 27	Sensor/Trip Point:	
	1-RM-90-136 1-RM-90-137	3.0 x normal full power background including N-16 contribution and HWC System injection.

(Page 1 of 1)

Sensor Location: Radiation monitor drawers are on Panel 1-9-10 in the control room.

Probable Cause:

A. Radiation is three times the normal full power background.
B. Sensor malfunctions.
C. SI (SR) in progress.

Automatic Action:

A. Mechanical vacuum pumps trip.
B. Vacuum pump suction valves 1-FCV-066-0036 and 1-FCV-066-0040 close.

Operator Action:

A. **VERIFY** the alarm on 1-RM-90-136 and 1-RM-90-137 on Panel 1-9-10. ☐
B. **CONFIRM** main steam line radiation level on recorder 1-RR-90-135, Panel 1-9-2. ☐
C. **IF** alarm is valid and Reactor Scram has not occurred, **THEN PERFORM** the following:
   1. **IF** core flow is above 60%, **THEN LOWER** core flow to between 50-60%. ☐
   2. **MANUALLY SCRAM** the Reactor. ☐
   3. **REFER TO** 1-AOI-100-1. ☐

D. **IF** plant conditions **DO NOT** require execution of 1-C-5, **THEN VERIFY** the MSIVs closed. ☐
E. **NOTIFY** RAD PRO. ☐
F. **VERIFY** actions of 1-ARP-9-3A Window 7 have been completed. ☐
G. **IF** Technical Specifications limits are exceeded, **THEN REFER TO** EPIP-1. ☐

References: 1-47E610-90-1 730E915-9, 10 1-45E620-5

BFN Unit 1	Off-Gas System	1-OI-66 Rev. 0022 Page 11 of 120

3.0 PRECAUTIONS AND LIMITATIONS (continued)

I. The Mechanical Vacuum Pumps are not to be used to purge the main condenser if hydrogen concentration is suspected of being present.

J. The Mechanical Vacuum Pumps are not to be used when reactor power is greater than 5% unless being electrically rotated for Preventive Maintenance.

The Mechanical Vacuum Pump(s) may be electrically rotated for Preventive Maintenance if the suction valve(s) is closed and seal water in-service to prevent seizing. This requires the automatic trip to be defeated by a step text Work Order. [BFPER 00-003819-000] [BFPER 02-014849-000]

K. Charcoal bed re-alignment during power operation is prohibited. Any major change in off-gas flow will disturb bed equilibrium and result in a temporary (8 to 12 day) rise in stack discharge activity.

L. Charcoal bed prefilter and afterfilter differential pressure is not to exceed 10" H_2O. Switching to standby filters is recommended when filter differential pressure reaches 8" H_2O.

M. A Mechanical Vacuum Pump will auto trip under any of the following conditions:

- Hotwell pressure is equal to or below -26" HG
- Hotwell pressure is equal to or below -22" HG, with reactor pressure greater than or equal to 600 psig (vacuum pumps suction valves also auto close)
- Main Steam Line radiation is greater than or equal to 3 times normal background at full load (vacuum pumps suction valves also close)
- Seal water pump trips
- Undervoltage

N. During SJAE operation, steam supply pressure is to be maintained between 190 and 225 psig. Insufficient steam pressure will result in improper dilution of hydrogen. Excessive steam pressure causes water droplet carryover which reduces recombiner efficiency.

O. During operation above 25% power, the discharge of the SJAEs is to be routed through the charcoal adsorber.

P. Mechanical Vacuum Pumps will not start unless a Seal Water Pump is running and hotwell pressure is above -26" Hg.

INSTRUCTOR NOTES

(2) High-High Radiation

(a) MAIN STEAM LINE RAD HIGH-HIGH / INOP (55-3A-27) alarm at a radiation level of 3 times the Normal Full Power Background radiation level

(b) RAD HIGH-HIGH / INOP Alarm signal is generated by MSL Rad Recorder (RR-90-135)

(3) Downscale

(a) MAIN STEAM LINE DOWNSCALE (55-3A-14) alarms when low detector output is sensed

(b) During normal power operation this indicates instrument malfunction

(c) This alarm is expected during conditions of very low Main Steam flow

(d) DOWNSCALE Alarm signal is generated by NUMAC Log Radiation Monitor

e. Trip — Obj. V.B.1, Obj. V.C.1, Obj. V.D.2

(1) Trip level - MAIN STEAM LINE RAD HIGH-HIGH / INOP 3 times normal full power background radiation from monitor or detector INOP

(2) Closes condenser vacuum pump suction valves FCV-66-36 and 40 and trips condenser mechanical vacuum pump — Obj. V.B.4.b, Obj. V.C.4.b

Examination Outline Cross-reference:	Level	RO	SRO
286000 Fire Protection	Tier #	2	------
G2.2.42 (10CFR 55.41.10)	Group #	2	------
Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	K/A #	286000G2.2.42	
	Importance Rating	3.9	------

Proposed Question: **# 63**

Which ONE of the following completes the statement for the MIMIMUM requirements for High Pressure Fire Pumps in accordance with Fire Protection Report Volume 1?

The High Pressure Fire Protection System shall be operable at **ALL** times with ______ aligned to the fire suppression header.

A. ONE Diesel Fire Pump

B. ONE Electric Fire Pump

C. TWO Electric Fire Pumps

D. ONE Diesel Fire Pump **AND** ONE Electric Fire Pump

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that typical system pressure demand is such that one Fire Pump will operate in case of a fire in the buildings.

B INCORRECT**:** Plausible in that typical system pressure demand is such that one Fire Pump will operate in case of a fire in the buildings and Electric Fire Pump Power Supplies are from 4 kV Shutdown Boards.

C INCORRECT: Plausible in that two Fire Pumps are required to be operable and Electric Fire Pump Power Supplies are from 4 kV Shutdown Boards.

D **CORRECT**: Two High Pressure Fire Pumps shall be operable and aligned to the high pressure fire header as specified in the Fire Protection Report Volume 1.

Justification: Candidate must know how many High Pressure Fire Pumps are required by Fire Protection Report Volume 1 to correctly answer the question.

Technical Reference(s): OPL171.049 Rev. 15 (Attach if not previously provided)

Fire Protection Report Volume 1 Rev. 6

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.049 V.B.6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

INSTRUCTOR NOTES

(3) Fire Pump motors are powered by 4kV shutdown boards. — Pumps) Obj. V.D.2/V.B.8.d V.C.9.d

b. Two strainers are provided on the discharge of the fire pumps to increase reliability of system operation. — Obj. V.E.3 480V WATER SUP BD

0-PCV-26-4 on discharge header maintains header pressure at 175 psig. — See 0-SIMI-26B

c. Two High Pressure Fire Pumps shall be operable and aligned to the high pressure fire header as specified in the Fire Protection Report Volume 1. — 1 ELECTRIC 1 DIESEL

d. All pumps discharge into a yard main which encircles the entire plant.

(1) Branch lines from the yard main distribute water to all plant buildings and supply automatic spray systems for main and station service transformers located in the yard and the hydrogen trailer port.

(2) Hydrants are located throughout the plant yard area.

(3) Wye-hose connections and fire hose racks are strategically located throughout the plant buildings.

(4) Hose and nozzle connections are compatible with equipment used by the local fire department.

(5) Two check valves 0-26-1015 and 0-26-977 are installed in the system to prevent the backflow of water outside the protected area.

e. Fire pump discharge

(1) Discharge of the fire pumps is the alternate supply of bearing lube water to the Cooling Tower Lift Pumps.

Manual #: Fire Protection Report Vol. 1	PLANT: BFN	UNIT(s): 1/2/3	PAGE: 56 of 894
TITLE: Fire Protection Plan		SECTION: 1	REV: 6

9.3/9.4 FIRE PROTECTION SYSTEMS LIMITING CONDITION FOR OPERATING AND SURVEILLANCE REQUIREMENTS (continued)

9.3.11.B FIRE PUMPS AND WATER DISTRIBUTION MAINS	9.4.11.B FIRE PUMPS AND WATER DISTRIBUTION MAINS
1. The High-Pressure Fire Protection System shall be OPERABLE at all times with: a. Two high-pressure fire pumps, one electric and one diesel, each with a capacity of 2,250 gpm, with their discharges aligned to the fire suppression header. b. An OPERABLE flow path capable of taking suction from Wheeler Reservoir and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe and the last valve ahead of the system valve on each spray system required to be OPERABLE per requirements 9.3.11.C, 9.3.11.E, and 9.3.11.F.	1. The High-Pressure Fire Protection System shall be demonstrated OPERABLE: a. At least quarterly by starting each electric-motor-driven high-pressure fire pump and operating it for at least 15 minutes on recirculation flow. b. Intentionally left blank. c. At least semiannually by performance of a system flush. d. At least biannually, the system shall be chemically treated. e. At least yearly by cycling each testable valve in the flow path through at least one complete cycle of full travel.
2. The High-Pressure Fire Protection System shall be OPERABLE at all times: a. With only the diesel and no electric fire pumps OPERABLE, restore at least one electric fire pump to OPERABLE status within 7 days or provide an alternate backup pump or supply. b. With at least one electric fire pump and no diesel fire pump OPERABLE, restore the diesel fire pump to OPERABLE status within 7 days or provide an alternate backup pump or supply.	f. At least once per 18 months, by performing a system functional test which includes simulated actuation of the system throughout its operating sequence, and:

INSTRUCTOR NOTES

6. When one or more of the following initiation signals are present the fire pump will automatically start:

   Obj. V.D.10
   Obj. V.E.11
   0-45E644-1
   0-45W643
   This initial fire pump auto-start is based on a sensed fire condition; NOT on low pressure.

   a. Transformers, Shunt Reactor, or Unit 2 HPCI room temperature equal to or greater than 225°F

   b. Turbine Building protected areas or Unit 1/3 HPCI room temperature rise of 12°F/min

   c. Transformer of Shunt Reactor differential, overcurrent or sudden pressure relays actuated

   d. Any spray, sprinkler or fog system manual or automatic actuation

   NOTE: Typical system pressure demand is such that one Fire Pump will operate in case of a fire in the buildings, and two pumps will operate in case of a transformer fire.

7. Starting any Fire Pump automatically closes the RSW Head Tank isolation valves (FCV-25-70 and FCV-25-32).

8. Stopping all operating Fire Pumps automatically opens RSW Head Tank isolation valve FCV-25-32. The operator must reopen FCV-25-70 at Panel 1-9-20 or locally.

9. Fire Pump Selector switch, XS-26-43 on Control Room Panel 1-9-20, should <u>NOT</u> be left in the OFF position because this would prevent the automatic start of a Fire Pump.

   Obj.V.B.5/V.C.4
   Obj.V.D.9/V.E.10

10. Due to the anti-pumping feature of the electric Fire Pump circuit breakers, the first pump in the selected sequence will <u>not</u> restart if it is manually tripped with an automatic start signal present.

   2-45E765-7
   Obj.V.B.5/V.C.4
   Obj.V.D.9/V.E.10

11. If system header pressure is not >120 psig 15 seconds after the initiation signal, a second Fire Pump automatically starts.

12. If system header pressure is not >120 psig 30 seconds after the initiation signal, the third Fire Pump automatically starts.

	INSTRUCTOR NOTES
(3) Fire Pump motors are powered by 4kV shutdown boards.	Pumps) Obj. V.D.2/V.B.8.d V.C.9.d
b. Two strainers are provided on the discharge of the fire pumps to increase reliability of system operation.	Obj. V.E.3 480V WATER SUP BD
0-PCV-26-4 on discharge header maintains header pressure at 175 psig.	See 0-SIMI-26B
c. Two High Pressure Fire Pumps shall be operable and aligned to the high pressure fire header as specified in the Fire Protection Report Volume 1.	1 ELECTRIC 1 DIESEL

d. All pumps discharge into a yard main which encircles the entire plant.

(1) Branch lines from the yard main distribute water to all plant buildings and supply automatic spray systems for main and station service transformers located in the yard and the hydrogen trailer port.

(2) Hydrants are located throughout the plant yard area.

(3) Wye-hose connections and fire hose racks are strategically located throughout the plant buildings.

(4) Hose and nozzle connections are compatible with equipment used by the local fire department.

(5) Two check valves 0-26-1015 and 0-26-977 are installed in the system to prevent the backflow of water outside the protected area.

e. Fire pump discharge

(1) Discharge of the fire pumps is the alternate supply of bearing lube water to the Cooling Tower Lift Pumps.

Examination Outline Cross-reference:

290003 Control Room HVAC

K4.01 (10CFR 55.41.7)

Knowledge of CONTROL ROOM HVAC design feature(s) and/or interlocks which provide for the following:

- System initiations/reconfiguration: Plant-Specific

Level	RO	SRO
Tier #	2	------
Group #	2	------
K/A #	290003K4.01	
Importance Rating	3.1	------

Proposed Question: **# 64**

Which ONE of the following completes the statements?

When **BOTH** CREV trains are operable, the preferred position for CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, is in **__(1)__** .

If the **SELECTED** CREV Train AUTO-INITIATE / TEST switch is placed to the INITIATE / TEST position the initiation sequence starts **__(2)__**.

A. **(1)** TRAIN "A".
(2) immediately.

B. **(1)** TRAIN "B".
(2) immediately

C. **(1)** TRAIN "A".
(2) after 30 seconds.

D. **(1)** TRAIN "B".
(2) after 30 seconds.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 correct - When both CREV trains are operable, the preferred position for CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, is in TRAIN "A" which makes the "A" CREV the lead train. Part 2 correct - If the selected train's switch is put in the INITIATE/TEST position that train will immediately enter its initiation sequence,

B INCORRECT: Part 1 incorrect – Train "A" is Normally the Selected Train. Plausible in that either Train of CREV can be selected as Lead Train. Part 2 correct – See Explanation A.

C INCORRECT: Part 1 correct – See Explanation A. Part 2 incorrect - If the selected train's switch is put in the INITIATE/TEST position that train will immediately enter its initiation sequence, Plausible in that the STANDBY train initiation sequence is delayed 30 seconds if the train's switch is put in the INITIATE/TEST position.

D INCORRECT: Part 1 and 2 are incorrect as explained above.

Justification: Question requires plant-specific knowledge of Control Room Emergency Ventilation interlocks which provide for system initiations and reconfiguration to successfully answer.

Technical Reference(s): OPL171.067 Rev. 16 (Attach if not previously provided)

0-OI-31 Rev. 136

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 0	Control Bay and Off-Gas Treatment Building Air Conditioning System	0-OI-31 Rev. 0136 Page 21 of 285

3.6 CREV and CREV instrumentation operability issues (continued)

B. The main control room boundary may be opened intermittently under administration controls. For openings other than normal entry and exit, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the main control room and whose task is to close the opening when main control room isolation is indicated. With both CREVs inoperable in Modes 1, 2 or 3, for other than a control room boundary issue, enter LCO 3.0.3 Immediately. With two CREV subsystems inoperable during OPDRVs, initiate action to suspend OPDRVs. Reference TS 3.7.3.

C. When there is an automatic actuation of CREVS, the following automatic isolation dampers and hatch is required to be closed for CREVS to be considered operable.

1. 0-FCO-31-150B, 0-FCO-31-150D, 0-FCO-31-150E, 0-FCO-31-150F, 0-FCO-31-150G.

2. Removable equipment hatch in U-3 Mechanical Equipment Room, floor Elevation 617'.

D. The CREV system utilizes 15.45 kW Duct heaters to control moisture buildup in the charcoal adsorber. A malfunction of the 15.45 kW duct heater makes the applicable CREV unit inoperable. [Reference Functional Evaluation in PER 74959 and 75680]

E. One of the UNIT 1 & 2 Control Bay Supply Fans and one of the UNIT 3 Control Bay Supply Fans and their associated power and control circuits is required to be operable for CREVS instrumentation (control bay high radiation)to be considered operable. Reference Tech Spec 3.3.7.1.

F. CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 may be placed in either the “A” or “B” position, depending on the operability status of the CREV trains. When a CREV train is inoperable, it will NOT be selected as lead. When both CREV trains are operable, the preferred position for CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, is in TRAIN "A" which makes the "A" CREV the lead train. In the event that "A" CREV is INOP, CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 is required to be placed in the TRAIN "B" position so the “B” CREV will initiate, without a time delay, as the lead train.

G. When one of the CREV trains is inoperable for testing, the CREV UNIT PRIMARY SELECTOR SWITCH, 0-XSW-031-7214 is required to be aligned to the train which is NOT under testing conditions to ensure the non-test train will initiate under an actual initiation signal.

a. Manual initiation of the emergency mode of operation can be performed from the control room by operation of the AUTO-INITIATE/TEST switch (putting switch in INITIATE/TEST position) at Pnl 2-9-22. This operation results in energizing the CR1 relay for that train/division and isolation of the control room dampers. Only one solenoid must be energized to close these dampers. Therefore, either test switch will initiate a damper isolation.

Adherence to procedures
INPO SER 03-05
Obj.V.B.4/V.B.2

b. One switch alone in the INITIATE/TEST position will NOT result in full functionality of the CREVS units. If that train is not the selected unit, then operation of that train will be delayed by approx. 30 seconds, waiting for the selected train. This delay will result in the operator waiting to see the result of his operation of the switch. The operator will see the amber light lit, indicating energization of the CR 1 relay and the solenoid for isolation damper closing, but will see no activity of the CREVS unit until the delay timer has timed out.

Normally “A” is selected unit via switch in CREV room. If “A” is inoperable, switch 0-XSW-031-7214 SYSTEM PRIORITY SELECTOR SWITCH is placed in TRAIN B position to start it without time delay.

c. If the selected train’s switch is put in the INITIATE/TEST position that train will immediately enter its initiation sequence, with the damper’s red light being lit as well as the green, indicating travel of the damper toward the open position. However, should there be any failure of the selected unit; the standby unit will not start. This is because the CR1 relay for the standby unit was not actuated.

d. Therefore, when manually initiating emergency operation of the new CREVS units, it is important to put the AUTO-INITIATE/TEST switches of BOTH trains to the INITIATE/TEST position.

Examination Outline Cross-reference:	Level	RO	SRO
290002 Reactor Vessel Internals	Tier #	2	------
K1.12 (10CFR 55.41.6)	Group #	2	------
Knowledge of the physical connections and/or cause-effect relationships between REACTOR VESSEL INTERNALS and the following:	K/A #	290002K1.12	
• SLC	Importance Rating	3.4	------

Proposed Question: **# 65**

Which ONE of the following completes the statement?

The Standby Liquid Control (SLC) System Injection Sparger's __**(1)**__ provides for sodium pentaborate injection __**(2)**__ the Reactor Core Plate.

A. **(1)** inner tube
(2) above

B. **(1)** inner tube
(2) below

C. **(1)** outer tube
(2) above

D. **(1)** outer tube
(2) below

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: First part correct; but, SLC injects below the Core Plate as detailed below in 'B'.

B **CORRECT:** (See attached Lesson Plan excerpt) The **inner** tube penetrates the vessel inside the shroud near the core edge, terminating as a perforated tube **below** the core plate. It acts as the sodium pentaborate injection line.

C INCORRECT: The outer tube penetrates the vessel inside the shroud near the core edge, and terminates in the core plate. SLC injects below the Core Plate as detailed below in 'B'.

D INCORRECT: It is not the outer tube, as discussed in 'C' above. The second part of the statement is correct.

RO Level Justification: Tests whether the candidate has knowledge of the physical connections between Reactor Vessel Internals and SLC. Non-Licensed Operators are only required to know the function of the Injection Sparger itself, not broken down to the level of an inner/outer tube relationship or its relationship to Reactor Vessel Internals.

Technical Reference(s): OPL171.039, Rev. 16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.5.b (As available)

Question Source:

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:
55.41 **X**
55.43

Comments:

OPL171.039
Revision 16
Page 16 of 48

INSTRUCTOR NOTES

6. Explosive Valves	
a) Two 100% capacity explosive (Squib) valves, FCV 63-8A and B, are installed in parallel.	TP-3 Obj. V.B.3.a Obj. V.C.2.a Obj. V.D.3.f Obj. V.E.3.f Obj. V.B.5.f Obj. V.C.4.e Obj. V.D.4 Obj. V.E.4
b) Provide a zero leakage seal between the boron solution and the reactor.	
c) Each valve contains two firing primers, powered by the 250V DC control power from the 480V Shutdown Boards A and B, (unit specific).	
d) Either primer is capable of actuating the valve.	
e) The primer is fired by taking the main control room handswitch, HS-63-6A, to the START PUMP A or START PUMP B position. This forces the ram outward, which shears the end cap off the valve fitting, allowing flow to pass through the valve.	Obj. V.B.3.b Obj. V.C.2.b
f) After firing, the ram remains extended. This prevents the sheared cap from obstructing flow through the valve.	
g) The primer requires a minimum current of 2 amps to fire, and fires within 2 milliseconds after this circuit is applied. All the explosion by-products are retained in the trigger explosive chamber.	Obj. V.B.7
h) Each valves firing circuit continuity is monitored by a blue indicating light on Panel 9-5 and a current meter located in the back of Panel 9-5.	Obj. V.B.3.a Obj. V.C.2.a
7. Injection Sparger	
a) The tube-within-a-tube design piping of the SLC injection line allows the line to be used for injection by the SLC System and also provides input signals to various systems for both below and above core plate pressures.	Obj. V.B.3.e Obj. V.D.3.g Obj. V.E.3.g Obj. V.C.2.e TP-4
b) The inner tube penetrates the vessel inside the shroud near the core edge, terminating as a perforated tube below the core plate. It performs the following functions:	Obj. V.C.4.c Obj. V.B.5.b
(1) Acts as a sodium pentaborate injection line.	
(2) Supplies a below core plate pressure signal to the core differential pressure instrument.	

	INSTRUCTOR NOTES
(3) Supplies a below core plate pressure signal to the non-fully instrumented jet pump differential pressure instruments	
(4) Supplies a below core plate pressure signal to the jet pump total developed head instrument.	
(5) Supplies a below core plate pressure signal to the RHR and Core Spray initiation and injection valve logic.	
c) The outer tube penetrates the vessel inside the shroud near the core edge, and terminates in the core plate. It performs the following functions:	Obj. V.B.5.b Obj. V.C.4.c
(1) Supplies an above core plate pressure signal to the core differential pressure instrument.	
(2) Supplies an above core plate pressure signal to the Core Spray system line-break detection circuitry.	Obj. V.B.5.a Obj. V.C.4.a
(3) Supplies an above core plate pressure signal to the CRD System for drive and cooling water differential pressure instruments.	
8. Isolation Valves and Piping	
a) Test Tank Isolation valve (HCV-63-14)	
(1) Normally locked closed.	
(2) It has a green indicating light on panel 9-5 in the Main Control Room (MCR).	
(3) Located on Reactor Bldg Elev 639, north.	
b) Drain Tank Isolation valve (HCV-63-13)	
(1) Normally locked closed.	
(2) Green indicating light on MCR Panel 9-5.	
(3) Located on the north wall, Reactor Bldg Elev 621.	
c) Inside Containment System Isolation Valve (HCV-63-12)	
(1) Allows isolation for system maintenance.	
(2) Normally locked open.	
(3) Red indicating light on MCR Panel 9-5.	
(4) Located as close as possible to the reactor vessel, on Elev 584 in the drywell.	

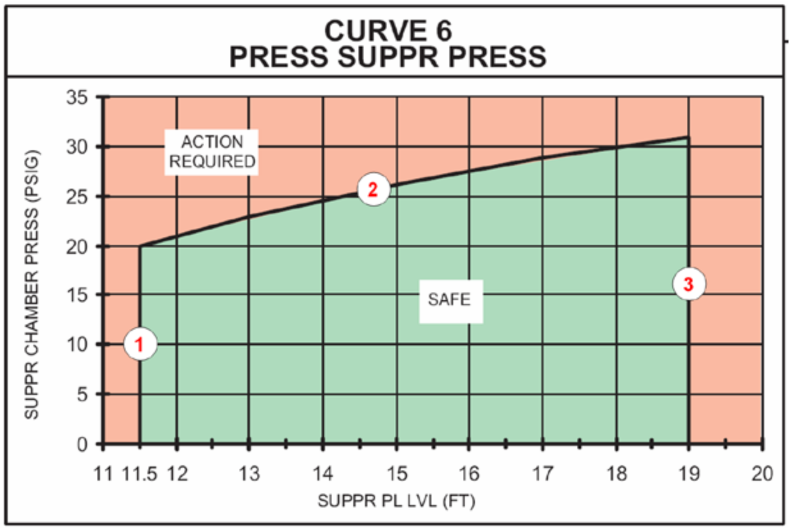
Examination Outline Cross-reference:

G2.1.25 (10CFR 55.41.10)

Ability to interpret reference materials, such as graphs, curves, tables, etc.

	RO	SRO
Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.1.25	
Importance Rating	3.9	------

Proposed Question: **# 66**



Which ONE of the following completes the statement?

In accordance with the EOI Program Manual derivation, Line 1 on Curve 6, "Pressure Suppression Pressure," above, corresponds to the Suppression Pool Water Level at which the ________.

A. Downcomer Vents become uncovered.

B. HPCI Turbine Exhaust opening becomes uncovered.

C. Safety Relief Valve (SRV) Tailpipe openings become uncovered.

D. Control Room Suppression Pool Water Narrow Range Level Indication goes off scale low.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (See attached excerpt) According to the EOI Program Manual, 11.5 feet (or Line 4) is the Suppression Pool Water Level which corresponds to the elevation of the downcomer vent openings.

B INCORRECT: The HPCI Turbine Exhaust becomes uncovered in the range of but above this value (at 12.75 feet) and is a significant direct Suppression Chamber Air Space pressurization event if HPCI remains running. PSP would be quickly exceeded.

C INCORRECT: SRV Tailpipes become uncovered around 5.5 feet. This is plausible because of the required ED at 11.5 feet. Normally, an ED on a parameter such as this is accomplished before you lose the ability to do so safely (within Safety Analyses assumptions).

D INCORRECT: Plausible because the X-Axis is based upon Suppression Pool Water Level and Narrow Range goes off-scale low at -25 inches which corresponds to approximately 13 feet.

RO Level Justification: Tests the candidate's ability to interpret Pressure Suppression Pressure Curve bounding limitations on Suppression Chamber Pressure versus Suppression Pool Level.

Technical Reference(s): OPL171.201, Rev. 7 (Attach if not previously provided)
EOI Program Manual Sect. 2-VI-H, Rev. 9
0-TI-394, Rev. 4

Proposed references to be provided to applicants during examination: Embedded EOI Curve 6 - PSP

Learning Objective: V.B.12 (As available)

Question Source: Bank # X
Modified Bank # Hatch 07 NRC Q#67 (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

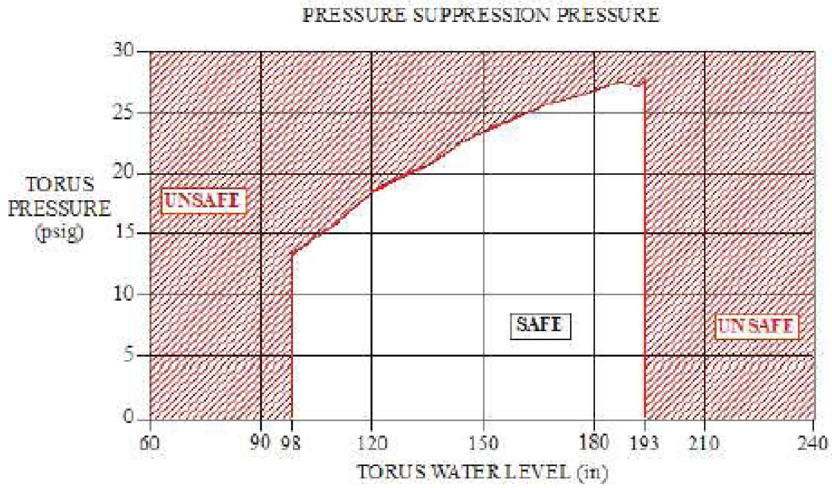
Question Cognitive Level: Memory or Fundamental Knowledge **X**
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments: This question was originally developed for an Audit Exam.

BANK QUESTION THAT WAS MODIFIED…

67. G2.1.25 001



Which ONE of the following components corresponds to the torus water level limit of 193" in graph 7, Pressure Suppression Pressure?

A. Top of the torus-to-drywell vacuum breakers

B✓ Bottom of torus ring header

C. Control room torus water level indicator is at the top of the band

D. Control room torus pressure instrument tap becomes covered

A. Incorrect because torus-to-drywell vacuum breakers are submerged at 197.5 ". Plausible because these vacuum breakers are in the suppression pool range being considered.

B. Correct.

C. Incorrect because the highest control room torus level instrument indication is 300." Plausible because the x-axis deals with torus level.

D. Incorrect because the level @ which the torus pressure instrument tap is covered is 40 feet . Plausible since the y-axis deals with torus pressure.

5.0 CALCULATIONS

The derivation of the PSP is shown graphically in Figure 2. Line 1 corresponds to the highest suppression chamber pressure which can occur without steam in the suppression chamber airspace. This pressure is determined by calculating the pressure that would exist as a function of suppression pool water level with all drywell noncondensibles purged to the suppression chamber and suppression pool temperature at the Heat Capacity Temperature Limit corresponding to the lowest SRV lift pressure. Higher suppression pool water levels result in higher pressures since the airspace volume is smaller.

Line 2 corresponds to the highest suppression chamber pressure from which an emergency depressurization will not raise suppression chamber pressure above Primary Containment Pressure Limit A before RPV pressure drops to the Minimum RPV Flooding Pressure. This curve is calculated by subtracting the rise in suppression chamber pressure during blowdown from Primary Containment Pressure Limit A. The calculation assumes the blowdown is initiated at the lowest SRV lift pressure and compensates for changes in suppression pool heat capacity with changes in suppression pool water level (as defined by the Heat Capacity Temperature Limit). As suppression pool water level increases, a larger heat sink is available to absorb blowdown energy. Consequently, the difference in suppression pool temperature before and after the blowdown decreases, causing the rise in suppression chamber pressure to decrease. Since Primary Containment Pressure Limit A is constant in this range, Line 2 rises with increasing suppression pool water level.

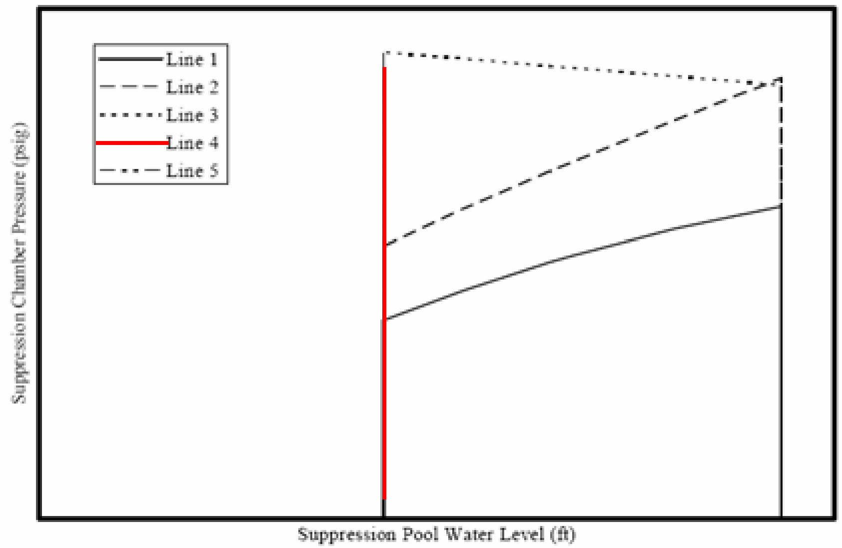
Line 3 corresponds to the highest suppression chamber pressure at which SRVs can be opened without exceeding the suppression pool boundary design load. This curve is the suppression pool boundary design pressure less (1) the suppression pool boundary loads imposed by SRV actuation and (2) the hydrostatic head between the suppression pool water level and the level assumed in the design calculation.

Line 4 is the suppression pool water level corresponding to the elevation of the downcomer vent openings. If suppression pool water level is below this elevation, the RPV may not be kept in a pressurized state since steam discharged through the vents may not be condensed. The PSP is therefore vertical at this elevation.

Line 5 is the suppression pool water level corresponding to the Maximum Pressure Suppression Primary Containment Water Level. Above this elevation, the pressure suppression function of the containment cannot be assured. The PSP is therefore vertical at this elevation.

The PSP is thus the envelope defined by Lines 4 and 5 and the most limiting values of Lines 1, 2, and 3. As shown in Figure 2, Line 1 is most limiting over the range of

Figure 2 – PSP Derivation



		INSTRUCTOR NOTES
	OR	
•	That initial suppression chamber pressure which, if RPV depressurization was initiated and allowed to continue until RPV pressure reaches the Minimum RPV Flooding Pressure (90/80/70 psig), would cause suppression chamber pressure to reach the Primary Containment Pressure Limit. This initial allowed pressure decreases with increasing suppression pool level due to the larger heat sink available.	not limiting
	OR	
•	That suppression chamber pressure which can be maintained without exceeding the suppression pool boundary design load if SRVs are opened. This pressure decreases with increasing suppression pool level.	not limiting
a.	The purpose of the Pressure Suppression Pressure Curve is to determine if the pressure suppression capability has been degraded and to preclude containment failure due to exceeding design loads and the primary containment pressure limit.	
b.	The Pressure Suppression Pressure Curve is comprised of three segments:	
c.	**Segment A-B**	
	(1) At suppression pool levels below the suppression chamber downcomer openings (11.5 ft), the pressure suppression function of the suppression chamber cannot be assured. Any steam produced as a result of a leak or break would be directed through the downcomers and pressurize the suppression chamber directly. If suppression pool level is found to be at this level, Emergency RPV depressurization is initiated.	

Browns Ferry Nuclear Plant

Unit 0

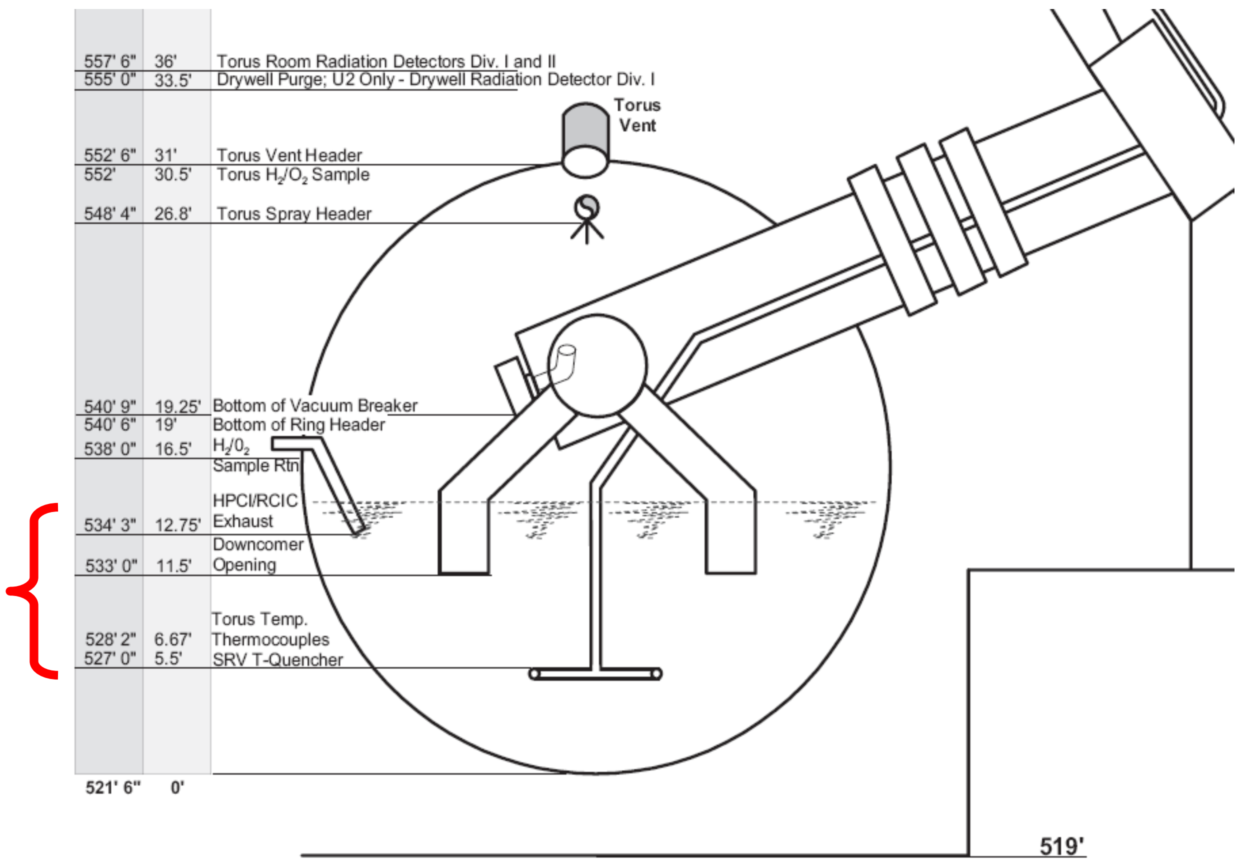
Technical Instruction

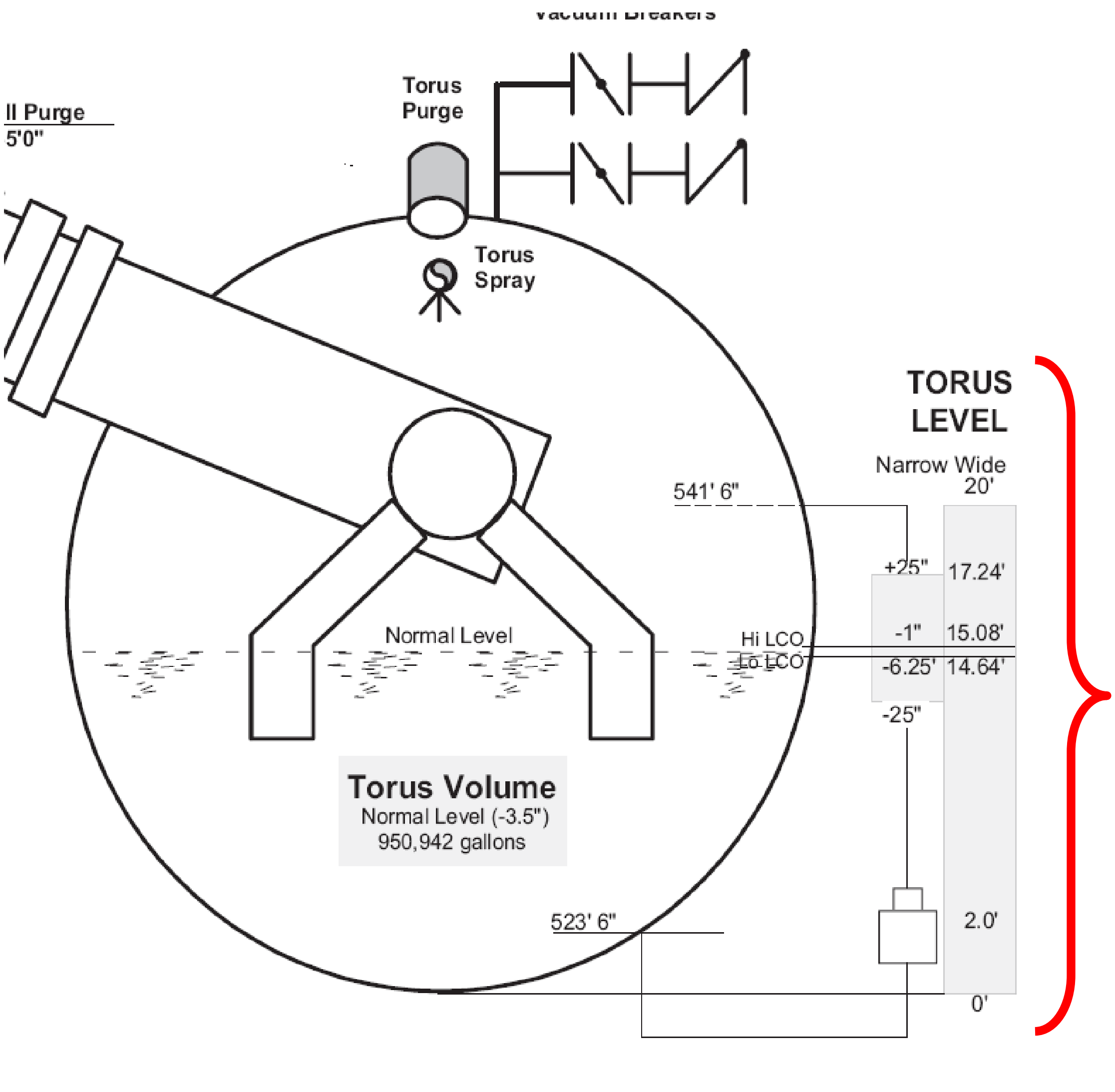
0-TI-394

Technical Support for Severe Accident Management Guidelines (SAMG)

Revision 0004

ILLUSTRATION 1 EXCERPT





Vacuum Breakers
ll Purge
5'0"
Torus Purge
Torus Spray
TORUS LEVEL
Narrow Wide
20'
541' 6"
+25"
17.24'
Normal Level
Hi LCO
Lo LCO
-1"
15.08'
-6.25'
14.64'
-25"
Torus Volume
Normal Level (-3.5")
950,942 gallons
523' 6"
2.0'
0'

Examination Outline Cross-reference:

G2.1.27 (10CFR 55.41.7)
Knowledge of system purpose and/or function.

	RO	SRO
Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.1.27	
Importance Rating	3.9	------

Proposed Question: **# 67**

Which ONE of the following completes the statement?

In accordance with 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) For Light-Water Nuclear Power Reactors," all of the following are design functions of Browns Ferry ECCS with the **EXCEPTION** of ________.

A. maintaining peak cladding temperature less than or equal to 2600 ºF.

B. maintaining core geometry such that the core remains amenable to cooling.

C. minimizing total cladding oxidation to less than or equal to 17% of the total cladding thickness prior to oxidation.

D. minimizing total hydrogen generation to less than or equal to 1% of the hypothetical amount possible if all of the cladding were to react chemically with water or steam.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** The calculated maximum fuel element temperature specified in 10CFR50.46 is 2200 ºF versus 2600 ºF, which is the melting point of Stainless Steel according to 0-TI-394.

B INCORRECT: This is a specified variable in 10CFR50.46.

C INCORRECT: This is a specified variable in 10CFR50.46.

D INCORRECT: This is a specified variable in 10CFR50.46.

RO Level Justification: Tests the candidate's ability to recall the specified criteria of 10CFR50.46 as it applies to Browns Ferry ECCS. Compounding the level of difficulty is the addition of wording (above and beyond what is normally memorized). For example, competent candidates should normally remember "cladding oxidation less than 17%;" as this is the way it is normally discussed; but, may not specifically recall in relation to what.

Technical Reference(s): 10CFR50.46 (Attach if not previously provided)

0-TI-394, Rev. 04

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.1 (OPL171.042 / 044 / 045) (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

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§ 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.

(b)(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200º F.

(2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

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Illustration 5
(Page 2 of 2)

SAMG Thumb Rules

DECAY POWER AFTER SHUTDOWN

Time (min.)	Decay Power (%)	Injection Rate to Maintain Constant Level * (gpm)
2	3.128	657
4	2.732	574
6	2.525	530
8	2.383	500
10	2.270	477
15	2.064	433
20	1.919	403
25	1.806	379
30	1.717	361
35	1.638	344
40	1.572	330
45	1.516	318
50	1.465	308
60	1.383	290
90	1.222	257
120	1.121	235
150	1.052	221
200	0.973	204
300	0.874	184
450	0.786	165
600	0.729	153
900	0.655	138
1200	0.607	127
1500	0.573	120
1800	0.547	115

* Injection Rate To Maintain Constant Level

210 gpm injection is roughly equivalent to 1% core power (Assumes Constant pressure, and injection temperature approximately 100 °F).

Melting Temperatures for Reactor Materials

Temperature (°F)	Condition
600	Design Temperature for Incore Detectors
1500	Fuel Rod Perforation Begins
1800	Significant Metal Water Reaction Begins
2600	Stainless Steel Melting point
3365	Zircaloy-4 Melting Point
4400	B_4C Melting Point
5145	UO_2 Melting Point

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.8 (10CFR 55.41.10)	Tier #	3	------
Ability to coordinate personnel activities outside the control room.	Group #	------	------
	K/A #	G2.1.8	
	Importance Rating	3.4	------

Proposed Question: **# 68**

In accordance with 2-OI-68, “Reactor Recirculation System,” which ONE of the following describes the **MINIMUM** qualification **AND** coordination requirements to perform LOCAL VFD Speed Control manipulations for the 2A Reactor Recirc Pump?

A. A Licensed Reactor Operator in communication with the Main Control Room may perform LOCAL VFD Speed Control manipulations.

B. A Licensed Reactor Operator may perform LOCAL VFD Speed Control manipulations with a Senior Reactor Operator supervising locally.

C. A qualified Assistant Unit Operator may perform LOCAL VFD Speed Control manipulations with a Licensed Reactor Operator supervising.

D. A qualified Assistant Unit Operator in communication with the Main Control Room may perform LOCAL VFD Speed Control manipulations.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** A Licensed Reactor Operator is the minimum qualification, with Main Control Room communications established.

B INCORRECT: Plausible in that it would be true if communications were established with the Main Control Room.

C INCORRECT: Plausible in that it would be true if communications were established with the Main Control Room and the AUO were enrolled in a License Training Program.

D INCORRECT: Plausible in that it would be true if AUO were enrolled in a License Training Program and supervised by a Licensed Operator.

Justification: Requires knowledge of MINIMUM qualification AND coordination requirements to perform LOCAL VFD Speed Control manipulations for Reactor Recirc Pumps.

Technical Reference(s): 2-OI-68 Rev. 134 (Attach if not previously provided)

OPDP-1 Rev. 15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.071V.B.6 (As available)

Question Source:

Bank # Duane Arnold 2007 #66

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Duane Arnold 2007

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

BFN Unit 2	Reactor Recirculation System	2-OI-68 Rev. 0134 Page 118 of 182

8.10 Local VFD Speed Control

CAUTION

Keying a Radio while the VFD control cabinet door is open has caused VFD trips.

[1] **ESTABLISH** communication between a licensed operator at the recirc drive and the Unit Operator in the control room. (A licensed Operator will <u>not</u> be required if the recirc pump is running uncoupled.) □

[2] **PERFORM** the following for local operation of Recirc Drive 2A:

[2.1] **VERIFY** in LOCAL, VFD 2A SPEED LOCAL/REMOTE XFER SW, 2-XS-068-2002. □

[2.2] **DEPRESS** MANUAL using the key pad located inside the Recirc Drive Cooling Cabinet Control Cabinet, START key □

[2.3] **DEPRESS** and **HOLD** "SHIFT" using the key pad located inside the Recirc Drive Cooling Cabinet Control Cabinet, **THEN**:

[2.3.1] **DEPRESS** and **RELEASE** "MAIN/5" key. □

[2.3.2] **RELEASE** the "SHIFT" key. (the MAIN menu should be displayed) □

[2.4] **IF** it is desired to raise recirc pump speed **THEN** (Otherwise N/A)

DEPRESS the up arrow using the key pad located inside the Recirc Drive Cooling Cabinet Control Cabinet, and **NOTE** a rise in the demand "DEMD" as displayed in the display window on the key pad. □

[2.5] **IF** it is desired to lower recirc pump speed **THEN** (Otherwise N/A)

DEPRESS the down arrow using the key pad located inside the Recirc Drive Cooling Cabinet Control Cabinet, and **NOTE** a lowering of the demand "DEMD" signal as displayed in the display window on the key pad. □

NPG Standard Department Procedure	Conduct of Operations	OPDP-1 Rev. 0015 Page 13 of 72

3.6 Reactivity Management (continued)

8. Ensure a member of Operations Management Staff, not assigned to the operating crew, shall be present in the control room during infrequent evolutions such as reactor and plant start-up, shutdown, and other significant changes to core reactivity.

B. The Unit Supervisor is responsible for all manipulations that affect reactivity and is charged to:

1. Promulgate and enforce the standards and expectations associated with reactivity management.

2. Give permission to Unit Operators to make reactivity changes. Personally oversee all reactivity changes or assign another SRO to oversee the reactivity change if unable to give his/her undivided attention. During periods of frequent reactivity manipulations or significant plant evolutions (such as startup or shutdown), another SRO may be assigned to perform the reactivity management oversight function

3. Utilize approved reactivity plans or notify the shift Reactor Engineer of unplanned reactivity changes greater than 1% thermal power.

4. Review and approve all planned reactivity changes or core alterations in accordance with approved procedures or instructions developed by Reactor Engineering. Ensure Reactor Engineering is available in the control room during planned significant reactivity evolutions.

5. Ensures that pre-job briefs for work activities address potential reactivity effects. Personnel involved in reactivity manipulations or working on reactivity control equipment must be properly trained, must understand their roles and responsibilities, and must be briefed on management expectations. (Refer to Attachment 5, Pre-Job Briefing Guidelines.)

C. Unit Operators are charged to:

1. Monitor reactor parameters to ensure the unit is operating within prescribed bands and monitor prescribed parameters and instrumentation to verify plant response is as expected during reactivity manipulations. If the unit is determined to be operating above its licensed core thermal power limit take prompt (typically no more than 10 minutes from time of determination) action to reduce power below the core thermal power limit.

2. Take conservative action, including manual scram/reactor trip, when abnormal reactor conditions are encountered, and does not rely solely on the Reactor Protection System to protect the reactor during reactivity events.

3. Monitors nuclear instrumentation during refueling activities that could affect the reactivity of the core so that abnormal reactivity events can be mitigated.

3.6 Reactivity Management (continued)

4. Directly supervise trainees manipulating reactivity related controls, as if the Unit Operator were performing the manipulation personally. The trainees must be enrolled in an approved licensed training program.

5. Knows and monitors the effects of the reactivity change. Make positive reactivity changes by only one method at a time.

6. Understands and compares reactivity management plan and actual plant performance during core maneuvers (BWR).

7. Stops and questions unexpected situations involving reactivity, criticality, power level, or core anomalies. Meets anomalous indication with conservative action.

3.7 Radiological Safety

A. The radiological protection requirements established within the TVA NPG for conduct in plant radiation areas are developed and implemented to protect all radiation workers from the harmful effects of radiation. As one of three safety focus areas, operators lead their station in following and enforcing the procedures and standards that have been established.

1. All Operations personnel are expected to demonstrate the following fundamental behaviors when working in and around radiologically protected areas:

   - Adhering to all radiological posting requirements
   - Using proper practices and precautions when working in a radiologically posted area
   - Being aware of and taking actions to reduce personal exposure through the use of fundamental concepts of time, distance and shielding
   - Knowing the requirements of the general or specific RWP controlling the work activity
   - Promptly reporting identified radiological hazards to Radiation Protection and the control room
   - Informing Radiation Protection prior to evolutions which have the potential to change radiological conditions in the plant
   - Coaching and correcting observed deviations in personnel radiological work practices
   - Maintaining overall awareness of normal plant radiological conditions
   - Minimizing waste generation by limiting packaging in the RCA. Controlling radioactive material and ensuring all material is surveyed before it leaves the RCA
   - Maintaining up-to-date awareness of personnel exposures
   - Ensuring that pre-job briefs discuss radiological conditions, planned exposure and specific actions to minimize dose and avoid contamination

RO 66	K/A Number Generic	Statement 2.1.8	IR 3.8	Origin N	Source Question NA
LOK F	10CFR55.41(b) 10	LOD (1-5)	Reference Documents OI-264 Rev 99		
Ability to coordinate personnel activities outside the control room.					

Per OI 264, Reactor Recirculation System, which ONE of the following describes the MINIMUM qualification and coordination requirements for local operation of a Recirculation MG Set Scoop Tube?

a. A qualified Nuclear Station Plant Equipment Operator may perform scoop tube position adjustment with a Licensed Reactor Operator supervising at the Recirculation MG Set.

b. A qualified Nuclear Station Plant Equipment Operator in communication with the Main Control Room may perform scoop tube position adjustment

c. A Licensed Reactor Operator may perform scoop tube position adjustment with a Senior Reactor Operator supervising at the Recirculation MG Set.

d. A Licensed Reactor Operator in communication with the Main Control Room may perform scoop tube position adjustment.

Correct Answer: D a Licensed Reactor Operator is the minimum qualification, with Main Control Room communications established.
Plausible Distractors: A is plausible: would be true if communications were established and the NSPEO were enrolled in a License Training Program. B is plausible: would be true if supervised by a Licensed Operator and the NSPEO were enrolled in a License Training Program. C is plausible: would be true if communications were established.
Objective Link: None

Examination Outline Cross-reference:

G2.2.37 (10CFR 55.41.7)
Ability to determine operability and/or availability of safety related equipment.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.37	
Importance Rating	3.6	------

Proposed Question: **# 69**

Which ONE of the following completes the statement?

In accordance with Unit 2 Tech Spec 3.4.1,"Recirculation Loops Operating," Recirculation Loop Jet Pump flow mismatch with **BOTH** Recirculation Loops in operation must be **LESS THAN OR EQUAL TO __(1)__** of rated core flow when operating at **LESS THAN** 70% rated core **__(2)__**.

A. **(1)** 5%
(2) flow.

B. **(1)** 10%
(2) flow.

C. **(1)** 5%
(2) power.

D. **(1)** 10%
(2) power.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – See Explanation C. Part 2 correct – See Explanation B.

B **CORRECT**: Part 1 and 2 correct - In accordance with Unit 2 Tech Spec 3.4.1,"Recirculation Loops Operating," SR 3.4.1.1, limit for recirculation loop jet pump flow mismatch with both recirculation loops in operation is ≤ 10% of rated core flow when operating at < 70% of rated core flow.

C INCORRECT: Part 1 incorrect – Plausible in that this is flow mismatch limit for > 70% rated core flow Part 2 incorrect – Plausible in that it is easily confused core flow versus power for mismatch criteria.

D INCORRECT: Part 1 correct – See Explanation B. Part 2 incorrect – See Explanation C.

RO Level Justification: To determine operability of Recirc, candidate must know Jet Pump Flow mismatch criteria.

Technical Reference(s): U2 TS 3.4.1 Amm 258 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

Question Source:

Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------------NOTE------------------------ Not required to be performed until 24 hours after both recirculation loops are in operation. --- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours

Examination Outline Cross-reference:

G2.2.22 (10CFR 55.41.5)

Knowledge of limiting conditions for operations and safety limits.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.2.22	
Importance Rating	4.0	------

Proposed Question: **# 70**

Which ONE of the following combinations of Reactor Power **AND** Reactor Pressure on Unit 1 constitute a Safety Limit violation?

	Reactor Power	Reactor Pressure
A.	15%	750 psig
B.	24%	770 psig
C.	28%	775 psig
D.	32%	810 psig

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: If Reactor Pressure greater than 785 psig, this would be a correct answer.

B INCORRECT: If Reactor Pressure greater than 785 psig, this would be a correct answer.

C **CORRECT:** With the reactor steam dome pressure < 785 psig or core flow< 10% rated core flow, THERMAL POWER shall be ≤ 25% RTP.

D INCORRECT: If Reactor Power was less than 25%, this would be a correct answer.

RO Level Justification: Tests the candidate's knowledge of safety limits.

Technical Reference(s): Unit 1 Tech Specs, Sect. 2.0, Am. 267 (Attach if not previously provided)
OPL171.087, Rev. 8

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.087 V.B.14 (As available)

Question Source: Bank # FITZ 08 #70
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Fitzpatrick 2008

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

THERMAL POWER shall be ≤ 25% RTP.

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

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

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

OPL171.087
Revision ~~7~~8
Page 4 of 5

a. Section 1.4 Frequency

Frequency specifies how often a surveillance requirement (SR) must be performed or met for continued compliance with the LCO. When “met” is specified, the SR need not be performed or current if there is no reason to suspect its acceptance criteria would not be met if it were performed. The examples specify all the special circumstances applied to frequencies and should be reviewed.

Review the section 1.4 examples

1. Section 2.0 SAFETY LIMITS

a. This section contains the requirements for meeting the four Safety Limits. The first two requirements ensure the Minimum Critical Power Ratio (MCPR) is not exceeded. The third ensures adequate core cooling. The last ensures the integrity of the vessel and associated piping.

Section 2.0 does not exist in TRM;

Obj. V.B.15

b. For all 3 units

1) Reactor steam dome pressure shall be $\leq$ 1325 psig.

2) Thermal power shall be $\leq$ 25% when steam dome pressure is < 785 psig OR core flow is < 10% of rated

3) Reactor water level shall be > Top of Active Fuel (TAF)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	G	
	Group #		
	K/A #	G 2.2.22	
	Importance Rating	3.4	

G. 2.2.22 Knowledge of limiting conditions for operations and safety limits (CFR: 43.2 / 45.2)

Proposed Question: # 70

Determine which of the following combinations of reactor power and reactor pressure constitute a Safety Limit violation.

	Reactor power	Reactor pressure
A.	15% RTP	750 psig
B.	24% RTP	770 psig
C.	28% RTP	775 psig
D.	32% RTP	810 psig

Proposed Answer: C

Explanation (Optional):

Technical Reference(s): __________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: __________

Learning Objective: __________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.4 (10CFR 55.41.7)	Tier #	3	------
(multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.	Group #	------	------
	K/A #	G2.2.4	
	Importance Rating	3.6	------

Proposed Question: **# 71**

Which ONE of the following completes the statement?

In accordance with 2/3-OI-3, "Reactor Feedwater System," the **MAXIMUM** speed for Unit 2 Reactor Feed Pump Turbines (RFPTs) is **__(1)__** **AND** the **MAXIMUM** speed for Unit 3 RFPTs is **__(2)__** .

A. **(1)** 5050 rpm
(2) 5050 rpm

B. **(1)** 5050 rpm
(2) 5850 rpm

C. **(1)** 5850 rpm
(2) 5050 rpm

D. **(1)** 5850 rpm
(2) 5850 rpm

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect –The MAXIMUM speed for Unit 2 RFPTs is 5850 rpm. Part 2 correct – MAXIMUM speed for Unit 3 RFPT is 5050 rpm. Combination of both RFPTs having same MAX speed is plausible in that Unit 1 and 2 have the same MAX speed.

B INCORRECT: Part 1 incorrect - The MAXIMUM speed for Unit 2 RFPTs is 5850 rpm. Part 2 incorrect - MAXIMUM speed for Unit 3 RFPTs is 5050 rpm.

C **CORRECT**: Part 1 correct - The MAXIMUM speed for Unit 2 RFPTs is 5850 rpm. Part 2 correct - MAXIMUM speed for Unit 3 RFPTs is 5050 rpm.

D INCORRECT: Part 1 is correct and Part 2 is incorrect as explained above.

Justification: Candidate must recognize Unit 2/3 differences for RFPT MAXIMUM speeds to correctly answer the question.

Technical Reference(s): 2-OI-3 Rev. 135 (Attach if not previously provided)
3-OI-3 Rev. 81

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

____

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

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

BFN Unit 2	Reactor Feedwater System	2-OI-3 Rev. 0135 Page 15 of 216

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Z. Startup bypass valve, 2-LCV-3-53, is air to open, spring to close. The valve fails as is on loss of air (65 psig). On a loss of power or loss of control signal, 2-LCV-3-53 fails closed.

AA. RFP 2A(2B)(2C) SUCTION VALVE, 2-FCV-2-83(95)(108), will not open if bearing oil pressure is low (less than 4 psig).

BB. Total Feedwater flow provides inputs to Feedwater Level Control System, Rod Worth Minimizer, Recirc Flow Control System, Feedwater Integrator, and Feedwater flow recorder.

CC. RX WATER LEVEL NARROW RANGE instruments, 2-LI-3-208A(D) on Panel 2-9-5 and 2-LI-3-208B(C) on Panel 2-9-3 should be monitored closely during reactor level transients. These instruments provide inputs to Main Turbine, Reactor Feedpump Turbine, HPCI, and RCIC high water level trip logic.

DD. [TSAR] Operation of Reactor Feed Pump Turbines with inoperable high level trip instrumentation may result in equipment damage or personnel hazard. [Item D-91]

EE. It is acceptable to Shut Down Vapor Extractor for up to 24 hours since this should not cause any detrimental effect to the system. If the Extractor needs to be out longer than 24 hours, an evaluation of the effects on the system should be done by System Engineering. [SEOPR 96-2/3 047-003]

FF. For operating Feed Pumps, check the following parameters and maintain them within the ranges described below.

1. RFPT Hydraulic Pressure: ≈ 200 psig (local indication).
2. Lube Oil Pressure to RFP Bearings: ≈ 15 psig (local indication).
3. Lube Oil Pressure to RFPT Bearings: ≈ 10 psig (local indication).
4. Bearing lube oil from cooler: 110°F to 120°F (obtained from Process Computer Point Id's 24-56, 24-54, and 24-52).
5. Bearing lube oil to cooler: 180°F maximum (obtained from Process Computer Point Id's TBD025, TBD032, and TBD039).
6. Maximum Oil Temp Rise across the Turbine Bearings: 50°F.
7. Vertical Vibration at RFP Bearing Supports: 2 mils double amplitude.
8. RFPT Speed: 5850 rpm maximum (Panel 2-9-6).

BFN Unit 3	Reactor Feedwater System	3-OI-3 Rev. 0081 Page 16 of 238

3.0 PRECAUTIONS AND LIMITATIONS (continued)

FF. For operating Feed Pumps, monitor and maintain the following parameters within ranges described below.

1. RFPT Hydraulic Pressure: ≈ 200 psig (local indication).
2. Lube Oil Pressure to RFP Bearings: ≈ 15 psig (local indication).
3. Lube Oil Pressure to RFPT Bearings: ≈ 10 psig (local indication).
4. Bearing lube oil from cooler: 110°F to 120°F (obtained from Process Computer Point Id's 24-56, 24-54, and 24-52).
5. Bearing lube oil to cooler: 180°F maximum (obtained from Process Computer Point Id's TBD025, TBD032, and TBD039).
6. Maximum Oil Temp Rise across the Turbine Bearings: 50°F.
7. Vertical Vibration at RFP Bearing Supports: 2 mils double amplitude.
8. RFPT Speed: 5050 rpm maximum (3-9-6).

GG. New Flow Control Valve, 3-FCV-3-53 Start up Bypass Valve, has a hand wheel associated with it which acts as a local locking device (Dogging device). With hand wheel all the way closed, valve will respond normally from the output air signal. When hand wheel is fully in open direction, then valve is locked in open position. This is a unique acting valve and close attention to detail is required when hand wheel is manipulated. This hand wheel is not for locking valve closed.

HH. Maintenance will be required to provide documentation for all leads lifted and re-landed in this procedure.

II. Attempting to reset a tripped turbine when speed is greater than 5000 RPMs may result in damage to the trip plunger.

JJ. The critical Steps warning represents a step or series of steps which require additional focus, attention, and increased awareness. The Operator performing these steps for the activity needs to ensure the Unit Supervisor and other Control Room staff are aware of the evolution. PEER checks are required for this activity and short briefs need to be made prior to performing the evolution. Included in the briefs are worst case scenario and contingencies. Examples are transition from Feed pumps to Condensate System for vessel inventory control, and ensuring the heaters are charged prior to opening the feedpump discharge valve when placing first feedpump in service.

BFN Unit 1	Reactor Feedwater System	1-OI-3 Rev. 0018 Page 14 of 216

3.0 PRECAUTIONS AND LIMITATIONS (continued)

Y. The RFW START-UP LCV, 1-LCV-003-0053, is air to open, spring to close. The valve fails as is on a loss of air (65 psig). On a loss of power or loss of control signal, RFW START-UP LCV, 1-LCV-003-0053 fails closed.

Z. RFP 1A(1B)(1C) SUCTION VALVE, 1-FCV-002-0083(0095)(0108), will NOT open if bearing oil pressure is low (less than 4 psig).

AA. Total Feedwater flow provides inputs to the Feedwater Level Control System, Rod Worth Minimizer, Recirc Flow Control System, Feedwater Integrator, and The Feedwater flow recorder.

BB. LEVEL (A)(D), 1-LI-3-208A(D) on Panel 1-9-5 and RX WATER LEVEL NARROW RANGE, 1-LI-3-208B(C) on Panel 1-9-3 should be monitored closely during reactor level transients. These instruments provide inputs to the Main Turbine, Reactor Feedpump Turbine, HPCI, and RCIC high water level trip logic.

CC. [TSAR] Operation of the Reactor Feed Pump Turbines with inoperable high level trip instrumentation may result in equipment damage or a personnel hazard. [Item D-91]

DD. It is acceptable to shut down the Vapor Extractor for up to 24 hours since this should NOT cause any detrimental effect to the system. If the Extractor needs to be out longer than 24 hours, then an evaluation of the effects on the system should be done by System Engineering. [SEOPR 96-2/3 047-003]

EE. For the operating Feed Pumps the following parameters should be within the ranges described below.

1. RFPT Hydraulic Pressure: ≈ 200 psig (local indication).
2. Lube Oil Pressure to RFP Bearings: ≈ 15 psig (local indication).
3. Lube Oil Pressure to RFPT Bearings: ≈ 10 psig (local indication).
4. Bearing lube oil from cooler: 110°F to 120°F (obtained from Process Computer Point IDs 24-56, 24-54, and 24-52).
5. Bearing lube oil to cooler: 180°F maximum (obtained from Process Computer Point IDs TBD025, TBD032, and TBD039).
6. Maximum Oil Temp Rise across the Turbine Bearings: 50°F.
7. Vertical Vibration at RFP Bearing Supports: 2 mils double amplitude.
8. RFPT Speed: 5850 rpm maximum (1-9-6).

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.14 (10CFR 55.41.12)	Tier #	3	------
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	Group #	------	------
	K/A #	G2.3.14	
	Importance Rating	3.4	------

Proposed Question: **# 72**

Unit 1 was at 35% Reactor Power when the Hydrogen Injection System was placed in service in Automatic / Power Determined mode in accordance with 1-OI-4, “Hydrogen Water Chemistry System.”

- Power is raised from 35% Reactor Power to 100% Reactor Power
- At 100% Reactor Power hydrogen flow rate indicates 20 scfm

Which ONE of the following completes the statements?

In accordance with 1-OI-4, hydrogen injection flow rate is **__(1)__** the normal 100% Reactor Power flow rate.

Radiation levels in the Condenser Bay will stabilize **__(2)__** expected normal full power radiation levels.

A. **(1)** above
(2) at

B. **(1)** below
(2) at

C. **(1)** above
(2) above

D. **(1)** below
(2) below

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See explanation C. Part 2 incorrect - radiation levels are expected to increase to above normal levels. Plausible because flow is higher than normal, but the candidate must understand the effects of H2 injection on Radiation levels.

B INCORRECT: Part 1 incorrect - H2 flow rate is NOT below normal. Plausible in that this injection flow rate is below the maximum allowed H2 injection rate of 25 scfm. Part 2 is incorrect -radiation levels are expected to increase to above normal levels.

C **CORRECT**: Part 1 correct - In the Automatic / Power Determined Mode, the hydrogen injection system is load following. It is normally placed in service above 25% Rx power. As Rx power is increased, the hydrogen flow rate is increased to the maximum amount that the controller is set for. Normal H2 Injection Rate (100% Reactor Power) is 14 scfm. Due to current EPU spanned software installed on U1 HWC computer, Chemistry has requested Ops to input a 16 scfm H2 injection value in order to receive the required 14 scfm actual H2 injection flow at 100% Power. Part 2 correct – Operation of HWC with injection rates above normal will cause a significant rise in radiation dose rates in steam affected areas.

D INCORRECT: Part 1 and 2 incorrect as explained above.

Justification: Requires knowledge of radiation hazards associated with changes in HWC System parameters to correctly answer. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 1-OI-4 Rev. 9 (Attach if not previously provided)
OPL171.220 Rev. 6

Proposed references to be provided to applicants during examination: ______

Learning Objective: OPL171.220 V.B.6 / 9_ (As available)

Question Source: Bank # Hatch 2009 # 72
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Hatch 2009

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 **X**
55.43

Comments:

INSTRUCTOR NOTES

(8) At the OIU, transfer mode to Automatic / Power Determined Setpoint

or responses when viewing different screens

(9) After reaching steady state oxygen flow and indication, verify offgas oxygen stabilizes at 21% ± 5%

Adjusting the oxygen controller ratio may be required.
Procedure USe

5. Normal Operation

a. During operation important injection flow rate values are:

(1) Minimum hydrogen injection rate allowed to be entered on the OIU: 3 SCFM. When 3 SCFM is entered in the OIU for Automatic/Power Determined mode, hydrogen injection rate will lower automatically to a new SCFM depending on new power level

Normally used when lowering HWC for ALARA or maintenance.

(2) Normal allowed hydrogen injection rate (100% reactor power): ~12-14 SCFM.

Unit Diff:
U1 / 2 = 14 scfm
stpt @ 16 scfm
U3 = 12 scfm

(3) Maximum hydrogen injection rate allowed: U1 / 3 = 25 SCFM, U2 = 20 SCFM

Value set by Chem Lab
This is the desired setpoint. This flowrate is post noble metals injection. However U-1 will utilize this flowrate pre-noble metal injection. Noble Metal injection planned to be performed 1st cycle of operation

b. Changes in reactor power can affect the HWC System's ability to operate

(1) Reactor power maneuvers <15%,: HWC System should control injection rates adequately

(2) Reactor power maneuvers >15% with a 15 minute wait period between each 30% change: HWC System should control injection rates adequately

INSTRUCTOR NOTES

a. These compounds are circulated through the reactor coolant systems and are ultimately removed by the RWCU System

b. A smaller fraction of the N^{16} is carried over in the steam in the form of nitrogen gas (N_2) and ammonia (NH_3) — Predominate contributor to background radiation levels

4. H_2 injection alters the N^{16} carryover ratio

a. Concentrations of NO_3, NO_2, and NO decrease

b. Concentration of NH_3 increases — Ammonia

(1) A gas

(2) High water solubility

5. The net production of N^{16} is not influenced by hydrogen injection

6. The increased dose rates are due to the increased ease with which N^{16} gets out of the reactor and into the steam pipes when in the NH_3 form — We can maintain up to 2.7 ppm injection concentration.

7. The initial U2 run was the first week in Nov. 1999. Up to 90 scfm hydrogen was injected. Average MSL radiation level increased approximately 5 times normal — MSL 'B' was highest at 5.2 times normal

8. Addition of noble metals to reactor water

a. Noble metals decompose during reactor startup or shutdown — Rubidium and Iridium

b. During this time it produces a thin layer of noble metal on wetted surfaces

c. The ECP on these surfaces are reduced significantly during subsequent operation

3.0 PRECAUTIONS AND LIMITATIONS (continued)

4. Important hydrogen injection flow rate values are as follows:

   a. Minimum H_2 Injection Rate allowed to be entered on the OIU is 5 scfm. This is the injection rate normally used when lowering HWC for ALARA considerations or maintenance purposes per Section 6.0, Normal Operations. When 5 scfm is entered in the OIU for Automatic/Power Determined Mode, H_2 Injection Rate will lower automatically to a new scfm depending on the new power level, i.e., 5 scfm for 100% power; when power is lowered to 90%, the injection rate will automatically roll back to 4.5 scfm and so on.

   b. Normal H_2 Injection Rate (100% Reactor Power) is 14 scfm. Due to current EPU spanned software installed on U1 HWC computer, Chemistry has requested Ops to input a 16 scfm H2 injection value in order to receive the required 14 scfm actual H2 injection flow at 100% Power. This value, 14 scfm, is determined by Chemistry with the performance of CI-13-1, Chemistry Program. Chemistry will notify Operations should this value change). “Off Normal” operating conditions may require other injection rates which are to be coordinated with the System Engineer, Chemistry, Radiation Protection, and approved by the Shift Manager.

   c. Maximum H_2 Injection Rate allowed: 25 scfm.

BFN Unit 1	Hydrogen Water Chemistry System	1-OI-4 Rev. 0009 Page 12 of 92

3.0 PRECAUTIONS AND LIMITATIONS (continued)

5. Personnel should be aware that changes in Reactor power can affect the HWC System's ability to operate. The following depict power changes and their expected HWC System response:

   a. Reactor power maneuver of less than 30%:

      HWC System should control injection rates adequately.

   b. Reactor power maneuver of greater than 30% with a 15 minute waiting period in between each 30% power change:

      HWC System should control injection rates adequately.

   c. Reactor power maneuver of greater than 30% without a 15 minute waiting period in between each 30% power change:

      HWC System may NOT be able to keep up with the power change. A rising power maneuver of this type can lead to a HWC System automatic shutdown on low Offgas O_2 concentration ($\leq 5\%$).
      A lowering power maneuver can lead to a HWC System shutdown on high Offgas O_2 concentration ($\geq 40\%$).

   d. Certain plant abnormal occurrences such as a Recirc Pump trip while at power can result in an automatic shutdown of the HWC System.

6. Reductions in reactor power while maintaining the Hydrogen injection rate in the Operator Determined Setpoint mode will cause a significant rise in radiation dose rates in steam affected areas above the dose rate for that area with Hydrogen injection in Power Determined Setpoint mode.

7. As time permits, the Radiation Protection Shift Supervisor should be notified prior to or immediately after lowering reactor power with the Hydrogen Water Chemistry System aligned for injection. The Radiation Protection Shift Coordinator is responsible for ensuring dose rates to personnel entering steam affected areas are properly monitored.

8. The oxygen controller directs oxygen flow to the Off Gas System (suction side of the SJAEs) and to the Condensate System (suction to each Condensate Pump). The controller is used in only one mode, the Automatic/Hydrogen Determined Setpoint mode. This mode has oxygen flow automatically following the hydrogen injection flow rate. A delay time exists in the oxygen controller logic to prevent undesirable oxygen concentrations from occurring in the Off Gas System.

HLT 4 NRC Exam

72. G2.3.14 001

Unit 1 was at 35% power when the Hydrogen Injection System was placed in service in AUTOMATIC - EXTERNAL mode IAW 34SO-P73-001-1, "Hydrogen and Oxygen injection and Control for HWC" section 7.1.2, "Placing 1P73-R025, Hydrogen Controller, in EXTERNAL."

- o Power is raised from 35% power to 100% power
- o At 100% power hydrogen flow rate indicates 40 SCFM

Which ONE of the following answers both of these statements?

IAW 34SO-P73-001-1, "Hydrogen and Oxygen injection and Control for HWC", hydrogen injection flow rate is ______ the normal 100% power flow rate.

Radiation levels in the Condenser Bay will stabilize ______ expected normal full power radiation levels.

A. above;
at

B. below;
at

C. below;
below

D✓ above;
above

Examination Outline Cross-reference:

G2.3.5 (10CFR 55.41.11)

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

	RO	SRO
Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.3.5	
Importance Rating	2.9	------

Proposed Question: **# 73**

Which ONE of the following completes the statement in accordance with EOI-4 Program Manual?

Per EOI-4, "Radioactivity Release Control," __**(1)**__ Building Ventilation is re-started, if shut down, to __**(2)**__.

A. **(1)** Reactor
(2) filter the ground level release.

B. **(1)** Turbine
(2) filter the ground level release.

C. **(1)** Reactor
(2) to ensure ability to monitor **AND** elevate the radioactive release.

D. **(1)** Turbine
(2) to ensure ability to monitor **AND** elevate the radioactive release.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that Reactor Building Ventilation direction to restart is provided in EOI-3 under certain conditions and system does provide filtration.

B INCORRECT: Plausible in that where a high potential for contamination exist, Turbine Building Ventilation exhaust fans with HEPA filters are installed to control the spread of contamination. However, this is not the bases for starting Turbine Building Ventilation IAW EOI-4.

C INCORRECT: Plausible in that Reactor Building Ventilation does is discharged through an elevated, monitored release point and direction to restart is provided in EOI-3 under certain condition.

D **CORRECT:** Per EOI Program Manual, 0-V-F, Operation of the Turbine Building Ventilation System preserves turbine building accessibility, and assures that radioactivity in turbine building areas is discharged through an elevated, monitored release point.

Justification: Ability to use radiation monitoring equipment to monitor Turbine Building Ventilation is tested.

Technical Reference(s):	EOI Program Manual O-V-F Rev. 1	(Attach if not previously provided)
	OPL171.204 Rev. 7	

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.204 V.B.10 (As available)

Question Source:	Bank #		
	Modified Bank #	DAEC 07 #57	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Duane Arnold 2007	

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:	55.41	**X**
	55.43	

Comments:

DISCUSSION: ENTRY CONDITIONS EOI-4

Entry into EOI-4 should be accomplished early enough to provide an appropriate transition from normal or abnormal operating procedures to emergency procedures. BFN Radiological Emergency Plan event classifications are based on NUMARC/NESP-007 guidelines. NUMARC/NESP-007 requires Unusual Event classification at 2 times the ODCM limits and entry into Alert classification at 200 times the ODCM limits. NUMARC/NESP-007 specifies a duration of 1 hour for Unusual Event classification releases and 15 minutes for Alert classifications. This methodology makes it possible to have an Alert event or Site Area Emergency event, by meeting the shorter time duration, while never having reached the Unusual Event classification due to the longer time limits.

The wording of the entry condition provides for entry into EOI-4 when radioactivity release exceeds event classification limits for Unusual Event or any higher classification. Unusual Event limits, coupled with the required duration, are substantially above the limits for normal plant operation and represent inability to control radioactivity release. Entry into Emergency Procedures is appropriate at this, or any higher event classification, to ensure procedure guidance is provided when the guidance of normal or abnormal plant procedures has failed to terminate or reverse the trend of increasing release. Refer also to Emergency Plan Implementing Procedure, BFN-EPIP-1, Emergency Plan Classification Logic.

DISCUSSION: STEP RR-1

This retainment override step applies throughout the performance of this procedure. The operator is directed to restart turbine building ventilation if turbine building ventilation is shutdown.

Operation of the Turbine Building Ventilation System preserves turbine building accessibility, and assures that radioactivity in turbine building areas is discharged through an elevated, monitored release point. Continued personnel access to the turbine building may be essential for responding to emergencies or transients which may degrade into emergencies. Since the turbine building is not an air-tight structure, a radioactive release inside the turbine building would not only limit personnel access, but would eventually lead to an unmonitored ground level release.

The wording of the entry condition provides for entry into EOI-4 when radioactivity release exceeds event classification limits for Unusual Event or any higher classification. Unusual Event limits, coupled with the required duration, are substantially above the limits for normal plant operation and represent inability to control radioactivity release. Entry into Emergency Procedures is appropriate at this, or any higher event classification, to ensure procedure guidance is provided when the guidance of normal or abnormal plant procedures has failed to terminate or reverse the trend of increasing release. Refer also to Emergency Plan Implementing Procedure, BFN-EPIP-1, Emergency Plan Classification Logic.

Obj. V.B 8, 9
Obj. V.C.8,9

Any EPIP-1,
Section II-4
Event Classification

3. Step RR-1

This retainment override step, applies throughout the performance of this procedure. The first override directs the operator to exit all EOI flowcharts and perform actions as directed by the SAM team only when SAMG entry is required and the TSC SAM team has assumed command and control. The second part of the override has the operator restart turbine building ventilation if turbine building ventilation is shutdown.

Operation of the Turbine Building Ventilation System preserves turbine building accessibility, and assures that radioactivity in turbine building areas is discharged through an elevated, monitored release point. Continued personnel access to the turbine building may be essential for responding to emergencies or transients which may degrade into emergencies. Since the turbine building is not an air-tight structure, a radioactive release inside the turbine building would not only limit personnel access, but would eventually lead to an unmonitored ground level release.

Obj. V.B.10
Obj. V.C.10

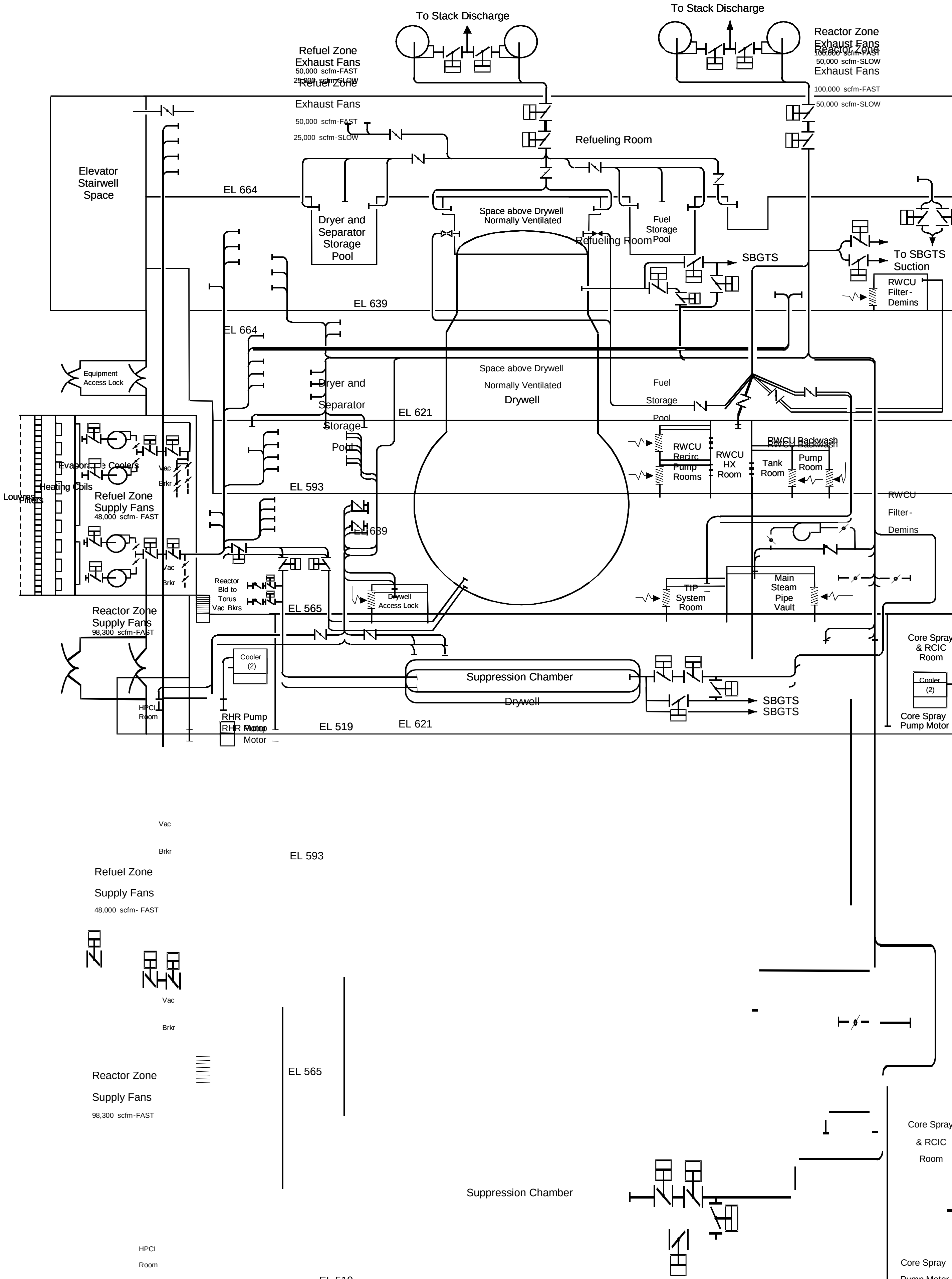
WHILE EXECUTING THIS PROCEDURE:

IF	THEN
SAMG ENTRY IS REQUIRED AND THE TSC SAM TEAM HAS ASSUMED COMMAND AND CONTROL	**EXIT** ALL EOI FLOWCHARTS AND **PERFORM** SAMG ACTIONS AS DIRECTED BY THE SAM TEAM
RX ZONE VENTILATION EXH RADIATION LVL IS ABOVE 72 MR/HR	**VERIFY** ISOLATION OF RX ZONE AND REFUEL ZONE VENTILATION AND **VERIFY** INIT OF SGTS
REFUEL ZONE VENTILATION EXH RADIATION LVL IS ABOVE 72 MR/HR	**VERIFY** ISOLATION OF REFUEL ZONE VENTILATION AND **VERIFY** INIT OF SGTS
RX ZONE VENTILATION IS ISOLATED AND RX ZONE VENTILATION EXH RADIATION LVL IS BELOW 72 MR/HR,	**RESTART** RX ZONE VENTILATION (APPX 8F). **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). **DEFEAT** ISOLATION INTERLOCKS IF NECESSARY (APPX 8E)

SCC-1

1-EOI-3	PAGE 1 OF 1
SECONDARY CONTAINMENT CONTROL	
UNIT 1 BROWNS FERRY NUCLEAR PLANT	
REV: 0	

1. Where a high potential for contamination exist, exhaust fans with HEPA filters are installed to control the spread of contamination. — Shown on Unit 1 Turbine Building Vent print
   a. Turbine decontamination chamber
   b. Outage hot tool room .
   c. Total turbine building exhaust capacity is approximately 269,000 cfm with a maximum supply of approximately 255,000 cfm, which allows the building to be kept under a slightly negative pressure to prevent leakage of contamination to environment, and provides for an elevated, monitored release. — Obj. V.B.2 Obj. V.C.2
2. Radiation Monitoring — Obj. V.B.2 Obj. V.C.12
   a. The turbine building exhaust flow is monitored by RM-90-250 CAM which also monitors the reactor building and refuel floor.
   b. The turbine room (roof) exhaust is monitored by RM-90-249 and RM-90-251 CAMS which must be in service anytime the fans are in service. Fan/monitor combinations are as follows: — PROCEDURE ADHERENCE
   c. Unit 1 & 2 RM 90-251 Fans A,B,C,D,E
   d. Unit 1 & 2 RM 90-249 Fans F,G,H,J
   e. Unit 3 RM 90-251 Fans A,B,C,D
   f. Unit 3 RM 90-249 Fans E,F,G,H,J
3. The following turbine area ventilation fans will trip on loss of air flow:
   a. Turbine spaces supply fan
   b. Mechanical spaces supply fans
   c. Electrical spaces supply fans
   d. Mechanical spaces exhaust fan
   e. Demineralizer spaces exhaust fans


To Stack Discharge
To Stack Discharge
Refuel Zone
Exhaust Fans
50,000 scfm-FAST
25,000 scfm-SLOW
Reactor Zone
Exhaust Fans
100,000 scfm-FAST
50,000 scfm-SLOW
Refueling Room
Elevator
Stairwell
Space
EL 664
Dryer and
Separator
Storage
Pool
Space above Drywell
Normally Ventilated
Fuel
Storage
Pool
SBGTS
To SBGTS
Suction
RWCU
Filter-
Demins
EL 639
Equipment
Access Lock
Drywell
EL 621
Evaporative Coolers
Heating Coils
Louvres
Filters
Vac
Brkr
Refuel Zone
Supply Fans
48,000 scfm- FAST
RWCU
Recirc
Pump
Rooms
RWCU
HX
Room
RWCU Backwash
Tank
Room
Pump
Room
EL 593
Reactor Zone
Supply Fans
98,300 scfm-FAST
Reactor
Bld to
Torus
Vac Bkrs
EL 565
Drywell
Access Lock
TIP
System
Room
Main
Steam
Pipe
Vault
Core Spray
& RCIC
Room
Cooler
(2)
Suppression Chamber
SBGTS
HPCI
Room
RHR Pump
Motor
EL 519
Core Spray
Pump Motor

TP-1: Reactor Building Ventilation

RO 57	K/A Number 295038	Statement 2.1.32	IR 3.4	Origin N	Source Question NA
LOK F	10CFR55.41(b) 10	LOD (1-5)	Reference Documents EOP 4 Bases Rev 8		
High Off-site Release Rate Conduct of Operations: Ability to explain and apply all system limits and precautions.					

Why does EOP-4, Radioactivity Release Control, require restarting Turbine Building Exhaust Fans?

Restarting Turbine Building Ventilation will:

a. ensure air is monitored and elevated prior to release to the environment.

b. provide additional air flow to dilute radioactivity prior to release to the environment.

c. provide cooling to promote condensation of steam leaks to minimize radioactivity release.

d. maintain a positive pressure in the Turbine Building to minimize leakage from the Reactor Building.

Correct Answer: A Exhaust Fans are restarted to ensure air is monitored and elevated prior to release to the environment.
Plausible Distractors: B is plausible: air flow will dilute effluent, but is not the reason stated in the EOP Bases. C is plausible: would be true if Supply Fans were restarted. D is plausible: would be true if Supply Fans were restarted.
Objective Link: None

Examination Outline Cross-reference:

G2.4.20 (10CFR 55.41.10)

Knowledge of the operational implications of EOP warnings, cautions, and notes.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.4.20	
Importance Rating	3.8	------

Proposed Question: **# 74**

In accordance with 1-EOI-1, "RPV Control," NOTE #1, which ONE of the following indications confirms that the **Unit 1** reactor will remain subcritical under **ALL** conditions without boron?

A. **ALL** control rods are at position 02.

B. **ALL** control rods full-in **EXCEPT** 2 at position 30.

C. Reactor Power is on range 7 of the IRMs **AND** lowering.

D. **ALL** control rods full-in **EXCEPT** 1 at position 02 **AND** 1 at position 48.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: In accordance with "RPV Control," 1-EOI-1 NOTE #1 for Unit 1, the reactor will remain subcritical under ALL conditions without boron when all control rods are inserted to or beyond position 02. This is true for Unit 1 only. Unit 2 and 3 require no more than 19 control rods inserted to or beyond position 02 and all other control rods fully inserted which supports the plausibility of the distracters.

B INCORRECT: In accordance with "RPV Control," 1-EOI-1 NOTE #1 for Unit 1, the reactor will remain subcritical under ALL conditions without boron when all control rods except one are inserted to or beyond position 00.

C INCORRECT: When used in the EOIs Subcritical is defined as Reactor power on range 7 and lowering of the IRMs with the IRMs inserted. However, this does not ensure that the reactor will remain subcritical under ass conditions without boron.

D INCORRECT: Does not meet criteria for ensuring the reactor will remain subcritical under ALL conditions without boron. Plausible misconception in that candidate may believe combination of SDM One Rod Out and Maximum Subcritical Bank Withdrawal Position will ensure reactor will remain subcritical under all conditions.

Justification: Requires knowledge of the operational implications of EOP NOTE #1 to successfully answer.

Technical Reference(s): 1-EOI-1 Rev. 0 (Attach if not previously provided)

EOI Program Manual 0-X-B Rev. 19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

____

Question Source: Bank #

Modified Bank # Oyster Creek 07 #5 (Note changes or attach parent)

New

Question History: Last NRC Exam Oyster Creek 2007

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

A.70 Maximum Subcritical Banked Withdrawal Position

UNIT 1

Reference/Philosophy

Fuel Parameter Memorandum L32 080924 801

PSTG and EOI Value

Position 02

UNIT 2

Reference/Philosophy

Fuel Parameter Memorandum L32 070109 800

PSTG and EOI Value

Any 19 control rods are at notch 02,
with all other control rods fully inserted.

UNIT 3

Reference/Philosophy

Fuel Parameter Memorandum L32 051118 800

PSTG and EOI Value

Any 19 control rods are at notch 02,
with all other control rods fully inserted.

1-EOI-1	PAGE 1 OF 1
RPV CONTROL	
UNIT 1 BROWNS FERRY NUCLEAR PLANT	
REV: 0	

NOTES

1. THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS WHEN:
   - ALL CONTROL RODS ARE INSERTED TO OR BEYOND POSITION **02**

     OR
   - ALL CONTROL RODS EXCEPT ONE ARE INSERTED TO OR BEYOND POSITION **00**

     OR
   - DETERMINED BY REACTOR ENGINEERING

2. TSC STAFF MAY RECOMMEND AN ALTERNATE CURVE FOR STATION BLACKOUT PER 0-AOI-57-1A

OYSTER CREEK 2007

OC ILT 07-1 RO NRC Exam KEY

Following an automatic scram from rated power, the Operator placed the REACTOR MODE SELECTOR switch in SHUTDOWN.

Which of the following indications, **ALONE,** allows the Reactor Operator to confirm that the reactor will remain shutdown under all conditions without boron?

A. All APRMs indicate < 2% power

B. All control rods indicate position **04.**

C. All LPRM amber lights on Panel **4F** are LIT.

D. **All** control rods full-in **EXCEPT** 2 control rods at position 30.

.

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.45 (10CFR 55.41.10)	Tier #	3	------
Ability to prioritize and interpret the significance of each annunciator or alarm.	Group #	------	------
	K/A #	G2.4.45	
	Importance Rating	4.1	------

Proposed Question: **# 75**

Which ONE of the following describes the meaning of a WHITE magnetic border being installed on a Main Control Room panel annunciator?

This type of border indicates that the annunciator ______.

A. has ONE **OR** more alarm inputs disabled.

B. is associated with ongoing testing **OR** maintenance.

C. is "NOT ABNORMAL" for current plant conditions.

D. is an entry condition for EOI-3, “Secondary Containment Control.”

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: In accordance with “Annunciator Disablement,” OPDP-4, a blue magnetic boarder indicates that an alarm is out of service.

B **CORRECT**: In accordance with “Annunciator Disablement,” OPDP-4, a white magnetic boarder indicates that an alarm is out of service for TESTING or MAINTENANCE.

C INCORRECT: In accordance with “Annunciator System,” 0-OI-55, a hot pink boarder indicates that an alarm is "NOT ABNORMAL" for current plant conditions.

D INCORRECT**:** Entry conditions for “Secondary Containment Control,” EOI-3 are designated by an orange rectangle on an annunciator window.

Justification: Requires knowledge of Control Room Annunciator designators / boarders to effectively prioritize and interpret significance of alarms.

Technical Reference(s): OPDP-4 Rev. 4 (Attach if not previously provided)

0-OI-55 Rev. 46

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________________ (As available)

____

Question Source: Bank #

Modified Bank # Hatch 07 # 74 (Note changes or attach parent)

New

Question History: Last NRC Exam Hatch 2007

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge **X**

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 **X**

55.43

Comments:

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5.0 DEFINITIONS

Disabled Input Indicator

- BFN -A blue magnetic border labeled "Disabled Alarm Input."
- SQN-A blue dot (sticker) attached to the window with the SER point written on it.
- WBN-An orange plastic lens cover labeled "Disabled Alarm" which snaps over the affected window and a blue plastic lens cover labeled "Disabled Input."

Out-of-Service Indicator

- BFN -A white magnetic border labeled "Testing/Maintenance."
- SQN-An orange sticker attached to the window.
- WBN-A green plastic lens cover labeled "Maintenance" which snaps over the affected window.

Maintenance Activities - Activities that restore components to their as-designed condition, including activities that implement approved design changes. Maintenance activities are not subject to 10 CFR 50.59. Maintenance activities include troubleshooting, calibration, refurbishment, maintenance-related testing, identical replacements, housekeeping and similar activities that do not permanently alter the design, performance requirements, operation or control of equipment. Maintenance activities also include temporary alterations to the facility or procedures that directly relate to and are necessary to support the maintenance. Examples of temporary alterations that support maintenance include jumpering terminals, lifting leads, placing temporary lead shielding on pipes and equipment, removal of barriers, and use of temporary blocks, bypasses, scaffolding and supports.

Nuisance Alarm - An alarm that comes in repetitively due to an instrumentation problem, or maintenance activity that detracts from the operator's ability to monitor and control the plant.

Valid Alarm - An alarm that is actuated when the monitored parameter exceeds the setpoint or meets the intent of a setpoint (e.g. if a high pressure alarm occurs at 580# and the alarm setpoint is 600# but pressure is normally zero or close to zero, that is a valid alarm. In a similar scenario, if pressure is normally 550#, the alarm may not be valid).

6.0 REQUIREMENTS AND REFERENCES

Requirements and References are contained in the "OPDP-4 REQ & REF" document.

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8.3 Identification of Out of Service Annunciators

- **REFERENCE** OPDP-4, Annunciator Disablement

NOTES

1) This Section applies to annunciators which alarm or are in alarm status due to the present plant conditions (i.e., Modifications, extended Maintenance, alarms due to plant Mode, etc.).
2) These borders signify "THESE ILLUMINATED ALARMS ARE ILLUMINATED DUE TO THE PRESENT PLANT CONDITIONS," and no operator action is required.
3) The diagonal bar in the "Hot Pink" border means "NOT ABNORMAL" for current plant conditions.

8.4 Identification of Lit Annunciators for Normal Plant Conditions

[1] **PLACE** "Hot Pink" identification border on each applicable annunciator window. ☐

[2] **WHEN** conditions of the plant change such that the annunciator will no longer remain illuminated as a normal condition, **THEN**

REMOVE the "Hot Pink" identification border from each applicable annunciator window. ☐

74. G2.4.45 001

Which ONE of the following describes the meaning of a white plastic frame being installed on an annunciator at the Reactor / Containment Cooling and Isolation panel 2H11-P601?

The white plastic frame means that the annunciator:

A. is the result of some plant evolution that is both known and expected by the operating crew, i.e., expected alarm flag.

B. has been disabled, i.e., “card is pulled.”

C. indicates an entry condition for 31EO-EOP-010-1, RC/RPV Control.

D✓ is an indicator of a potential radiological condition.

Note: The meaning of the white plastic frame is only defined in 73EP-EIP-018-0, Prompt Offsite Dose Assessment.

A. Incorrect because white outline is alarm with potential for being an indication of a radiological condition. Plausible since "expected" alarms are specifically identified too. (i.e., yellow flag)

B. Incorrect because white outline is alarm with potential for being an indication of a radiological condition. Plausible since disabled alarms are specifically identified too. (i.e., yellow magnet dot)

C. Incorrect because white outline is alarm with potential for being an indication of a radiological condition. Plausible since alarms associated with Secondary Containment Control Table 5 are also specifically identified too. (i.e., they have a label immediately adjacent to the annunciator, e.g., SC/L-1).

D. Correct.