Examination Outline Cross-reference:	Level	RO	SRO
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	Tier #	1	------
AK1.02 (10CFR 55.41.8 TO 41.10)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :	K/A #	295001AK1.02	
• Power/flow distribution	Importance Rating	3.3	------

Proposed Question: **# 1**

Unit 1 is at 100% Reactor Power **AND** Core Flow is 92%. A trip of 1A Recirc Pump results in Operation in Region II of the Core Power to Flow Map.

Which ONE of the following completes the statement below?

The required action(s) in accordance with 1-AOI-68-1A, "Recirc Pump Trip / Core Flow Decrease," is (are) to **IMMEDIATELY** ________.

A. insert a Manual Reactor Scram

B. raise Core Flow until Region II of the Power to Flow Map is exited

C. insert Control Rods until Region II of the Power to Flow Map is exited

D. insert Control Rods until Load Line is < 95.2%; then, raise Core Flow to > 45%

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that IF both Recirc Pumps are tripped in Modes 1 or 2, THEN 1-AOI-68-1A requires the Reactor to be Scrammed.

B INCORRECT: Plausible in that immediately raising core flow would be an expeditious method to exit instability regions. If load line was less than 95.2% following the Recirc Pump trip, this would be the correct answer.

C INCORRECT: Plausible in that Control Rod are required to be immediately inserted if in Region I or II but the crew will stop inserting Control Rods when Load Line is < 95.2%. That is, Control Rod insertion will stop prior to exiting the Region and raising core flow will complete the exit from Region II. If core flow was greater than 45% following the Recirc Pump Trip, this would be the correct answer.

D **CORRECT:** In accordance with 1-AOI-68-1A, IF Region I or II of the Power to Flow Map is entered due to a trip of a Recirc Pump, THEN IMMEDIATELY take actions to insert control rods to less than 95.2% loadline. Then, RAISE core flow to greater than 45% in accordance with1-OI-68.

KA Justification:

The KA is met because it tests candidate's knowledge of operational implications of Reactor Power / Flow distribution with a partial loss of core circulation as a result of a Recirc Pump trip.

Question Cognitive Level:

Question rated as C/A because Candidates' must process multiple pieces of data to determine correct actions in accordance with 1-AOI-68-1A. Candidate must recognize that with core flow of 92% at Reactor Power of 100% that Load Line is greater 100% and will remain greater than 100% following the Recirc Pump trip. Also, must recognize that following the trip, Core Flow will be less than 45% requiring increase in core flow also.

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: OPL171.007 V.B.28 (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	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] ☐

PLAUSIBILITY SUPPORT

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. ☐

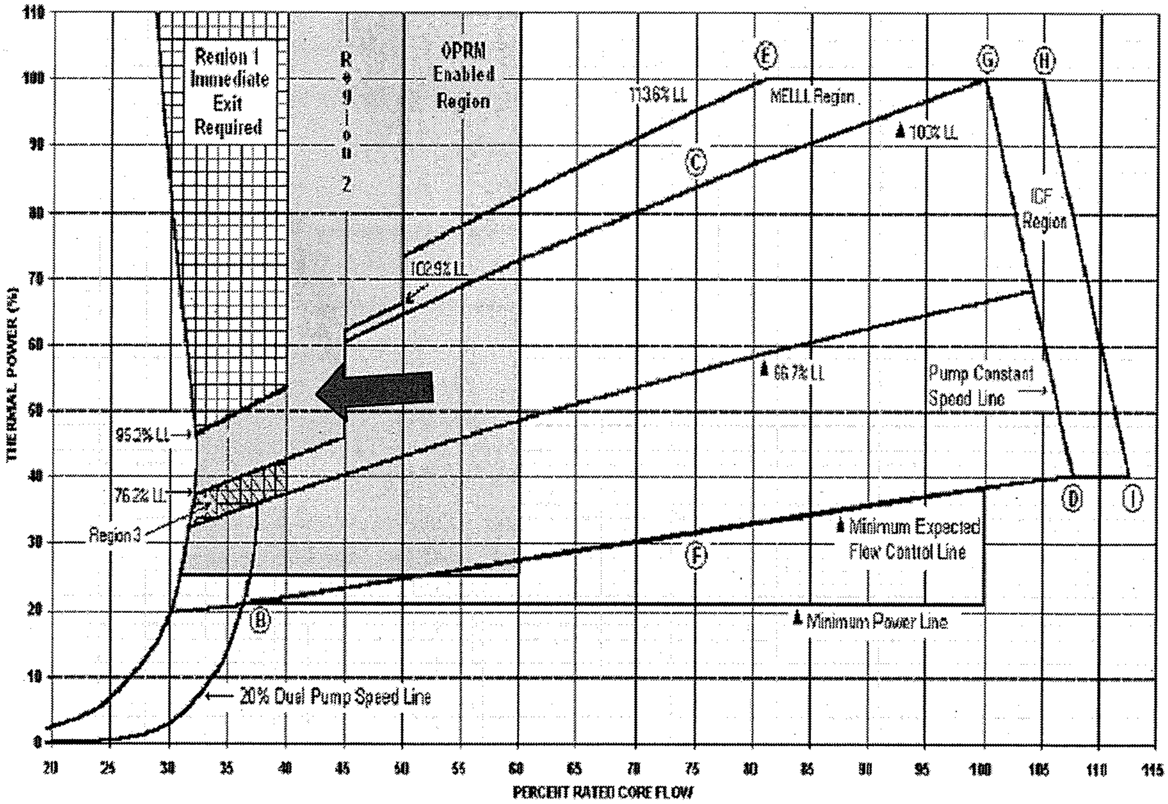
[1.3] **IF** the ΔT between the Rx vessel bottom head temperature and the moderator temperature precludes restart of a Recirc pump, **OR** forced Recirculation flow CANNOT be established for any reason, **THEN** (Otherwise NA)

[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. ☐

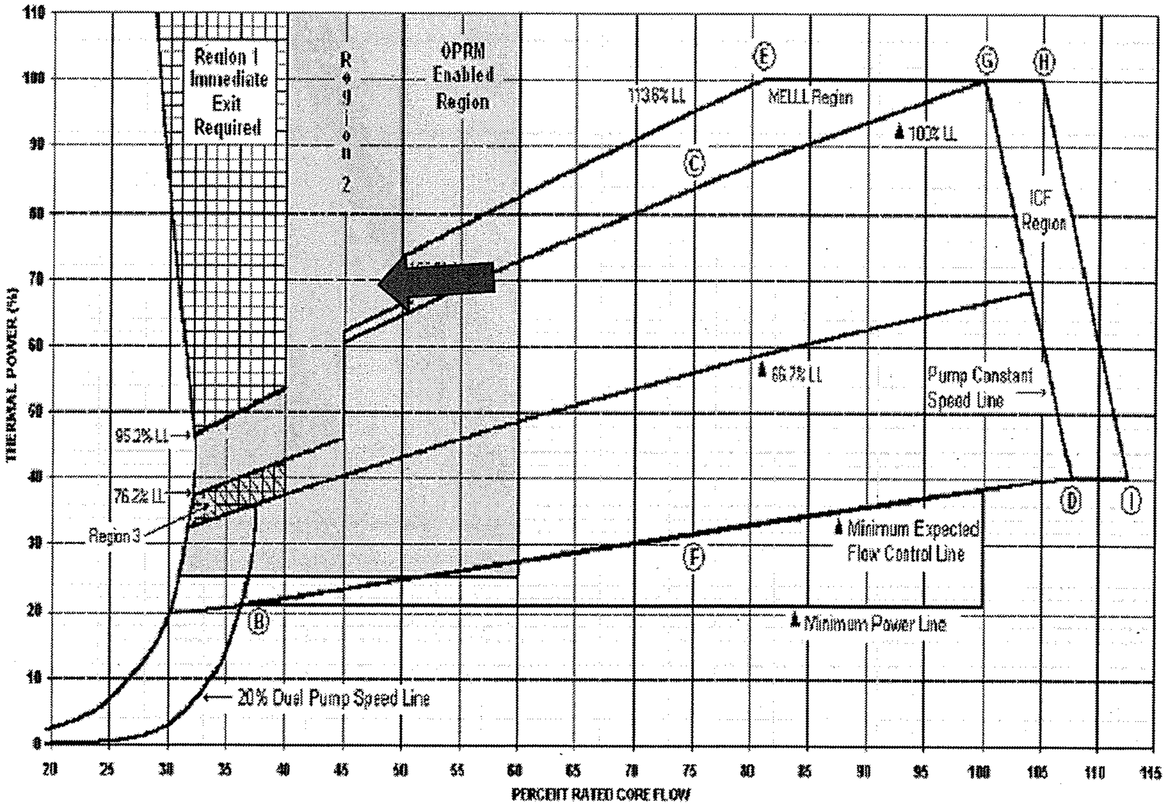
PLAUSIBILITY SUPPORT

If Load Line was < 95.2% following the Recirc Pump Trip, Distractor B would be the correct answer.



PLAUSIBILITY SUPPORT

If Core Flow was >45% following the Recirc Pump Trip, Distractor C would be the correct answer.



Examination Outline Cross-reference:	Level	RO	SRO
295003 Partial or Complete Loss of A.C. Power / 6	Tier #	1	------
G2.4.6 (10CFR 55.41.10)	Group #	1	------
Knowledge of EOP mitigation strategies.	K/A #	295003 G2.4.6	
	Importance Rating	3.7	------

Proposed Question: **# 2**

A leak in the Unit 1 Drywell results in the following conditions:

- Drywell Temperature is 170° F and rising
- A Lockout occurs on 4kV Shutdown Board C
- Reactor Level is (+) 10 inches and stable
- Suppression Pool Level is 15 feet

Which ONE of the following completes the statements below?

In accordance with 1-EOI-2, "Primary Containment Control," Drywell Spray must be initiated before MAXIMUM Drywell Temperature of __**(1)**__ . Assuming no manual electric board transfers are performed, RHR __**(2)**__ is (are) available for Drywell Spray from the control room.

A. **(1)** 200° F
(2) Loop I **ONLY**

B. **(1)** 200° F
(2) Loop I **AND** Loop II

C. **(1)** 280° F
(2) Loop I **ONLY**

D. **(1)** 280° F
(2) Loop I **AND** Loop II

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 Drywell Temperature of 200° F is a recognizable value of 1-EOI-2, Drywell Temp Leg requiring entry into EOI-1. Part 2 incorrect – Plausible in that Unit 2 480 V Shutdown Board B is supplied from 4 kV S/D Board D. On Unit 2 this would be the correct answer.

C **CORRECT**: Part 1 correct – 1-EOI-2 directs Drywell Spray prior to Drywell Temp of 280° F. Part 2 correct – Loop II Drywell Spray valves are powered from 480 RMOV Board B which is powered from 480 V S/D Board B. This Board is powered from 4 kV S/D Board C on Unit 1 which is locked out. Although one pump is available on Loop 2, Spray Valves can not be opened from the control room.

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

KA Justification:

The KA is met because question tests knowledge of EOI mitigation strategies with partial loss of AC Power.

Question Cognitive Level:

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. Candidate must determine effect of a Lockout on 4kV Shutdown Board C on ability to Spray the Drywell.

Technical Reference(s): OPL171.036 Rev. 12 / 1-EOI-2 Rev. 1 (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.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:

(2) Tube side fluid - RHRSW @ 4500 gpm

d. Tube side pressure should be kept higher than shell side if possible to minimize the potential leakage of RHR water to the RHRSW. This limits the potential for radioactive discharge to the environment. RHRSW discharge is monitored for radioactivity prior to discharge to the river. No automatic actions occur due to a high radioactivity condition in the RHRSW.

4. Valves — Obj. V.B.8

a. Power supplies - All RHR motor-operated valves are powered from the 480V Reactor MOV Boards except as noted. The Reactor MOV Board power supplies are as follows. Note divisional separation maintained. — Obj. V.E.6

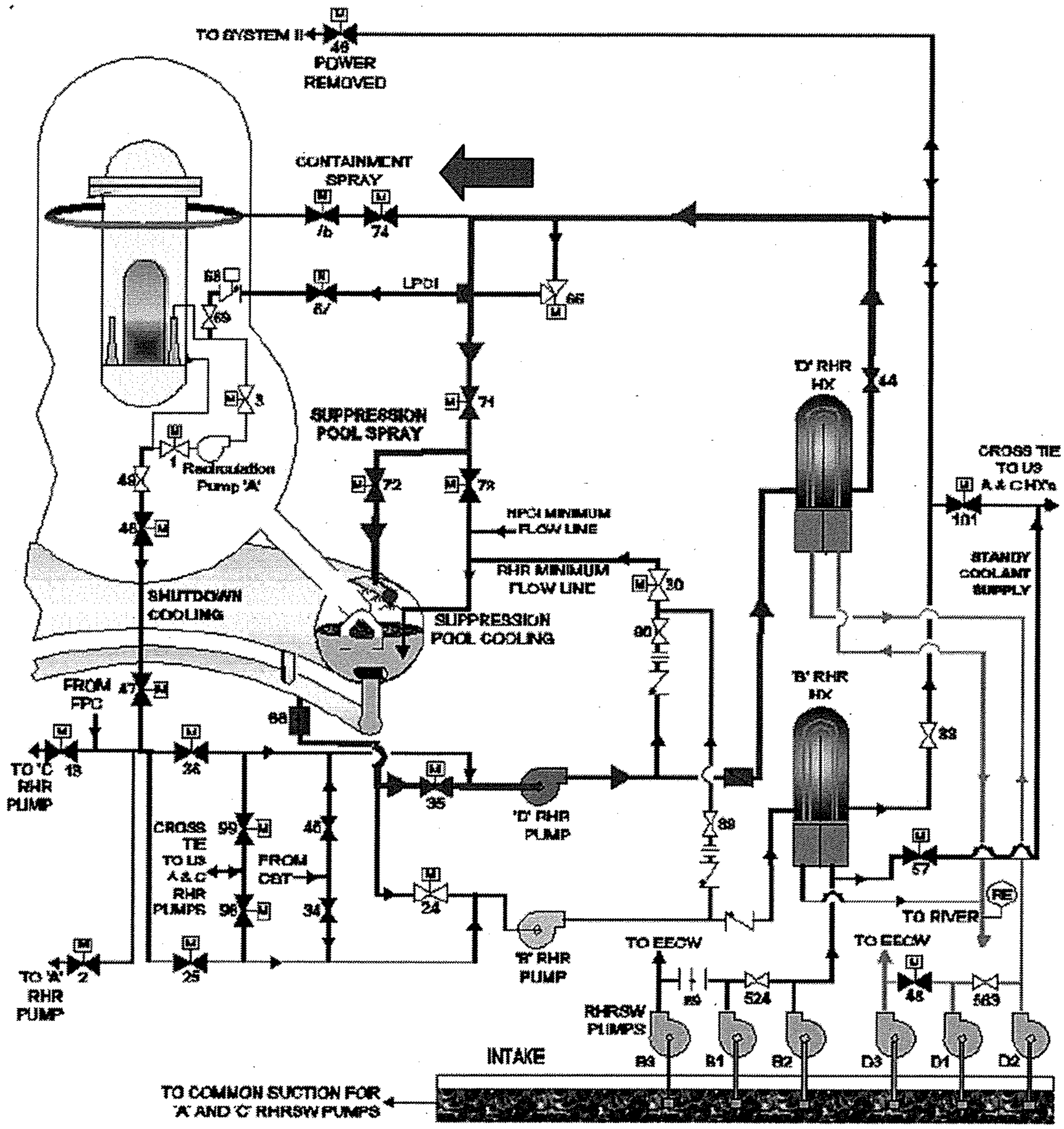
480 RMOV BOARD	NORMAL POWER	DIV	ALT POWER	DIV	VALVES
"A"	480V S/D "A"	I	480V S/D "B"	II	RHR Sys I valves (except as noted)
"B"	480V S/D "B"	II	480V S/D "A"	I	RHR Sys II valves (except as noted)
"C"	480V S/D "B"	II	480V S/D "A"	I	none
"D" **	"DN" MG Set	I	"DA" MG Set	II	74-7 & 53
"E" **	"EN" MG Set	II	"EA" MG Set	I	74-30 & 67

- ** Unit 1 does not have RMOV Bd 'D' or 'E'. The loads on 'D' Bd are fed from 1A Bd and 'E' are fed from 1B Bd
- Outboard Shutdown Cooling Isolation Valve FCV-74-47 is powered from 250 VDC MOV Board A. — Obj. V.B.8

OPL171.044
Revision 17
Appendix C
Page 97 of 146

VALVE#	VALVE NAME	INTERLOCKS
53/67	LPCI Inboard Injection	• NOT a throttle valve – normally closed • Cannot open valve if Outboard Injection is open and Rx pressure is >450# • Auto opens on LPCI initiation signal when Rx pressure is <450#. Remains open until LPCI initiation signal is clear and reset • Auto close if Inboard and Outboard SDC Isolation Valves open and Group 2 isolation signal received. Seals in and must be reset with SDC Isolation reset pushbutton. Valve will not open on LPCI initiation signal until reset. • NORMAL/EMERGENCY switch in EMERGENCY bypasses >450# and Outboard Injection valve open interlock. Also prevents auto open and close signal from logic and allows valve operation ONLY at breaker. Applies to 2/3-74-53 and 1-74-67. • EMERGENCY OPEN switch bypasses all interlocks. Applies to 2/3-74-53 and 1-74-67.
60/61 74/75	Drywell (Containment) Spray	• No auto open logic • Cannot open Inboard valve normally unless Outboard valve is fully closed. • Cannot open Outboard valve normally unless Inboard valve is fully closed. • Auto closed on LPCI initiation signal • The Sup. Pool/Chamber Isol. valve and LPCI initiation signal interlocks can be bypassed if Rx level >-183" AND Drywell pressure is >1.96 psig AND LPCI initiation signal present AND CONTAINMENT SPRAY OVERRIDE switch is in SELECT • Rx level and LPCI initiation signal can be overridden with 2/3 CORE HEIGHT OVERRIDE switch • NORMAL/EMERGENCY switch in EMERGENCY bypasses all interlocks and allows valve operation ONLY at breaker. Applies to 2/3-74-60 and 1-74-74.

OPL171.044
Revision 17
Appendix D
Page 104 of 146



TP 5 RHR SYSTEM SUPPRESSION POOL SPRAY FLOW DIAGRAM UNIT 2 SYSTEM II

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

4. 480V Reactor MOV Boards

Unit 2 Reactor MOV Boards A, B, C, D, and E are discussed here. Unit 3 Boards are similar. Valves on D & E boards, on Unit 1, were moved to A & B boards. Unit 1 does not have D & E boards.

Obj. V.B.6.g
Obj. V.C.1.g
Obj. V.D.6.g
UNIT DIFFERENCE

a. Reactor MOV Boards serve the smaller 480V loads that are important to plant safety. Each MOV board has two incoming sources, one from each 480V shutdown board. Reactor MOV Boards A and D feed normally from 480V Shutdown Board A and alternately from 480V Shutdown Board B. The normal supply for Reactor MOV Boards B, C, and E is 480V Shutdown Board B with A being the alternate.

Obj V.D.5

b. Boards D and E, the "LPCI Valve Boards," are fed through motor-generator sets for both their normal and alternate supplies. Unit 1 LPCI MG sets have been removed. Loads that were on U-1 D/E board are now on A & B boards (Unit difference)

Examples: Recirculation discharge valves, LPCI inboard injection valves, & RHR min-flow valves

c. Boards A, B, and C have manual transfer of power supplies. Boards D and E transfer automatically from normal to alternate on undervoltage; transfer back is manual.

Obj. V.B.8.g
Obj. V.C.2.g
Obj. V.D.8.g

LER 2-85-007

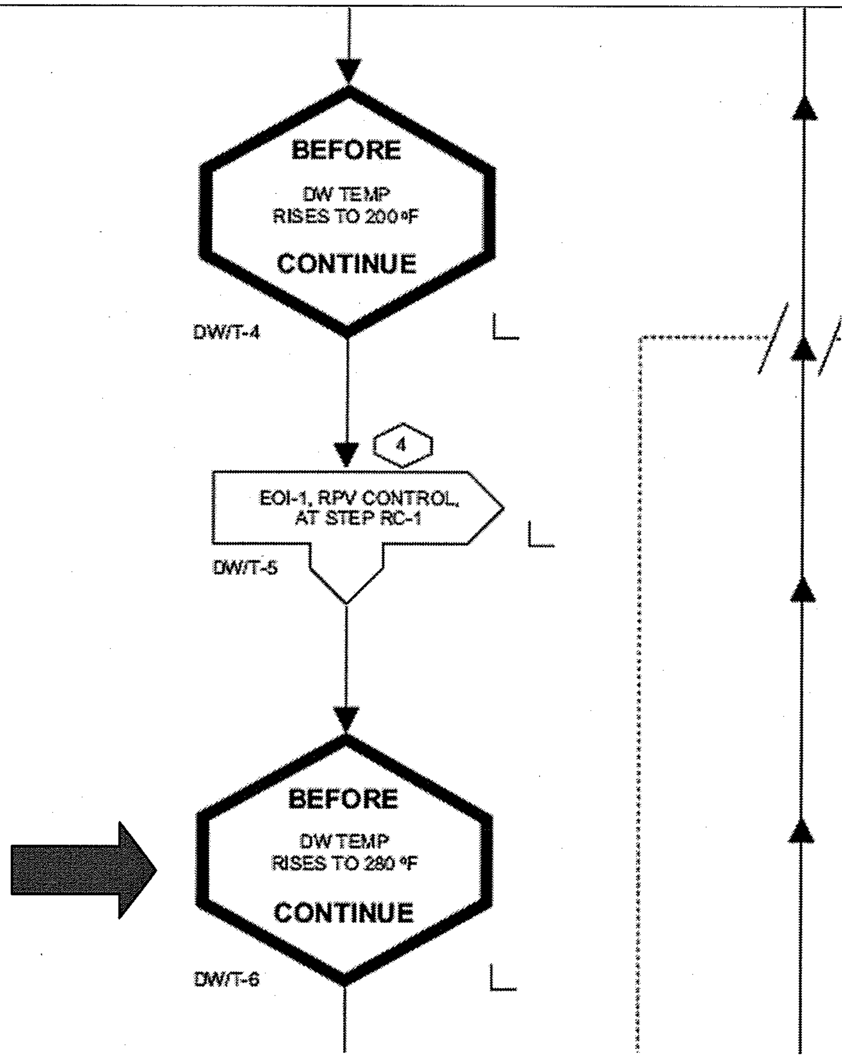
d. Selected feeder breakers have normal/emergency selector switches to allow local operation of the associated component.

5. 480 volt board indications and controls

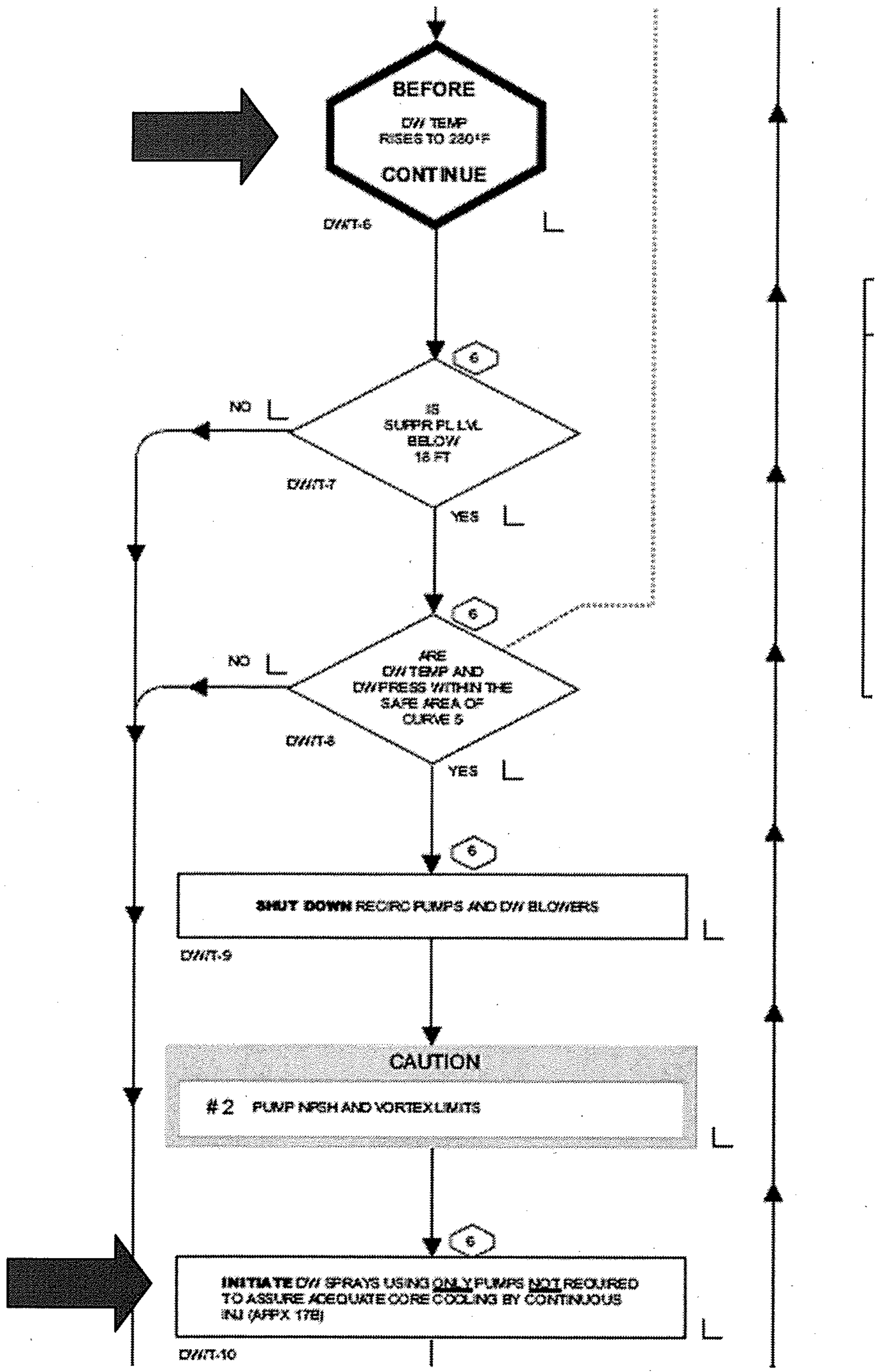
a. Panel 9-8 indications

(1) 480V Shutdown Bd. B voltage

(2) 480V Unit Boards voltage and amperage



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



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

DISTRACTOR PLAUSIBILITY SUPPORT

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

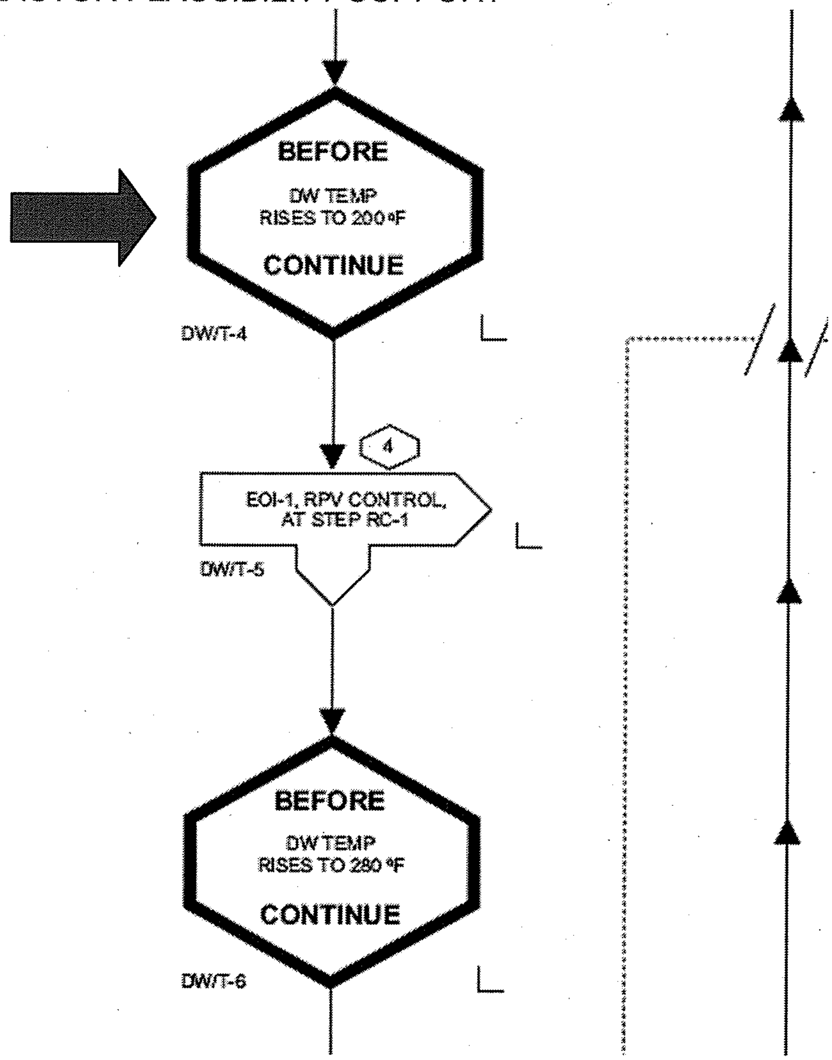
1. 480V Shutdown Boards

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

DISTRACTOR PLAUSIBILITY SUPPORT



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 below?

480V Shutdown Board 2B 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: **C**

Explanation (Optional):

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

B INCORRECT: Part 1 incorrect - 480v Shutdown Board 2B remains energized with the loss of 4kV Shutdown Board B. Plausibility based on misconception 480v Shutdown Board B normal power supply would be from 4kV Shutdown Board B. If this was Unit 1 480 V and 4Kv A Shutdown Boards, this would be the correct answer. 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

C **CORRECT**: Part 1 correct - 480v Shutdown Board 2B remains energized with the loss of 4kV Shutdown Board B. 4kV Shutdown Board D is the normal feeder to the 480v S/D Bd 2B. 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.

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

KA Justification:

The KA is met because 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 2B.

Question Cognitive Level:

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 189

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 # BFN 1006 #3 (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 2010

(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:

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. 0189 Page 106 of 112

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		

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 0	480V/240V AC Electrical System	0-OI-57B Rev. 0189 Page 106 of 112

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		

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 0	Switchyard and 4160V AC Electrical System	0-OI-57A Rev. 0141 Page 188 of 201

Illustration 1
(Page 5 of 7)

Auxiliary Power Supplies and Bus Transfer Schemes

REMARKS: Automatic delayed transfer from the normal to alternate 1 source is initiated by undervoltage on the normal source and automatic return is initiated by normal voltage on normal source. These transfers are blocked after time delay in the presence of an accident signal. When an accident signal is present, alternate 1 source breakers are tripped. Also, on 4kV Shutdown Bd A, B, C and D, the common accident signal auto trip from U-3 bus tie breakers (Alternate 3), has been removed. All diesel generators are automatically started by an accident signal, loss of voltage on its shutdown board for 1.5 seconds or degraded voltage for 4 seconds on its shutdown board. After five (5) seconds with no voltage on the shutdown board, all its supply breakers and all its loads except 4160-480V transformers are automatically tripped. Alternate 2 source is then automatically connected. A second level voltage protection is provided for each 4kV shutdown board which will operate an undervoltage relay. If voltage reduces to that board and after 7.43 seconds (from the initial time zero) the feed to the board is tripped, the auto transfer is blocked and motor breakers on the board are tripped. 1.36 seconds later the DG breaker closes in on that shutdown board. Manual return to the normal auxiliary power system is permitted if normal auxiliary power system voltage returns and if a unit is **NOT** in early stage of accident. Units 1 and 2 shutdown boards can be manually tied to their respective 3 unit shutdown board. When doing this, Unit 3's breaker must be closed in on a dead line (interlocked to prevent closing in on an energized line) then Units 1 and 2 respective shutdown breaker can be synchronized to tie the two boards together. Provision is included for backfeeding diesel generator power from the shutdown boards into the 4160V unit boards for hot standby shutdown cooling if all plant power, other than diesel generator power, is lost. For this purpose, means are provided to manually synchronize 4kV shutdown boards.

BFN 1006 #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.

Examination Outline Cross-reference:	Level	RO	SRO
295005 Main Turbine Generator Trip / 3	Tier #	1	------
AK1.01 (10CFR 55.41.8 to 41.10)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP :	K/A #	295005AK1.01	
• Pressure effects on reactor power	Importance Rating	4.0	------

Proposed Question: **# 4**

Unit 3 is operating at 20% Reactor Power with the Main Turbine online when a pipe rupture results in loss of **ALL** EHC:

Which ONE of the following completes the statement below?

Reactor Pressure will __**(1)**__ **AND** the Reactor __**(2)**__ Scram.

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

B. **(1)** lower
(2) will

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

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

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** With the failure of EHC, the Main Turbine Trips and Bypass Valves will fail closed. Reactor Pressure will rise until the Reactor High Pressure Scram setpoint is reached.

B INCORRECT: Plausibility based on misconception that Bypass Valves fail open on loss of EHC and subsequent scram on MSIV closure. Failing open is plausible in that there are EHC failures which will result in Bypass Valves failing open. For example, with EHC Control System in HEADER PRESSURE CONTROL, a single Header Pressure input failing high would result in Main Turbine Control Valves and Bypass Valves opening in attempt lower Reactor Pressure. Additionally, 3-AOI-47-2, "Turbine EHC Control System Malfunctions," addresses EHC System Failures which result in lowering Reactor Pressure.

C INCORRECT: Plausible in that if candidate considers only Main Turbine Trip actuation of RPS, this would be the correct answer since it is bypassed at this power level.

D INCORRECT: Plausibility based on misconceptions that Bypass Valves fail open on loss of EHC as discussed in detail above and subsequent scram on MSIV closure is bypassed at this power level or candidate considers only Main Turbine Trip actuation of RPS.

KA Justification:

The KA is met because the question tests knowledge of the operational implications of Pressure effects on reactor power as they apply to Main Turbine Generator Trip due to loss of EHC.

Question Cognitive Level:

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): OPL171.010, Rev. 12 (Attach if not previously provided)
3-OI-99 Rev. 47

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.010, V.B.6 (As available)
OPL171.010, V.B.23

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:

F. Turbine Bypass Valves (Nos. 1 through 9)

TP-1 and TP-7,8
≈ 3 % per BPV

1. Purposes

Obj.V.B.6.d
Obj.V.C.2.d
Obj.V.E.27

a. Routes steam not needed by the turbine to the condenser during the following conditions:

(1) Reactor Startup

(2) Turbine Roll

(3) Turbine Trips

(4) Reactor cooldown

b. Works in conjunction with the turbine control valves to maintain a constant reactor pressure for a given reactor power level.

c. Provides the capability to prevent over pressurization of the reactor if the MSIVs are open.

2. Location

The nine bypass valves are physically located above the turbine throttle in the moisture separator room near the main turbine stop and control valves.

3. Bypass Valve Design

a. Bypass valves are hydraulically operated, reverse seating globe valves.

b. The valves are positioned as required by a Control PAC and Servo-valves.

c. Valves fail closed upon loss of hydraulics.

BFN Unit 3	Reactor Protection System	3-OI-99 Rev. 0047 Page 63 of 80

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 (≤ 154psig 1st stage pressure)(TR 3.3.1)
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 (≤154psig 1st stage pressure)(TR 3.3.1)
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

DISTRACTOR PLAUSIBILITY SUPPORT

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.

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 3	Turbine EHC Control System Malfunctions	3-AOI-47-2 Rev. 0006 Page 6 of 8

4.0 OPERATOR ACTIONS

NOTE

If necessary, place keeping marks may be made directly in the Control Room copy of this instruction. Management Services should be contacted for a replacement copy when time permits.

4.1 Immediate Actions

[1] **IF** Reactor Pressure lowers to or below 900 psig, **THEN**

MANUALLY SCRAM the Reactor and **CLOSE** the MSIVs. [PER 03-006187-000] ☐

4.2 Subsequent Actions

[1] **IF** ANY EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). ☐

[2] **VERIFY** Automatic Actions have occurred. ☐

[3] **IF** a Group 1 isolation has occurred, **THEN**

PLACE EHC PUMP 3A and 3B, 3-HS-47-1A and 3-HS-47-2A, to PULL TO LOCK. ☐

[3.1] **WHEN** the turbine bypass valves close, **THEN**

RESET the Group 1 PCIS isolation and **OPEN** MSIVs, as desired. **REFER TO** 3-OI-1. ☐

[4] **USE** EHC WORKSTATION computer to aide in diagnosing the problem. ☐

[5] **REQUEST** assistance from Site Engineering. ☐

[6] **IF** necessary, **THEN**

TROUBLESHOOT the EHC Control System. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 3	Reactor Protection System	3-OI-99 Rev. 0047 Page 63 of 80

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 (≤ 154psig 1st stage pressure)(TR 3.3.1)
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 (≤154psig 1st stage pressure)(TR 3.3.1)
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

Examination Outline Cross-reference:	Level	RO	SRO
295006 SCRAM / 1	Tier #	1	------
AA1.05 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to SCRAM :	K/A #	295006AA1.05	
• Neutron monitoring system	Importance Rating	4.2	------

Proposed Question: **# 5**

With Unit 2 in Mode 2, Intermediate Range Monitors (IRMs) indicate 29.1 on Range 3 **AND** Reactor Period is 90 seconds.

Which ONE of the following identifies approximately how long it will take to reach the IRM Scram setpoint?

A. 35 seconds

B. 65 seconds

C. 125 seconds

D. 180 seconds

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that this would be half the time to the first doubling.

B INCORRECT: Plausible in that this would be the time to the first doubling.

C **CORRECT**: C is correct as with a reactor period of 90 and 2 doubling times, (29.1-58.2 and 58.2-116.4). This time would be 62.28 seconds times 2. The scram setpoint would be reached in 124.56 seconds.

D INCORRECT: Plausible in that this would be twice the period.

KA Justification:

The KA is met because the question tests candidates' ability to monitor IRMs as they apply to Scram.

Question Cognitive Level:

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. Candidates must determine doubling time based on Reactor Period then calculate time to reach IRM Scram setpoint.

Technical Reference(s): OPL171.020, Rev. 11 / 2-OI-92A, Rev. 28 (Attach if not previously provided)
2-GOI-100-1A Rev. 145

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.020 V.B.7 (As available)

Question Source: Bank # Monticello 07 #43
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Monticello 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:

INSTRUCTOR NOTES

E. Trips — TP-10

1. Rod blocks — Obj.V.D.7, V.B.5; Obj. V.C.3.,

Block	Setpoint	When Bypassed	Instructor Notes
Downscale	≤ 7.5	Range 1 or RUN	Obj. V.B.6. Obj.V.C.4 Obj. V.B.5
High	≥ 90/104.6	RUN Mode	Unit Difference: IRM high setpoint is 90 at Unit 2 and 104.6 on Unit 1 and Unit 3
INOP	-HV low (<90v) -Module unplugged -Function switch not in OPERATE -Loss of ±24VDC	RUN Mode	
Detector Wrong Position	Detector Not Full IN	RUN Mode	Obj.V.B.13

2. Scrams — TP-11

Scrams	Setpoint	When Bypassed	Instructor Notes
High-High	≥ 116.4	In RUN Mode	Obj. V.B.7. Obj. V.C.5. Obj.V.D.8
INOP	-HV low (<90v) -Module unplugged -Function switch not in OPERATE -Loss of ±24VDC	In RUN Mode	

F. Controls Provided

1. Panel 9-5

a. Recorder switches select between IRM channels, and APRM/RBM channels have been removed. All units now contain digital recorders, which do not require operation of selector switches. These switches have been removed.

b. Range switches allow operator to select appropriate IRM range to maintain indications between 25 to 75 on 0-125 scale. 0-40 scale is no longer utilized.

BFN Unit 2	Intermediate Range Monitors	2-OI-92A Rev. 0028 Page 14 of 14

Illustration 1
(Page 1 of 1)

IRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
IRM High	> 90 ON 125 SCALE	Rod block unless REACTOR MODE SWITCH in RUN
IRM Inop	A. Module unplugged B. Mode switch **NOT** in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless REACTOR MODE SWITCH in RUN Reactor Scram unless REACTOR MODE SWITCH in RUN
IRM Downscale	< 7.5 on 125 SCALE	Rod block unless IRMs on range 1 unless REACTOR MODE SWITCH in RUN
IRM Detector Wrong Position	detector NOT full in	Rod block unless detector full-in, or REACTOR MODE SWITCH in RUN
IRM High-High	> 116.4 ON 125 SCALE	Reactor Scram unless REACTOR MODE SWITCH in RUN

BFN Unit 2	Unit Startup and Power Operation	2-GOI-100-1A Rev. 0145 Page 88 of 178

5.4 **Withdrawal of Control Rods while in Mode 2 (continued)**

NOTE
Period is measured directly from IRM's, using one of the following methods: 1) MULTIPLY time for 10% power rise by 10.5. 2) MULTIPLY doubling time by 1.445. 3) DIVIDE time for decade rise by 2.3. 4) Directly, time for power to rise from 25 to 68.

[15.2] **RECORD** the following in the Narrative Log:

A. Critical Data

- Period ________ ☐
- Time ________ ☐
- Rod Group ________ ☐
- Rod Number ________ ☐
- Rod Notch ________ ☐

(R) ________ ________ ________
Initials Date Time

B. Recirc Pump 2A and 2B Temperatures using either of the following: (N/A indication for a pump that is OOS and in Single Loop Operation.)

- RECIRC PUMPS DISCH FLOW & TEMP PMP-2A (PMP-2B), CH 3 (CH 4) on 2-XR-68-2/5 on Panel 2-9-4.
- RECIRC PMP A (B) SUCT TEMP 68-6A (68-83A) on ICS.
- RECIRC PMP A (B) DISCHARGE TEMP 68-2 (68-78) on ICS.

________ / ________
2A LOOP °F / 2B LOOP °F

(R) ________ ________ ________
Initials Date Time

Examination Outline Cross-reference:	Level	RO	SRO
295016 Control Room Abandonment / 7	Tier #	1	------
G2.1.28 (10CFR 55.41.7)	Group #	1	------
Knowledge of the purpose and function of major system components and controls.	K/A #	295016G2.1.28	
	Importance Rating	4.1	------

Proposed Question: **# 6**

Which ONE of the following functions can be performed at Backup Control Panel 2-25-32?

A. Close **ALL** MSIVs

B. Operate **ALL** ADS Valves

C. Suppression Chamber Spray

D. Control Reactor Level with HPCI

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: BOTH Inboard and Outboard MSIVs can be closed from Backup Control Panel 2-25-32.

B INCORRECT: Plausible in that Four ADS valves can be controlled from Panel 25-32. Six SRVs (Non-ADS) have disconnect switches at Panel 25-32.

C INCORRECT: Plausible in that indications for RHR are on 2-25-32 and 2-AOI-100-2, "Control Room Abandonment," provides instruction for Suppression Pool Cooling and Shutdown Cooling.

D INCORRECT: Plausible in that Reactor Level can be controlled with RCIC at Pnl 2-25-32.

KA Justification:

The KA is met because it tests the candidate's knowledge of function of major system components associated with Control Room Abandonment procedure and the Backup Control Panel.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s):	2-AOI-100-2, Rev. 54	(Attach if not previously provided)
	OPL171.208, 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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0054 Page 10 of 96

4.2 Unit 2 Subsequent Actions (continued)

CAUTION

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

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

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

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

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

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0054 Page 8 of 96

4.2 Unit 2 Subsequent Actions

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

DIRECT an operator to Unit 2 Auxiliary Instrument Room to perform Attachment 11. ☐

NOTES

1) The following transfers Reactor Pressure Control to Panel 2-25-32 to allow for pressure control while completing the Panel Checklist.

2) Attachment 9, Alarm Response Procedure Panel 2-25-32, provides for any alarms associated with this instruction.

CAUTION

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

2) [NER/C] Operation from Panel 2-25-32 bypasses logic and interlocks normally associated with the components. [GE SIL 326,

[2] **PLACE** the following MSRV control switches in CLOSE/AUTO at Panel 2-25-32:

Switch No.	Description	
2-HS-1-22C	MAIN STM LINE B RELIEF VALVE	☐
2-HS-1-5C	MAIN STM LINE A RELIEF VALVE	☐
2-HS-1-30C	MAIN STM LINE C RELIEF VALVE	☐
2-HS-1-34C	MAIN STM LINE C RELIEF VALVE	☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0054 Page 9 of 96

4.2 Unit 2 Subsequent Actions (continued)

[3] **PLACE** the following MSRV disconnect switches in DISCT at Panel 2-25-32:

Switch No.	Description	
2-XS-1-18	MAIN STM LINE B RELIEF VALVE DISCT	☐
2-XS-1-4	MAIN STM LINE A RELIEF VALVE DISCT	☐
2-XS-1-42	MAIN STM LINE D RELIEF VALVE DISCT	☐
2-XS-1-23	MAIN STM LINE B RELIEF VALVE DISCT	☐
2-XS-1-41	MAIN STM LINE D RELIEF VALVE DISCT	☐
2-XS-1-180	MAIN STM LINE D RELIEF VALVE DISCT	☐

[4] **PLACE** the following MSRV transfer switches in EMERG at Panel 2-25-32:

Switch No.	Description	
2-XS-1-22	MAIN STM LINE B RELIEF VALVE XFR	☐
2-XS-1-5	MAIN STM LINE A RELIEF VALVE XFR	☐
2-XS-1-30	MAIN STM LINE C RELIEF VALVE XFR	☐
2-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 2-25-32:

A. 2-HS-1-22C, MAIN STM LINE B RELIEF VALVE ☐

B. 2-HS-1-5C, MAIN STM LINE A RELIEF VALVE ☐

C. 2-HS-1-30C, MAIN STM LINE C RELIEF VALVE ☐

D. 2-HS-1-34C, MAIN STM LINE C RELIEF VALVE ☐

DISTRACTOR PLAUSIBILITY SUPPORT

9. Trip reactor feed pumps as necessary to prevent tripping on high water level. — Obj. V.B.8; Obj. V.C.5

10. Start the diesel generators. (9-8 Switch starts respective units D/G only)

11. Verify each EECW header has one pump in service.

12. Announce to all plant personnel that the Control Room is being evacuated and all operators are to report to their assigned backup control stations.

13. Obtain hand held radios from the control room.

14. Proceed to the Backup Control Panel (25-32)

F. Subsequent Actions — See AOI-100-2 for details for actions; HU Tools: Procedure Use

1. If rods failed to fully insert and RPS did not deenergize, an operator is directed to pull RPS fuses. However, this is beyond the actual design bases. — Obj V.C.2; See AOI-100-2 Attachment 11

2. Transfer reactor pressure control to Panel 25-32 to allow for pressure control while the rest of the panel checklist is being completed. — Note: System Status prior to abandonment maintained by GOI-300-1 checklists.

3. Before any transfer switch is placed in EMERGENCY, its associated control switch must be verified to be in the proper position. Placing a transfer switch in the EMERGENCY position enables the local control switch, and the device will assume the condition called for by the local control switch. For example, if a transfer switch for an ADS valve is placed in EMERGENCY with the local control switch in OPEN, the ADS valve will open. — Obj. V.B.2; Obj. V.B.3.

   a. Place the transfer switches for the ADS valves, and the disconnect switches for the non-ADS valves in EMERGENCY after making sure the control switches are in the AUTO position. This action disables the Control Room hand switches and the ADS function and is performed to prevent spurious blowdown of the primary system. The other 3 SRVs are disabled by opening their breakers on 250VDC RMOV board 2B(3B). — TP-1; Obj. V.B.7

   Four ADS valves can be controlled from Panel 25-32. Six SRVs (Non-ADS) have only disconnect switches at Panel 25-32. — Obj. V.B.8; Obj. V.B.7

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0054 Page 11 of 96

4.2 Unit 2 Subsequent Actions (continued)

NOTES

1) Attachment 1 provides normal backup control stations and available communications.
2) Attachment 10 provides PAX extensions and locations.

[7] **ESTABLISH** communication with the following personnel and **DIRECT** attachments be completed as follows:

- U-2 Unit Operator complete Attachment 2, Part A. ☐
- U-2 Rx Bldg AUO complete Attachment 3, Part A. ☐
- U-2 Turb Bldg AUO complete Attachment 4, Part A. ☐

[8] Upon completion of attachments, **RE-ESTABLISH** communication using the best available means and continue procedure. ☐

CAUTION

1) RCIC TURBINE STEAM SUPPLY VALVE, 2-FCV-71-8, transfer switch has been placed in EMERGENCY and will **NOT** trip on Reactor Water Level High (+51 inches). Failure to maintain level below this value may result in equipment damage.
2) RCIC will still trip on low suction pressure, high turbine exhaust pressure, mechanical overspeed, and trip push button on pnl 25-32.

[9] **INITIATE** RCIC as follows:

[9.1] **CHECK OPEN** 2-FCV-71-9 RCIC TURB TRIP/THROT VALVE RESET, 2-HS-71-9D At Panel 2-25-32. (Red Light above switch) ☐

[9.2] **PLACE** RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, 2-HS-071-0034C, in OPEN at 250V DC RMOV Bd 2B, compt. 5D. (Unit 2 Turbine Building AUO) ☐

[9.3] **PLACE** RCIC TURB STM SUPPLY VALVE EMER HAND SWITCH, 2-HS-071-0008C, in OPEN at 250V DC RMOV Bd 2C, compt. 4B. (Unit 2 Reactor Building AUO) ☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0054 Page 12 of 96

4.2 Unit 2 Subsequent Actions (continued)

> **NOTE**
>
> RCIC Turbine should start and flow should stabilize at 620 gpm.

[9.4] **CHECK** turbine speed 2100 rpm or above using RCIC TURBINE SPEED, 2-SI-71-42B at Panel 2-25-32. ☐

[9.5] **PLACE** RCIC PUMP MIN FLOW VALVE EMER HAND SWITCH, 2-HS-071-0034C, in CLOSE at 250V DC RMOV Bd 2B, compt. 5D. (Unit 2 Turbine Building AUO) ☐

[9.6] **ADJUST** flowrate as necessary using RCIC SYSTEM FLOW/CONTROL, 2-FIC-71-36B at Panel 2-25-32. ☐

[9.7] **MAINTAIN** Reactor Water Level between +2 and +50 inches using RX WATER LEVEL A & B, 2-LI-3-46A & B at Panel 2-25-32. ☐

> **NOTE**
>
> The following step prevents HPCI operation and automatic opening of HPCI MAIN PUMP MINIMUM FLOW VALVE, 2-FCV-73-30.

[10] At 250V Reactor MOV Bd 2A, **PERFORM** the following:

[10.1] **VERIFY CLOSED** HPCI STEAM SUPPLY VALVE TO TURB FCV-73-16 at Compt. 3D. (MO 23-14). ☐

[10.2] **PLACE** HPCI TURBINE STEAM SUP VLV TRANS, 2-XS-73-16, in EMERG at Compt. 3D. ☐

[10.3] **IF** desired to verify HPCI MIN FLOW BYPASS TO SUPPRESSION CHAMBER VALVE, 2-FCV-73-30, closed prior to opening breaker, **THEN** (Otherwise N/A)

DIRECT operator to verify locally. ☐

[10.4] **PLACE** HPCI MAIN PUMP MIN FLOW VLV FCV-73-30, breaker in OFF Compt. 8D. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0054 Page 16 of 96

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

[15] **INITIATE** RHR Suppression Pool Cooling as follows:

CAUTIONS

1) The RHRSW and EECW Systems are common to all three units. Coordination and communication between the Unit Operators on all three units is required whenever configuration changes to the RHRSW and/or the EECW Systems are made.

2) Communication between 4160V Shutdown Bd A and 480V RMOV Bd 2A is necessary for establishing RHRSW flow and to prevent exceeding 53 amps on RHRSW Pump C2.

[15.1] **PLACE** RHRSW PUMP C2 MOTOR, 0-HS-23-12C, in CLOSE at 4160V Shutdown Bd B, compt. 15, to start RHR SERVICE WATER PUMP C2. ☐

[15.2] **THROTTLE OPEN** RHR HX 2C OUTLET VLV, 2-HS-023-0040C at 480V RMOV Bd 2A, compt. 18C. ☐

[15.3] **WHEN** between 48 and 52 amps on RHR SERVICE WATER PUMP C2, **THEN:**

STOP throttling, RHR HX 2C OUTLET VLV, 2-HS-023-0040C. ☐

[15.4] **VERIFY OPEN** RHR SYSTEM I MINIMUM FLOW VALVE, 2-FCV-74-7, at either of the following:

- 480V RMOV Bd 2D, compt. 5E, RHR SYSTEM I MINIMUM FLOW VLV, **OR**
- Rx Bldg - SW Quad - El 541' local control switch RHR SYSTEM I MINIMUM FLOW VALVE, 2-HS-74-7B. ☐

[15.5] **PLACE** RHR PUMP 2C, 2-HS-074-0016C, in CLOSE to start RHR PUMP 2C at 4160V Shutdown Bd B, compt. 17. ☐

[15.6] **PLACE** RHR SYSTEM I SUPP POOL SPRAY/TEST ISOL VLV, 2-HS-74-57C, in OPEN at 480V RMOV Bd 2A, compt. 11C. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Control Room Abandonment	2-AOI-100-2 Rev. 0054 Page 23 of 96

4.2 Unit 2 Subsequent Actions (continued)

NOTE

Unit 2 and Unit 3 both align RHR Loop I for Shutdown Cooling. **IF** both Unit 2 and Unit 3 Control Rooms have been abandoned, **THEN** coordinate the initiation of Shutdown Cooling such that one unit uses RHR Pump A and the other Unit uses RHR Pump C. This will allow minimum flow protection to be maintained for the RHRSW Pumps and prevent flow adjustments on one Unit from affecting the opposing Units Cooldown rate.

[20] **INITIATE** RHR Shutdown Cooling as follows: (Otherwise N/A)

[20.1] **VERIFY** REACTOR PRESSURE B, 2-PI-3-79, less than 50 psig, at Panel 2-25-32. ☐

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

PERFORM the following: (Otherwise N/A)

[20.2.1] **PLACE** RHR SYSTEM I TEST VLV, 2-HS-074-0059C, in CLOSE, at 480V RMOV Bd 2A, compt. 19C5. ☐

[20.2.2] **PLACE** RHR SYSTEM I SUPP POOL SPRAY/TEST ISOL VLV, 2-HS-74-57C, in CLOSE, at 480V RMOV Bd 2A, compt. 11C. ☐

[20.2.3] **VERIFY CLOSED** RHR SYSTEM I OUTBD INJECTION VLV, 2-HS-74-52C at 480V RMOV Bd 2A, Compt. 2B. ☐

[20.2.4] **PLACE** RHR PUMP C, 2-HS-74-16C, in TRIP to stop RHR PUMP 2C, at 4160V Shutdown Bd B, compt. 1. ☐

[20.2.5] **PLACE** RHR PUMP 2A, 2-HS-074-0005C, in TRIP, at 4160V Shutdown Bd A, compt. 19, to stop RHR PUMP 2A. ☐

Examination Outline Cross-reference:	Level	RO	SRO
295018 Partial or Complete Loss of Component Cooling Water / 8	Tier #	1	------
AA2.01 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER :	K/A #	295018 AA2.01	
• Component temperatures	Importance Rating	3.3	------

Proposed Question: **# 7**

Unit 3 is operating at 100% Reactor Power when the following alarms **AND** indications are received:

- A Partial Loss of Reactor Building Closed Loop Cooling Water occurs.
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH, (3-9-4B, Window 17) is in alarm.
- RWCU Non- Regenerative Heat Exchanger Discharge Temperature is 140° F.

Which ONE of the following describes the effect of this condition, if any, on the operation of the Reactor Water Cleanup (RWCU) Pumps?

A. TRIP immediately due to isolation valve position.

B. TRIP directly due to the high temperature signal.

C. CONTINUE to operate since no trips are received.

D. TRIP after a low flow condition exists for 30 seconds.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** RWCU Non- Regenerative Heat Exchanger Discharge Temperature at 140° F isolates RWCU. When RWCU isolation valve FCV 69-1or 2 Not Full Open, RWCU Pumps trip.

B INCORRECT: B is plausible; identifies misconception about RWCU Pump Trip directly from High Temperature signal. High Temperature initiates a PCIS Isolation. When the Isolation Valve is NOT FULLY OPEN, the RWCU Pump TRIPS. Also, Pump cooling water outlet high temperature (RBCCW) 140°F after a 30-second time delay trips RWCU Pumps.

C INCORRECT: Plausible in that the purpose of the 140°F isolation is to protect ion exchange resin from high temp damage but the F/Ds Design temperature of 150° F has not yet been reached.

D INCORRECT: Plausible in that System Low Flow of 56 gpm with a time delay of 30 seconds will trip RWCU Pumps. With the Isolation Trip coming with valves just off full open, they would cause the trip prior to low flow condition.

KA Justification:

This question satisfies the K/A statement by testing candidates' ability to interpret RWCU Temperatures as they apply to Partial Loss of RBCCW. Partial loss of RBCCW results in RWCU Non- Regenerative Heat Exchanger Discharge Temperature at 140° F which isolates RWCU. When RWCU isolation valve FCV 69-1or 2 Not Full Open, RWCU Pumps trip.

Question Cognitive Level:

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-OI-69 Rev. 79 (Attach if not previously provided)

OPL171.013 Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.013 V.B.3 (As available)

Question Source: Bank # Nine Mile 2 08 #45

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Nine Mile 2 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 3	Reactor Water Cleanup System	3-OI-69 Rev. 0079 Page 15 of 138

3.8 **RWCU Isolation Signals**

A. Reactor water level low (LEVEL 3).

B. Non-regenerative heat exchanger outlet high temperature 140°F.

C. RWCU Pump Room 3A high temperature 148°F.

D. RWCU Pump Room 3B high temperature 148°F.

E. Main Steam Tunnel/RWCU Piping high temperature 197°F.

F. RWCU System Pipe Trench 131°F.

G. RWCU Heat Exchanger Room Pipe Chase Area high temperature 166°F.

H. RWCU Heat Exchanger Room high temperature 139°F.

I. Standby Liquid Control system initiation.

CORRECT ANSWER AND DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 3	Reactor Water Cleanup System	3-OI-69 Rev. 0079 Page 14 of 138

3.6 **Pumps (continued)**

C. RWCU is required to be operated with the following restrictions with reactor pressure ≤ 50 psig (modes 2 or 3) or any time the unit is in mode 4, mode 5, or defueled:

1. One pump in operation, pump can be operated to its maximum flow capacity.

2. Two pumps in operation, maximum flow limited to ≤ 100 gpm per pump (200 gpm total).

D. Leaving an idle pump pressurized can damage the seals by erosion paths across the seal faces.

E. If conditions listed below are satisfied, the Unit 3 RWCU pumps may be operated with 0 gpm seal water flow, after pump start. However, RWCU pump operation with 0 gpm seal water flow will reduce seal life and the seal will most likely need to be replaced before operating the seal at its design parameter.

1. CRD pumps are not available and RWCU seal water is supplied from the CS&S System.

2. Reactor Vessel is at atmospheric pressure.

3. RWCU seal flow is 1.8 to 2 gpm prior to pump start.

3.7 **RWCU Pump Trip Signals**

A. Low flow 56 gpm (30 second time delay if control switch in NORMAL after start).

B. Cooling water high temperature 140°F (7 sec. time delay).

C. RWCU INBD SUCT ISOLATION VALVE, 3-FCV-69-1 not full open.

D. RWCU OUTBD SUCT ISOLATION VALVE, 3-FCV-69-2 not full open.

E. RWCU RETURN ISOLATION VALVE, 3-FCV-69-12 fully closed.

CORRECT ANSWER AND DISTRACTOR PLAUSIBILITY SUPPORT

OPL171.013
Revision 18
Page 29 of 47

(4) High temperature at the outlet of the NRHX (140°F, TIS-69-11) to protect the ion exchange resin from damage due to high temperature. An alarm is provided on Panel 9-4 from TIC-69-10.

Not a Safety function.

(5) Loss of RPS A will result in an inboard and outboard Group 3 (RWCU) isolation. Loss of RPS B will result in an outboard Group 3 (RWCU) isolation.

b. RWCU Recirc Pumps Trips

Obj. V.B.5
Obj. V.D.6
Obj. V.E.10

(1) RWCU isolation valve FCV 69-1or 2 not full open.

(2) RWCU return isolation valve FCV 69-12 full closed.

UNIT DIFFERENCE
TACF 2-08-001-069 disables 2B pump trip on 2-FCV-69-12 closure

(3) Pump flow 56 gpm for 30 seconds with the control switch in NORMAL-AFTER-START position.

(4) Pump cooling water outlet high temperature (RBCCW) 140°F after a 30-second time delay, if the control switch is in the NORMAL-AFTER-START position.

Right unit/train/component

(5) 480V Load Shed Logic

c. Filter/Demineralizers

(1) Holding pump auto starts at 40 gpm decreasing

(2) Alarm on high differential pressure at 25 psid; alarm on resin trap differential pressure of 20 psid.

(3) Automatic isolation at 25 psid (F/D); 20 psid resin trap

NOTE: Will only close the effluent valves

DISTRACTOR PLAUSIBILITY SUPPORT

4. Filter/Demineralizers (F/Ds) — Obj. V.B.2, Obj. V.C.2., Obj. V.E.3.d

a. Are used to maintain water purity by mechanical and chemical filtration. They remove insoluble solid particles and dissolved solids from the water. Each of the units is of the pressure precoat type, which uses finely ground mixed ion exchange medium. The F/Ds operate in parallel at 50% of the total system capacity. Design temperature (°F) is 150°F. Water to the F/Ds should be maintained less than 130°F to maximize resin efficiency and to avoid resin damage. Water Temperature of 150° - 200°F reduces the impurity removal capacity. Temperature >200°F causes the Powdex to decompose. Resin traps are provided to prevent carryover into the reactor system of filter or resin material due to filter element failure. — Monitor Critical Plant Parameters

b. Resin introduction into the reactor coolant can cause conductivity increase, pH decrease, and sulfate concentration increase (which propagates pitting corrosion and intergranular stress corrosion cracking). — See Plant/Industry Experience (Section X. G)

c. Holding pumps are provided to maintain the filter charged until the unit is in service. Pump will automatically start if flow through the filter drops to 40 gpm to prevent the precoat from dropping off the filter elements. — TP-3; PCR 08003930 added steps to remove Demins from service prior to RPS Bus xfer

d. A Flow control valve maintains constant flow rate through each F/D for varying pressure drops. This is set at local F/D control station. (NOTE: Flow control valves are normally operated in manual.). Flow should never exceed 135 gpm/unit (170 gpm/unit on Unit 1/3) Test results has shown that U-1 will not exceed 160 gpm. — **UNIT DIFFERENCE**

Nine Mile Point Unit 2 Reactor Operator Written Examination
Draft Submittal

RO 45	Tier 1	K/A Number 295018	Statement AA2.01	IR 3.3	Origin N	Source Question NA
LOK H	Grp 1	10 CFR 55.41(b) 7	LOD (1-5)	Reference Documents ARP 602319		
Ability to determine and/or **interpret** the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Component temperatures						

QUESTION 45

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

- Loss of Reactor Building Closed Loop Cooling Water occurs
- Annunciator 602319, RWCU FILTER DEMIN INLET TEMP HI-HI alarms

Which one of the following describes the affect of this condition, if any, on the operation of the Reactor Water Cleanup Pumps?

A. CONTINUE to operate since no trips are received.

B. TRIP immediately due to isolation valve position.

C. TRIP directly due to the high temperature signal.

D. TRIP after a low flow condition exists for 15 minutes.

Correct Answer: B When 602319 alarms, at WCS*MOV112, CLEANUP SUCT OUTBOARD ISOL VLV closes. When the valve is NOT FULLY OPEN, the WCS Pumps TRIP immediately.

Plausible Distractors:
A is plausible; would be true for RWCU F/D Inlet (NRHX Outlet) temperature below 140°F.
C is plausible; identifies misconception about RWCU Pump Trip directly from High Temperature signal. High Temperature initiates a PCIS Isolation. When the Isolation Valve is NOT FULLY OPEN, the RWCU Pump TRIPS.
D is plausible; would be true for valve closures OTHER THAN WCS*MOV112. A Low Flow condition would develop which initiates a time delayed RWCU Pump TRIP.

Examination Outline Cross-reference:	Level	RO	SRO
295019 Partial or Total Loss of Inst. Air / 8	Tier #	1	------
AK3.03 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : • Service air isolations: Plant-Specific	K/A #	295109AK3.03	
	Importance Rating	3.2	------

Proposed Question: **# 8**

Control Air Header Pressure is lowering due to a rupture in the system.

Which ONE of the following identifies the **HIGHEST** Control Air Pressure that will result in Service Air Isolation Valve, 0-FCV-33-1, closing **AND** the reason?

A. 30 psig;
To isolate non-essential Service Air loads.

B. 30 psig;
Due to insufficient air pressure to keep the valve open.

C. 50 psig;
To isolate non-essential Service Air loads.

D. 50 psig;
Due to insufficient air pressure to keep the valve open.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: 1st part correct – See B Explanation. 2nd Part incorrect – See C Explanation.

B **CORRECT:** Service air supply valve from control air header (0-FCV-33-1). The valve automatically opens if control air pressure falls to 85 psig and closes at 30 psig (due to insufficient air pressure to keep the valve open).

C INCORRECT: Recognizable pressure associated with loss of Control Air as the pressure that Condensate Demin Bypass Valve Fails open. Plausible in that it is logical to isolate non-essential Service Air loads with a loss of Control Air similar to RBCCW Sectionalizing Valve closing on low header pressure to isolate non-essential RBCCW loads.

D INCORRECT: 1st part incorrect – see C Explanation. 2nd Part Correct – See B Explanation.

KA Justification:

This question satisfies the K/A statement by testing knowledge of the reason and the setpoint for Service air isolation Valve, 0-FCV-33-1, closing as a result of a rupture in the Control Air System and lowering pressure.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 0-OI-32 Rev 127 (Attach if not previously provided)
OPL171.054 Rev 15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.054 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:

(b) Service air supply valve from control air header (0-FCV-33-1). Can be operated from panel 1-9-20 and/or 3-9-20. The switch positions are **CLOSE-AUTO-OPEN**, with position indication lamps just above each control switch. The valve automatically opens if control air pressure falls to 85 psig and closes at $\approx$ 30 psig (due to insufficient air pressure to keep the valve open).

(c) A manual bypass valve can be utilized if 0-FCV-33-1 should fail to open.

c. System Annunciators

ARP usage

(1) AIR COMPRESSOR ABNORMAL alarm on PANEL 1-9-20 ONLY. Any alarm annunciated on panel 0-LPNL-925-0118

(2) SERVICE AIR XTIE VALVE OPEN alarm PANEL 1-9-20 and 3-9-20 (PCV 33-1 opens at 85 psig)

(3) CONTROL AIR PRESSURE LOW alarm on each units 9-20 panel at 70 psig

(4) CONTROL AIR DEW POINT HIGH at -20°F on 2-9-20 panel and -28.9°C on 1-9-20 panel.

(5) Two local alarms annunciate to indicate primary controller failure or backup controller failure due to loss of power to controller or software failure.

d. Control Room Indication Panel 9-20 Unit Control Air Header Pressure

e. **G** Air Compressor amps are indicated on panel 1-9-20.

f. **A-D** Control Air Compressor Normal Operating Parameters

(1) Operator Setpoints screen

(a) Lead Offline Pressure 60-128 psig

(b) Lead Online pressure 50-118 psig

(c) Lag Offset 0-45 psig

(d) Load time delay 0-60 seconds

(e) Condensate Interval 60-360 seconds

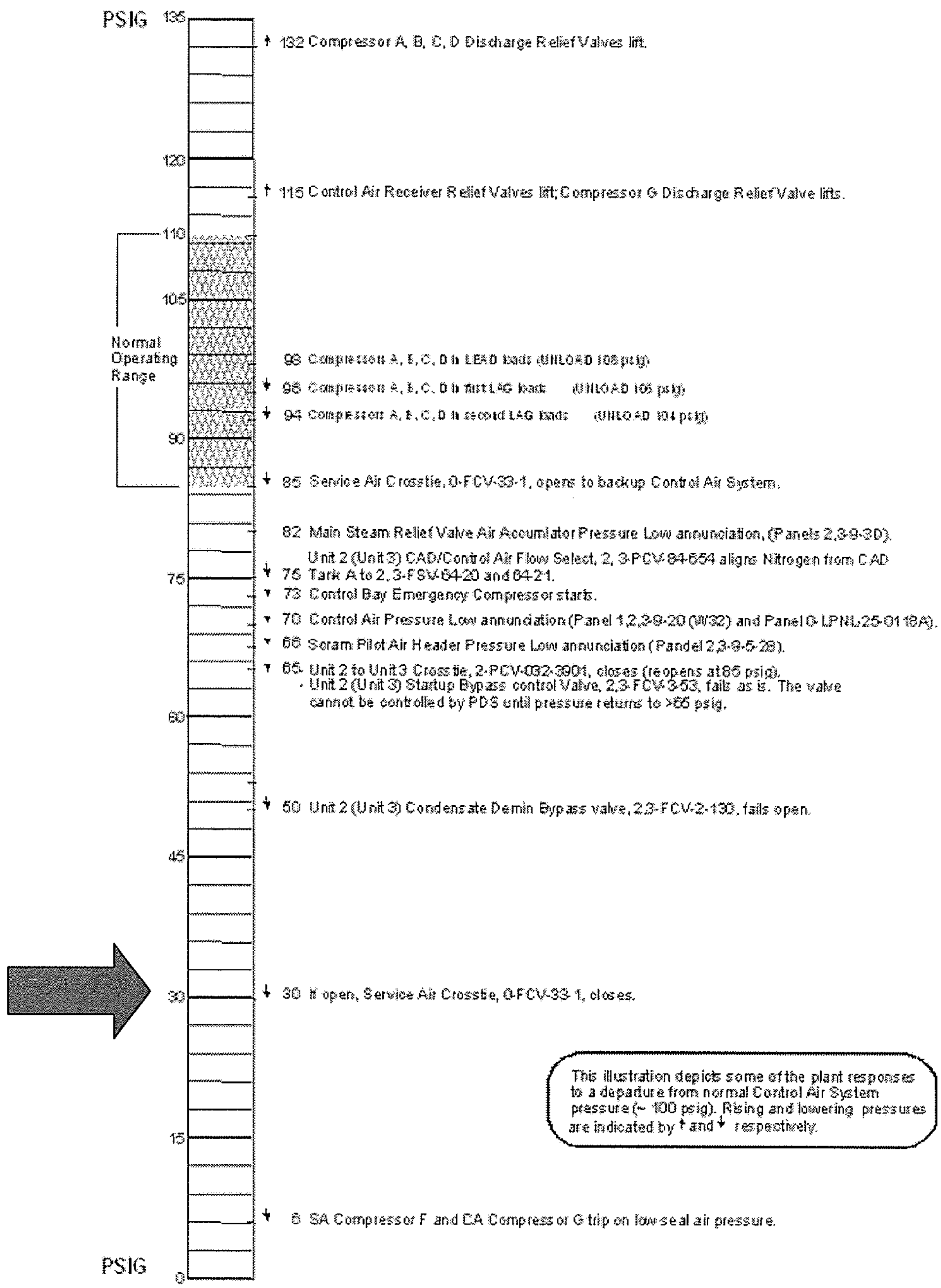
(f) Condensate discharge time 2-20 seconds

(g) Max. First State Temperature 300-440°F

BFN Unit 0	Control Air System	0-OI-32 Rev. 0127 Page 67 of 113

Illustration 1
(Page 1 of 1)

Control Air System Pressure Spectrum

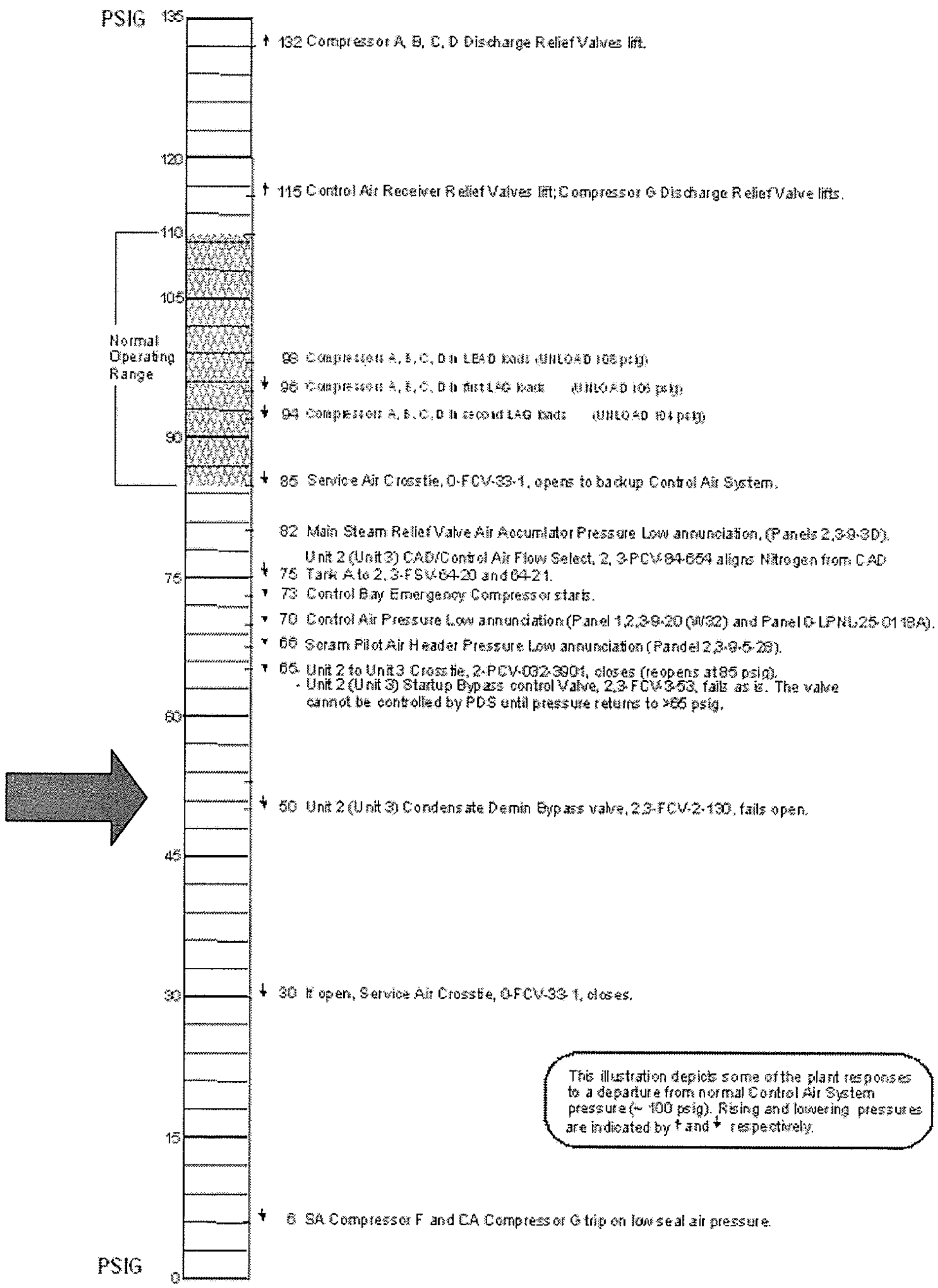


PLAUSIBILITY SUPPORT

BFN Unit 0	Control Air System	0-OI-32 Rev. 0127 Page 67 of 113

Illustration 1
(Page 1 of 1)

Control Air System Pressure Spectrum



PLAUSIBILITY SUPPORT

6. Motor Operated Valves (MOVs)

a. The spare RBCCW Pump has 6 MOVs which can be used to line it up to Unit 1, 2, or 3.

Right Unit, Right train, Right Component

(1) These MOVs are controlled from Panel 9-4, Unit 1 Control Room.

INPO SER 30-05 Attention to Detail and Intrusiveness

(2) These MOVs are interlocked to permit alignment of the spare pump to only one unit at a time, to prevent cross-tying of the Unit RBCCW Systems.

ROTORK valves have open, closed, and mid position. Mid does not indicate a % of valve open or closed

b. FCV-70-48 controls the RBCCW supply to the non-essential equipment loop.

(Referred to as the **SECTIONALIZING** valve)

(1) U1/2 FCV-70-48 automatically closes on:

Obj. V.B.4

(a) Initiation of U1/2 480V Load Shed Logic.(Loss of normal AC power with any U1/2 diesel generator tied to a U1/2 4kV shutdown board as a sole source, in conjunction with an accident signal)

Obj. V.C.1
Obj. V.D.5

(CAS signal 2.45 psig DW press with 450 psig Rx press, or -122" Level)

UNIT DIFFERENCE Unit 3, 70-48 closes on low pressure as a result of the pump trips from load shed.

(b) All three units FCV-70-48 close on low RBCCW supply header pressure of 57 psig,

(corresponds to an actual header pressure of 50 psig)

Examination Outline Cross-reference:	Level	RO	SRO
295021 Loss of Shutdown Cooling / 4	Tier #	1	------
AK2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following:	K/A #	295021AK2.01	
• Reactor water temperature	Importance Rating	3.6	------

Proposed Question: **# 9**

Unit 3 is in Mode 4 with the following conditions:

- Reactor Level band is (+) 78 inches to support testing.
- **ALL** Reactor Recirc **AND** RWCU Pumps are isolated and tagged out.
- 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.

If Reactor Coolant Stratification occurs, it is indicated by __**(2)**__ .

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

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

C. **(1)** is
(2) Differential temperatures of 40°F between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle

D. **(1)** is **NOT**
(2) Differential temperatures of 40°F between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle

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 Reactor pressure > 0 psig with any Reactor Coolant temperature indication < 212°F

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 Differential temperatures of **>** 50°F between Reactor Vessel Bottom Head **AND** any Reactor Vessel Feedwater Nozzle.

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

KA Justification:

The KA is met because to successfully answer the question, the candidate must demonstrate knowledge of the interrelationship between loss of shutdown cooling and Reactor Water Temp.

Question Cognitive Level:

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 19

Proposed references to be provided to applicants during examination: NONE

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

Question Source: Bank #

Modified Bank # BFN 1006 #9 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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. 0019 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]

[6] **PLOT** heatup/cooldown rate as necessary. **REFER TO** 3-SR-3.4.9.1(1). ☐

[7] **REQUEST** the SRO to **ESTIMATE** the following times at least once per shift until a method of decay heat removal is restored:

[7.1] **DETERMINE** the time since shutdown. ☐

[7.2] **DETERMINE** the current RPV heat-up rate from 3-SR-3.4.9.1(1), or, if reactor coolant stratification is suspected, use Illustration 1.

[7.2.1] **IF** additional information is required to determine the heat-up rates, **THEN**

NOTIFY Reactor Engineer. ☐

[7.3] **DETERMINE** the reactor coolant temperature or use the last valid reactor coolant temperature available. ☐

BFN Unit 3	Loss of Shutdown Cooling	3-AOI-74-1 Rev. 0019 Page 13 of 26

4.2 Subsequent Actions (continued)

CAUTION

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

[13] [NER/C] **IF** forced circulation has been lost **AND** vessel cavity is less than 80 inches, **THEN**

PERFORM the following: (Otherwise N/A)

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

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

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

[14] **IF** the affected loop of RHR cannot be placed back in Shutdown Cooling, **THEN**

PLACE the alternate loop of RHR in Shutdown Cooling. **REFER TO** 3-OI-74. (Otherwise N/A) ☐

[15] **IF** no Unit 3 RHR loop can be placed in Shutdown Cooling, **THEN**

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. (Otherwise N/A) ☐

[16] **IF** no RHR loops can be placed in service, **THEN**

VERIFY a Recirculation Pump in service. **REFER TO** 3-OI-68. (Otherwise N/A) ☐

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

PERFORM 3-SR-3.4.9.5-7, RPV Head Temperature Monitoring. (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:

OPL171.074
Revision 8
Page 8 of 16

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.

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 3	Loss of Shutdown Cooling	3-AOI-74-1 Rev. 0019 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]

[6] **PLOT** heatup/cooldown rate as necessary. **REFER TO** 3-SR-3.4.9.1(1). ☐

[7] **REQUEST** the SRO to **ESTIMATE** the following times at least once per shift until a method of decay heat removal is restored:

[7.1] **DETERMINE** the time since shutdown. ☐

[7.2] **DETERMINE** the current RPV heat-up rate from 3-SR-3.4.9.1(1), or, if reactor coolant stratification is suspected, use Illustration 1.

[7.2.1] **IF** additional information is required to determine the heat-up rates, **THEN**

NOTIFY Reactor Engineer. ☐

[7.3] **DETERMINE** the reactor coolant temperature or use the last valid reactor coolant temperature available. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

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.

BROWNS FERRY 1006 NRC #9

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

	RO	SRO
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.

If Reactor Coolant Stratification occurs, it 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

Examination Outline Cross-reference:	Level	RO	SRO
295023 Refueling Acc / 8	Tier #	1	------
AA1.03 (10CFR 55.41.7)	Group #	1	------
Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS:	K/A #	295023AA1.03	
• Fuel handling equipment	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 in accordance with Refueling AOIs?

A. Verify Secondary Containment is intact.

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

C. Raise the fuel bundle from the core location **AND** traverse to the area of the cattle chute.

D. If SLC is operable place SLC PUMP 1A/1B, 1-HS-63-6A control switch in START A **OR** START B.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: This is plausible because it 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 **CORRECT:** In order to answer this question correctly the candidate must determine the appropriate condition and Immediate Action required by 1-AOI-79-2.

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

KA Justification:

This question satisfies the *K/A* 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.

Question Cognitive Level:

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. Candidate must recognize that inadvertent criticality has occurred based on indications and select appropriate immediate actions.

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 # BFN 1006 #10

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam BFN 2010

(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. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

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. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

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.

0610 NRC Exam
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:	Level	RO	SRO
295024 High Drywell Pressure / 5	Tier #	1	------
EK3.08 (10CFR 55.41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE :	K/A #	295024EK3.08	
• Containment spray: Plant-Specific.	Importance Rating	3.7	------

Proposed Question: **# 11**

Unit 2 was at 100% Reactor Power when a spurious Group I Isolation occurred. The pressure transient caused a small-break LOCA to occur inside the Drywell.

Which ONE of the following describes the basis for actions with respect to 12 psig Suppression Chamber Pressure?

A. Drywell sprays must be initiated prior to this pressure to prevent opening the Suppression Chamber to Reactor Building vacuum breakers **AND** de-inerting the containment.

B. Drywell sprays must be initiated above this pressure because almost **ALL** of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus **AND** chugging is possible.

C. Above this pressure indicates that almost **ALL** of the nitrogen **AND** other non-condensable gases in the torus have been transferred to the drywell air space **AND** Suppression Chamber Sprays will be ineffective.

D. Above this pressure indicates that almost **ALL** of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus so initiating Drywell Sprays may result in containment failure.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This is plausible because initiation of DW sprays at high SC pressure could reduce pressure low enough to open the Suppression Chamber to Reactor Building Vacuum Breakers. However, this is part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

B **CORRECT**: Drywell sprays must be initiated above this pressure because almost all of the nitrogen **AND** other non-condensable gases in the drywell have been transferred to the torus **AND** chugging is possible. The basis for the Pressure Suppression Pressure Limit of 12 psig Suppression Chamber pressure.

C INCORRECT: This is plausible if the LOCA occurred inside the Suppression Chamber and NOT the Drywell as given in the stem.

D INCORRECT: This is plausible because initiating SC sprays with high temperature non-condensable gases in the SC will result in evaporative cooling and a rapid pressure drop. However, the SC to DW vacuum relief system is capable of compensating for this pressure drop. This is also part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

KA Justification:

The KA is met because it tests knowledge of the reasons for Drywell Spray as it applies to High Drywell Pressure.

Question Cognitive Level:

This question is rated as Memory due to the requirement to recall or recognize discrete bits of information.

Technical Reference(s): EOIPM Section 0-V-D Rev. 0 (Attach if not previously provided)

OPL171.203 Rev. 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 V.B.5 (As available)

Question Source: Bank # BFN 0610 #62

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 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:

1. Step PC/P-2

   a. This decision step has the operator evaluate present and future performance of venting the drywell or suppression chamber using CAD and SGTS, in relation to the current value and trend of drywell and suppression chamber pressure, to determine if primary containment pressure can be maintained below the high drywell pressure scram setpoint. — Questioning Attitude

   b. If primary containment pressure can be maintained below the high drywell pressure scram setpoint, the operator returns to Step PC/P-1 until EOI-2 can be exited or primary containment pressure cannot be maintained below the high drywell pressure scram setpoint.

   c. If containment pressure control systems are unable to maintain primary containment pressure below the high drywell pressure scram setpoint, then further control actions, beginning at Step PC/P-3, need to be addressed.

2. Step PC/P-3 — Obj.V.B.5, V.C.5

   a. This before decision step has the operator evaluate present and future performance of venting the drywell or suppression chamber using CAD and SGTS, in relation to the current value and trend of suppression chamber pressure, to determine if suppression chamber pressure can be maintained below 12 psig (Suppression Chamber Spray Initiation Pressure).

OPL171.203
Revision 7
Page 26 of 7

b. Engineering calculations have determined that if suppression chamber pressure exceeds 12 psig, Suppression Chamber Spray Initiation Pressure, there is no assurance that chugging will be prevented at downcomer openings of the drywell vents.

c. Suppression Chamber Spray Initiation Pressure is defined to be the lowest suppression chamber pressure that can occur when 95% of the noncondensables in the drywell have been transferred to the airspace of the suppression chamber.

d. Scale model tests have demonstrated that chugging will not occur so long as the drywell atmosphere contains at least 1% noncondensables.

e. To prevent the occurrence of conditions under which chugging may happen, the Suppression Chamber Spray Initiation Pressure is conservatively defined by specifying 5% noncondensables.

f. Chugging is the cyclic condensation of steam at downcomer openings of the drywell vents. Chugging occurs when steam bubbles collapse at the exit of the downcomers. The rush of water that fills the void (some of which is drawn up into the downcomer pipe) induces a severe stress at the junction of the downcomer and vent header.

Obj.V.B.6a
Obj.V.C.6a
SER 03-05

g. Repeated application of this stress can cause these joints to experience fatigue failure, thereby creating a pathway that bypasses the pressure suppression function of primary containment.

SER 03-05

h. Subsequent steam that discharges through the downcomers would then exit through fatigued cracks and directly pressurize the suppression chamber air space, rather than discharging to and condensing in the suppression pool.

3. Step PC/P-4

a. This action step directs the operator to manually place those pumps not required to assure adequate core cooling, in the suppression chamber spray mode. Because this step is prioritized with a miniature before decision step PC/P-3 symbol, this action must be performed before suppression chamber pressure reaches 12 psig, Suppression Chamber Spray Initiation Pressure.

EOI Appendix 17C provides step-by-step guidance for operating RHR in the suppression chamber spray mode.

b. This step only addresses initiation of suppression chamber sprays. Instructions for terminating suppression chamber spray operation, once initiated, are provided by Step PCC-2.

4. Step PC/P-5

a. This contingent action step requires the operator to wait until the stated condition has been met before continuing in EOI-2. Subsequent actions in this section of EOI-2 will not be performed until suppression chamber pressure exceeds Suppression Chamber Spray Initiation Pressure.

b. Although operation of suppression chamber sprays by itself will not prevent chugging, the requirement to wait to initiate drywell sprays until reaching Suppression Chamber Spray Initiation Pressure assures that suppression chamber spray operation is attempted before operation of drywell sprays.

DISCUSSION: STEP PC/P-3

This before decision step has the operator evaluate present and future performance of venting the drywell or suppression chamber using CAD and SGTS, in relation to the current value and trend of suppression chamber pressure, to determine if suppression chamber pressure can be maintained below Suppression Chamber Spray Initiation Pressure. The before decision step requires that this determination and subsequent actions be performed before suppression chamber pressure reaches Suppression Chamber Spray Initiation Pressure.

Engineering calculations have determined that if suppression chamber pressure exceeds <A.65>, Suppression Chamber Spray Initiation Pressure, there is no assurance that chugging will be prevented at downcomer openings of the drywell vents. This value is rounded off in the EOI to use the closest, most conservative value that can be accurately determined on available instrumentation.

Suppression Chamber Spray Initiation Pressure is defined to be the lowest suppression chamber pressure that can occur when 95% of the noncondensables in the drywell have been transferred to the airspace of the suppression chamber. Scale model tests have demonstrated that chugging will not occur so long as the drywell atmosphere contains at least 1% noncondensables. To prevent the occurrence of conditions under which chugging may happen, the Suppression Chamber Spray Initiation Pressure is conservatively defined by specifying 5% noncondensables.

Chugging is the cyclic condensation of steam at downcomer openings of the drywell vents. Chugging occurs when steam bubbles collapse at the exit of the downcomers. The rush of water that fills the void (some of which is drawn up into the downcomer pipe) induces a severe stress at the junction of the downcomer and vent header. Repeated application of this stress can cause these joints to experience fatigue failure, thereby creating a pathway that bypasses the pressure suppression function of primary containment. Subsequent steam that discharges through the downcomers would then exit through fatigued cracks and directly pressurize the suppression chamber air space, rather than discharging to and condensing in the suppression pool.

Although operation of suppression chamber sprays by itself will not prevent chugging, initiation before reaching the Suppression Chamber Spray Initiation Pressure assures that this method of primary containment pressure reduction is attempted before the operation of drywell sprays is directed in subsequent steps of the procedure.

If suppression chamber pressure can be maintained below <A.65>, Suppression Chamber Spray Initiation Pressure, the operator returns to Step PC/P-1. If suppression chamber pressure cannot be maintained below Suppression Chamber Spray Initiation Pressure, the operator continues at Step PC/P-4 before suppression chamber pressure actually reaches Suppression Chamber Spray Initiation Pressure.

Browns Ferry 0610 #62

1. RO 295020AK3.08 1

Unit 2 was at 100% rated power when a spurious Group I Isolation occurred. The pressure transient caused a small-break LOCA to occur inside the Drywell.

Which ONE of the following describes the basis for actions with respect to 12 psig Suppression Chamber Pressure?

A. Drywell sprays must be initiated prior to this pressure to prevent opening the Suppression Chamber to Reactor Building vacuum breakers and de-inerting the containment.

B. Drywell sprays must be initiated above this pressure because almost all of the nitrogen and other non-condensible gases in the drywell have been transferred to the torus and chugging is possible.

C. Above this pressure indicates that almost all of the nitrogen and other non-condensible gases in the torus have been transferred to the drywell air space and Suppression Chamber Sprays will be ineffective.

D. Above this pressure indicates that almost all of the nitrogen and other non-condensible gases in the drywell have been transferred to the torus so initiating Drywell Sprays may result in containment failure.

Answer: B

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

1. The basis for the Pressure Suppression Pressure Limit of 12 psig Suppression Chamber pressure.

B - correct:

A - incorrect: This is plausible because initiation of DW sprays at high SC pressure could reduce pressure low enough to open the Suppression Chamber to Reactor Building Vacuum Breakers. However, this is part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

C - incorrect: This is plausible if the LOCA occurred inside the Suppression Chamber and NOT the Drywell as given in the stem.

D - incorrect: This is plausible because initiating SC sprays with high temperature non-condensable gases in the SC will result in evaporative cooling and a rapid pressure drop. However, the SC to DW vacuum relief system is capable of compensating for this pressure drop. This is also part of the bases for the Drywell Spray Initiation Pressure Limit Curve #5.

Examination Outline Cross-reference:	Level	RO	SRO
295025 High Reactor Pressure / 3	Tier #	1	
EA1.04 (10CFR 55.41.7)	Group #	1	
Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: • HPCI: Plant-Specific	K/A #	295025EA1.04	
	Importance Rating	3.8	

Proposed Question: **# 12**

Unit 1 HPCI is in operation in Pressure Control Mode per 1-EOI Appendix 11C, "ALTERNATE RPV PRESSURE CONTROL SYSTEMS HPCI TEST MODE."

- Reactor Pressure is 1050 psig
- 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, is in Automatic

Which ONE of the following completes the statement below?

To lower Reactor Pressure, the operator is required to use __**(1)**__ **AND** __**(2)**__ in accordance with 1-EOI Appendix 11C.

A. **(1)** 1-FCV-73-36, HPCI/RCIC CST TEST VLV,
(2) throttle it in the CLOSE direction

B. **(1)** 1-FCV-73-36, HPCI/RCIC CST TEST VLV,
(2) throttle it in the OPEN direction

C. **(1)** 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL,
(2) LOWER the setpoint

D. **(1)** 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL,
(2) RAISE the setpoint

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that 1-FCV-73-35, HPCI PUMP CST TEST VLV is adjusted in accordance with 1-EOI Appendix 11C to control HPCI pump discharge pressure at or below 1100 psig.

B INCORRECT: See Explanation A.

C INCORRECT: Second Part is incorrect – Plausibility based on misconception that lowering setpoint will result in lowering Reactor Pressure.

D **CORRECT:** Both parts are correct – Per 1-EOI Appendix 11C, ADJUST 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller to control RPV pressure. Raising set point will lower reactor pressure, per the appendix..

KA Justification:

The KA is met because the question tests the candidates' ability to operate and monitor HPCI in pressure control mode as it applies to high Reactor Pressure.

Question Cognitive Level:

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-EOI Appendix 11C Rev. 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.042 V.B.10 (As available)

Question Source:			
	Bank #	Hatch 09 #52	
	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:

BFN UNIT 1	ALTERNATE RPV PRESSURE CONTROL SYSTEMS HPCI TEST MODE	1-EOI APPENDIX-11C Rev. 1 Page 3 of 3

6. **VERIFY** proper HPCI minimum flow valve operation as follows:

   a. IF ....HPCI flow is above 1200 gpm,

      THEN..........**VERIFY CLOSED** 1-FCV-73-30, HPCI PUMP MIN FLOW VALVE. _____

   b IF ....HPCI flow is below 600 gpm,

      THEN..........**VERIFY OPEN** 1-FCV-73-30, HPCI PUMP MIN FLOW VALVE. _____

7. **THROTTLE** 1-FCV-73-35, HPCI PUMP CST TEST VLV, to control HPCI pump discharge pressure at or below 1100 psig. _____

8. **ADJUST** 1-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller to control RPV pressure. _____

9. IF ....HPCI injection to the RPV becomes necessary,

   THEN..........**ALIGN** HPCI to the RPV as follows:

   a. **OPEN** 1-FCV-73-44, HPCI PUMP INJECTION VALVE. _____

   b. **THROTTLE** 1-FCV-73-35, HPCI PUMP CST TEST VLV, to control injection. _____

   c. **GO** TO EOI Appendix 5D. _____

LAST PAGE

HATCH 09 #52

HLT 4 NRC Exam

52. 295025G2.1.23 001

31EO-EOP-107-2, "ALTERNATE RPV PRESSURE CONTROL" is in progress.

- The HPCI system is being used to control reactor pressure.
- The 2E41-R612, "HPCI flow controller," is in automatic, with the setpoint at 3000 gpm.

To INCREASE the reactor cooldown rate (CDR), the operator is required to use _______ and ______ IAW 31EO-EOP-107-2.

A.✓ 2E41-R612, "HPCI flow controller,"
RAISE the setpoint

B. 2E41-R612, "HPCI flow controller,"
LOWER the setpoint

C. 2E41-F011, "Test to CST VLV,"
throttle it in the CLOSE direction

D. 2E41-F011, "Test to CST VLV,"
throttle it in the OPEN direction

Description; While HPCI is in pressure control mode with the controller in automatic, per procedure the cooldown rate (CDR) is controlled by throttling 2E41-F008, "Test to CST VLV". 31EO-EOP-107-2 specifies that throttling F008 in the closed direction will increase the CDR if the controller is in auto. If the controller is in Manual, throttling F008 will have minimal effect on CDR. In Manual the CDR is increased by increasing the controller output and decreased by reducing the controller output.

This concept has been difficult for some students to master (which direction to throttle the valve to increase CDR).

A. **Correct;** see description above.

B. **Incorrect,** 1st part is correct, 2nd is not correct, opening the valve will reduce the CDR. **Plausible** if the candidate assumes that opening the valve results in more water flow, which would require more steam flow.

C. **Incorrect,** 1st part is not correct (wrong valve). 2nd part is correct. **Plausible** if the student does not remember which valve is throttled to control CDR. The valves (F008 & F011) are in series and have the same name.

D. **Incorrect,** 1st part is not correct (wrong valve). 2nd part is not correct. **Plausible** if the candidate assumes that opening the valve results in more water flow, which would require more steam flow.

Examination Outline Cross-reference:	Level	RO	SRO
295026 Suppression Pool High Water Temp. / 5	Tier #	1	------
EK2.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following:	K/A #	295026EK2.02	
• Suppression pool spray: Plant-Specific	Importance Rating	3.6	------

Proposed Question: **# 13**

Unit 3 has experienced a LOCA **AND** the following conditions exist:

- Suppression Chamber Pressure is 5 psig
- Suppression Pool level is 14.5 feet
- Drywell Pressure is 7.5 psig
- Suppression Pool Temperature is 200° F
- **BOTH** RHR Loop I Pumps are in Suppression Chamber / Drywell Spray with Loop flow of 11,500 gpm
- Core Spray Pump 2A flow is 4000 gpm
- **NO** other ECCS Pumps are running

Based on the above conditions, which ONE of the following identifies the ECCS Pump(s), if any, that has (have) sufficient NPSH for continued operation?

[REFERENCE PROVIDED]

A. **NONE**

B. RHR Loop I Pumps **ONLY**

C. Core Spray Pump 2A **ONLY**

D. Core Spray Pump 2A **AND** RHR Loop I Pumps

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausible in that If RHR is plotted for the loop flow and not the pump flows, it would be in the unsafe region of Curve 2 making this the correct answer.

B **CORRECT:** Operating point for RHR Loop I Pumps is within the safe region of Curve 2.

C INCORRECT: Core Spray Pump 2A above the safe region of NPSH Limits Curve 1. Plausible in that if Drywell pressure is used to plot Curve 1, Pump would be operating in the safe region of curve 1 and if RHR is Plotted for Loop flow, it would be in the Unsafe of Curve 2.

D INCORRECT: RHR Loop I Pumps have adequate NPSH. However, CS Pump 2A does not. Plausible in that if Drywell pressure is used to plot both Curves, all Pumps would be operating in the safe regions and this would be the correct answer.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the interrelationship between High Suppression Pool Temperature and RHR Spray Operation.

Question Cognitive Level:

This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question and use a reference to solve a problem.

Technical Reference(s): 3-EOI-1 Curve 1 / Curve 2 Rev. 8 (Attach if not previously provided)
OPL171.201 Rev. 7

Proposed references to be provided to applicants during examination: CS NPSH Limit Curve 1
RHR NPSH Limit Curve 2

Learning Objective: OPL171.201 V.B.13 (As available)

Question Source: Bank #
Modified Bank # BFN 1006 #15 (Note changes or attach parent)
New

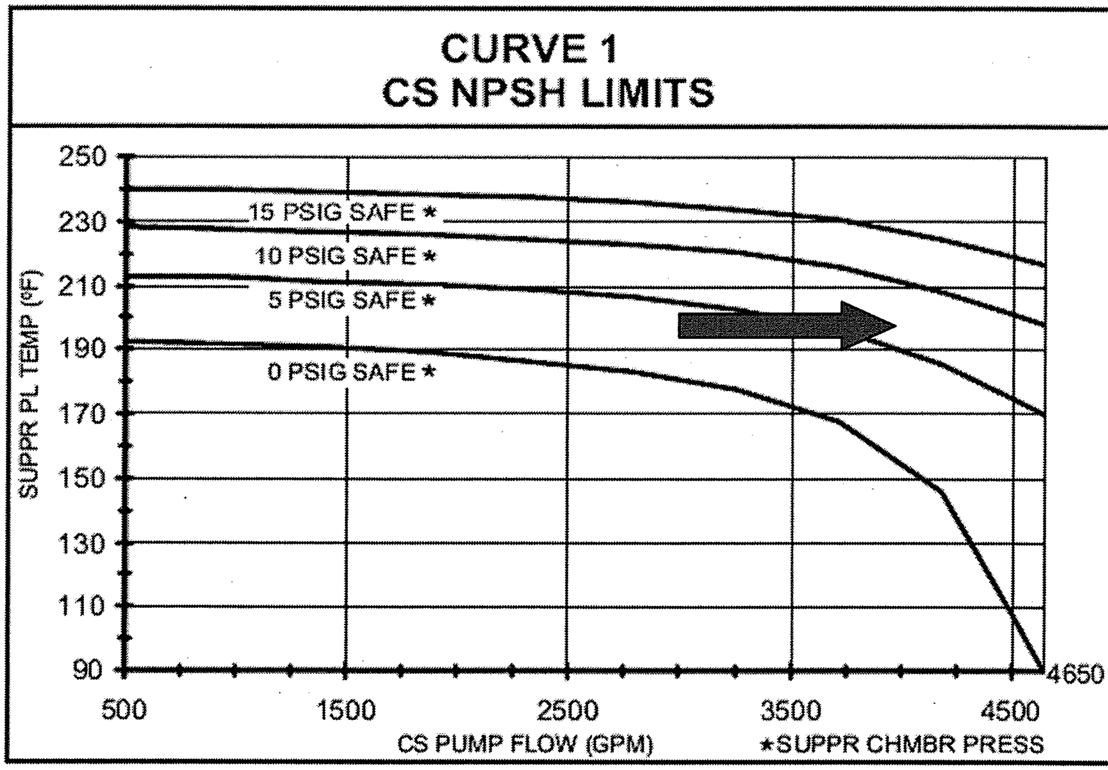
Question History: Last NRC Exam Browns Ferry 1006

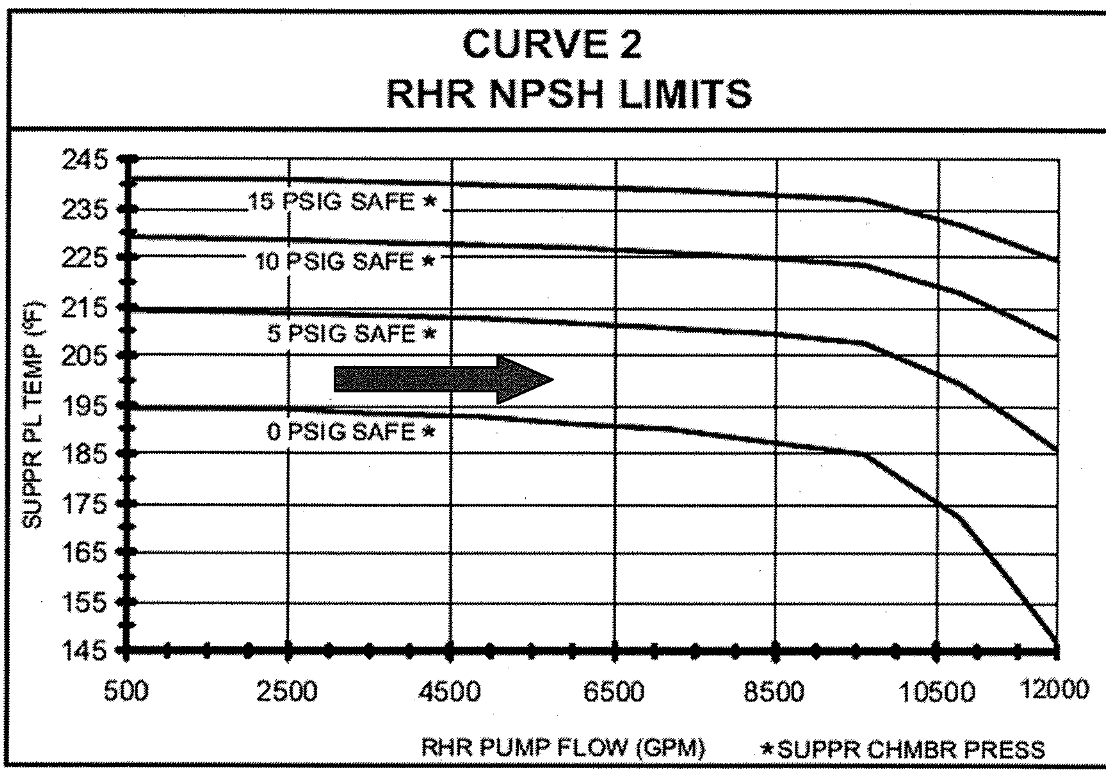
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

Comments:


CURVE 1
CS NPSH LIMITS
15 PSIG SAFE *
10 PSIG SAFE *
5 PSIG SAFE *
0 PSIG SAFE *
SUPPR PL TEMP (ºF)
250
230
210
190
170
150
130
110
90
500
1500
2500
3500
4500
4650
CS PUMP FLOW (GPM)
*SUPPR CHMBR PRESS


CURVE 2
RHR NPSH LIMITS
15 PSIG SAFE *
10 PSIG SAFE *
5 PSIG SAFE *
0 PSIG SAFE *
SUPPR PL TEMP (ºF)
245
235
225
215
205
195
185
175
165
155
145
500
2500
4500
6500
8500
10500
12000
RHR PUMP FLOW (GPM)
*SUPPR CHMBR PRESS

****

a. 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.

2. **Caution #2** — See EOI flow charts

"Operation of RHR or CS with suction from the Suppr pl may result in equipment damage if:

- 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

BFN 1006 #15

Examination Outline Cross-reference:	Level	RO	SRO
295030 Low Suppression Pool Wtr Lvl / 5	Tier #	1	------
EK1.02 (10CFR 55.41.8)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: • Pump NPSH	K/A #	295030EK1.02	
	Importance Rating	3.5	------

Proposed Question: **# 15**

Unit 3 has experienced a LOCA **AND** the following conditions exist:

- Suppression Pool Level is (-) 5.5 inches
- Suppression Chamber Pressure is 5 psig
- Drywell Pressure is 10 psig
- Suppression Pool Temperature is 200° F
- RHR Pump 2A flow is 11,500 gpm
- Core Spray Loop II flow is 4,000 gpm
- **NO** other ECCS Pumps are running

Based on the above conditions, which ONE of the following identifies the ECCS Pump(s), if any, that has (have) sufficient NPSH for continued operation?

[REFERENCE PROVIDED]

A. **NONE**

B. RHR Pump 2A **ONLY**

C. Core Spray Loop II Pumps **ONLY**

D. Core Spray Loop II Pumps **AND** RHR Pump 2A

Proposed Answer: **C**

Examination Outline Cross-reference:	Level	RO	SRO
295028 High Drywell Temperature / 5	Tier #	1	------
EK1.01 (10CFR 55.41.8)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE:	K/A #	295028EK1.01	
• Reactor water level measurement	Importance Rating	3.5	------

Proposed Question: **# 14**

Given the following Unit 2 plant conditions:

- Reactor pressure is being maintained at 50 psig
- Temperature near the water level instrument run in the Drywell is 220° F
- The Shutdown Vessel Flooding Range Instrument (2-LI-3-55) is reading (+) 35 inches

Which ONE of the following is the **HIGHEST** Drywell Run Temperature at which the 2-LI-3-55 reading (+) 35 inches is considered valid?

[REFERENCE PROVIDED]

A. 200° F

B. 250° F

C. 270° F

D. 300° F

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This is plausible since 200°F is a valid indication; however the question calls for the HIGHEST temperature.

B **CORRECT**: In order to answer this question correctly, the candidate must use EOI Caution #1 to determine operable RPV water level instruments.

C INCORRECT: This is plausible if the candidate interpolates the Caution #1 table, however this is NOT permissible.

D INCORRECT: This is plausible if the candidate uses only Curve 8.

KA Justification:

The KA is met because it tests knowledge of the operational implications of Reactor water level measurement with High Drywell Temperature near the water level instruments runs.

Question Cognitive Level:

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): OPL171.201 Rev 7 (Attach if not previously provided)

2-EOI-1 Rev 12 (Including version / revision number)

Proposed references to be provided to applicants during examination: **2-EOI Caution #1 and Curve 8**

Learning Objective: OPL171.201 V.B.13 (As available)

Question Source: Bank # BFN 0610 #73

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 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

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

2-EOI-1	PAGE 1 OF 1
RPV CONTROL	
UNIT 2 BROWNS FERRY NUCLEAR PLANT	
REV: 12	

CAUTIONS

CAUTION #1

- AN RPV WATER LVL INSTRUMENT MAY BE USED TO DETERMINE OR TREND LVL <u>ONLY WHEN</u> IT READS ABOVE THE MINIMUM INDICATED LVL ASSOCIATED WITH THE HIGHEST MAX DW OR SC RUN TEMP.
- IF DW TEMPS, OR SC AREA TEMPS (TABLE 6), AS APPLICABLE, ARE OUTSIDE THE SAFE REGION OF CURVE 8, THE ASSOCIATED INSTRUMENT <u>MAY</u> BE UNRELIABLE DUE TO BOILING IN THE RUN.

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A, B	EMERGENCY -155 TO +60	ON SCALE	N/A	BELOW 150
		-145	N/A	151 TO 200
		-140	N/A	201 TO 250
		-130	N/A	251 TO 300
		-120	N/A	301 TO 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	NORMAL 0 TO +60	ON SCALE	N/A	BELOW 150
		+5	N/A	151 TO 200
		+15	N/A	201 TO 250
		+20	N/A	251 TO 300
		+30	N/A	301 TO 350
LI-3-52 LI-3-62A	POST ACCIDENT -268 TO +32	ON SCALE	N/A	N/A
LI-3-55	SHUTDOWN FLOODUP 0 TO +400	+10	BELOW 100	N/A
		+15	100 TO 150	N/A
		+20	151 TO 200	N/A
		+30	201 TO 250	N/A
		+40	251 TO 300	N/A
		+50	301 TO 350	N/A
		+65	351 TO 400	N/A

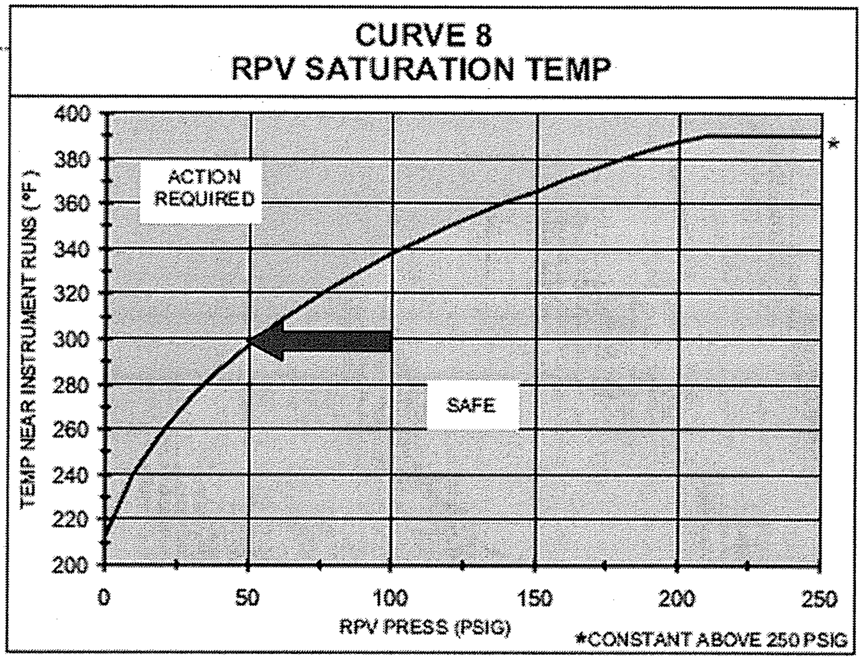


TABLE 6
SECONDARY CONTMT INSTRUMENT RUNS

INSTRUMENT	SC TEMP ELEMENTS AND LOCATIONS			
	EL 621 (74-95F)	EL 593 (74-95C AND D)	EL 565 (69-835A THRU D)	RWCU HXRM (69-29F, G, H)
LI-3-58A	°F	°F	N/A	°F
LI-3-58B	°F	°F	N/A	N/A
LI-3-53	°F	°F	N/A	°F
LI-3-60	°F	°F	N/A	N/A
LI-3-206	°F	°F	N/A	°F
LI-3-253	°F	°F	N/A	N/A
LI-3-52	°F	°F	°F	N/A
LI-3-62A	°F	°F	°F	N/A
LI-3-55	°F	°F	N/A	N/A
LI-3-208A, B	°F	°F	N/A	°F
LI-3-208C, D	°F	°F	N/A	N/A

1. **Caution #1** — See EOI flow charts

a. RPV water level instrument systems sense liquid level in the vessel downcomer region by measuring differential pressure (dP) between a variable leg water column and a reference leg water column. The reference leg **remains** full of water from steam condensing in the chamber located at the top of the reference leg water column. Excess condensate drains back into the RPV. To ensure reference leg water remains gas free a trickle flow of CRDH water is continuously injected into the 4 primary reference legs.

SER 03-05

CRDH injection prevents reference leg notching which can occur if the reference legs are filled with non-condensable gas super saturated water, then depressurized.

b. When water level in the reactor vessel lowers, variable leg height of water decreases, sensed dP increases, and indicated RPV water level lowers. The converse occurs when water level in the reactor vessel increases; variable leg height of water increases, sensed dP decreases, and indicated RPV water level increases.

SER 03-05

c. Changes in height or density of water in the instrument reference leg can cause changes in indicated RPV water level. For example: if actual RPV water level is constant at some on-scale value and the instrument reference leg head of water (height and/or density) decreases, sensed dP decreases and indicated RPV water level increases. Under extreme conditions, a high and increasing drywell or containment temperature can decrease the density of water in the reference leg such that the instrument falsely indicates an on-scale and steadily increasing water level even though the actual RPV water level is decreasing and well below the elevation of the instrument variable leg tap.

SER 03-05

d. It is important to note that the information presented in Caution #1 is not just a simple accommodation for inaccuracies in RPV water level indication which occur when plant conditions are different from those for which the instruments are calibrated. Rather, the caution defines conditions under which the displayed value and the indicated trend of RPV water level cannot be relied upon. SER 03-05

e. Part B of Caution #1 identifies the limiting conditions beyond which water in instrument legs may boil. Water in the RPV water level instrument legs is maintained in a liquid state by cooling action of the surrounding atmosphere and pressure in the reactor vessel. Water in the instrument legs will boil, however, if its temperature exceeds saturation temperature for the existing RPV pressure. SER 03-05

f. Boiling is a concern in both horizontal and vertical reference and variable instrument leg runs. Boil-off from reference leg water inventory reduces the reference head of water, decreases dP sensed by the instrument, and results in an erroneously high indicated RPV water level. Boiling in the instrument's variable leg exerts increased pressure on the variable leg side of the dP cell. This effect results in a lower sensed dP and an erroneously high indicated RPV water level. SER 03-05

g. Part B of Caution #1 references the RPV Saturation Temperature Curve (Curve 8) The RPV Saturation Temperature Curve is generic, based simply on the properties of water. The axis for RPV pressure is plotted from atmospheric pressure to the pressure setpoint of the lowest lifting MSRV. Note that the temperature axis of the RPV Saturation Temperature Curve is not simply drywell temperature. Depending upon the relative location of instrument reference legs and variable legs, indications from monitors near instrument runs must be considered. SER 03-05

h. Because BFN does not have the capability of directly reading temperature indications near instrument runs located in secondary containment, the RPV Saturation Temperature Curve (Curve 8) is supplemented with Table 6, Secondary Containment Instrument Runs. Table 6 identifies the temperature elements and general locations for the instrument runs to each RPV water level instrument.

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

j. Part A of Caution #1 allows the operator to determine if each indicated RPV water level range is reliable by being above the Minimum Indicated Level for each of a series of instrument run temperature ranges. Engineering calculations have determined that when indicated RPV water level is above the Minimum Indicated Level, the operator is assured that actual RPV water level is above the instrument variable leg tap, and trends are valid.

The instrument will indicate high by the amount of this offset throughout its range.

k. The Minimum Indicated Level is defined to be the highest RPV water level instrument indication which results from off-calibration instrument run temperature conditions when RPV water level is actually at the elevation of the instrument variable leg tap. Separate levels are provided for each RPV water level instrument.

SER 03-05

l. The table in Part A is structured to give a Minimum Indicated Level corresponding to several temperature ranges for each of the RPV water level instrument ranges. This yields more usable instrument range than would be available if single values were used.

Examination Outline Cross-reference:	Level	RO	SRO
295030 Low Suppression Pool Wtr Lvl / 5	Tier #	1	-------
G2.1.31 (10CFR 55.41.10)	Group #	1	-------
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	K/A #	295030G2.1.31	
	Importance Rating	4.6	-------

Proposed Question: **# 15**

Unit 3 was at 100% Reactor Power when a leak from the Torus resulted in Suppression Pool Level of 11.4 feet. Required actions of the EOIs have been performed.

Which ONE of the following completes the statement below?

Two minutes after initiating required EOI actions, Wide Range Reactor Pressure Indication(s) available on Control Room Panel(s) __**(1)**__ will be __**(2)**__.

A. **(1)** 3-9-5 **ONLY**
(2) stable

B. **(1)** 3-9-5 **ONLY**
(2) lowering

C. **(1)** 3-9-3 **AND** 3-9-5
(2) stable

D. **(1)** 3-9-3 **AND** 3-9-5
(2) lowering

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that this would be the correct answer if the question asked where Narrow Range Pressure indication is available. Part 2 incorrect – Plausible in that in accordance with 3-EOI-2, reactor scram is required if Suppression Pool can not be maintained >11.5 feet. Two minutes after the scram, reactor pressure would be stable. However, this is incorrect since 3-EOI-2 also required ED for this condition.

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

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

D **CORRECT**: Part 1 correct – Wide Range Pressure indication is available on both 3-9-3 and 3-9-5. Part 2 correct – Per 3-EOI-2, if Suppression Pool Level can not be maintained > 11.5 feet, Reactor Scram and Emergency Depressurization are required.

KA Justification:

The KA is met because the question tests candidates' ability to locate control room wide range pressure indications, and to determine that they correctly reflect the desired plant lineup which is lowering pressure due to requirement to ED on Low Suppression Pool Level.

Question Cognitive Level:

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-EOI-1 Rev. 8 / 3-EOI-2 Rev. 8 (Attach if not previously provided)
OPL171.003 Rev. 19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.203 V.B.13 (As available)

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

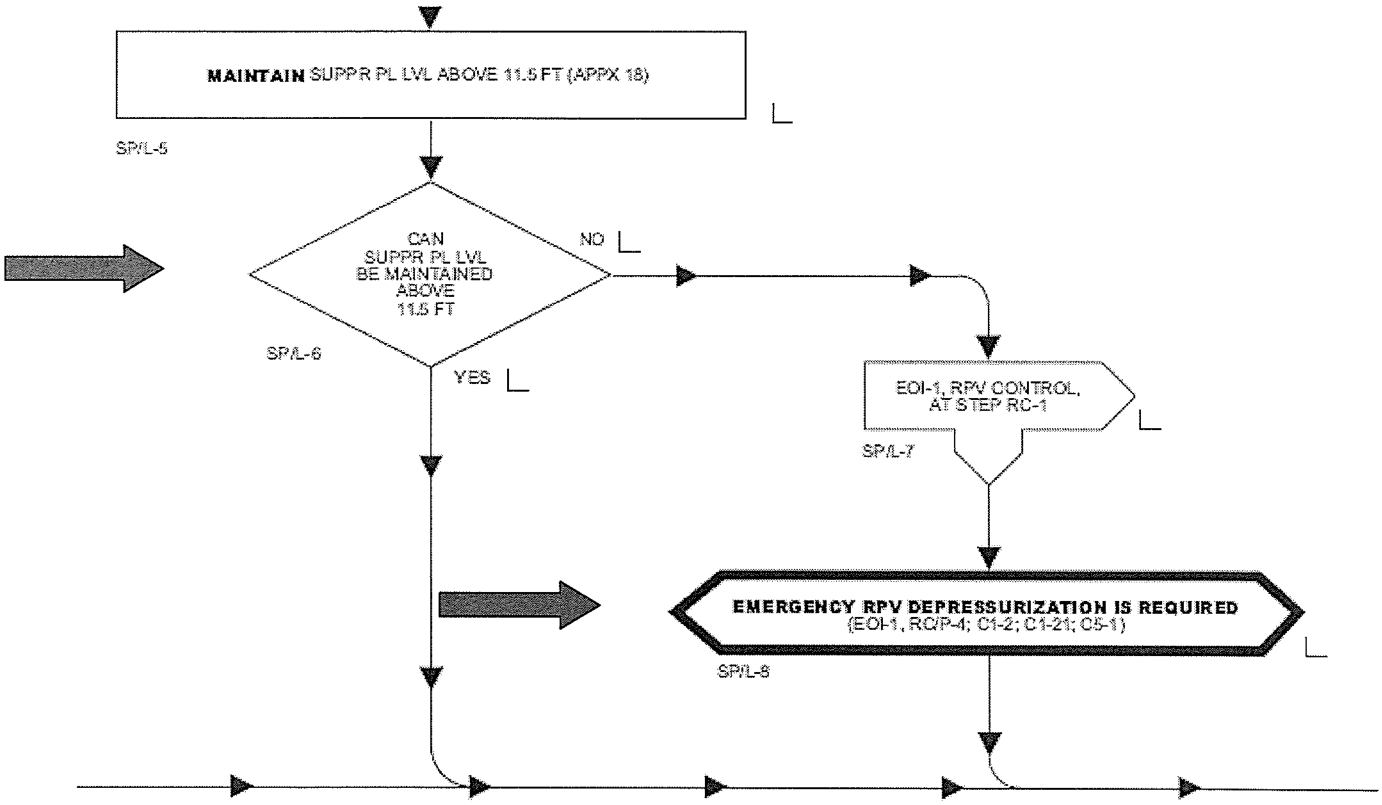
Question History: Last NRC Exam

(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:



3-EOI-2 PAGE 1 OF 1

PRIMARY CONTAINMENT CONTROL

UNIT 3
BROWNS FERRY
NUCLEAR PLANT

REV: 8

INSTRUCTOR NOTES

(a) Provides the high reactor vessel pressure signal (1148 psig) to initiate an ATWS ARI/RPT. — TP-6

Also see PIP-95-71, this PIP gives Instr. Racks, local panels, instr.#, Master/Slave trip TVA (GE), panel in AIR, function, & power supply.

(b) Provides input for opening logic for all SRVs. Uses slave relays (4 per SRV) set at 1135, 1145, or 1155 psig in a 2 of 2 once logic to open SRV.

(c) Provides input to EHC for Reactor Pressure Control — A and C OR B and D

(d) Provides pressure indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86).

(e) Pressure input is from Steam space — Obj. V.B.14, V.D.8, V.E.4

(2) PT-3-22-AA, -BB, -C, -D

(a) Provide the reactor vessel high pressure (1073 psig) signal to RPS for reactor scram.

(b) Provide reactor pressure indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86).

(c) Pressure input is from the Steam space — Obj. V.B.14, V.D.8, V.E.4

(3) PIS-3-22A and B:

(a) Trip mechanical vacuum pumps if reactor pressure is >600 psig and condenser vacuum is >22 inches Hg.

(b) Pressure input is from Steam space. — Obj. V.B.14, V.D.8, V.E.4

(4) PT-3-54, -61, -207

(a) Provide pressure input to the FWLCS

INSTRUCTOR NOTES

(b) Provides reactor pressure indication on recorder PR-3-53 (Panel 9-5) over a range of 0-1500 psig (average pressure). Reactor high pressure alarm is actuated at 1058 psig. PT-3-54, -61, -207 provide reactor pressure indication on Panel 9-5.

(c) Pressure input is from the Steam space — Obj. V.B.14, V.D.8, V.E.4

(5) PT-3-74A/B, (reference columns) PT-68-95/96 (SLC diffuser piping)

(a) Provides reactor pressure permissive signal (< 450 psig) for opening Core Spray and LPCI admission valves.

(b) In conjunction with high drywell pressure provides Core Spray and LPCI automatic initiation signal.

(c) Provides recirc discharge valve auto closure at 230 psig.

(d) Provides reactor pressure indication on Panel 9-3.

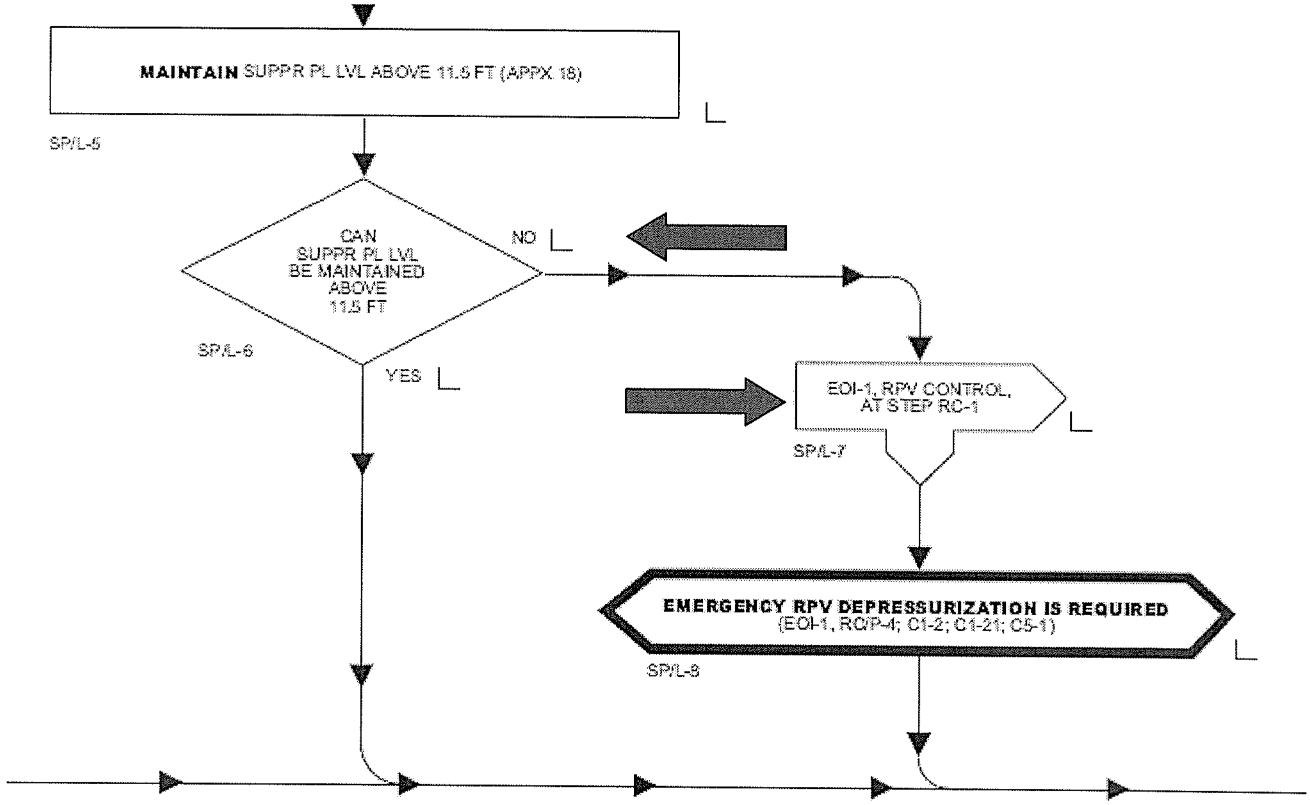
(e) Provides reactor pressure indicator on the ATU cabinets (9-81, 9-82).

(f) PT-3-74 pressure input is from Steam space. PT-68-95/96 pressure input is from liquid space below core plate. — Obj. V.B.14, V.D.8, V.E.4

(6) PT-3-59

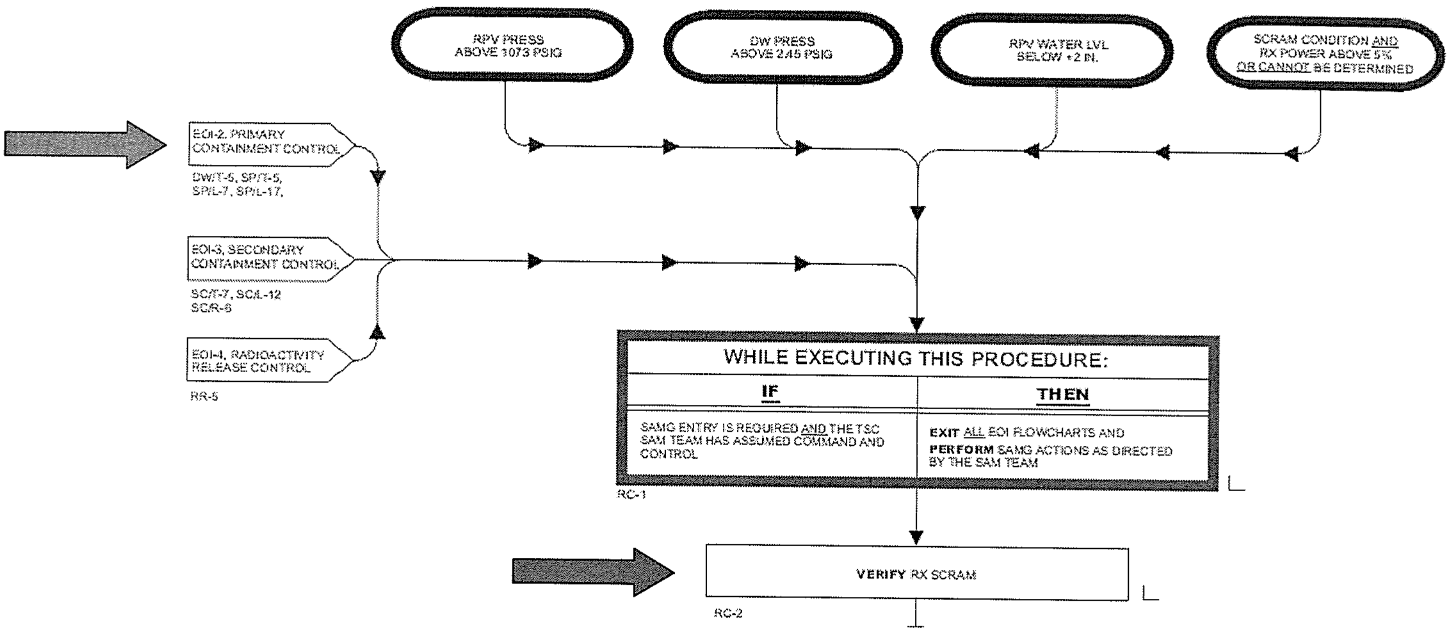
(a) Provides a narrow range (850-1100 psig) reactor pressure indication on Panel 9-5 recorder.

DISTRACTOR PLAUSIBILITY SUPPORT



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

DISTRACTOR PLAUSIBILITY SUPPORT



3-EOI-1	PAGE 1 OF 1
RPV CONTROL	
UNIT 3 BROWNS FERRY NUCLEAR PLANT	
REV: 8	

DISTRACTOR PLAUSIBILITY SUPPORT

INSTRUCTOR NOTES

(b) Provides reactor pressure indication on recorder PR-3-53 (Panel 9-5) over a range of 0-1500 psig (average pressure). Reactor high pressure alarm is actuated at 1058 psig. PT-3-54, -61, -207 provide reactor pressure indication on Panel 9-5.

(c) Pressure input is from the Steam space — Obj. V.B.14, V.D.8, V.E.4

(5) PT-3-74A/B, (reference columns) PT-68-95/96 (SLC diffuser piping)

(a) Provides reactor pressure permissive signal (< 450 psig) for opening Core Spray and LPCI admission valves.

(b) In conjunction with high drywell pressure provides Core Spray and LPCI automatic initiation signal.

(c) Provides recirc discharge valve auto closure at 230 psig.

(d) Provides reactor pressure indication on Panel 9-3.

(e) Provides reactor pressure indicator on the ATU cabinets (9-81, 9-82).

(f) PT-3-74 pressure input is from Steam space. PT-68-95/96 pressure input is from liquid space below core plate. — Obj. V.B.14, V.D.8, V.E.4

(6) PT-3-59

(a) Provides a narrow range (850-1100 psig) reactor pressure indication on Panel 9-5 recorder.

Examination Outline Cross-reference:	Level	RO	SRO
295031 Reactor Low Water Level	Tier #	1	------
K3.01 (CFR 41.5)	Group #	1	------
Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL :	K/A #	295031K3.01	
• Automatic depressurization system actuation	Importance Rating	3.9	------

Proposed Question: **# 16**

Given the following Unit 1 plant conditions:

- HPCI 120VAC POWER FAILURE, (1-9-3F, Window 7) is in alarm
- Reactor Water Level is (-) 122 inches and lowering
- Drywell Pressure is 1.8 psig and steady
- Assume **NO** operator action

Which ONE of the following describes the time that must elapse before ADS automatically initiates **AND** the reason for this response?

ADS will initiate in __**(1)**__ . This actuation is in response to a LOCA __**(2)**__.

A. **(1)** 265 seconds
(2) inside the Drywell

B. **(1)** 360 seconds
(2) inside the Drywell

C. **(1)** 265 seconds
(2) outside the Drywell

D. **(1)** 360 seconds
(2) outside the Drywell

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - This time delay is associated with -122 inches received without a high DW pressure (>2.45 psig), which is given in the stem. However, once this timer times out, if ECCS pumps are running, a 95 second timer initiates and must time out before ADS initiates. This makes the total time 360 seconds. Part 2 incorrect - This is the basis for ADS initiation with BOTH high DW pressure AND low RPV level.

B INCORRECT: Part correct as stated in D. Part 2 incorrect as stated in A above.

C INCORRECT: Part 1 incorrect as stated in A above. Part 2 correct. ADS initiation in the absence of high DW pressure is due to decay heat boil-off following a LOCA outside the Drywell with MSIV isolation.

D **CORRECT:** Part 1 correct - Time delay associated with -122 inches received without a high DW pressure >2.45 psig (265 sec), plus the 95 second timer makes the total time 360 seconds. Part 2 correct. ADS initiation in the absence of high DW pressure is due to decay heat boil-off following a LOCA outside the Drywell with MSIV isolation.

KA Justification:

The KA is met because the question tests knowledge of the reason for Automatic Depressurization system actuation as it applies to Low Reactor Water Level.

Question Cognitive Level:

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)

1-OI-1 Rev. 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.043 V.B.4 (As available)

Question Source: Bank # BFN 0707 #54

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 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: The addition of "Assume NO operator action" was added due to procedural guidance which would inhibit ADS initiation under this condition. In this condition, 1-EOI-1 flowchart path RC/L would allow ADS to be inhibited below -100 inches. In addition, 1-EOI-C1 would be entered below approximately -120 inches and direct that ADS be inhibited. In fact, there are no foreseeable circumstances where ADS would be allowed to auto initiate by procedure.

The HPCI 120VAC Power Failure annunciator is to provide realistic conditions where ADS would auto initiate. If HPCI were operable, ADS would not be required under these conditions.

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

and HPCI) fail to maintain vessel level

4) The -122" setpoint will also initiate 265 second timers that seal in and will run even if water level is restored to >-122". The timers can be reset (if Rx. Level >-122") using pushbuttons in the auxiliary instrument room.

Timer logic in ECCS ATU drawing series "45E670".

5) Once these timers have timed out, the drywell pressure contacts are bypassed, but other relays (that are not sealed in) must still sense reactor level <-122"

6) If so, and the other conditions are met (<+2" and low pressure pumps running), the 95 second timers will start.

7) This feature is based on a LOCA outside of the drywell which has been isolated. Level is below -122" and inventory is boiling off due to decay heat.

8) General Electric calculations have determined that the core will remain covered for 15 minutes after the -122" level is reached. Our system will initiated within the 15 minutes calculated by GE

9) The +2" water level signal is a confirmatory low level signal

Obj. V.B.4
Obj. V.C.3
Obj. V.C.4
Obj. V.D.3
Obj. V.E.4

d. The 95-second timer allows the primary high pressure ECCS system (HPCI) to function and relieve conditions that would require ADS

1) If during the 95-second timer run-out the water level signals clear, the timer resets automatically

Obj. V.C.4
Obj. V.D.4

2) The operator can use timer reset pushbuttons on Panel 9-3 to delay automatic opening of the SRVs

Obj. V.C.4
Obj. V.D.4

3) The operator can use the keylock inhibit switches (Panel 9-3) to prevent the initiation of the 95-second timers and thereby prevent an ADS actuation.

Keylock XS-1-159 A Logic
XS-1-161 B Logic

BFN Unit 1	Main Steam System	1-OI-1 Rev. 0011 Page 12 of 63

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, Sections 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 signal (+2.0 inches), REACTOR LEVEL LOW ADS BLOWDOWN PERMISSIVE, 1-XA-55-9-3C, 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, 1-XA-55-9-3C, Window 33 and RX WTR LVL LOW LOW LOW ECCS/ESF INIT 1-LA-3-58A, 1-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 1-LA-3-58A, 1-XA-55-9-3C Window 28, for 265 seconds (High drywell pressure bypass), and

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, 1-XA-55-9-3C, Window 10.

4. When ALL of the above logic is satisfied, then a 95 second timer starts and ADS BLOWDOWN TIMERS INITIATED, 1-9-3C, Window 11, alarms, and the timer must be timed out to initiate ADS blowdown.

D. Depressing 1-XS-1-159 and -161 on Panel 1-9-3 resets the ADS Blowdown Timers. They also reset the 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 77 seconds.

BROWNS FERRY 0707

Examination Outline Cross-reference:	Level	RO	SRO
295031EK3.01	Tier #	1	
Knowledge of the reasons for the following responses as they apply to Reactor Low Water Level: Automatic Depressurization System actuation.	Group #	1	
	K/A #	295031EK3.01	
	Importance Rating	3.9	4.2

Proposed Question: **RO # 54**

Given the following Unit 1 plant conditions:

- HPCI 120VAC POWER FAILURE (9-3F W7) is in alarm.
- A LOCA has occurred initiating a scram on Low Reactor Water Level.
- Reactor water level (-) 122 inches and lowering
- Drywell pressure 1.8 psig and steady
- A Pre-Accident Signal (PAS) has just been received and all ECCS equipment respond as designed.
- Assume NO operator actions.

Which ONE of the following describes the time that must elapse before ADS automatically initiates and the reason for this response?

ADS will initiate in ___(1)___. This actuation is in response to a _____(2)_____.

	(1)	(2)
A.	265 seconds	LOCA inside the Drywell
B.	360 seconds	LOCA inside the Drywell
C.	265 seconds	LOCA outside the Drywell
D.	360 seconds	LOCA outside the Drywell

Examination Outline Cross-reference:	Level	RO	SRO
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	Tier #	1	------
EA2.06 (10CFR 55.41.10)	Group #	1	------
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN :	K/A #	295037EA2.06	
• Reactor pressure	Importance Rating	4.0	------

Proposed Question: **# 17**

An ATWS has occurred on Unit 1 with the following time line **AND** conditions:

- At 1200 Reactor Power is 15%
- At 1210 SLC is initiated
- At 1235 SLC Storage Tank Level is 67%
- At 1300 SLC Storage Tank Level is 43%

Which ONE of the following completes the statements below?

In accordance with 1-EOI-1, "RPV Control," __**(1)**__ is the earliest time the crew must commence depressurizing the Reactor below the Shutdown Cooling Reactor Pressure interlock.

Cooldown rate of 100° F per hour __**(2)**__ be exceeded.

A. **(1)** 1235
(2) can

B. **(1)** 1235
(2) **CANNOT**

C. **(1)** 1300
(2) can

D. **(1)** 1300
(2) **CANNOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Level must be 43% to commence cooldown. Plausible in that 67% tank level is Hot Shutdown weight for SLC. Part 2 incorrect – Plausible in that under certain conditions in EOI-1, cooldown is performed irrespective of cooldown rates.

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

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

D **CORRECT**: Part 1 correct – In accordance with 1-EOI-1, when SLC has been injected into the RPV to a tank level of 43%, depressurize the RPV below the shutdown cooling pressure interlock. Part 2 correct – Must maintain cooldown rate < 100° F per hour.

KA Justification:

The KA is met because the question tests the candidates' ability to determine when Reactor Pressure is lowered in accordance with the EOIs with an ATWS condition present.

Question Cognitive Level:

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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Technical Reference(s): 1-EOI-1, Rev. 0 (Attach if not previously provided)
OPL171.202 Rev. 8

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.039, V.B.6 (As available)
OPL171.202, V.B.9

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:

Excerpt from 1-EOI-1, "RPV Control," RC/P leg

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 °F

DEPRESSURIZE THE RPV BELOW THE SHUTDOWN COOLING RPV PRESS INTERLOCK WITH ONE OR MORE OF THE FOLLOWING DEPRESSURIZATION SYSTEMS. **MAINTAIN** COOLDOWN RATE BELOW 100 °F/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

DISTRACTOR PLAUSIBLITY SUPPORT

OPL171.202
Revision 8
Page 30 of 6

a. The subsequent steps in this procedure depressurize and cool down the RPV to cold shutdown conditions. If no boron has been injected into the RPV, depressurization and cooldown may proceed as long as control rod insertion is sufficient to shut down the reactor. Such action is permitted even though the existing margin to criticality is small. The positive reactivity added during cooldown may return the reactor to criticality. Should this condition occur, the operator is directed to return to Step RC/P-10, to terminate the cooldown and stabilize RPV pressure, until the reactor can once again be made subcritical.

2. Step RC/P-13

a. This contingent actions step requires the operator to wait until one or more of the stated conditions have been met before continuing in this procedure.

b. After RPV pressure is stabilized, it is appropriate to ensure that the reactor is subcritical prior to performing a normal RPV depressurization and cooldown. Otherwise, the positive reactivity added during forced cooldown, below the saturation temperature for low RPV pressure, may cause the reactor to return to power. Any one of three conditions will ensure that the reactor is subcritical.

c. The first condition requires that the reactor will remain subcritical without boron under all conditions. — See Note 1 Bases in OPL171.201

d. The second condition requires that the SLC System has injected into the RPV to at least all but 43% of the SLC tank level. This SLC tank level corresponds to the Cold Shutdown Boron Weight of boron. The Cold Shutdown Boron Weight is defined to be the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. — Obj.V.B.4.c

e. The third condition allows RPV depressurization and cooldown to proceed as long as control rod insertion is sufficient to maintain the reactor subcritical under present conditions. As used in the EOIs, the term "subcritical" means that reactor power is below the heating range and not increasing. This condition is applicable only if no boron has been injected into the RPV. Such action is permitted even though the existing margin to criticality may be small. A return to criticality under these conditions is acceptable because termination of the cooldown will stop the reactor power increase. Direction to terminate the cooldown is provided in Step RC/P-12.

3. Step RC/P-14

a. This action step directs the operator to use any of the depressurization sources listed in Steps RC/P-10 and RC/P-11 to depressurize the RPV.

b. Once it has been determined that the reactor is subcritical, the operator is directed to depressurize the RPV ensuring that the Technical Specification cooldown rate of
100 °F/hour is observed to maintain RPV metal ductility limits. The cooldown rate is also controlled to avoid an inadvertent, rapid return to criticality, if the margin to subcriticality is small.

c. If MSRVs are being used to depressurize the RPV and the continuous pneumatic supply to the MSRV actuators is isolated or unavailable. Even though MSRV accumulators contain a reserve pneumatic supply, leakage through in-line valves, fittings, and actuators may deplete the reserve capacity. Thus, subsequent to loss of the continuous MSRV pneumatic supply, there is no assurance as to the number of MSRV operating cycles remaining. Obj.V.B.7 Obj.V.C.3

Examination Outline Cross-reference:	Level	RO	SRO
295038 High Off-Site Release Rate	Tier #	1	------
EK2.10 (10CFR 55.41.7)	Group #	1	------
Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: • Condenser air removal system	K/A #	295038EK2.10	
	Importance Rating	3.2	------

Proposed Question: **# 18**

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

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

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

A. **NO** valves will reposition

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

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

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

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: With Off Gas Treatment Select Switch, 2-XS-66-113, not in AUTO, the Radiation High will not result in automatic alignment of Offgas Charcoal Adsorbers.

B INCORRECT**:** Plausibility based on misconception that only Adsorber Bypass Valve, 2-FCV-66-113B will close on High Radiation and that the function remains in force with the Off Gas Treatment Select Switch, 2-XS-66-113, is in BYPASS.

C INCORRECT**:** If Off Gas Treatment Select Switch, 2-XS-66-113, was in AUTO, this would be the correct answer. Adsorber Bypass Valve (FCV-66-113B) will close, and Adsorber Inlet Valve (FCV-66-113A) will open when one channel reaches OG POST-TREATMENT RADIATION HIGH. Plausible in that the 3 X High Radiation Offgas isolation will occur with the Off Gas Treatment Select Switch, 2-XS-66-113 in any position.

D INCORRECT: Plausibility based on misconception that Charcoal Adsorber Train 2 Inlet Valve, 2-FCV-66-118 will open on High Radiation and that the function remains in force with the Off Gas Treatment Select Switch, 2-XS-66-113, is in BYPASS. Plausible in that when aligning charcoal filters for parallel operation, 2-OI-66 directs opening of this valve.

KA Justification:

The KA is met because the question tests knowledge of the interrelations between High Off-Site Release Rate as indicated by Offgas Post Treat Radiation High and the Condenser air removal system including the response of Adsorber Bypass Valve, FCV-66-113B, **AND** the Adsorber Inlet Valve, FCV-66-113A. Since there is no procedural guidance for operation with the Off Gas Treatment Select Switch, 2-XS-66-113, in AUTO in any conditions, the question is posed with the Select Switch in BYPASS for operational validity.

Question Cognitive Level:

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): OPL171.033 Rev. 13 (Attach if not previously provided)

OPL171.030 Rev. 18

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

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

INSTRUCTOR NOTES

(4) Uses same sample as pretreatment monitor

(5) Used as an expanded scale device for locating ruptured or failed fuel elements — Obj. V.B.1,3.b Obj. V.C.1,3.b

(6) One pen recorder (RM-90-160)

(7) No alarm functions

c. Off-Gas post-treatment radiation monitoring system (RM-90-265A/266A) — TPs 8, 9

(1) Two Gamma sensitive scintillation detectors, powered from $\pm$ 24 VDC Neutron Monitoring Batteries. — Obj. V.D.3.d Obj. V.B.2 Obj. V.C.2

(2) Post-treat Radiation monitors (RM-90-265A/266A) - Feed a two-pen recorder on Control Room panel 9-2 (RR-90-265).

(3) Samples are drawn from Off-Gas flow just downstream of the charcoal beds and returns sample to inlet of charcoal beds

(4) Alarm/trip signals come from the radiation monitors and the following alarms and protective functions: — Obj. V.B.5 Obj. V.C.5

(a) OG POST-TREATMENT RADIATION HIGH (55-4C-33) alarms at $6.2X10^4$ cps.

(i) If the Off Gas Treatment Select Switch (XS-66-113) is in AUTO and the High Radiation alarm is received

(ii) Adsorber Bypass Valve (FCV-66-113B) will close, and Adsorber Inlet Valve (FCV-66-113A) will open

DISTRACTOR PLAUSIBLITY SUPPORT

INSTRUCTOR NOTES

(5) Off-Gas isolation is a two-out-of-two logic — Obj. V.B.4.b; Obj. V.C.4.a

(a) Downscale, Hi-Hi-Hi or INOP on RM-90-265A

AND

Downscale, Hi-Hi-Hi or INOP on RM-90-266A

will automatically isolate the Off-Gas system after a 5 second time delay. (FCV-66-28 closes)

3. Stack-Gas Radiation Monitoring System (RM-90-147 & 148) — Obj. V.D.7; Obj. V.B.3.b; Obj. V.C.3.b

a. Purpose

(1) Used to indicate and record release rates from the stack during normal operation and to alarm whenever limits are reached

(2) To monitor the stack gas effluent, a sample is drawn through an isokinetic probe which is located two-thirds of the way up the stack — Note: isokinetic probe explained in section 9 of this lesson

b. The stack receives exhaust gases from following:

(1) Steam Jet Air Ejector (SJAE)

(2) Steam Packing Exhauster (SPE)

(3) Mechanical vacuum pump

(4) Standby Gas Treatment (SGT)

(5) Stack Gas Analyzer Room Vent

OPL171.033
Revision 13
Page 73 of 75

Appendix 2 - Monitor Summary

Monitor	ID No	Tech Spec	Monitor Power	Type	Indications		Function
Main Steam Lines	136 137	None	RPS A RPS A	Ion	CR NUMAC digital display (2) Recorder (RR-90-135) 1 selector switche	Alarms	- DNSCL - HIGH (1.5 x NFLB) - HI/HI/INOP (3 x NFLB) Vac Pmp Vlvs, Vac Pmps,
Off Gas Pre-Treatment	157	None	I&C 'A'	Ion	CR Meter Recorder RR-90-157	Alarms	- OG AVG ANNUAL RELEASE LIMIT EXCEEDED - OG PRETREATMENT HIGH - OG PRETREATMENT DNSC - OG SAMPLE FLOW ABNML
Flux Tilt	160	None	NMS	Ion	CR Meter Recorder; One pen RR-90-160		Indication only
Stack Gas	147 148	ODCM 1.1.2	NMS	Scintillation	Indicator(U1) Recorder(U1)	Alarms	- HIGH - HIGH/HIGH - DNSCL/INOP - FLOW ABNORMAL
Off-Gas Post Treatment	265/ 266	ODCM 1.1.2	NMS	Scintillation	Indicators(2) Recorder RR-90-265	Alarms	Alarms Only-No Trips - HIGH (Aligns charcoal beds if in AUTO) - HIGH-HIGH -Alarm Only - HIGH-HIGH-HIGH/INOP(2 of 2 isolates Off-Gas; any combination) - DOWNSCALE
Turb/Rx/Refuel Ventilation	250	ODCM 1.1.2	I&CA	Scintillation	C.T.(U1) (For all units)	Alarms	- HIGH - DOWNSCALE

OPL171.030
Revision 18
Page 44 of 74

INSTRUCTOR NOTES

d. Any Channel Hi

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

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

5. Condenser Vacuum Low (≤ 25" Hg Vacuum)

a. Initiates auto start of selected SJAE

Auto swap inhibited by procedure on U2

b. The following valves should respond:

(1) The steam admission valve (motor- and air-operated) for the standby SJAE should open.

Auto swap capability removed on U3. DCN 51323

(2) The condensate inlet and outlet valves (motor-operated) for the standby SJAE should open.

(3) The outlet valve for the standby SJAE should open when steam press ≥ 173 psig if control switch is in OPEN.

(4) The steam admission valves for the running SJAE should shut (MOVs, PCV, and outlet valve).

6. Condenser Vacuum High (≥ 26" Hg Vacuum)

PS between suction valve & pump

a. Prevents operation of condenser vacuum pump when it is improper to do so

b. Trips the condenser vacuum pump

7. Condenser Vacuum High (≥ 22" Hg Vacuum) with Reactor Pressure High (≥ 600 psig)

Prevents operation of condenser vacuum pump and isolates vacuum pump suction valve

DISTRACTOR PLAUSIBLITY SUPPORT

BFN Unit 2	Off-Gas System	2-OI-66 Rev. 0099 Page 50 of 135

5.11 **Aligning Charcoal Filters for Parallel Flow**

NOTE

The charcoal beds can be aligned for either parallel or series flow, but normally parallel flow is preferred. Performing the following steps at Panel 9-53 aligns the charcoal beds for parallel flow. If series alignment is preferred, Section 8.10 is required to be performed in lieu of the following steps.

CAUTION

The charcoal adsorbers are required to be aligned in the treatment mode prior to reaching 25% power.

[1] **PLACE** OFFGAS TREATMENT SELECT handswitch, 2-XS-66-113, in TREAT. ☐

[2] **OPEN** CHARCOAL ADSORBER TRAIN 2 INLET VALVE, using 2-HS-66-117. ☐

[3] **OPEN** CHARCOAL ADSORBER TRAIN 1 DISCH VALVE, using 2-HS-66-118. ☐

[4] **CLOSE** CHARCOAL ADSORBER TRAINS SERIES VLV, using 2-HS-66-116. ☐

[5] **CHECK** dewpoint temperature on OFFGAS REHEATER TEMPERATURE recorder, 2-TRS-66-108, indicates 45°F or less (Blue Pen). ☐

CAUTION

A Reheater Inlet Dewpoint Temperature above 48°F may cause wetting of the charcoal beds.

[6] **IF** the Off-Gas System is intended to be operated with charcoal beds in parallel with the charcoal beds on another (shutdown) unit, **THEN** (Otherwise N/A)

COMPLETE Section 8.11. ☐

Examination Outline Cross-reference:

600000 Plant Fire On Site / 8

AA2.13 (10CFR 55.41.10)

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE:

- Need for emergency plant shutdown

	RO	SRO
Level	RO	SRO
Tier #	1	-------
Group #	1	-------
K/A #	600000AA2.13	
Importance Rating	3.2	-------

Proposed Question: **# 19**

With **ALL** 3 Units operating at 100% Reactor Power, a fire at 4 kV Shutdown Board A has resulted in the following:

- Failure of Unit 1 RHR Pump 1A **AND** Core Spray Pump 1A
- Shift Manager has declared an Appendix R Fire

In accordance with Safe Shutdown Instructions, which ONE of the following identifies which, if any, Reactor(s) is (are) required to be scrammed?

A. **NO** Reactor Scram is required

B. Unit 1 **ONLY**

C. Unit 1 **AND** Unit 2 **ONLY**

D. **ALL** 3 Units

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Plausible in that no conditions have been identified which would require a Reactor Scram in accordance with AOIs (including 0-AOI-26-1, "Response to Fires"), EOIs or Tech Specs. If candidate considers only these Abnormal / Emergency Procedures, this would be the correct answer.

B INCORRECT: Plausible in that ONLY Unit 1 has equipment that has been damaged by the fire.

C INCORRECT: Plausible in that 4 kV Shutdown Board A supplies loads on Unit 1 and Unit 2.

D **CORRECT**: Per Safe Shutdown Instructions, if SSIs are entered for an Appendix R Fire, ALL 3 Units must be scrammed.

KA Justification:

The KA is met because it tests the candidate's ability to determine need to emergency shutdown Units based plant fire on site.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.031 Rev 13 (Attach if not previously provided)

0-SSI-5 Rev. 7

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	**Unit 1, 4KV Electric Board Room 1A**	**0-SSI-5 Rev. 0007 Page 6 of 117**

INITIALS

2.0 UNIT 2 CONTROL ROOM OPERATOR ACTIONS

TBD-2

[1] **DIRECT** Unit 3 Unit Supervisor to perform Section 3.0 of 0-SSI-5 to Scram Unit 3, AND **PROCEED TO** cold shutdown. ________

[2] **DIRECT** Unit 1 Unit Supervisor to perform Section 4.0 of 0-SSI-5 to Scram Unit 1, AND **PROCEED TO** cold shutdown. ________

NOTE
The following instruments are those which have been credited for safe shutdown, and must be referenced when executing manual actions for this fire area:
2-LI-3-58A and 2-PI-3-74A for reactor level and pressure
2-TI-64-52AB and 2-PI-64-67B for drywell temperature and pressure
TBD-81
2-LI-64-159A and 2-TI-64-161 for the suppression pool level and temperature
2-LI-2-161A for Condensate Storage Tank 2

(0 Min)

[3] **DIRECT** Unit 2 Operator to perform the following:

TBD-3
TBD-1

[3.1] **VERIFY** reactor Scram AND **RECORD** current time (SSI time of entry).

Time ____________ ________

	INSTRUCTOR NOTES
B. Brief Overview of the Procedure (Generic)	
The Shift Manager (SM) determines when the entry conditions are met and uses 0-SSI-001 to determine which subsection (fire area) to perform.	
SM initiates the procedure and confirms with the Unit 1, 2, and 3 Operators that the unit is scrammed, the MSIVs are closed/checked closed, and a fire pump is verified running.	T=0 at verification of each Reactor trip (TBD-1)
Use of equipment fed from a shutdown board whose D/G is considered unreliable is allowed while executing the SSI; however, prompt action may be required to secure equipment which is determined to be operating spuriously, or the board may be lost to a fault at any time.	Conservative Decision Making
Appropriate reliable diesels are started from the Control Rooms as required for the particular power system alignment for the subsection (Electrical Alignment Illustration).	
HPCI and/or RCIC will be used to maintain reactor water level and the MSRVs used to control reactor pressure.	Some areas also have HPCI/RCIC unavailable
Unit Operator begins a rapid depressurization of the reactor using MSRVs for the affected unit. Aligns the electrical distribution per the sub instruction. The final plant system lineup has RHR flooding the vessel with the flow path recirculating water into vessel out the MSRVs to the torus to the RHR pumps and RHR heat exchangers with RHRSW as the cooling medium.	V.B.4 TP-1
Entry into the SSI, when the entry conditions are met, CANNOT be delayed. The time-lines associated with implementation are measured from the time the SSI is entered. As such, delay into entry could cause the analysis to be invalidated.	The SSI provide a methodology to protect the health and safety of the public during a fire

Examination Outline Cross-reference:	Level	RO	SRO
700000 Generator Voltage and Electric Grid Disturbances / 6	Tier #	1	------
AK2.07 (10CFR 55.41.5)	Group #	1	------
Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following	K/A #	700000AK2.07	
• Turbine/generator control	Importance Rating	3.6	------

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 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?

A. Depress the EHC load set RAISE pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are **IN** service.

B. Depress the EHC load set RAISE 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 RAISE 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 RAISE position; check the 161 kV Capacitor Banks are **OUT** of service.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - Depress the EHC load set RAISE pushbutton will have no affect on load or voltage at current power levels. Plausible in that raising load would aid in mitigating the grid low voltage condition. Part 2 is correct as required by 0-AOI-57-1E

B INCORRECT: Part 1 and 2 incorrect – 161 kV Capacitor Banks out of service will not aid in restoring system voltage. Plausible in that it is an action directed under certain conditions for Grid Instability in 0-AOI-57-1E

C **CORRECT:** Part 1 correct – Per 0-AOI-57-1E, RAISE reactive power to system voltage returns to 520KV OR UNTIL Generator Reactive Power reaches +200 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 In Service

D INCORRECT: Part 1 is correct and Part 2 is incorrect.

KA Justification:

The KA is met because the question tests knowledge of the interrelations between low system voltage due to Grid Disturbance and Generator Field Voltage Auto Adjust (90P), 3-HS-57-26.

Question Cognitive Level:

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-AOI-57-1E Rev 7 (Attach if not previously provided)
3-OI-47 Rev 91 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.036 V.B.13 (As available)

Question Source: Bank # BFN 0801 #20
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 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:

BFN Unit 3	Turbine-Generator System	3-OI-47 Rev. 0091 Page 87 of 241

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. ☐

BFN 0801 #20

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?

A. Depress the EHC load set RAISE pushbutton, 3-HS-47-75C; Check the 161 kV Capacitor Banks are **IN** service.

B. Depress the EHC load set RAISE 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 RAISE 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 RAISE position; check the 161 kV Capacitor Banks are **OUT** of service.

Answer: C

Examination Outline Cross-reference:

295002 Loss of Main Condenser Vac / 3

AK1.03 (10CFR 55.41.10)

Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM :

- Loss of heat sink

	RO	SRO
Level	RO	SRO
Tier #	1	------
Group #	2	------
K/A #	295002AK1.03	
Importance Rating	3.6	------

Proposed Question: **# 21**

Unit 3 is operating at 28% Reactor Power, when a lightning strike results in a loss of **ALL** Condenser Circulating Water Pumps. Immediate Actions of 3-AOI-100-1, "Reactor Scram," are complete.

Which ONE of the following identifies the **AUTOMATIC** protective actions that will occur?

A. Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure **ONLY**

B. MSIV Closure, Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure **ONLY**

C. Main Turbine trip, Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure **ONLY**

D. MSIV Closure, Main Turbine trip, Reactor Feed Pump Turbine trip **AND** Main Turbine Bypass Valve closure

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausibility based on misconception that the Main Turbine trip is bypassed at <30% Reactor Power. The subsequent Reactor Scram due to Turbine Trip is what is bypassed at < 30% Reactor Power.

B INCORRECT: Plausibility based on misconception that the Main Turbine trip is bypassed at <30% Reactor Power along with misconception that MSIV closure would result from loss of condenser vacuum. See discussion of MSIV Closure in D explanation.

C **CORRECT:** Main Turbine will trip at condenser vacuum of 21.8" Hg. Both Reactor Feed Pump Turbine Trip and Main Turbine Bypass Valve closure occur at 7" Hg Condenser Vacuum.

D INCORRECT: Plausibility based on misconception that MSIV closure would result from loss of condenser vacuum. The automatic functions associated with degrading condenser vacuum primarily exist to prevent condenser overpressurization. Even after all the automatic functions occur, the condenser is still vulnerable to overpressurization with the MSIVs open. Therefore, it is very logical that an automatic isolation of MSIVs would occur under these conditions and thus removing all sources of Nuclear Steam to the condenser. To make a comparison, there are several examples that can be found on NRC exams that utilize MSIV closure in response to High-High MSL Radiation. One could not really even argue that it is plausible because it was a Group 1 isolation previously since most plants eliminated the function so long ago. However, It is plausible because it is logical that an automatic isolation of MSIVs would occur under these conditions. Additionally, this was a distractor suggested by the chief on our previous NRC exam for a loss of condenser vacuum question. Plausibility also based on if Mode Switch is not taken to Shutdown, the MSIVs could close as a result of this transient due to Reactor Pressure < 850 psig with Mode Switch in Run.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the operational implications (Main Turbine trip / RFPT Trip / MT Bypass Valve closure) of loss of heat sink (all the Condenser Circ Water Pumps tripping) as it applies to Loss of Main Condenser Vacuum.

Question Cognitive Level:

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. Candidate must recognize that 3-AOI-100-1 Immediate Actions require Operator to place the Mode Switch to Shutdown. Then, with Mode Switch in Shutdown, recognize MSIV closure at 850 psig is bypassed.

Technical Reference(s): 3-AOI-47-3, Rev. 11 (Attach if not previously provided)
3-OI-47, Rev. 91
OPL171.010, Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.010 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.

BFN Unit 3	Turbine-Generator System	3-OI-47 Rev. 0091 Page 17 of 241

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-92565C]

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)

F. Turbine Bypass Valves (Nos. 1 through 9)

TP-1 and TP-7,8
≈ 3 % per BPV

1. Purposes

Obj.V.B.6.d
Obj.V.C.2.d
Obj.V.E.27

a. Routes steam not needed by the turbine to the condenser during the following conditions:

(1) Reactor Startup

(2) Turbine Roll

(3) Turbine Trips

(4) Reactor cooldown

b. Works in conjunction with the turbine control valves to maintain a constant reactor pressure for a given reactor power level.

c. Provides the capability to prevent over pressurization of the reactor if the MSIVs are open.

2. Location

The nine bypass valves are physically located above the turbine throttle in the moisture separator room near the main turbine stop and control valves.

3. Bypass Valve Design

a. Bypass valves are hydraulically operated, reverse seating globe valves.

b. The valves are positioned as required by a Control PAC and Servo-valves.

c. Valves fail closed upon loss of hydraulics.

Page 30

d. Valves close automatically on loss of vacuum, (7" mercury), to protect the Condenser from over pressurization.

e. The valves route steam from the main steam crosstie header directly to the main condenser.

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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	
p.	Loss of condenser vacuum	≤21.8" Hg	Indicative of loss of heat sink - the turbine is not designed to operate at low vacuum conditions.	processor Indicated value will be ~ 24.3 in Hg when turbine trip. See OI-47 precaution & limitation

OPL171.010
Revision 12
Appendix E
Page 67 of 80

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
295014 Inadvertent Reactivity Addition	Tier #	1	------
G2.4.50 (10CFR55.41.10)	Group #	2	------
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	K/A #	295014G2.4.50	
	Importance Rating	4.2	------

Proposed Question: **# 22**

Unit 1 is performing a startup per 1-GOI-100-1A, "Unit Startup." When the Operator At The Controls (OATC) placed the rod movement control switch to the single notch out position for the next control rod, the rod quickly moved 3 notches beyond its intended position. The following indications are received:

- SRM PERIOD, (1-9-5A, Window 20), in alarm
- SRM period indicates 25 seconds on 1-XI-92-7/44A - D

Which ONE of the following completes the statement below?

The OATC is required to __________.

A. **INSERT** Control Rods until the Reactor is brought subcritical.

B. **SHUT DOWN** the Reactor until a thorough assessment has been performed.

C. **REINSERT** the last Control Rod withdrawn to obtain a stable period greater than 60 seconds.

D. **STOP** Control Rod withdrawal until a stable period of greater than 100 seconds is observed.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Per 1-ARP-9-5A and GOI-100-1A, IF withdrawing control rods and a period less than 30 seconds is observed, THEN INSERT rods until subcriticality is observed.

B INCORRECT: Plausible in that this is the correct action if a 5 second period indication is observed.

C INCORRECT: Plausible in that this is the correct action for indication of < 60 but >30 second period.

D INCORRECT: Plausible in that 1-GOI-100-1A directs - WITHDRAW control rods to maintain a period of 100 seconds or greater.

KA Justification:

The KA is met because to successfully answer this question Operator must be able to verify that the SRM Period alarm as a result of the inadvertent reactivity addition is valid based on period indication. Then, recognize the need to insert control rods until the reactor is subcritical in accordance with the ARP.

Question Cognitive Level:

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-9-5A Rev 16
1GOI-100-1A Rev 23
OPL171.059 Rev 11
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.059 V.B.5 (As available)

Question Source: Bank # 1006 Audit # 69
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 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0016 Page 25 of 44

SRM PERIOD (20)	Sensor/Trip Point:	
	Relay K21	30 seconds period

(Page 1 of 1)

Sensor Location: Panel 1-9-12, MCR.

Probable Cause:
A. Electrical noise.
B. Rx power rising on a period of ≤ 30 sec.
C. SI (or SR) in progress.
D. Malfunction of sensor.

Automatic Action: None

Operator Action:

A. **CHECK** reactor period meter reading and amber indicating light illuminated on Panel 1-9-5. ☐

B. **IF** withdrawing control rods and a period less than 30 seconds is observed, **THEN**
INSERT rods until subcriticality is observed and **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission before pulling any more rods. ☐

NOTE
Periods less than 5 seconds are reportable to the NRC within 24 hours.

C. **REFER TO** 1-AOI-79-2, if applicable. ☐

D. **REFER TO** Tech. Spec. Sect. 3.3.1.2, Table 3.3.1.2-1, TRM Tables 3.3.4-1 and 3.3.5-1. ☐

References: 1-45E620-6-1 1-730E237-8

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0023 Page 79 of 171

5.0 INSTRUCTION STEPS (continued)

[24.4] **VERIFY** the following Panel 1-9-5 SRM display lights extinguished:

- HIGH HIGH. ☐
- HIGH OR INOP. ☐
- DNSCL. ☐
- BYPASSED (Will be illuminated if channel bypassed.) ☐
- RETRACT PERMIT (N/A if above setpoint.) ☐
- PERIOD. ☐

(R) ________ ________ ________
Initials Date Time

NOTE

The following steps apply for all Control Rod Withdrawals and do not require an Operator signoff for the steps. The actions should be reviewed by all personnel involved with withdrawing control rods.

[25] **MONITOR** Reactor Power during rod withdrawals and perform the following for the associated conditions.

[25.1] **IF** single-notch withdrawals result in a Reactor period of less than 60 seconds, **THEN**

PERFORM the following:

[25.1.1] **REINSERT** the last control rod withdrawn to obtain a stable period greater than 60 seconds.

[25.1.2] **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal.

[25.2] **IF** a Reactor period of less than 30 seconds is observed, **THEN**

PERFORM the following:

[25.2.1] **INSERT** control rods in accordance with 1-SR-3.1.3.5(A).

[25.2.2] **VERIFY** Reactor subcritical.

DISTRACTOR PLAUSIBILITY SUPPORT

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OPL171.059
Revision 11
Page 13 of 23

	INSTRUCTOR NOTES
i. Performance of the heautp and cooldown rate monitoring surveillance is required 15 minutes prior to heatup and pressurization.	2-SR-3.4.9.1(1) Use "HUR" on ICS
3. Review instruction steps 5.29 through 5.42	SRO in CR
a. If a single notch withdrawal results in a reactor period of less than 60 seconds, the last control rod pulled will be reinserted until a period of greater than 60 seconds is obtained, the Reactor Engineer, Reactivity Manager, and SM approval is required to resume rod withdrawal.	Obj. V.B.5.b Obj. V.C.5.b
b. If a reactor period of less than 30 seconds is observed, control rods shall be inserted until the reactor is subcritical, and obtain the Reactor Engineer, Reactivity Manager, and SM approval to resume rod withdrawal.	Obj. V.B.5.c Obj. V.C.5.c
c. If a reactor period of less than 5 seconds is observed, the reactor shall be shut down and cannot be restarted until an assessment has been performed.	Obj. V.B.5.d Obj. V.C.5.d
d. Near end of core life, criticality may occur before five doublings due to a stronger top peak flux and the buildup of plutonium.	Obj. V.B.4 Obj. V.C.4
e. Single notch withdrawal must begin when the SRM count rate has increased by a factor of 16 (four doublings), and may be stopped after reaching the heat range.	Obj. V.B.3 Obj. V.C.3 (e through g)

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0023 Page 79 of 171

5.0 **INSTRUCTION STEPS** (continued)

[24.4] **VERIFY** the following Panel 1-9-5 SRM display lights extinguished:

- HIGH HIGH. ☐
- HIGH OR INOP. ☐
- DNSCL. ☐
- BYPASSED (Will be illuminated if channel bypassed.) ☐
- RETRACT PERMIT (N/A if above setpoint.) ☐
- PERIOD. ☐

(R) ______ Initials ______ Date ______ Time

NOTE

The following steps apply for all Control Rod Withdrawals and do not require an Operator signoff for the steps. The actions should be reviewed by all personnel involved with withdrawing control rods.

[25] **MONITOR** Reactor Power during rod withdrawals and perform the following for the associated conditions.

[25.1] **IF** single-notch withdrawals result in a Reactor period of less than 60 seconds, **THEN**

PERFORM the following:

[25.1.1] **REINSERT** the last control rod withdrawn to obtain a stable period greater than 60 seconds.

[25.1.2] **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal.

[25.2] **IF** a Reactor period of less than 30 seconds is observed, **THEN**

PERFORM the following:

[25.2.1] **INSERT** control rods in accordance with 1-SR-3.1.3.5(A).

[25.2.2] **VERIFY** Reactor subcritical.

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0023 Page 80 of 171

5.0 **INSTRUCTION STEPS** (continued)

[25.2.3] **OBTAIN** Reactor Engineer, Reactivity Manager, and Shift Manager permission prior to subsequent control rod withdrawal.

[25.3] **IF** a Reactor period of less than 5 seconds is observed, **THEN**

SHUT DOWN the Reactor until a thorough assessment has been performed. REFER TO 1-GOI-100-12A.

CAUTION

Criticality should be expected at all times.

[25.4] **COMMENCE** rod withdrawal. REFER TO 1-OI-85 and 1-SR-3.1.3.5(A).

(R) ______ Initials ______ Date ______ Time

[25.5] **CHECK** coupling integrity by performing 1-SR-3.1.3.5(A) as each control rod is withdrawn.

(R) ______ Initials ______ Date ______ Time

[25.6] [INPO/C] **MONITOR** SRM/IRM instrumentation closely during rod withdrawal while approaching criticality, pausing between rod withdrawals as needed for neutron level stabilization. [INPO SER 89-006]

(R) ______ Initials ______ Date ______ Time

[25.7] **CONTINUE** withdrawing control rods in accordance with 1-SR-3.1.3.5(A).

(R) ______ Initials ______ Date ______ Time

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0023 Page 82 of 171

5.0 **INSTRUCTION STEPS** (continued)

CAUTIONS

1) Criticality should be expected at all times.
2) Extended operation close to the point of criticality could result in inadvertent criticality and must be avoided.

[27] **WHEN** in a configuration that is expected to be near critical **AND** Nuclear Instrument response is **NOT** as expected, **THEN**

NOTIFY Reactor Engineer and Shift Manager.

_______ Initials _______ Date _______ Time

[28] **IF** operation is to be suspended for greater than one hour near the point of criticality, **THEN** (Otherwise N/A)

PLACE the Reactor core sufficiently subcritical as directed by the Shift Manager and as advised by the Reactor Engineer, to avoid an inadvertent criticality.

_______ Initials _______ Date _______ Time

[29] **WITHDRAW** control rods to maintain a period of 100 seconds or greater as indicated on the following indicators on Panel 1-9-5:

- CHANNEL A PERIOD, 1-XI-92-7/44A. ☐
- CHANNEL B PERIOD, 1-XI-92-7/44B. ☐
- CHANNEL C PERIOD, 1-XI-92-7/44C. ☐
- CHANNEL D PERIOD, 1-XI-92-7/44D. ☐

(R) _______ Initials _______ Date _______ Time

Examination Outline Cross-reference:	Level	RO	SRO
295022 Loss of CRD Pumps / 1	Tier #	1	------
AA1.01 (10CFR 55.41.7)	Group #	2	------
Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: • CRD hydraulic system	K/A #	295022AA1.01	
	Importance Rating	3.1	------

Proposed Question: **# 23**

Unit 1 is at 100% Reactor Power when Control Rod Drive (CRD) Pump 1A trips. During the start of CRD Pump 1B, the following occurs:

- Control Rod 30-23 moves from position 16 to position 14
- Control Rod 38-31 moves from position 16 to position 12

Which ONE of the following identifies the required action(s) in accordance with CRD AOIs?

A. Immediately Scram the Reactor.

B. Reduce Reactor Power to 90%

C. Reduce Core Flow to 60% **AND** then Scram the Reactor.

D. Insert Control Rods 30-23 **AND** 38-31 to position 00 using CONTINUOUS IN.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** In accordance with 1-AOI-85-6, if more than 1 CR drifts, insert a reactor Scram Immediately

B INCORRECT: Plausible in that this is correct AOI actions for a single Control Rod Drifting out and unable to insert the control rod.

C INCORRECT: Plausible in that AOI action requiring a Reactor Scram are typically preempted with Core Flow reduction to 60%.

D INCORRECT: Plausible in that this is correct AOI actions for a single Control Rod Drifting out.

KA Justification:

The KA is met because the question tests the candidate's ability to monitor the CRD hydraulic system as it applies to Loss of the in service CRD Pump. Trip of CRD pump requires start of the standby pump. During start of the standby Pump, the CRD Hydraulic system is susceptible to inadvertent control rod drift if flow is raised rapidly or there is significant seat leakage on the in service CRD flow control valve.

Question Cognitive Level:

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. Operator must diagnose multiple control rod drifts based on indication and select appropriate action.

Technical Reference(s): 1-AOI-85-5 Rev. 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.074 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
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BFN Unit 1	Rod Drift In	1-AOI-85-5 Rev. 0001 Page 5 of 10

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** multiple rods are drifting into core, **THEN**

MANUALLY SCRAM Reactor. **REFER TO** 1-AOI-100-1. ☐

4.2 Subsequent Actions

[1] **IF** the Control Rod is moving from its intended position without operator actions, **THEN**

INSERT the Control Rod to position 00 using CONTINUOUS IN. (Otherwise N/A) ☐

[2] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. ☐

[3] **IF** another Control Rod Drift occurs before Reactor Engineering provides a verbal or written evaluation, **THEN**

MANUALLY SCRAM Reactor and enter 1-AOI-100-1. ☐

[4] **CHECK** Thermal Limits on ICS by running OFFICIAL 3D. ☐

[5] **ADJUST** control rod pattern as directed by Reactor Engineer and **CHECK** Thermal Limits on ICS (RUN OFFICIAL 3D).. ☐

[6] **IF** CRD Cooling Water Header DP is excessive and causing the control rod drift, **THEN**

ADJUST CRD SYSTEM FLOW CONTROL, 1-FIC-85-11, and CRD DRIVE WATER PRESS CONTROL VLV, 1-HS-85-23A, as required to establish the following: (Otherwise N/A)

- CRD DRIVE WTR HDR DP, 1-PDI-85-17A, between 250 and 270 psid ☐
- CRD SYSTEM FLOW CONTROL, 1-FIC-85-11, between 40 and 65 gpm ☐
- CRD CLG WTR HDR DP, 1-PDI-85-18A, at about 20 psid or as close as possible while maintaining flow and Header pressure. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 1	CRD System Failure	1-AOI-85-3 Rev. 0004 Page 9 of 12

4.2 **Subsequent Actions (continued)**

[2] **IF** Reactor Pressure is greater than or equal to 900 psig **AND**

- Charging Water Pressure can NOT be restored and maintained greater than 940 psig within 20 minutes, **AND**
- Two or more Scram accumulators are INOP with associated control rod NOT fully inserted, **THEN**

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

[2.1] **IF** core flow is above 60%, **THEN**

REDUCE core flow to between 50-60%. ☐

[2.2] **MANUALLY SCRAM** Reactor and **IMMEDIATELY PLACE** the Reactor Mode Switch in the SHUTDOWN position. ☐

[2.3] **REFER TO** 1-AOI-100-1. ☐

DISTRACTOR PLAUSIBLITY SUPPORT

BFN Unit 1	Rod Drift Out	1-AOI-85-6 Rev. 0001 Page 5 of 9

4.0 **OPERATOR ACTIONS**

4.1 Immediate Actions

[1] **IF** multiple Control Rod drifts are identified, **THEN**

MANUALLY SCRAM the Reactor and enter 1-AOI-100-1. ☐

4.2 Subsequent Actions

[1] **IF** a Control Rod is moving from its intended position without operator actions, **THEN**

SELECT the drifting control rod and **INSERT** to the FULL IN (00) position. ☐

CAUTION

[NRC/C] Operations outside of the allowable regions shown on the Recirculation System Operating Map could result in thermal-hydraulic power oscillations and subsequent fuel damage. 1-GOI-100-12A should be referenced for required actions and monitoring to be performed during a power decrease. [NCO 940245010]

[2] **IF** Control Rod Drive does NOT respond to INSERT signal, **THEN**

PERFORM the following: (Otherwise N/A)

[2.1] **REDUCE** Total Core Flow, as indicated on TOTAL CORE FLOW/CORE PRESS DROP, 1-XR-68-50 on Panel 1-9-5, by approximately 10% to control possible power increase. ☐

[2.2] [NER/C] **IF** drifting control rod is causing Reactor power to rapidly rise at a rate which can NOT be controlled by reducing recirculation flow, **THEN**

MANUALLY SCRAM the Reactor. (Otherwise N/A) [INPO SER 90-015] ☐

[3] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. ☐

Examination Outline Cross-reference:	Level	RO	SRO
295029 High Suppression Pool Wtr Lvl / 5	Tier #	1	------
EK2.02 (10CFR 55.41.7)	Group #	2	------
Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following:	K/A #	295029EK2.02	
• HPCI: Plant-Specific	Importance Rating	3.4	------

Proposed Question: **# 24**

Unit 1 Suppression Pool Level is (+) 7 inches.

Which ONE of the following completes the statements below?

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

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

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

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

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

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

Proposed Answer: **B**

Explanation (Optional):

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

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

C INCORRECT: Part 1 incorrect – Plausible in that this would be true for RCIC. Part 2 incorrect – Plausible in that this would be true for HPCI.

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

KA Justification:

The KA is met because the question tests knowledge of the interrelations between High Suppression Pool Water Level and HPCI

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.040 Rev. 23 / 1-OI-73 Rev. 17 (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.1 (As available)

OPL171.040 V.B.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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

INSTRUCTOR NOTES

(4) Exhaust through check valve to suppression pool

c. Water path — TP-1

(1) Normal condensate path from CST to the HPCI pump, to the A Feedwater line and into the reactor vessel

(2) Alternate suction path from suppression pool

(3) Automatic swapover to suppression pool on high suppression pool level +5.25" or low CST level Elev ≤552'6"

Approximately 7000 gallons left in CST piping when auto swap occurs on low CST Level

NOTE: There is a 5-second time delay for suction swap on high suppression pool level.

(4) Test flow path to CST

(5) Test line orifice which provides a discharge head to simulate reactor pressure

d. Drain System (discussed in detail in section B.4)

e. Turbine auxiliaries (discussed in detail in section B.2)

(1) Gland seal condenser for leakoffs

(2) Cooling water for gland seal condenser

(3) Gland seal condenser blower

(4) Gland seal condenser condensate pump

(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)

BFN Unit 1	High Pressure Coolant Injection System	1-OI-73 Rev. 0017 Page 10 of 78

3.0 **PRECAUTIONS AND LIMITATIONS** (continued)

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

1. Suppression Pool Level High at +5.25 in.

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

F. When HPCI SUPPR POOL OUTBD SUCT VLV, 1-FCV-073-0027 and HPCI SUPPR POOL INBD SUCT VLV, 1-FCV-073-0026 are fully open, HPCI CST SUCTION VALVE, 1-FCV-073-0040 automatically closes.

G. When either HPCI SUPPR POOL OUTBD SUCT VLV, 1-FCV-073-0027, or HPCI SUPPR POOL INBD SUCT VLV, 1-FCV-073-0026, is FULL OPEN, the HPCI/RCIC CST TEST VLV, 1-FCV-073-0036, and HPCI PUMP CST TEST VLV, 1-FCV-073-0035, close.

H. When the HPCI TURBINE STEAM SUPPLY VALVE, 1-FCV-073-0016, is opened, the following valves close:

1. HPCI HOTWELL PUMP INBD ISOL VLV, 1-FCV-073-0017A

2. HPCI HOTWELL PUMP OUTBD ISOL VLV, 1-FCV-073-0017B

3. HPCI STM LINE INBD DRAIN VLV, 1-FCV-073-0006A

4. HPCI STM LINE OUTBD ISOL VLV, 1-FCV-073-0006B

I. The HPCI PUMP MIN FLOW VALVE, 1-FCV-073-0030, automatically opens when system flow is at or below 900 gpm (lowering) if a system initiation signal is present. and automatically closes when system flow is at or above 1255 gpm (rising) regardless of presence of initiation signal.

J. HPCI PUMP MIN FLOW VALVE, 1-FCV-073-0030, opens 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.

K. When a HPCI System isolation signal is reset, the steam line isolation valves do not automatically open, and are required to be opened via handswitch operation, even if a system initiation signal is present.

L. HPCI turbine operation below 2,400 rpm should be minimized to ensure adequate oil pressure from the turbine driven oil pump, to reduce system vibration, and prevent possible water hammer in the exhaust line.

Examination Outline Cross-reference:	Level	RO	SRO
295034 Secondary Containment Ventilation High Radiation / 9	Tier #	1	------
EA2.02 (10CFR 55.41.10)	Group #	2	------
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION :	K/A #	295034EA2.02	
• Cause of high radiation levels	Importance Rating	3.7	------

Proposed Question: **# 25**

Unit 1 is at 100% Reactor Power with the following system line ups:

- Reactor Building Closed Cooling Water (RBCCW) Pumps 1A **AND** 1B are in service
- Reactor Water Cleanup (RWCU) Pumps 1A **AND** 1B are in service
- Fuel Pool Cooling and Cleanup (FPCC) Pump 1A is in service

Unit 1 Reactor **Scrams AND** the following alarms / indications are received:

- 480 V Shutdown Board 1A is locked out
- RBCCW SURGE TANK LEVEL HIGH, (1-9-4C, Window 6)
- RBCCW EFFLUENT RADIATION HIGH, (1-9-3A, Window 17)
- RX BLDG,TURB BLDG, RF ZONE EXH RADIATION HIGH, (1-9-3A, Window 4)

Which ONE of the following is a potential cause of the alarms?

Leakage into RBCCW from ____.

A. Reactor Recirc Pump seal coolers

B. Fuel Pool Cooling Heat Exchangers

C. Reactor Water Cleanup Pump Seal Coolers

D. Reactor Water Cleanup Non-Regenerative Heat Exchangers

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** With the isolation of RWCU at (+) 2 inches due to the scram and the loss of FPCC due to the lock out of Shutdown Board 1A, this remains the only choice that is not tripped and/or isolated. RBCCW Pump 1B remains in service supplying Reactor Recirc Pump seal coolers. Therefore, this is a potential source of inleakage into RBCCW and source of the high radiations alarms.

B INCORRECT: Fuel Pool Cooling Pump A power supply is from 480 V Shutdown Board 1A which is locked out. Any FPCC Heat Exchanger leakage would result in leakage into the FPCC system and not into RBCCW. Therefore, this could NOT be the source of the inleakage into RBCCW and the resulting high radiation alarms. Plausible in that FPCC is an RBCCW load and with the absence of the power loss, this could be the source of the Radiation Alarms.

C INCORRECT: With a Scram from 100% power, Reactor Level drops less than (+) 2 inches, RWCU is isolated and the Pumps are tripped. Therefore, this could NOT be the source of the inleakage into RBCCW and the resulting high radiation alarms. Plausible in that RWCU is an RBCCW load and with the absence of the isolation, this could be the source of the Radiation Alarms.

D INCORRECT: With a Scram from 100% power, Reactor Level drops less than (+) 2 inches, RWCU is isolated and the Pumps are tripped. Therefore, this could NOT be the source of the inleakage into RBCCW and the resulting high radiation alarms. Plausible in that RWCU is an RBCCW load and with the absence of the isolation, this could be the source of the Radiation Alarms.

KA Justification:

The KA is met because it tests the candidate's ability to assess the status of RBCCW and its loads to determine the cause of high radiation levels indicated in Secondary Containment and Secondary Containment Ventilation.

Question Cognitive Level:

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): OPL171.047 Rev. 12 (Attach if not previously provided)

1- ARP-9-3A Rev. 40 / 1- ARP-9-4C Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033 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:

FPCC LESSON PLAN

INSTRUCTOR NOTES

11. Circulating pumps

a. Purpose – To provide forced circulation of water through the system and back to the pool

(1) Quantity - 2

(2) Type - centrifugal horizontal

(3) Capacity - 600 gpm each

(4) Electrical supplies — Obj. V.B.8.d, Obj. V.C.2.d

(a) Pump 1A from 480 V Shutdown Board 1A (similar for Unit 2 & 3) — Obj. V.D.7, Obj. V.E.7

(b) Pump 1B from 480 V Shutdown Board 1B (Similar for Unit 2 & 3)

(5) Control of the pumps is from either the control room (panel 9-4) or the local panel by the pumps in the reactor building on elevation 621. — Procedural directions use the MCR switch on panel 9-4.

(6) Operation - under normal conditions system flow will be ~500 gpm (utilizing one pump). To handle the maximum normal heat load, both pumps are required to be operating, each at ~500 gpm flow. Pump disch pressure normally ~140 psig. — both heat exchangers and the "D" demin as well. Procedure Adherence

(7) System design flow rate is 600 gpm. System maximum flow rate is 1200 gpm.

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.

5. Significant Interlocks and Trip Logic — Obj. V.B.3; V.D.4 Obj. V.C.3; V.D.5

a. System isolation (closure of the inboard and outboard inlet isolation valves FCV 69-1 and 69-2 and the return isolation valve FCV 69-12) will occur on any of the following signals or conditions. — Obj. V.B.5; V.E.8 Obj. V.C.4; V.E.9 Note: 69-12 closes on isol. Signal but is not a PCIS valve.

(1) Low reactor water level (level 3) to protect the core in case of a break in RWCU System piping or equipment. One-out-of-two taken twice logic. — "Level 3" is Tech Spec terminology (>528" above vessel zero). Also, stated as 0" indicated level.

<u>Note</u>: Tech Specs low level setpoint $\geq$0". <u>Current actual setpoint is still +2"</u>. — LT-3-203A thru D

(2) High temperature in areas occupied by RWCU equipment and piping to isolate system in case of a piping break. — Refer to ARP's for latest setpoints

(a) Hi Temp PCIS Isolation Logic is triggered by at least two of twenty-four temperature switches. These switches cause alarms on Panel 9-5. These switches are installed in the following locations and can be read in the Aux Instrument Room.

- Main Steam Tunnel — 834 A thru D
- Pipe Trench — 835 A thru D
- "A" Pump Room — 836 A thru D
- "B" Pump Room — 837 A thru D
- East Wall Hx Room — 838 A thru D
- West Wall Hx Room — 839 A thru D

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0040 Page 26 of 52

RBCCW EFFLUENT RADIATION HIGH
1-RA-90-131A
17

(Page 1 of 2)

Sensor/Trip Point:

	HI	HI-HI
1-RM-90-131D	(Note 1)	(Note 1)

(1) ChemLab should be contacted for current setpoints per 0-TI-45.

Sensor Location: 1-RE-090-0131A (off-line) RBCCW HX, Rx Bldg, EL 593', R-R2

Probable Cause: Hx tube leak into RBCCW system.

Automatic Action: None

Operator Action:

A. **DETERMINE** cause of alarm by observing the following:
   1. **CHECK** RBCCW EFFLUENT OFFLINE RAD MON, 1-RM-90-131D, Panel 1-9-10. ☐

B. **NOTIFY** Chemistry to sample RBCCW for total gamma activity to verify condition. ☐

C. **DETERMINE** if source of leak is RWCU Non-Regenerative Heat Exchanger, Fuel Pool Cooling, or Reactor Water Sample, Recirc Pump A or B Seal Water Heat Exchanger(s). ☐

D. [NER/C] **CHECK** the following for indication of Recirc Pump Seal Heat Exchanger leak:
   - LOWERING in Reactor Recirculation Pump 1A(1B) N0. 1 or 2 SEAL, 1-PI-68-64A or 1-PI-68-63A (1-PI-68-76A or 1-PI-68-75A) on Panel 1-9-4. ☐
   - Temperature rise on CLG WTR FROM SEAL CLG 1-TE-68-54, on RECIRC PMP MTR 1A WINDING AND BRG TEMP temperature recorder, 1-TR-68-58, on Panel 1-9-21. ☐
   - Temperature rise on CLG WTR FROM SEAL CLG 1-TE-68-67, on RECIRC PMP MTR 1B WINDING AND BRG TEMP temperature recorder, 1-TR-68-84, on Panel 1-9-21. ☐

Continued on Next Page

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0040 Page 9 of 52

RX BLDG,TURB BLDG, RF ZONE EXH RADIATION HIGH 1-RA-90-250A

4

(Page 1 of 1)

Sensor/Trip Point:

1-RM-90-250

Gas HIGH ALARM - 6594 CPM
ALERT - 3297 CPM

Sensor Location: El 664' Refuel Floor R-4 P-Line

Probable Cause:
A. Daily source check.
B. High radiation in the Reactor Building, Turbine Building, Refuel Zone exhaust ventilation ducts.
C. Dry Cask storage activities in progress.

Automatic Action: None

Operator Action:
A. **CHECK** 1-RM-90-250 on Panel 1-9-2 (0-MON-90-361) and **MONITOR** activity levels on recorder AIR PARTICULATE MONITOR CONTROLLER 1-MON-90-50 on Panel 1-9-2. ☐
B. **IF** high activity is conformed, **THEN** **NOTIFY** RAD PRO. ☐
C. **REQUEST** Chemistry perform analysis to determine source. ☐
D. **IF** Dry Cask storage activities are in progress, **THEN** **NOTIFY** CASK Supervisor. ☐
E. **IF** the TSC is **NOT** manned, **THEN** **EVACUATE** personnel from affected areas. ☐
F. **IF** the TSC is manned, **THEN** **REQUEST** the TSC to evacuate non-essential personnel from affected areas. ☐
G. **REFER TO** 1-AOI-79-1 or 1-AOI-79-2 if applicable. ☐
H. **MONITOR** release rate for ODCM compliance. ☐
I. **IF** ODCM Limits are exceeded, **THEN** **REFER TO** EPIP-1. ☐
J. **IF** Eberline is operable, **THEN** **REFER TO** 1-OI-90, to reset alarms. ☐

References: 0-47W600-80 1-47E610-90-1 45E620-3
1-SIMI-90B TVA Calc NDQ00902005008/EDC63693

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0018 Page 12 of 43

RBCCW SURGE TANK LEVEL HIGH 1-LA-70-2A 6	Sensor/Trip Point:
	1-LS-070-0002A 4 Inches Above Center Line of Tank

(Page 1 of 2)

Sensor Location: RBCCW surge tank on the fourth floor in the M-G set room.

Probable Cause:
A. Makeup valve 1-FCV-70-1 open.
B. Bypass valve 1-2-1369 leaking.
C. Leak into the system.

Automatic Action: None

Operator Action:
A. **VERIFY** make-up valve 1-FCV-70-1 closed, using RBCCW SYS SURGE TANK FILL VALVE, 1-HS-70-1, on Panel 1-9-4. ☐
B. **CHECK** RBCCW PUMP SUCTION HDR TEMP, 1-TIS-70-3, indicates water temperature is 100°F or less, on Panel 1-9-4. ☐
C. **DISPATCH** personnel to verify high level, ensure bypass valve, 1-2-1369, is closed and observe sight glass level. ☐
D. **OPEN** surge tank drain valve, 1-70-609, then **CLOSE** valve when desired level is obtained. ☐
E. **REQUEST** Chemistry to pull and analyze a sample for total gamma activity and attempt to qualify source of leak. ☐
F. **CHECK** activity reading on RM-90-131D. ☐

Continued on Next Page

DISTRACTOR PLAUSIBILITY SUPPORT

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.

DISTRACTOR PLAUSIBILITY SUPPORT

- RBCCW flow is in the opposite direction to RCW flow.

c. The spare RBCCW heat exchanger has manual isolation valves which allow it to be lined up to Unit 1, 2, or 3.

4. Chemical Feeder

   a. Provides for addition of chemicals

   b. It is the bypass type. The RBCCW pumps provide the DP for chemical feed injection.

   c. Sodium nitrite is injected as a rust inhibitor and for pH control.

5. Expansion Tank — RB EL 639'

   a. Allows for water expansion from temperature and pressure changes within RBCCW System

   b. Provides adequate NPSH to RBCCW pumps

   c. Provides a place to add makeup water to RBCCW from demineralized water, through FCV-70-1, or through a manual bypass valve. — Pnl 9-4

   d. Provides an overflow to Reactor Building floor drain sump.

   e. High and low levels in the expansion tank alarm in the MCR. This allows leaks to be detected. A high level alarm with no high radiation alarm means RCW is probably leaking into the system. A high level alarm with a high radiation alarm indicates in-leakage from a heat load such as Fuel Pool Cooling or RWCU. — ± 4" above/below centerline

   f. The tank is provided with a vent pipe at the top which prevents pressure buildup in the RBCCW System.

Examination Outline Cross-reference:	Level	RO	SRO
295036 Secondary Containment High Sump/Area Water Level / 5	Tier #	1	
EK3.01 (10CFR 55.41.5)	Group #	2	
Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL :	K/A #	295036EK3.01	
• Emergency depressurization	Importance Rating	2.6	

Proposed Question: **# 26**

A HPCI Steam Supply leak has resulted in elevated Secondary Containment temperatures **AND** area water levels. HPCI Steam Supply Isolation valves have failed to isolate **AND CANNOT** be manually closed. Two Secondary Containment Water Levels are above their Maximum Safe Value requiring Emergency Depressurization.

Which ONE of the following completes the statement below?

In accordance with EOI-3, "Secondary Containment Control Bases," **ALL** of the following are reasons for requiring Emergency Depressurization with the **EXCEPTION** of ________.

A. to place the primary system in the lowest possible energy state

B. to reject decay heat to the suppression pool, rather than secondary containment

C. to reduce driving head and flow of primary systems that are unisolated and discharging into secondary containment

D. to allow access into the Reactor Building by the Emergency Response Organization to locate and manually isolate the leak

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This is one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

B INCORRECT: This is one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

C INCORRECT: This is one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

D **CORRECT**: This is NOT one of the four reasons specified in EOIPM Section 0-V-E for Emergency Depressurizing with 2 or more area water levels above the Maximum Safe Operating Value with a Primary System discharging into Secondary CTMT.

KA Justification:

The KA is met because the question test knowledge of the reasons for Emergency Depressurization as it applies to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVELS.

Question Cognitive Level:

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.

Justification:

Technical Reference(s): OPL 171.204 Rev. 7 (Attach if not previously provided)

EOIPM 0-V-E Rev. 1

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:

DISCUSSION: SC/L-14

This signal step informs the operator that actions to control RPV pressure must immediately change because of present plant conditions.

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

This step has been reached because water levels in two or more secondary containment areas have exceeded maximum safe operating value, and a direct threat exists relative to secondary containment integrity, equipment located in secondary containment, and continued safe operation of the plant. The RPV must be rapidly depressurized for the following reasons:

- To reduce/prevent further increase in secondary containment levels.
- To place the primary system in the lowest possible energy state.
- To reduce driving head and flow of primary systems that are unisolated and discharging into secondary containment.
- To reject decay heat to the suppression pool, rather than secondary containment.

OPL171.204
Revision 7
Page 31 AND 32 of 52

o. SC/L-14

This signal step informs the operator that actions to control RPV pressure must immediately change because of present plant conditions.

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

This step has been reached because water levels in two or more secondary containment areas have exceeded maximum safe operating value, and a direct threat exists relative to secondary containment integrity, equipment located in secondary containment, and continued safe operation of the plant. The RPV must be rapidly depressurized for the following reasons: Obj.V.B.7 Obj.V.C.7

- To reduce/prevent further rises in secondary containment levels.
- To place the primary system in the lowest possible energy state.
- To reduce driving head and flow of primary systems that are unisolated and discharging into secondary containment.
- To reject decay heat to the suppression pool, rather than secondary containment.

Examination Outline Cross-reference:	Level	RO	SRO
500000 High Containment Hydrogen Concentration	Tier #	1	------
EK2.09 (10CFR 55.41.7)	Group #	2	------
Knowledge of the interrelations between HIGH CONTAINMENT HYDROGEN CONCENTRATIONS the following:	K/A #	500000EK2.09	
• Drywell nitrogen purge system	Importance Rating	3.0	------

Proposed Question: **# 27**

Unit 2 was operating at 100% Reactor Power when a LOCA occurred. Plant conditions are as follows:

- Drywell H_2 is 3% increasing
- Drywell O_2 is 4% increasing
- Suppression Chamber H_2 is 2% steady
- Suppression Chamber O_2 is 3% steady

Which ONE of the following completes the statement below?

Based on the above conditions, Nitrogen must be lined up to _______.

A. the Drywell

B. the Suppression Chamber

C. the Drywell **AND** Suppression Chamber

D. **NO** primary containment area; the Primary Containment EOI entry condition for hydrogen concentration has **NOT** been exceeded

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** 2-EOI-2 directs monitoring and controlling Drywell and Suppression Chamber, H2 at or below 2.4% AND O2 at or below 3.3%. The Drywell is above both values. 3% H2 in the Drywell is greater than 2.3%, the minimum detectable value. 2-EOI Appendix 14A states to continue in the procedure when H2 or O2 concentration(s) are increasing. The stem states both are increasing in the Drywell. It then directs the operator to determine which area has the highest H2 or O2 concentrations and directs adding nitrogen to that area to reduce the concentration(s).

B INCORRECT: Suppression Chamber H2 and O2 are below the control limits, NO change is occurring, and lower than the Drywell. Plausible if the candidate doesn't know the control parameter values. H2 value is below the BFN min detectable value of 2.3%.

C INCORRECT: You never add to both areas at once. Procedure adds to one area at a time. Plausible since both areas have elevated H2 and O2 concentrations.

D INCORRECT: Procedure addresses correcting area before 3%. Drywell is above control parameters and increasing. Candidate may not know the EOI entry condition for primary containment hydrogen concentration.

KA Justification:

The KA is met because the question tests knowledge of the interrelations between elevated Primary Containment Hydrogen levels and Nitrogen makeup to Containment.

Question Cognitive Level:

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. The RO has to know the primary containment entry condition for high hydrogen concentration and deduce which area has the worst degrading conditions based on that fact.

Technical Reference(s): 2-EOI-2 Rev 10, OPL171.032 Rev 12 (Attach if not previously provided)
2-EOI-Appendix 14A Rev 7

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

PC H2 CONCENTRATION ABOVE 2.4%

SUPPR PL TEMP ABOVE 95 °F

WHILE EXECUTING THIS PROCEDURE:	
IF	**THEN**
SAMG ENTRY IS REQUIRED <u>AND</u> THE TSC SAM TEAM HAS ASSUMED COMMAND AND CONTROL	**EXIT** ALL EOI FLOWCHARTS AND **PERFORM** SAMG ACTIONS AS DIRECTED BY THE SAM TEAM
SAMG ENTRY IS REQUIRED	**ENTER** SP/L AT STEP SP/L-1

PCC-1

CAUTION

\# 4 PC PRESS VS PUMP NPSH

L

IRRESPECTIVE OF THE ENTRY CONDITIONS, **EXECUTE** DW/T, PC/P, PC

PC/H

WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
H2/02 ANALYZERS ARE IN STANDBY <u>AND</u> NO PCIS GROUP 6 ISOLATION EXISTS	1. **RESET** ANALYZER ISOLATION, IF NECESSARY 2. **SELECT** DW OR SUPPR CHMBR AND MOMENTARILY PULL OUT SELECT SWITCH HANDLE TO START SAMPLE PUMPS
PCIS GROUP 6 ISOLATION EXISTS	1. **PLACE** ANALYZER ISOLATION BYPASS KEYLOCK SWITCHES TO BYPASS 2. **SELECT** DW OR SUPPR CHMBR AND MOMENTARILY PULL OUT SELECT SWITCH HANDLE TO START SAMPLE PUMPS
H2 AND O2 MONITORING SYSTEM IS INOPERABLE	**NOTIFY** CHEM LAB TO SAMPLE FOR DW AND SUPPR CHMBR H2 AND O2

PC/H-1

MONITOR AND **CONTROL** DW AND SUPPR CHMBR

- H2 AT OR BELOW 2.4%

<u>AND</u>

- O2 AT OR BELOW 3.3%

USING THE N2 MAKEUP SYSTEM (APPX 14A)

L

PC/H-2

2-EOI APPENDIX-14A

NITROGEN MAKEUP TO PRIMARY CONTAINMENT

LOCATION: Unit 2 Control Room

ATTACHMENTS: None (√)

1. IF.............PCIS Group 6 Isolation signal exists,
   THEN.......**PERFORM** Appendix 8E concurrently with this procedure. ____

2. **MONITOR** Drywell and Suppression Chamber Hydrogen and Oxygen concentrations with H2/O2 CONCENTRATION recorders 2-XR-76-110A or 2-XR-76-110B (Panel 9-54 or 9-55). ____

3. IF.............Drywell or Suppression Chamber Hydrogen or Oxygen analyzers are or become inoperable,
   THEN.......**NOTIFY** Chem Lab to sample Drywell or Suppression Chamber for Hydrogen or Oxygen using CI-644. ____

4. WHEN......Drywell or Suppression Chamber Hydrogen or Oxygen concentration is increasing,
   THEN.......**CONTINUE** in this procedure. ____

NOTE: This procedure assumes that the normal makeup is from Nitrogen Storage Tank B.

5. **DISPATCH** personnel to N_2 Storage Tank B to **CHECK** 0-PI-76-331, N2 STORAGE TANK B PRESSURE, between 160 and 190 psig. ____

6. **DETERMINE** area with highest Hydrogen or Oxygen concentrations and **INFORM** SRO. ____

7. **VERIFY** PCIS RESET (Panel 9-4). ____

8. IF..............It is desired to makeup Nitrogen to the Suppression Chamber, THEN.......**CONTINUE** in this procedure at Step 10. ____

9. **CONTROL** Drywell Hydrogen or Oxygen as follows:

   a. **OPEN** the following valves to admit Nitrogen to Drywell (Panel 9-3):

      - 2-FCV-76-18, DRYWELL N2 MAKEUP INBD ISOLATION VLV ____
      - 2-FCV-76-17, PRI CTMT N2 MAKEUP OUTBD ISOLATION VLV. ____

   b. **SLOWLY** ADJUST 2-PC-76-14, DW/SUPPR CHBR N2 MU PRESS CONTROL (Panel 9-3), to maintain between 55 and 60 scfm, or as directed by SRO if SAMG execution is in progress. ____

   c. **VERIFY** 2-XR-76-14, DW/SUPPR CHBR N2 MAKEUP FLOW/PRESS (Panel 9-3), indicates below 60 scfm on the red pen, or as directed by SRO if SAMG execution is in progress. ____

   d. **CONTINUE** Nitrogen admission to the Drywell UNTIL Drywell Hydrogen and Oxygen are below desired values. ____

   e. **CONTINUE** in this procedure at Step 11. ____

10. **CONTROL** Suppression Chamber Hydrogen or Oxygen as follows:

   a. **OPEN** the following valves to admit Nitrogen to the Suppression Chamber (Panel 9-3):

      - 2-FCV-76-19, SUPPR CHBR N2 MAKEUP INBD ISOLATION VLV ____
      - 2-FCV-76-17, PRI CTMT N2 MAKEUP OUTBD ISOLATION VLV. ____

   b. **SLOWLY** ADJUST 2-PC-76-14, DW/SUPPR CHBR N2 MU PRESS CONTROL (Panel 9-3), to maintain between 55 and 60 scfm, or as directed by SRO if SAMG execution is in progress. ____

Examination Outline Cross-reference:	Level	RO	SRO
203000 RHR/LPCI: Injection Mode (Plant Specific)	Tier #	2	------
K3.04 (CFR 41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following:	K/A #	203000K3.04	
• Adequate core cooling	Importance Rating	4.6	------

Proposed Question: **# 28**

An accident occurred on Unit 2 **AND** resulted in the following conditions:

- Reactor water level indicates (–) 200 inches on Post Accident Range
- Reactor pressure is 400 psig
- **ALL** RHR / LPCI are lost
- **ONLY** ONE CRD Pump **AND** ONE Core Spray pump are running

Which ONE of the following completes the statement below?

Adequate core cooling ______.

[REFERENCE PROVIDED]

A. does **NOT** exist

B. is provided by Spray Cooling

C. is provided by Steam Cooling

D. is provided by Core Submergence

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: is incorrect because adequate core cooling exists. The candidate that fails to correct fuel zone level would believe that the core is no longer adequately cooled.

B INCORRECT: is incorrect because reactor pressure is too high for CS to inject. Plausible in that candidate may fail to recognize reactor pressure greater than the shutoff head (330 psig) of the CS pump.

C INCORRECT: is incorrect because the core is submerged with actual level above top of active fuel.

D **CORRECT:** The indicated parameter place corrected water level above TAF. With water level above TAF, adequate core cooling is assured by submergence.

KA Justification:

The KA is met because the question tests knowledge of the affect of Loss of RHR / LPCI on adequate core cooling.

Question Cognitive Level:

Question is rated as C/A because it involves the multi-part mental process of assembling, sorting, and use reference to solve a problem.

Technical Reference(s): OPL171.201 Rev. 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: 2-LI-3-52/62 Correction Curve

Learning Objective: OPL171.201 V.B.10 (As available)

Question Source: Bank # CNP 08 #17

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Cooper 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:

A. Key Words and Terms — Obj. V.B.10

1. Section I-C to the Program Manual (see Attachment 1) provides definitions for terms, phrases, and acronyms used in the EOIs. The following terms/phrases are to be highlighted in this lesson:

   a. Adequate Core Cooling — Obj. V.B.10.a

      Any of the following conditions (1-4):

      (1) Submergence: Reactor water level is verified at or above TAF, and based on present and past trends and plant conditions, is expected to remain above TAF.

      (2) Spray Cooling: During the execution of C1, the following conditions are met:

      - The reactor can be determined to be shutdown without boron (note 1)

        AND

      - One Core Spray subsystem is injecting at or above 6250 gpm. — One spray ring for design pattern

        AND

      - RPV water level can be determined to be above -215 inches (2/3 core height)

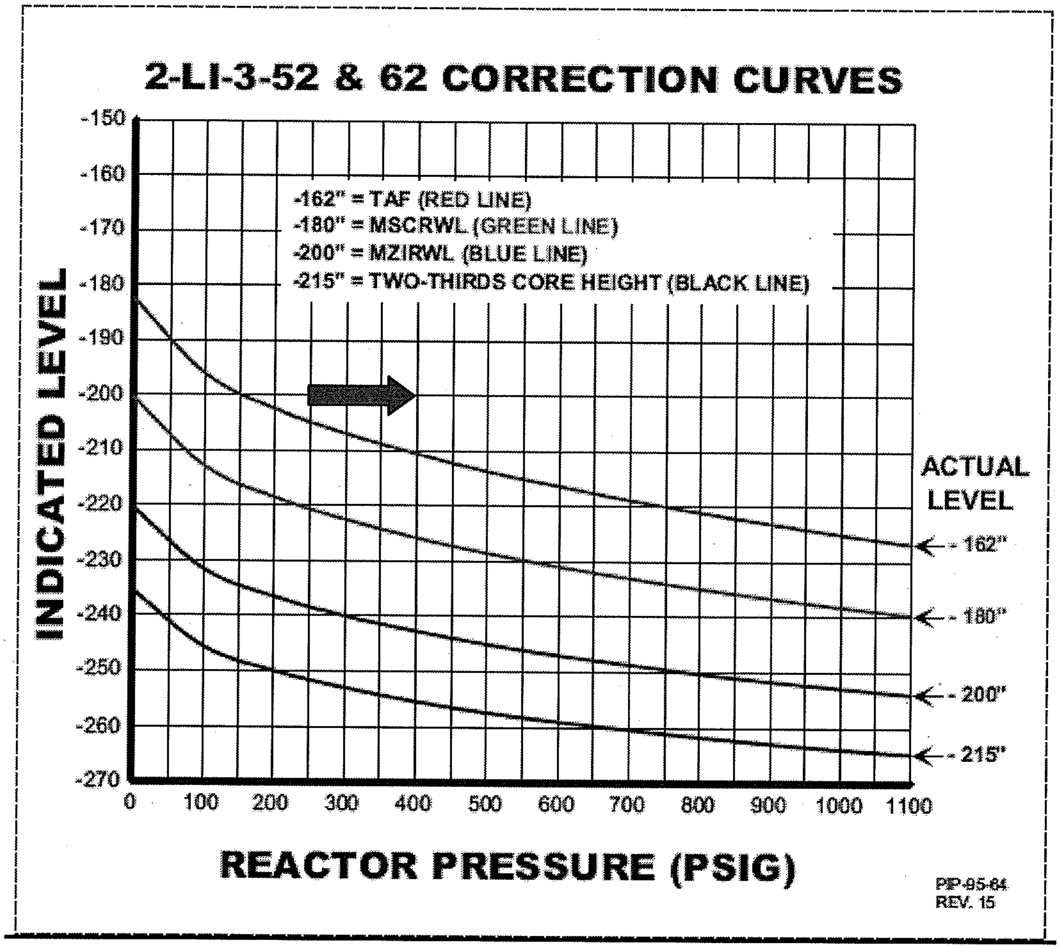
      (3) Steam Cooling With Injection:

      - During execution of C5 and C1, RPV water level can be maintained above the lower water level band allowed by the procedure, [Minimum Steam Cooling Water Level (MSCWL) - 180 inches]. — This will maintain PCT < 1500 °F

        OR

      - Reactor pressure can be maintained above MARFP following reactor depressurization.

OPL171.201
Revision 7
Appendix D
Page 4 of 6



TP-28 Figure 4 LI-3-52 &-62 CORRECTION CURVES

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

QUESTION: NRC RO 17

An accident occurred and resulted in the following conditions:

- Reactor water level is -21" (Indicated FZ) steady.
- Reactor pressure is 400 psig (stable).
- Only one (1) Control Rod Drive Hydraulic Pump and one CS pump are running.
- LPCI and CS initiation signals are present.

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What, if anything, ensures Adequate Core Cooling at this time?

Adequate core cooling...

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a. does not exist.

b. is provided by spray cooling.

c. is provided by core submergence.

d. is provided by steam updraft through the core.

ANSWER: NRC RO 17

c. is provided by core submergence.

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Explanation:

The indicated parameter place corrected water level at TAF. With water level at TAF adequate core cooling is assured.

Distractors:

a. is incorrect because adequate core cooling exists. The candidate that fails to correct fuel zone level would believe that the core is no longer adequately cooled.
b. is incorrect because reactor pressure is to high for CS to inject the candidate that fails to recognize reactor pressure greater than the shutoff head of the CS pump.
d. is incorrect because the core is submerged with actual level at 5 inches above top of active fuel.

Provide EOP graph 14.

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Examination Outline Cross-reference:	Level	RO	SRO
205000 Shutdown Cooling	Tier #	2	------
G2.2.22 (10CFR 55.41.5)	Group #	1	------
Knowledge of limiting conditions for operations and safety limits.	K/A #	205000G2.2.22	
	Importance Rating	4.0	------

Proposed Question: **# 29**

Unit 1 is in Mode 4 with RHR Pump 1B in Shutdown Cooling.

Which ONE of the following completes the statements below?

In accordance with Tech Spec 3.5.2, "ECCS - Shutdown," RHR Pump 1B __**(1)**__ Operable for the ECCS function.

The **MAXIMUM** allowed RCS cooldown rate per Tech Spec 3.4.9, "RCS Pressure and Temperature (P/T) Limits," is __**(2)**__ in any 1 hour.

A. **(1)** is
(2) 90° F

B. **(1)** is
(2) 100° F

C. **(1)** is **NOT**
(2) 90° F

D. **(1)** is **NOT**
(2) 100° F

Proposed Answer: **B**

Explanation (Optional):

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

B **CORRECT:** Part 1 correct - Per Tech Spec 3.5.2, A LPCI subsystem may be aligned for decay heat removal and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode. Part 2 correct – per Tech Spec 3.4.9, RCS cooldown shall be ≤ 100° F in any one hour

C INCORRECT: Part 1 incorrect – Plausible in that ECCS systems are normally required to start, align and inject in response to a system initiation signal to be considered operable. The provision to allow manual realignment is an exception for the conditions. Part 2 incorrect – Plausible in that this is the administrative RCS Cooldown limit.

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

KA Justification:

The KA is met because the question tests knowledge of limiting conditions for operations associated with Shutdown Cooling.

Question Cognitive Level:

Question is rated as C/A because it involves the multi-part mental process of assembling, sorting, and using knowledge and its meaning to solve a problem.

Technical Reference(s): U1 TS 3.4-21 Amm 234 (Attach if not previously provided)
U1 TS 3.4-24 Amm 234/ 3.4-26 Amm 256

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:

BASES (continued)

LCO

Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems include CS subsystems and LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the reactor pressure vessel (RPV). Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. The necessary portions of the Emergency Equipment Cooling Water System are also required to provide adequate cooling to each required ECCS subsystem.

An LPCI subsystem may be aligned for decay heat removal and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

APPLICABILITY

OPERABILITY of the low pressure ECCS injection/spray subsystems is required in MODES 4 and 5 to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of an inadvertent draindown of the vessel. Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1. ECCS subsystems are not required to be OPERABLE during MODE 5 with the spent fuel storage pool gates removed and the water level maintained at ≥ 22 ft above the RPV flange. This provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. --------------NOTE------------ Required Action A.2 shall be completed if this Condition is entered. --------------------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1	Restore parameter(s) to within limits.	30 minutes
	AND		
	A.2	Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-------------------------NOTES---------------------- 1. Only required to be performed during RCS heatup and cooldown operations or RCS inservice leak and hydrostatic testing when the vessel pressure is > 312 psig. 2. The limits of Figure 3.4.9-2 may be applied during nonnuclear heatup and ambient loss cooldown associated with inservice leak and hydrostatic testing provided that the heatup and cooldown rates are ≤ 15°F/hour. 3. The limits of Figures 3.4.9-1 and 3.4.9-2 do not apply when the tension from the reactor head flange bolting studs is removed. --- Verify: a. RCS pressure and RCS temperature are within the limits specified by Curves No. 1 and No. 2 of Figures 3.4.9-1 and 3.4.9-2; and b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period.	30 minutes
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.9-1, Curve No. 3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

3.9 REFUELING OPERATIONS

3.9.7 Residual Heat Removal (RHR) - High Water Level

LCO 3.9.7 One RHR shutdown cooling subsystem shall be OPERABLE and in operation.

-----NOTE-----
The required RHR shutdown cooling subsystem may not be in operation for up to 2 hours per 8 hour period.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and the water level ≥ 22 ft above the top of the RPV flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1	Verify an alternate method of decay heat removal is available.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

PLAUSIBLITY SUPPORT

i. Core Spray System

(1) Combines with at least two RHR pumps to meet ECCS cooling requirements on design basis LOCA.

(2) Shares divisional separated electrical power supplies.

(3) Load shedding interlocks and time delays prevent overloading power supplies.

(4) The Keep-fill System from Core Spray keeps the LPCI injection path full from the pump discharge check valve to the inboard LPCI injection valve.

j. Automatic Depressurization System (ADS) — Obj. V.B.17; Obj. V.E.10

(1) Receives an input from RHR pump discharge pressure switches for an initiation permissive.

(2) Provides RPV depressurization on a small break LOCA to allow LPCI injection.

k. High Pressure Coolant Injection System (HPCI)

Provides small break depressurization makeup if LPCI is not needed.

F. Modes of Operation

1. Shutdown Cooling - all manual operation — TP-8

a. Normal cooldown from rated conditions — Procedural compliance will prevent exceeding limitations.

Bypass steam to main condenser until SDC Rx pressure interlock is met (105 psig) then line up RHR pump in the SDC mode.

Maximum cooldown rate 90°F/hr — 100°F/hr Tech Spec

b. Refer to OI-74and EOI Appendix 17D for requirements for placing Shutdown Cooling in service.

2. Containment Cooling/Spray — TP-4, 5, and 6

a. When core is reflooded after a LOCA, flow can be diverted to the containment spray header. — Obj. V.B.3

b. Control of valves is gained by placing a selector switch in SELECT position.

PLAUSIBLITY SUPPORT

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.9

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This SR also ensures that the HPCI System will automatically restart on an RPV low-low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience with these components supports performance of the Surveillance at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
206000 High Pressure Coolant Injection System	Tier #	2	------
A3.05 (CFR: 41.7)	Group #	1	------
Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including:	K/A #	206000A3.05	
• Reactor water level:	Importance Rating	4.3*	------

Proposed Question: **# 30**

Unit 1 was operating at 100% Reactor Power when a LOCA occurred which resulted in the following conditions:

- RPV water level lowered to (-) 50 inches and is currently (+) 55 inches and slowly lowering.

Which ONE of the following is the **FIRST** condition that would cause an **AUTOMATIC** restart of HPCI?

A. Level lowers to (+) 27 inches.

B. Level lowers to (+) 2 inches.

C. Level lowers to (-) 45 inches.

D. Drywell Pressure greater than 2.45 psig.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: With level at (+) 27 inches, the level 8 (+) 51 inches signal will be clear. However, the Level 8 Turbine Trip will still be sealed in, unless manually reset. The candidate may select this if he/she doesn't realize the turbine trip relay seals itself in, and needs to be manually reset. Also (+) 27 is a recognizable value in that it is set point for Reactor Level Low alarm.

B INCORRECT: HPCI does **NOT** initiate on a Level 3 signal, (+) 2 inches. HPCI will **NOT** restart, if reactor water level lowers to this value, because of the sealed in Level 8 Turbine Trip. Level 3 is below Level 8 and the candidate may select this as a safe value. PCIS isolations and other events happen at Level 3. HPCI could be restarted with this condition, if the Level 8 reset pushbutton was depressed on the control room panel.

C **CORRECT:** HPCI will initiate on a Level 2 signal, (-) 45 inches, even though the Level 8 trip, (+) 51 inches, has **NOT** been manually reset. The Level 2 signal opens contacts that de-energize the Level 8 trip relay, which enables the HPCI Turbine to auto restart.

D INCORRECT: A drywell pressure of 2.45 psig is a normal HPCI initiation signal, and the signal seals in. However, the HPCI Turbine Trip is sealed in and will **NOT** reset on this initiation signal. Since this is an initiation signal, the candidate may think the HPCI Turbine will automatically restart.

KA Justification:

K/A is matched because question is on the HPCI system and monitoring automatic operation, based on water level conditions. The question asks what water level condition will allow HPCI to auto restart, based on the conditions of the stem.

Question Cognitive Level:

The candidate must know several facts: HPCI initiates on Level 2 (-) 45 inches reactor water level and on High Drywell Pressure (+) 2.45 psig. The stem also states level is (+) 55 inches and the candidate must determine that water level is above level 8 (+) 51 inches. The candidate must also know that the HPCI level 8 Turbine Trip Logic seals in and does not automatically reset. Operator action is required to manually reset it, unless Level 2 is reached. The sealed in Level 8 HPCI Turbine Trip will NOT allow the sealed in HPCI Initiation Signal Hi Drywell press to restart the system unless the Trip is manually reset or level again lowers to Level 2. Level 2 contacts will open and de-energize the L8 Turbine Trip relay, which will facilitate an automatic restart. To solve the problem posed by the question, the candidate must use a multi-part mental process to assemble, sort, and integrate parts of the HPCI and HPCI Logic systems.

Technical Reference(s): OPL171.042 Rev 20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.3.c (As available)

Question Source:	Bank #	Fermi 2	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

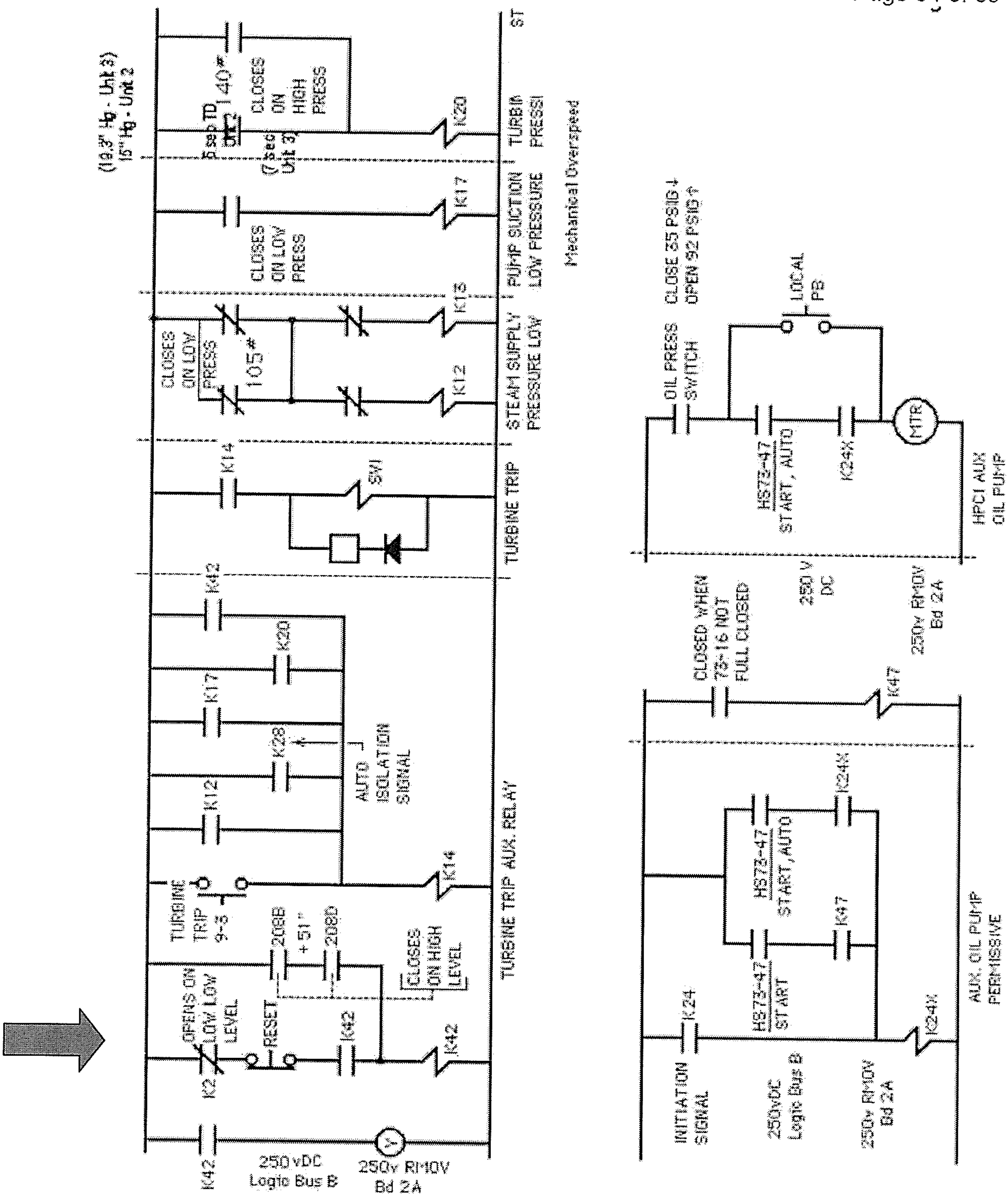
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 X
55.43

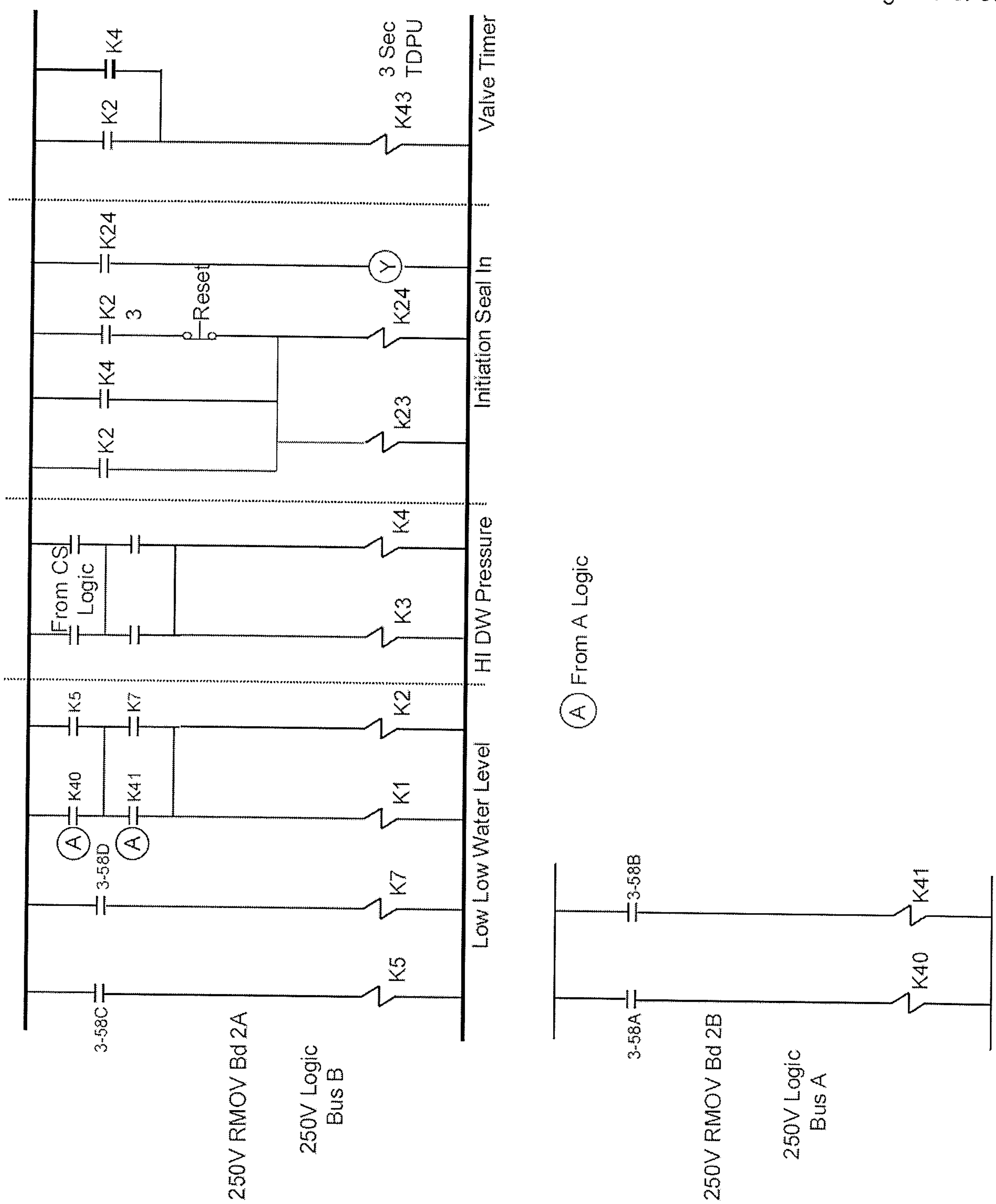
Comments: References attached.

OPL171.042
Revision 20
Appendix C
Page 64 of 69



TP-9: HPCI Turbine Trip Logic

OPL171.042
Revision 20
Appendix C
Page 63 of 69



TP-8: HPCI Initiation Logic

Examination Outline Cross-reference:	Level	RO	SRO
209001 Low Pressure Core Spray System	Tier #	2	------
K1.07 (10 CFR 55.41.2 to 41.9)	Group #	1	------
Knowledge of the physical connections and/or cause effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following:	K/A #	209001K1.07	
• D.C. electrical power	Importance Rating	2.5	------

Proposed Question: **# 31**

Unit 2 was operating at 100% Reactor Power, when a plant event resulted in a reactor scram **AND** loss of 250 VDC RMOV BD 2A. Degrading plant conditions have resulted in the following:

- Reactor Pressure is 325 psig and stable
- A few minutes later, Drywell Pressure is 2.8 psig

Based on the above conditions, which ONE of the following predicts how Core Spray will be affected by the bus loss?

A. **ALL** Core Spray pumps will start **AND ALL** injection valves will open.

B. **ONLY** the Loop 1 Core Spray pumps will start **AND** Loop 1 injection valves will open.

C. **ONLY** the Loop 2 Core Spray pumps will start **AND** Loop 2 injection valves will open.

D. **NO** Core Spray pumps will start **AND NO** injection valves will open.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Loop 2 pumps will not start and injection valves will not open. Candidate misconception that logic failure causes valves to fail open and pump start will NOT be affected.

B **CORRECT**: Loop 1 pumps will start and injection valves will open. SYS I Initiation Logic is still energized.

C INCORRECT: Loop 2 pumps will not start and injection valves will not open. Candidate misconception that logic failure causes valves to fail open and pump start will NOT be affected. Candidate misconception that 250 VDC RMOV BD 2A is a division 1 feed and affects Loop 1 pumps and valves.

D INCORRECT: Loop 1 pumps will start and injection valves will open. Candidate misconception that there is only one logic system for both loops of Core Spray so both would be affected and Loop 2 logic would be for UNIT 2.

KA Justification:

K/A requires cause effect relationship between Core Spray System and DC power. Question is about Core Spray system and the loss of DC power to one portion of its initiation logic and its effect on the system.

Question Cognitive Level:

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. The candidate must know the power supply to the Core Spray loop 2 logic and the effects of its loss. He/she must understand the system and logic interrelationships.

Technical Reference(s): OPL171.045 Rev 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: OPL171.045 Obj 4.d (As available)

Question Source: Bank #

Modified Bank # VY 2007 NRC Q6 (Note changes or attach parent)

New

Question History: Last NRC Exam Vermont Yankee 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:

VY 2007 NRC Exam

A plant event has resulted in a reactor scram and loss of Bus DC-2C. Degrading containment conditions has resulted in the following:

- Reactor Pressure is at 325 psig
- Drywell Pressure is at 2.8 psig

Based on the above conditions, how will Core Spray be affected by the bus loss?

A. All Core Spray pumps will start and All injection valves will open.

B. ONLY the Loop 'A' Core Spray pump will start and its injection valves will open.

C. ONLY the Loop 'B' Core Spray pump will start and its injection valves will open.

D. No Core Spray pumps will start and NO injection valves will open.

Proposed Answer: C

Explanation (Optional):

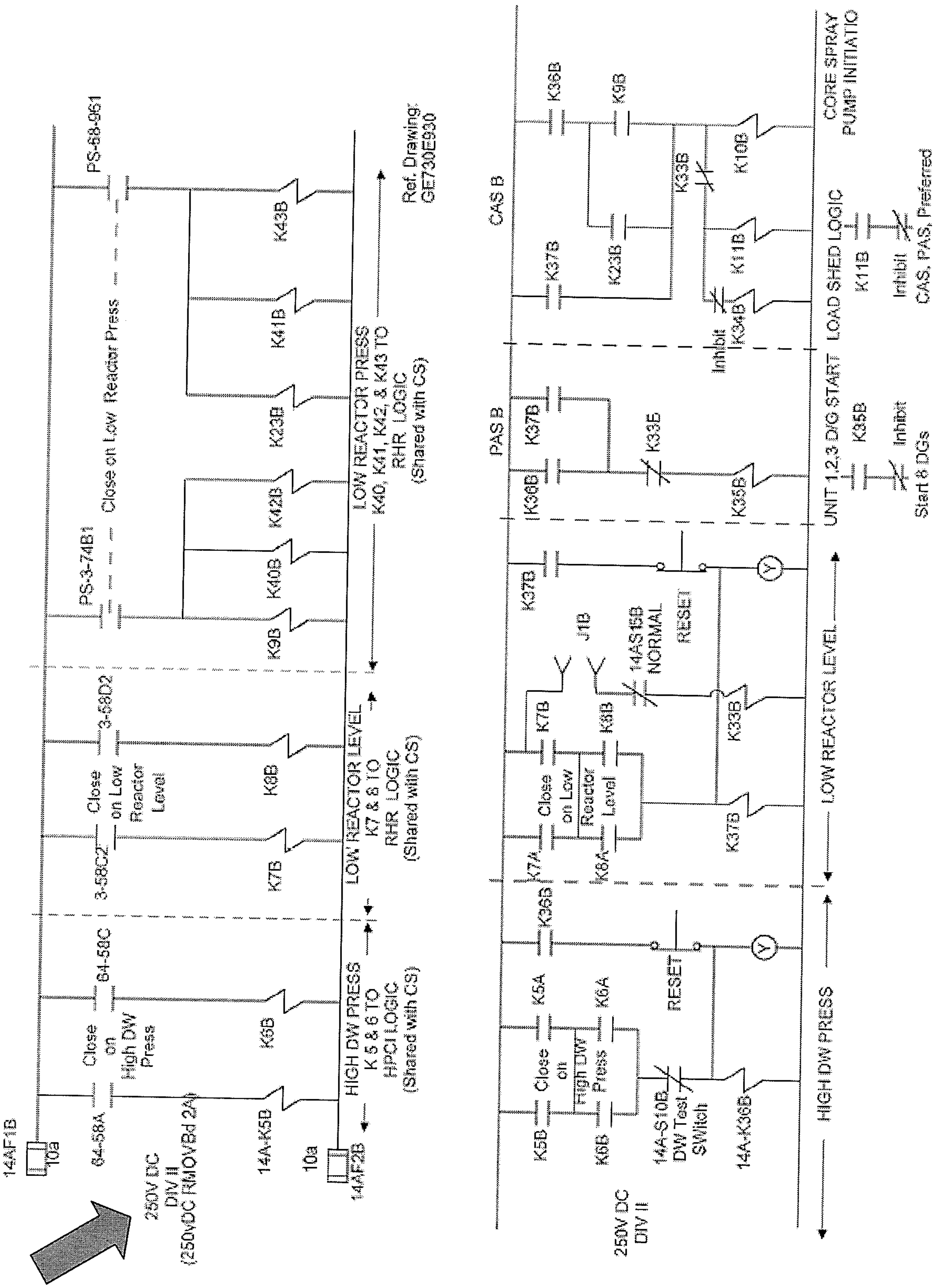
A. Incorrect – Loop 'A' pumps and valves have lost initiation logic power.

B. Incorrect – Bus DC-2C provides 125 VDC power to 'A' loop pump and valve initiation Logic. Only 'B' loop would have power.

C. Correct – Only Loop 'B' pumps and valves have initiation logic power.

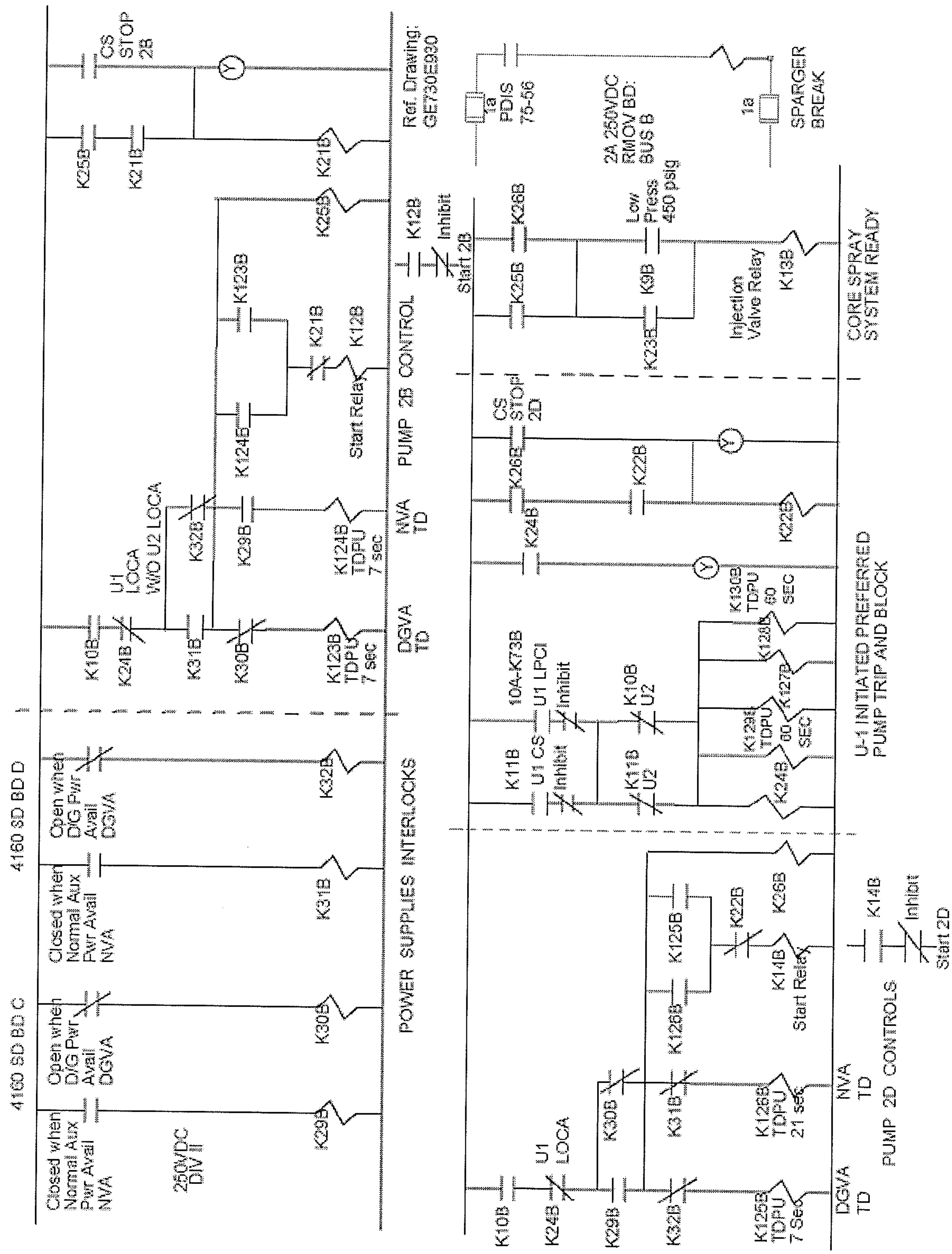
D. Incorrect – Loop 'B' pumps and valves still have initiation logic power.

OPL171.045
Revision 14
Appendix C
Page 47 of 50



TP-6 CORE SPRAY INITIATION LOGIC (SHEET 3 OF 4)

OPL171.045
Revision 14
Appendix C
Page 48 of 50



TP-7 CORE SPRAY INITIATION LOGIC (SHEET 4 OF 4)

Examination Outline Cross-reference:	Level	RO	SRO
211000 Standby Liquid Control System	Tier #	2	
A2.07 (10CFR 55.41.5)	Group #	1	
Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	211000A2.07	
• Valve closures	Importance Rating	2.9	

Proposed Question: **# 32**

Unit 1 is executing 1-EOI-1, "RPV Control," due to a Scram **AND** an ATWS. The Unit Operator (UO) is directed to inject Standby Liquid Control (SLC) per 1-EOI-1 Appendix 3A, "SLC Injection."

The UO places the SLC Pump control switch in the 'START-A' position.

Given the following plant conditions:

• SLC SQUIB VALVE CONTINUITY LOST, (1-9-5B, Window 20)	Extinguished
• SQUIB VALVE A and B CONTINUITY, blue lights on Panel 1-9-5	Illuminated
• SLC Pump 1A red light	Illuminated

Which ONE of the following describes the status of SLC **AND** the correct action(s) to take?

A. **ONE** squib valve has fired;
Place SLC Pump 1A in Stop, start the SLC Pump 1B, **AND** verify proper operation.

B. **NO** squib valves have fired;
Place SLC Pump 1A in Stop, start the SLC Pump 1B, **AND** verify proper operation.

C. **ONE** squib valve has fired;
Verify proper system operation by observing the SLC tank level lowering by ~1% per minute.

D. **BOTH** squib valves have fired;
Verify proper system operation by observing the SLC tank level lowering by ~1% per minute.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: 'A' pump did start by indication of RED light illuminated. Neither squib valve has fired; as indicated by the lack of the alarm and the blue lights are still lit. The squib valves are arranged in Parallel, so 1 firing would allow injection into RPV. Starting 'B' would allow the squib valves to be fired from the other primer.

B **CORRECT:** 'A' pump did start by indication of RED light illuminated. Neither squib valve has fired as indicated by the lack of the alarm and the blue lights are still lit. Starting 'B' would allow the squib valves to be fired from the other primer.

C INCORRECT: 'A' pump did start by indication of RED light illuminated. It is not required in the EOI's to dispatch personnel to the area. Starting 'B' would allow the squib valves to be fired from the other primer.

D INCORRECT: 'A' pump did start by indication of RED light illuminated. Neither squib valve has fired as indicated by the lack of the alarm and the blue lights are still lit. No flow so tank level will not decrease.

KA Justification:

The KA is met because the question tests the ability to predict the impact of valve closures on the SLC System. Based on the indications provided, candidate must conclude that following system initiation both Squib Valves remain closed and recognize the impact on SLC Injection. Based on the Squib Valves failing to open, the candidate must use 1-EOI-1 Appendix 3A to correct the consequences of this abnormal condition.

Question Cognitive Level:

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. Candidate must diagnose the system condition based on indications provided and then determine appropriate action to take to correct the abnormal condition.

Technical Reference(s): 1-EOI Appendix 3A rev 0 (Attach if not previously provided)
OPL171.039 rev 16 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.4 / V.B.5 (As available)

Question Source: Bank # BFN 0801 #33
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 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: The 'A' SLC pump has started and neither squib valve has fired as indicated by the lack of the alarm and the blue lights are still lit. The proper action iaw EOI-app 3A is to start the other pump and verify proper operation.

BFN UNIT 1	SLC INJECTION	1-EOI APPENDIX-3A Rev. 0 Page 1 of 2

LOCATION:	Unit 1 Control Room	
ATTACHMENTS:	None	(√)

1. **UNLOCK** and **PLACE** 1-HS-63-6A, SLC PUMP 1A/1B, control switch in START-A or START-B 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 1-EA-63-8 Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 20). ____
- 1-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. ____
- System flow, as indicated by 1-IL-63-11, SLC FLOW, red light illuminated on Panel 1-9-5, ____
- SLC INJECTION FLOW TO REACTOR 1-FA-63-11, Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 14). ____

3. IF .... Proper system operation <u>CANNOT</u> be verified,

   THEN........... **RETURN** to Step 1 and **START** other SLC pump. ____

4. **VERIFY** RWCU isolation by observing the following:

- RWCU Pumps 1A and 1B tripped ____
- 1-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed ____
- 1-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed ____
- 1-FCV-69-12, RWCU RETURN ISOLATION VALVE closed. ____

5. **VERIFY** ADS inhibited. ____

6. **MONITOR** reactor power for downward trend. ____

OPL171.039 r16

1. Explosive Valves
   a) Two 100% capacity explosive (Squib) valves, FCV 63-8A and B, are installed in parallel.
   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.
   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.
   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.

1. Main Control Room Instrumentation (Panel 9-5)

Parameter	Device	Range	Normal Indication
SLC Storage Tank Level	Level Indicator	0 - 100%	61 - 69%
SLC Pump Disch. Pressure	Pressure Indicator	0 - 2000 psig	0 psig with system in standby, ~1250 psig with pump running
HCV-63-12 Position	Red Light	ON when valve is open	ON
HCV-63-13 Position	Green Light	ON when valve is closed	ON
HCV-63-14 Position	Green Light	ON when valve is closed	ON
Squib valve firing circuit continuity	Blue Light (One for each squib)	ON when squib valve firing power is available and circuit continuity is maintained	ON

System Flow	Red Light	ON when sensed flow downstream of squibs is >40 gpm	OFF for normal system standby
SLC Pump	Red Light (One for each pump)	ON when pump is running	OFF for normal system standby
SLC Pump	Green Light (One for each pump)	ON when pump is stopped	ON for normal system standby

Examination Outline Cross-reference:	Level	RO	SRO
212000 Reactor Protection System	Tier #	2	------
A4.03 (10 CFR 55.41.7)	Group #	1	------
Ability to manually operate and/or monitor in the control room: • Provide manual select rod insertion	K/A #	212000A4.03	
	Importance Rating	3.9	------

Proposed Question: **# 33**

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

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

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

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

Verify CRD Pump operating, ____**(1)**____ , direct manually opening CRD Flow Control Valve (2-FCV-85-11A or B), verify Mode Switch in SHUTDOWN, bypass the Rod Worth Minimizer, CRD Power Switch ON, select control rod, **AND** place CRD ____**(2)**____.

A. **(1)** reset ARI
(2) Control Switch in ROD IN, until green 00 is lit, on the four rod display

B. **(1)** reset ARI
(2) Notch Override Switch in EMERG IN, until the control rod stops moving inward

C. **(1)** direct closure of CHARGING WATER SHUTOFF, 2-SHV-85-586
(2) Control Switch in ROD IN, until the green 00 is lit, on the four rod display

D. **(1)** direct closure of CHARGING WATER SHUTOFF, 2-SHV-85-586
(2) Notch Override Switch in EMERG IN, until the control rod stops moving inward

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: A loss of Control Air occurred, so scram and ARI cannot be reset. Also CRD Notch Override Switch is placed in Emergency In, in an ATWS. Procedure directs insert until movement stops. Candidate misconception that scram and ARI can be reset with NO Control Air available. Also misconception that CRD Control Switch is used in an ATWS when driving rods. NOT used due to RMCS settle function requirements between rods, would delay rod insertion in this emergency.

B INCORRECT: A loss of Control Air occurred, so ARI and scram cannot be reset. Part 2 is correct; the procedure directs insert until movement stops and use of Notch Override Switch in EMERGENCY IN until rod stops moving. Candidate misconception that scram and ARI can be reset with NO Control Air available.

C INCORRECT: A loss of Control Air occurred. Part 1 is correct because cannot reset scram or ARI. Candidate misconception that CRD Control Switch is used in an ATWS when driving rods. NOT used due to RMCS settle function requirements between rods, would delay rod insertion in this emergency. CRD Notch Override Switch is placed in Emergency In to insert the control rod, in an ATWS

D **CORRECT:** A loss of Control Air occurred. Scram and ARI cannot be reset because no air pressure. Charging water shutoff valve needs to be closed to direct water from Charging header to Drive Water Header to move rods. The CRD Flow Control Valve has lost air and needs to be manually opened to provide Drive Water Pressure to drive control rods. Emergency In is used to bypass the settle function on the Reactor Manual Control Sys, so the control rods can be inserted without waiting between rod selections, therefore taking less time to insert in the ATWS emergency.

KA Justification:

The K/A is matched because the question and K/A require how to manually select, insert, and determine (monitor) when the control rods are inserted.

Question Cognitive Level:

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. The candidate must deduce that an ATWS has occurred. He/she must determine that the loss of Control Air caused the scram and Recirc Pump Trip. The loss of Control Air will not allow reset of the scram or ARI. It complicates control rod movement because of loss of air to the CRD Flow Control Valve. Because of the ATWS, control rod movement will be with the ROD Notch Override Switch instead of the CRD Control Switch.

Technical Reference(s): 2-EOI Appendix 1D Rev 6 (Attach if not previously provided)
2-AOI-32-2 Rev 32

Proposed references to be provided to applicants during examination: None

Learning Objective: V.B.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:

2-EOI APPENDIX-1D

INSERT CONTROL RODS USING REACTOR MANUAL CONTROL SYSTEM

LOCATION: Unit 2 Control Room, Panel 9-5

ATTACHMENTS: 1. Tools and Equipment
2. Core Position Map (√)

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

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

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

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

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

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

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

   a. **SELECT** control rod. ____

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

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

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

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

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

END OF TEXT

DISTRACTOR PLAUSIBILITY SUPPORT
(Appendix normally performed for ATWS. Will not be effective due to loss of Control Air)

BFN UNIT 1	MANUAL SCRAM	1-EOI APPENDIX-1F Rev. 1 Page 1 of 7

LOCATION:	Unit 1 Control Room	
ATTACHMENTS:	1. Tools and Equipment 2. 1-PNLA-009-0015, Rear 3. 1-PNLA-009-0017, Rear	(√)

1. **VERIFY** Reactor Scram and ARI reset. ____

   a. IF ....ARI <u>CANNOT</u> be reset,

      THEN ....**EXECUTE** EOI Appendix 2 concurrently with Step 1.b of this procedure. ____

   b. IF ....Reactor Scram <u>CANNOT</u> be reset,

      THEN ....**DISPATCH** personnel to Unit 1 Auxiliary Instrument Room to defeat <u>ALL</u> 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 1-PNLA-009-0015, Rear:

         a) Relay 5A-K10A (DQ) Terminal 2 to Test Terminal 1-TX-099-05A-K12E (Bay 1). ____

         b) Relay 5A-K10C (AT) Terminal 2 to Test Terminal 1-TX-099-05A-K12G (Bay 3). ____

      3) **REFER** to Attachment 3 and **JUMPER** the following relay terminals in 1-PNLA-009-0017, Rear:

         a) Relay 5A-K10B (DQ) Terminal 2 to Test Terminal 1-TX-099-05A-K12F (Bay 1). ____

         b) Relay 5A-K10D (AT) Terminal 2 to Test Terminal 1-TX-099-05A-K12H (Bay 3). ____

2. WHEN ....RPS Logic has been defeated,

   THEN..........**RESET** Reactor Scram. ____

Examination Outline Cross-reference:	Level	RO	SRO
215003 Intermediate Range Monitor (IRM) System	Tier #	2	
K2.01 (10 CFR 55.41.7)	Group #	1	
Knowledge of electrical power supplies to the following:	K/A #	215003K2.01	
• IRM channels/detectors	Importance Rating	2.5	

Proposed Question: **# 34**

Unit 2 is performing a startup with the following conditions:

- Mode Switch is in STARTUP
- Reactor is critical
- IRMs are steady on Range 2

Which ONE of the following identifies the IRM power source **AND** the effect of a loss of power to a single IRM?

	IRM Power Source	Effect of Power Loss to IRM
A.	24 VDC Battery	Rod Block **ONLY**
B.	24 VDC Battery	Rod Block **AND** Half Scram
C.	250 VDC Battery	Rod Block **ONLY**
D.	250 VDC Battery	Rod Block **AND** Half Scram

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: An INOP half scram is also processed, as well as a rod block. Candidate misconception that scram function bypassed on range 2.

B **CORRECT:** 24 VDC supplies IRM detector voltage. With a loss of power, the detector will indicate downscale and receive an INOP trip. The INOP trip enforces both a rod block and a half scram on the corresponding RPS channel.

C INCORRECT: 24 VDC supplies IRM detector voltage. An INOP half scram is also processed. Candidate misconception that 250 VDC supplies IRMs. It does supply the neutron monitoring battery chargers.

D INCORRECT: 24 VDC supplies IRM detector voltage. Candidate misconception that 250 VDC supplies IRMs. It does supply the neutron monitoring battery chargers.

KA Justification:

K/A is matched because the question asks for power supply to the IRMs and affect of loss of the power supply. K/A asks for knowledge of electrical power supply to the IRMs channels/detectors.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.020 Rev 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.11 (As available)

Question Source: Bank #

Modified Bank # Nine Mile 2 /Q23 (Note changes or attach parent)

New

Question History: Last NRC Exam Nine Mile 2 / 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:

Nine Mile 2 NRC 2008

Tier # 2
Group # 1
K/A # 215003, K2.01
Importance Rating 2.5

(K&A Statement) - Knowledge of electrical power supplies to the following: IRM channels/detectors
Proposed Question: Common 23

The plant is performing a startup with the following conditions:

- Mode Switch in STARTUP
- Reactor critical
- IRMs steady on Range 2

Which one of the following will result from the failure of the 24 VDC Power Supply Fuses to a single IRM?

Rod Block	Half Scram
A. IRM INOP	None
B. IRM DOWNSCALE	None
C. IRM INOP	IRM INOP
D. IRM DOWNSCALE	IRM DOWNSCALE

Proposed Answer: C.

C. Power Supplies

Obj.V.B.11

1. The IRM power supplies receive unregulated +24 VDC power from the neutron monitoring battery and convert it to regulated voltages of proper magnitude for use by the IRM detectors and logic circuits. A loss of 24VDC power will give an inop trip, additionally there will be a loss of IRM indication.

2. Neutron monitoring battery chargers are fed from it's units 250V Battery Board, Panel 8, which in turn is fed from I&C A and B regulating transformers.

3. Detector Drives are from I&C A power supply. A loss of this power supply would result in an inability to move IRM's.

INSTRUCTOR NOTES

E. Trips

TP-10

1. Rod blocks

Obj.V.D.7, V.B.5
Obj. V.C.3.,

Block	Setpoint	When Bypassed
Downscale	≤ 7.5	Range 1 or RUN
High	≥ 90/104.6	RUN Mode
INOP	-HV low (<90v) -Module unplugged -Function switch not in OPERATE -Loss of ±24VDC	RUN Mode
Detector Wrong Position	Detector Not Full IN	RUN Mode

Obj. V.B.6.
Obj.V.C.4
Obj. V.B.5
Unit Difference
IRM high setpoint is 90 at Unit 2 and 104.6 on Unit 1 and Unit 3

Obj.V.B.13

2. Scrams

TP-11

Obj. V.B.7.
Obj. V.C.5. Obj.V.D.8

Scrams	Setpoint	When Bypassed
High-High	≥ 116.4	In RUN Mode
INOP	-HV low (<90v) -Module unplugged -Function switch not in OPERATE -Loss of ±24VDC	In RUN Mode

F. Controls Provided

1. Panel 9-5

a. Recorder switches select between IRM channels, and APRM/RBM channels have been removed. All units now contain digital recorders, which do not require operation of selector switches. These switches have been removed.

b. Range switches allow operator to select appropriate IRM range to maintain indications between 25 to 75 on 0-125 scale. 0-40 scale is no longer utilized.

DISTRACTOR PLAUSIBLITY SUPPORT

INSTRUCTOR NOTES

C. Power Supplies

Obj.V.B.11

1. The IRM power supplies receive unregulated ±24 VDC power from the neutron monitoring battery and convert it to regulated voltages of proper magnitude for use by the IRM detectors and logic circuits. A loss of 24VDC power will give an inop trip, additionally there will be a loss of IRM indication.

2 Neutron monitoring battery chargers are fed from it's units 250V Battery Board, Panel 8. which in turn is fed from I&C A and B regulating transformers.

± 24 VDC Neutron Monitor Battery powers cabinets and detectors.

3. Detector Drives are from I&C A power supply. A loss of this power supply would result in an inability to move IRM's.

D. Instrumentation

1. Control Room Instrumentation

SER 03-05 'Controlling Plant Evolutions Precisely' Monitor all available indications during reactivity changes

Item	Device	Range
Reactor Power	Dual Recorders (4)	0 to 125 Located 9-5
Reactor Power	8 meters, dual scale	0 to 40 not used, located 9-12
		0 to125 , located 9-12

2. Annunciators, alarm indication

Obj.V.B.5/ V.B.7

a. Annunciators

Annunciator	Function/Remarks
IRM Downscale	Rod Block (Range 1-bypassed)
IRM High	Rod block
IRM High-High or INOP	Scram (RPS Channel A/B)

Obj.V.D.3

b.. Alarms other than annunciators on Panel 9-5

Each IRM channel has a set of 4 lights above the IRM range switches

(1) Hi-Hi/Inop (red)

(2) High (amber)

(3) Downscale (White)

(4) Bypassed (White)

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor (SRM) System	Tier #	2	------
K5.01 (10 CFR 55.41.5)	Group #	1	------
Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM : • Detector operation	K/A #	215004K5.01	
	Importance Rating	2.6	------

Proposed Question: **# 35**

Which ONE of the following completes the statement below?

The applied voltage to the SRM detector is __**(1)**__ than the applied voltage used for the IRM detector **AND** the SRM electrode generates an electrical signal __**(2)**__ proportional to neutron flux in the core.

A. **(1)** lower
(2) directly

B. **(1)** higher
(2) directly

C. **(1)** lower
(2) inversely

D. **(1)** higher
(2) inversely

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: SRM voltage is higher. Candidate misconception that SRM detectors detect lower power therefore the voltage detector power requirement is lower.

B **CORRECT:** The SRM (IRM) detector is a fission chamber that has an applied voltage to the electrode of approximately 350 (100) volts. The operating chamber is pressurize with Argon to about 213 (17) psia. They generate an electrical signal proportional to the neutron flux level in the core.

C INCORRECT: SRM voltage is higher and the signal is not inversely proportional. Candidate misconception that SRM detectors detect lower power therefore the voltage detector power requirement is lower. Candidate misconception that Campbeling correction (square root effect) is used by the SRM, and this makes the signal inversely proportional.

D INCORRECT: the signal is not inversely proportional. Candidate misconception that Campbeling correction (square root effect) is used by the SRM, and this makes the signal inversely proportional.

KA Justification:

K/A is met by question asking knowledge of the SRM detector operation. RO knowledge Task. Memory knowledge because RO must recall facts about SRM detector operation.

Question Cognitive Level:

The question tests for the total recall of discrete facts or bits of information, for a single system.

Technical Reference(s): OPL171.019 Rev 13 (Attach if not previously provided)
OPL171.020 Rev 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.D.2 (As available)

Question Source:
Bank # Brunswick 07 #12
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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

ES-401 Sample Written Examination Question Worksheet Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 K5.01	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM : Detector operation

Proposed Question: Common 12

The Source Range Monitor (SRM) detectors are fission chambers that have an applied voltage to an electrode. The applied voltage to the SRM detector is ___________.

A. higher than the applied voltage used for the IRM detector and the SRM electrode generates an electrical signal inversely proportional to neutron flux in the core.

B. lower than the applied voltage used for the IRM detector and the SRM electrode generates an electrical signal inversely proportional to neutron flux in the core.

C. higher than the applied voltage used for the IRM detector and the SRM electrode generates an electrical signal proportional to neutron flux in the core.

D. lower than the applied voltage used for the IRM detector and the SRM electrode generates an electrical signal proportional to neutron flux in the core.

Proposed Answer: C

A. Incorrect – the signal is not inversely proportional.
B. Incorrect – SRM voltage is higher and the signal is not inversely proportional.
D. Incorrect – SRM voltage is higher

Technical Reference(s): SD 09.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____ (As available)

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 ____

OPL171.019
Revision 13
Page 10 of 51

Instructor Notes

(4) For a core of <20,000 Mwd/T exposure, spontaneous fission of Cm -242 is the primary intrinsic neutron source.

(5) For a core of > 20,000 Mwd/T exposure, spontaneous fission of Cm -244 is the primary intrinsic neutron source.

3. Detection Chamber

a. The purpose of the detection chamber is to generate an electrical signal proportional to the neutron flux level in the core.

(3) Ionization chamber

Obj. V.B.3
Obj. V.D.2

(a) The inner electrode of the ionization chamber is supplied with 350 VDC by a high voltage power supply.

OPL171.020
Revision 11
Page 9 of 44

(4) Operating voltage is 100V DC (350V DC for SRM)

Examination Outline Cross-reference:	Level	RO	SRO
215004 Source Range Monitor (SRM) System	Tier #	2	
K5.03 (10 CFR 55.41.5)	Group #	1	
Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM : • Changing detector position	K/A #	215004K5.03	
	Importance Rating	2.8	

Proposed Question: **# 36**

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)	$9.5x10^3$	125	150	$8.0x10^3$

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: **A**

Explanation (Optional):

A **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.

B INCORRECT: Plausible in that with SRM C Not Full in and associated IRM E not on range 3, candidate may believe that it must also be inserted to clear the Rod Block. However, although SRM C is not full in, it is above the Rod Block set point of 145 cps so the Rod Block is bypassed.

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.

KA Justification:

K/A is matched because in the question operational conditions/implications have arisen from the mis-positioning of the SRM detectors. The candidate must determine which detector is causing the conditions and based on his/her knowledge resolve the situation. Knowledge involves recognizing the interaction between the SRM/IRM systems, including consequences and implications.

Question Cognitive Level:

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-OI-92 Rev. 14 (Attach if not previously provided)
OPL171.019 Rev 13

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.019 V.B.8 (As available)

Question Source: Bank #
Modified Bank # BFN 1006 #37 (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 1006

(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:

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)	$9.5x10^3$	95	80	$8.0x10^3$

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. 0014 Page 14 of 14

Illustration 1
(Page 1 of 1)
SRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
SRM High	= 6.8 X 10^4 counts per second	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Downscale	5 counts per second	Rod block unless IRMs on range 3 (or higher) or REACTOR MODE SWITCH in RUN
SRM Detector Wrong Position	145 counts per second	Rod block unless detector full-in, IRMs on range 3 (or higher), or REACTOR MODE SWITCH in RUN
SRM High-High	= 2 x 10^5 counts per second	Scram if shorting links removed

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)

(b) High/INOP (amber)

Refer to OI-92 for current setpoints (2×10^5 cps)

(c) Downscale (white)

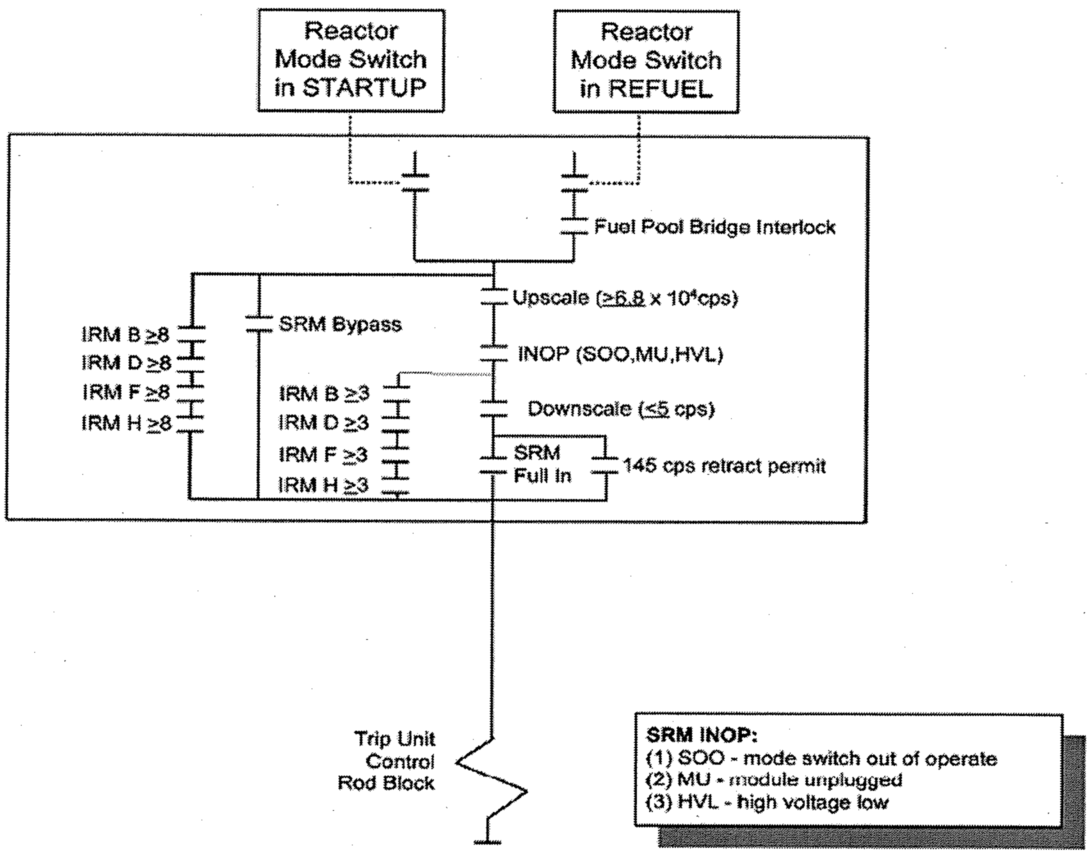
(d) Bypassed (white)

(2) Each SRM channel has a white RETRACT PERMISSIVE light above its respective LCR meter. (≥ minimum cps energizes the light.)

Obj. V.B.6

Set points in OI-92

OPL171.019
Revision 13
Appendix C
Page 50 of 51



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

Examination Outline Cross-reference:	Level	RO	SRO
215005 APRM / LPRM	Tier #	2	------
A3.08 (10CFR 55.41.7)	Group #	1	------
Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including:	K/A #	215005A3.08	
• Control rod block status	Importance Rating	3.7	------

Proposed Question: **# 37**

Unit 2 APRM Channel 3 has a total of 18 LPRM inputs.

Which ONE of the following statements identifies the expected response to this condition?

A. The APRM will produce a Rod Block signal **ONLY**.

B. **NO** Rod Block **OR** Reactor Scram signals are generated.

C. The APRM will produce a Rod Block signal **AND** a Scram signal input to **EACH** 2/4 logic voter module.

D. The APRM will produce a Rod Block signal **AND** a Scram signal input to **ITS RESPECTIVE** 2/4 logic voter module **ONLY.**

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: If the number of un-bypassed LPRM inputs exceeds the minimum number required in the APRM average (<20 total or <3 per level), an APRM INOP condition is applied. This results in a Rod Block only - manual trip must be inserted for inoperable condition.

B INCORRECT: Plausibility based on misconception that since no Reactor Scram signal is generated with this Inop condition, likewise, no Control Rod Block is generated. Also plausible that the candidate may believe the minimum number of LPRM inputs is still available and conditions are not met for Rod Block or Scram Signal.

C INCORRECT: Plausible in that < 20 LPRM inputs to an APRM results in INOP Condition. ALL other APRM Inop signals do result in an APRM Trip. This would be the correct answer for any other APRM Inop Signal.

D INCORRECT: Plausibility based on the misconception that a Scram Signal would result with < 20 LPRMs input into the APRM and that the resultant scram signal would input only into associated logic voter module.

KA Justification:

The KA is met because the question tests ability to monitor automatic operations of the Average Power Range Monitoring System including Control rod block status and scram signal input to voter logic given less than the required 20 LPRM inputs into an APRM.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.148 Rev 12 (Attach if not previously provided)

2-OI-92B Rev. 38

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.148 V.B.7/31 (As available)

Question Source: Bank # OPL171.148 #58

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:

OPL171.148
Revision 12
Page 18 of 106

INSTRUCTOR NOTES

(4) When the LPRM signal is ≤ 3%.

b. The LPRM signals can be manually bypassed from the APRM average flux calculation. This operation can only performed on panel 9-14.

Password entry is required

c. An LPRM may be manually bypassed with either the high voltage applied (BYP/HV ON) (Still have LPRM indication) OR with the high voltage off (BYP/HV OFF). (No LPRM indication avail)

Obj. V.C.1.c
Indication preserved

No Indication

d. A bypassed LPRM will not be included in the APRM average and will not indicate downscale or upscale conditions.

e. For the APRM channel, the total number of LPRM inputs that may be bypassed is 23 before reaching an INOP condition.

Obj. V.B.7.b.(1)
Obj. V.C.1.b.(1)
More later

(1) If the number of un-bypassed LPRM inputs exceeds the minimum number required in the APRM average (<20 total or <3 per level), an APRM INOP condition is applied.

Rod Block only - manual trip must be inserted for inoperable condition.

f. Operability for all aspects of the PRNM system needs to be assessed when bypassing LPRMs.

9. Keylock Mode Switch

- One keylock switch per LPRM/APRM instrument
- Has two positions "OPER" and "INOP".
- The key is removable in either position

10. LPRM alarms (Panel 9-5)

Obj. V.B.5

a. LPRM Upscale and LPRM Downscale

Total Scale = 0 to 125%

(1) The upscale and downscale set point markers are displayed inside the bargraphs and a status indication is displayed above the bargraphs. The solid box above the bargraph indicates that the set point marker is presently exceeded while a hollow box indicates a past condition.

Downscale is less than or equal to 3% of scale AND upscale is greater than or equal to 100% of scale

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 Reactor Core Isolation Cooling System (RCIC)	Tier #	2	
K2.02 (10 CFR 55.41.7)	Group #	1	
Knowledge of electrical power supplies to the following:	K/A #	217000K2.02	
• RCIC initiation signals (logic)	Importance Rating	2.8	

Proposed Question: **# 38**

Which ONE of the following identifies the RCIC initiation logic power supply?

A. 250 VDC RMOV BD A

B. 250 VDC RMOV BD B

C. Div 1 ECCS ATU inverter

D. Div 2 ECCS ATU inverter

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This supplies the Channel/Bus B Isolation Logic. Easily confused by candidates.

B **CORRECT:** This supplies the Initiation Logic and Channel/Bus A Isolation Logic.

C INCORRECT: This supplies 125 VAC to the RCIC Flow controller and various RCIC indicators. HPCI and RCIC system components and power supplies are easily confused by the examinees.

D INCORRECT: This supplies 125 VAC to the HPCI Flow controller and various HPCI indicators. HPCI and RCIC system components and power supplies are easily confused by the examinees.

KA Justification:

K/A asks for knowledge of electrical power supply to RCIC initiation logic. Question is designed to ask directly for the RCIC initiation logic power supply.

Question Cognitive Level:

Question requires recall of discrete information and is therefore a memory or low cognitive question.

Technical Reference(s): OPL171.040 Rev 23 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.7 (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:

8. Failure Modes

a. Loss of Power to the Flow Controller — Obj. V.C.4

(1) Div I ECCS ATU Inverter

(2) Loss of Power causes the controller to go to zero milliamp output and turbine speed would lower to minimum (~600 rpm). However, on Units 1&3, the Div I ECCS Inverter also powers to EGM Control Box which would result in overspeed on Units 1& 3 only. — Reference P&L 3.23 / 3.0.W

b. Loss of control air — Obj.V.B.7 Obj. V.C.4

(1) RCIC steam line steam trap bypass valve (FCV-71-5) fails closed (Unit 3)

(2) RCIC steam line condensate drain valves (FCV 71-6A and 6B) fail closed

(3) RCIC condensate pump Clean Radwaste discharge valves (FCV-71-7A and 7B) fail closed

c. Loss of electrical power to valves — Obj. V.B.7. Obj. V.C.4.

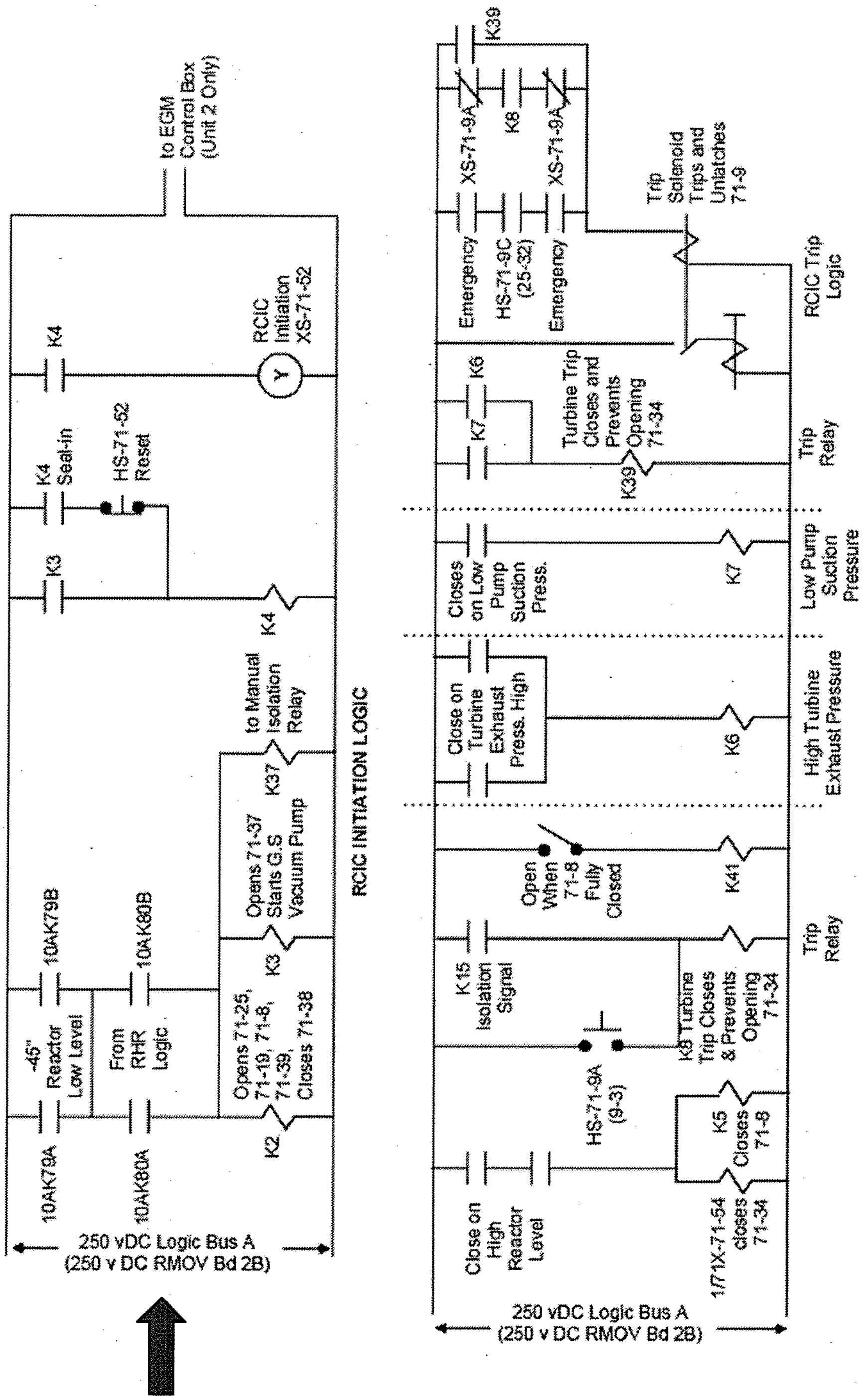
All motor-operated isolation valves remain in the last position upon failure of valve power. Solenoid operated valve FCV 71-5 (Unit 2) fails closed.

d. Loss of Power to Relay Logic — Obj. V.B.7. Obj. V.C.4

(1) If Bus A fails, the automatic initiation circuit and turbine trip solenoid will not operate. Channel A isolation logic circuit is lost. Power is lost to EG-M control box and this causes FCV-71-10 trip governor valve to go wide open (if RCIC is operating). - Unit 2 (Unit 3 EGM power is from DIV I Inverter) — UNIT DIFFERENCE

(2) If power is lost to the EGM Control Box, Springs will re-position the 71-10 servo to fully open the governor valve (Unit 2 only). — UNIT DIFFERENCE

OPL171.040
Revision 23
Appendix C
Page 69 of 74



TP-9: RCIC Initiation and Trip Logic

Examination Outline Cross-reference:	Level	RO	SRO
217000 Reactor Core Isolation Cooling System (RCIC)	Tier #	2	------
K2.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following: • Gland seal compressor (vacuum pump)	K/A #	217000K2.04	
	Importance Rating	2.6	------

Proposed Question: **# 39**

Which ONE of the following completes the statement?

The power supply to the Unit 2 RCIC Vacuum Pump is ________.

A. 250 VDC RMOV BD 2A

B. 250 VDC RMOV BD 2C

C. 480 VAC RMOV BD 2A

D. 480 VAC RMOV BD 2B

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: This is, in fact a power supply to RCIC components; just not the RCIC Vacuum Pump. Refer to attached PRESTARTUP REQUIREMENTS.

B **CORRECT:** 250 VDC RMOV BD 2C is the power supply to the RCIC Vacuum Pump. See Attached Electrical Lineup Checklist.

C INCORRECT: This is, in fact a power supply to RCIC components; just not the RCIC Vacuum Pump. Refer to attached PRESTARTUP REQUIREMENTS.

D INCORRECT: This is, in fact a power supply to RCIC components; just not the RCIC Vacuum Pump. Refer to attached PRESTARTUP REQUIREMENTS.

KA Justification:

The KA is met because the question tests candidate knowledge of power supplies to RCIC Vacuum 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.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 2-OI-71, Rev. 61 / 2-OI-71 Att. 3 Rev. 58 (Attach if not previously provided)

OPL171.040 Rev. 23

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:

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Reactor Core Isolation Cooling	2-OI-71 Rev. 0061 Page 12 of 73

4.0 PRESTARTUP/STANDBY READINESS REQUIREMENTS

NOTE
When Section 4.0 is required to be verified by subsequent Sections, Section 4.0 will be performed.

[1] **VERIFY** the following related system requirements are satisfied:

A. Oil level is visible in RCIC turbine pedestal sight glass. ☐

B. The following panels are energized. (REFER TO 0-OI-57B, 0-OI-57C, and 0-OI-57D)

- 250VDC Reactor MOV Board 2A ☐
- 250VDC Reactor MOV Board 2C ☐
- 240V Lighting Board 2A ☐
- 480V Reactor MOV Board 2A ☐
- 480V Reactor MOV Board 2B ☐
- Panel 2-9-9, Cabinet 2 ☐
- Panel 2-9-9, Cabinet 3 ☐
- Panel 2-9-9, Cabinet 4 ☐
- Panel 2-9-9, Cabinet 5 ☐
- 1E ECCS ATU Inverter (Division I) ☐

ii. Noncondensables are removed by a DC-powered vacuum pump discharging to the suppression pool. If the vacuum is excessive, a valve controlled by condenser pressure, in the vacuum pump discharge line, will automatically open and release noncondensables back to the condenser. The vacuum pump automatically starts on system initiation. — "C" 250VDC RMOV Bd

(c) During operation, liquid from the spray and condensed steam is collected in a receiver section of the barometric condenser and pumped by a DC powered condensate pump back to the suction of the RCIC pump. — "C" 250 VDC RMOV Bd

i. Pump cycles on high and low level signals from the barometric condenser.

ii. Pump is rated at 3 hp.

(d) During periods of system non-use, the barometric condenser is continually drained to Clean Radwaste through two air-operated valves in the condensate pump discharge line. The valves operate off level in the condenser (FCV-71-7A and 7B). These valves automatically close when 71-8 is not fully closed.

(e) High pressure in the barometric condenser alarms at approximately 8" Hg (Only if normally shut steam supply valve FCV-71-8) is not fully closed. (15 sec. TD))

BFN Unit 0	Attachment 3 Reactor Core Isolation Cooling Electrical Lineup Checklist	2-OI-71/ATT-3 Rev. 0058 Page 7 of 7

4.0 ATTACHMENT DATA (continued)

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
Reactor Bldg. - 250V RMOV Bd 2C - El 565'			
8B	2-BKR-071-0017 RCIC SUPPR POOL INBD SUCT VALVE BREAKER (GE-13-41)	ON	____ ____
8D	2-BKR-071-0025 RCIC LUBE OIL COOLING WATER VALVE BREAKER (GE-13-132)	ON	____ ____
10E	2-BKR-071-0031 RCIC TURB BAROMETRIC CNDR VAC PUMP BREAKER	ON	____ ____
Electric Board Room 2B - 250V RMOV Bd 2B - El 593'			
8E1	2-BKR-071-002B/8E1 RCIC SYS LOGIC DIV I-2 PNL 2-25-31	ON	____ ____
5D	2-BKR-071-0034 RCIC PUMP MIN FLOW VALVE BREAKER (GE-13-27)	ON	____ ____
5B	2-BKR-071-0003 RCIC STMLINE OUTBD ISOL VALVE BREAKER (GE-13-16)	ON	____ ____

Examination Outline Cross-reference:	Level	RO	SRO
218000 ADS	Tier #	2	
G2.1.7 (10CFR 55.41.5)	Group #	1	
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	K/A #	218000G2.1.7	
	Importance Rating	4.4	

Proposed Question: **# 40**

Unit 2 was operating at 100% Reactor Power with RHR Pump 2D tagged out of service. A Loss of Coolant Accident with a subsequent Loss of Off Site Power has resulted in the following plant conditions:

- Reactor Water Level is (-)125 inches
- Drywell Pressure is 4.1 psig
- A **AND** C 4KV Shutdown Boards are de-energized

Which ONE of the following identifies the **MINIMUM** action(s), if any, that will prevent the Automatic Depressurization System (ADS) from an Auto-Initiation?

A. **NO** action is required

B. Place **ONLY** ADS Logic Inhibit Switch 'A' to INHIBIT

C. Place **ONLY** ADS Logic Inhibit Switch 'B' to INHIBIT

D. Place **BOTH** ADS Logic Inhibit Switches 'A' **AND** 'B' to INHIBIT

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: RHR Pump C running meets Pump running permissive for System 1 and 2 ADS logic. Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running is required. RHR C Pump is running and NO Core Spray Pumps are running.

B INCORRECT: Plausible in that different combinations of ECCS Pumps operating meet the pump running permissive for different ADS logic channels.

C INCORRECT: Plausible in that different combinations of ECCS Pumps operating meet the pump running permissive for different ADS logic channels.

D **CORRECT**: RHR Pump C running meets Pump running permissive for System 1 and 2 ADS logic. Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running is required. RHR C Pump is running and NO Core Spray Pumps are running

KA Justification:

The KA is met because the question tests candidates' ability to evaluate plant performance and make operational judgments for the ADS System based on operating characteristics, reactor behavior, and instrument interpretation including Reactor Level, Drywell Pressure and Electrical Distribution indications. Based on those indications, candidate must make operational judgment regarding the status of ADS logic.

Question Cognitive Level:

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. 47

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.043 V.B.4 (As available)

Question Source:	Bank #		
	Modified Bank #	BFN 1006 #40	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	Browns Ferry 1006	

(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. 0047 Page 12 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.

BROWNS FERRY 1006

Proposed Question: **# 40**

Unit 2 was operating at 100% Reactor Power with RHR Pump 2B tagged. A Loss of Coolant Accident with a subsequent Loss of Off Site Power has resulted in the following plant conditions:

- Reactor Water Level is (-)125 inches
- Drywell Pressure is 4.1 psig
- B **AND** D 4KV Shutdown Boards are de-energized
- RHR Pump 2A tripped

Which ONE of the following identifies the **MINIMUM** action, if any, that will prevent the Automatic Depressurization System (ADS) from an Auto-Initiation?

A. **NO** action is required

B. Place **ONLY** ADS Logic Inhibit Switch 'A' to INHIBIT

C. Place **ONLY** ADS Logic Inhibit Switch 'B' to INHIBIT

D. Place **BOTH** ADS Logic Inhibit Switches 'A' **AND** 'B' to INHIBIT

Proposed Answer:

Explanation (Optional):

A INCORRECT: Pump running permissive is not met with only Core Spray Pumps A and B running. It is the same permissive for System 1 and 2 ADS logic. Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running is required. No RHR Pumps are running and only A and B Core Spray Pumps are running.

B INCORRECT: Plausible in that channel A logic is made up. However channel C logic is not so there is no requirement to Inhibit System 1 logic.

C INCORRECT: Plausible in that channel B logic is made up. However channel D logic is not so there is no requirement to Inhibit System 1 logic.

D INCORRECT: Plausible in that if the right combination of Core Spray Pumps were running on any RHR Pump running, this would be correct answer.

Justification: To correctly answer this question, candidate must recognize condition not met for automatic initiation of ADS to determine no action is required to prevent inadvertent initiation of ADS logic. 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.

Examination Outline Cross-reference:	Level	RO	SRO
223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off	Tier #	2	------
K4.02 (10 CFR 55.41.7)	Group #	1	------
Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: • Testability	K/A #	223002K4.02	
	Importance Rating	2.7	------

Proposed Question: **# 41**

Which ONE of the following explains the response of the isolation logic for Reactor Water Cleanup Suction Isolation Valves?

A trip of **BOTH** division 1 (A, C) low level sensor relay(s) within a logic trip channel will cause a __**(1)**__ isolation **AND** __**(2)**__ closure.

A. **(1)** half
(2) NO valve

B. **(1)** half
(2) inboard valve

C. **(1)** full
(2) inboard valve

D. **(1)** full
(2) inboard **AND** outboard valve

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Typical PCIS logic is designed so each valve has 2 trip channels, each containing 4 level sensor relays two from division 1 (A and C contacts in series) and two from division 2 (B and D contacts in series) with both sets of contacts in parallel. The trip of one or both division 1 low level sensor relays in a single channel will cause a half isolation on the Inbd and Obrd valves and no valve closure. The isolation is said to be half-cocked. A trip of one or both low level sensor relays in each division will cause a full isolation and valve closure. (Inbd and Obrd valves)

B INCORRECT: Half is correct, but no valve closure will occur. It would take a trip of a sensor relay in the other low level sensor division to affect closure.

C INCORRECT: Full is incorrect. There would be no valve closure for the conditions given. Misconception by candidate that a trip of any two sensor relays would cause valve closure.

D INCORRECT: Full is incorrect. Neither valve would move under the given conditions. Misconception of logic operation.

KA Justification:

The K/A is matched because the stem asks for knowledge of how a trip channel is tested and how an isolation does not occur. This is RO level knowledge because Instrument technicians test isolation instrumentation daily in the plant without isolations occurring. Knowledge is covered in lesson plan.

Question Cognitive Level:

Examinee must know discrete bits of information about the system.

Technical Reference(s): OPL171.017 Rev 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

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

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

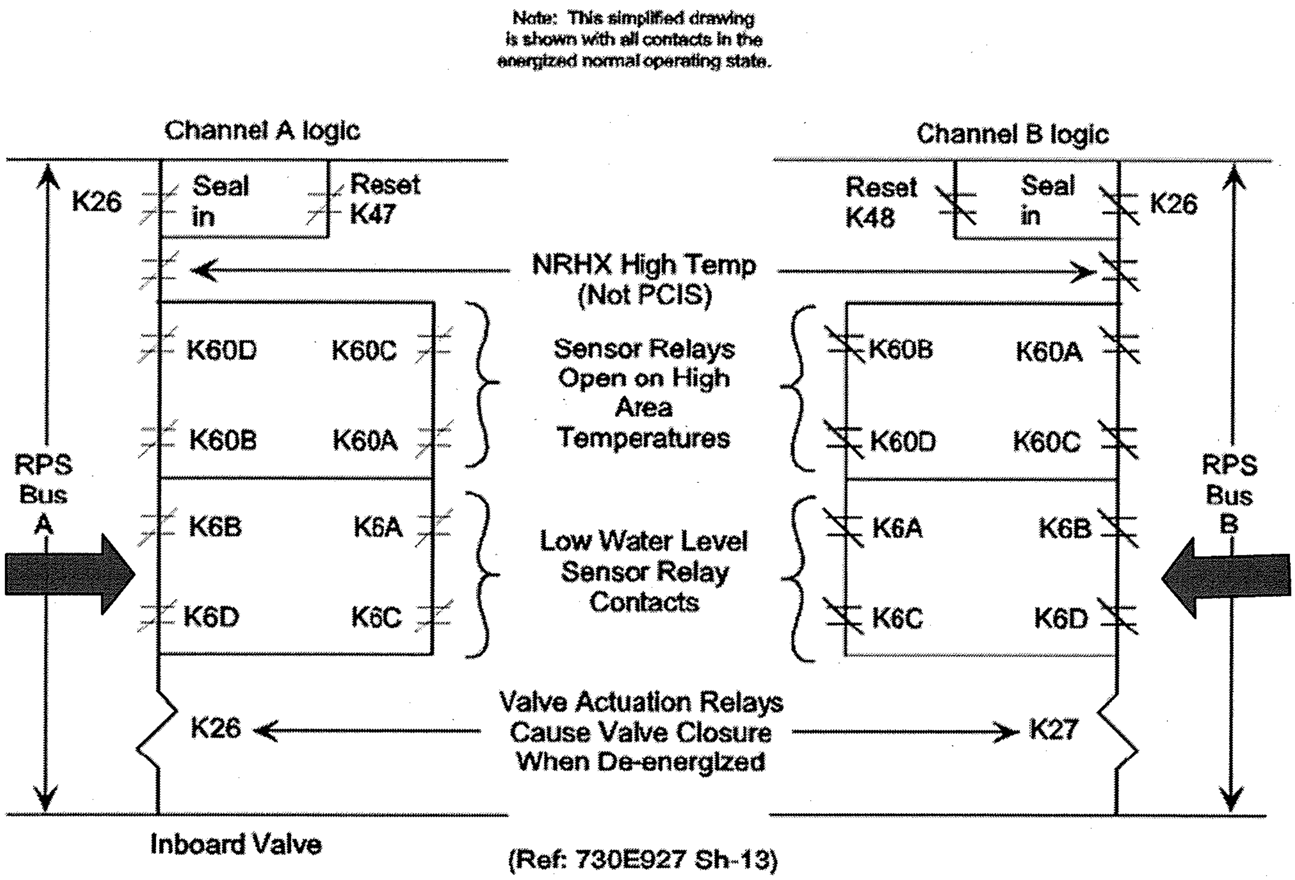
Question History: Last NRC Exam

(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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:


Note: This simplified drawing is shown with all contacts in the energized normal operating state.
Channel A logic
Channel B logic
K26
Seal in
Reset K47
Reset K48
Seal in
K26
NRHX High Temp (Not PCIS)
K60D
K60C
K60B
K60A
Sensor Relays Open on High Area Temperatures
K60B
K60A
K60D
K60C
RPS Bus A
RPS Bus B
K6B
K6A
K6D
K6C
Low Water Level Sensor Relay Contacts
K6A
K6B
K6C
K6D
K26
Valve Actuation Relays Cause Valve Closure When De-energized
K27
Inboard Valve
(Ref: 730E927 Sh-13)

RWCU Valves

C. Typical PCIS Isolation Logic

1. A typical logic arrangement for the PCIS valves (except MSIVs) is shown in TP-1. This figure shows that two separate trip channels (A and B) are each provided with two sensor relay contacts (A/C and B/D).

   PCIS de-energizes to isolate (except HPCI/RCIC)

   Obj. V.B.1
   Obj. V.C.1

   a. This arrangement creates trip subchannels A1/A2 and B1/B2.

   b. A trip of either sensor relay within a trip channel will cause opening of the associated contact and de-energization of the associated relay. This condition will create a "half isolation" signal within both logic channels but NO VALVE MOVEMENT.

      HPCI/RCIC are energize to actuate

      Obj. V.B.3
      Obj. V.C.3

   c. Should a trip of either sensor relay in the other trip channel occur, conditions will exist to de-energize the valve actuation relays in each logic channel, causing both isolation valves to close.

      PCIS logic is arranged as follows:

A1 **OR** A2
AND
B1 **OR** B2

= **Inboard AND Outboard** valve closure

Note: Most PCIS logic is assembled as above. The MSL drains however are an exception.

The MSL drain logic is as follows:

A1 **AND** B1 = I/B valve closure

A2 **AND** B2 = O/B valve closure

2. The Channel A logic is powered from RPS Bus A, and contains the valve actuation relay associated with the Inboard Valve.

   A1 A2

INSTRUCTOR NOTES

B1 B2

3. The Channel B logic is powered from RPS Bus B, and contains the valve actuation relay associated with the Outboard Valve.

4. It is noteworthy to point out that while RPS A &B supplies power to their respective logic channels, a loss of "A" RPS would result in the closure of the I/B MSL drain valve FCV-55, and a loss of "B" RPS would result in the closure of the O/B MSL drain valve FCV-56. This is due to a loss of power to their respective relays rather than satisfying the logic.

Examination Outline Cross-reference:	Level	RO	SRO
223002 PCIS/Nuclear Steam Supply Shutoff **K4.05 (10CFR 55.41.7)** Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: • Single failures will not impair the function ability of the system	Tier #	2	------
	Group #	1	------
	K/A #	223002K4.05	
	Importance Rating	2.9	------

Proposed Question: **# 42**

Unit 2 is starting up following a refueling outage with Reactor Pressure at 80 psig.

RPS MG Set A has tripped. RPS Distribution Panel A has **NOT** yet been transferred to its alternate source.

The 3 X Low Reactor Water Level instrument providing input to PCIS Channel B2 fails downscale.

Which ONE of the following describes the response of MSIVs **AND** Main Steam Line Drains?

A. **ONLY** the Inboard Steam Line Drain valve **AND ALL** MSIVs close.

B. **ONLY** the Outboard Steam Line Drain valve **AND ALL** MSIVs close.

C. Inboard **AND** Outboard Steam Line Drain valves **AND ALL** MSIVs close.

D. Inboard **AND** Outboard Steam Line Drain valves close, **AND ALL** MSIVs remain open.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that Loss of RPS A will close MSL Inboard Drain Valve AND deenergize MSIV AC solenoids. However with B2 failed downscale and RPS A deenergized, both A and B logic are made up to deenergize both AC and DC solenoids and provides an isolation signal to the outboard MSL drain. If B1 channel had failed, this would be the correct answer.

B INCORRECT: Plausibility based on misconception that only outboard will isolate as result of combination of logic power and failure of B2. The inboard valve will close as a result of loss of relay power with loss of RPS A. If RPS B had failed, this would be the correct answer.

C **CORRECT**: Channel B2 tripped would give a Group 1 logic *B/D* tripped, loss of RPS A would remove power from Group 1 logic A/C and result in a full MSIV isolation. A2 (Loss of RPS) and B2 closes outboard steam line drain. Loss of A logic power from RPS A will close the Inboard steam line drains.

D INCORRECT: Plausibility based on misconception that DC Pilot Solenoids would remain energized and therefore MSIVs remain open since either solenoid energized maintains the valves open. If B logic was also powered from 250 VDC, like the DC solenoids, this would be the correct answer.

KA Justification:

The KA is met because the question tests candidate's knowledge of Primary Containment Isolation System design features and interlocks which provide for single failures not impairing the function ability of the system.

Question Cognitive Level:

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-1, Rev. 47 (Attach if not previously provided)

OPL171.017, Rev.15

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.017 V.B.3 (As available)

Question Source:	Bank #	Brunswick 07 #17	
	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:

C. Typical PCIS Isolation Logic

1. A typical logic arrangement for the PCIS valves (except MSIVs) is shown in TP-1. This figure shows that two separate trip channels (A and B) are each provided with two sensor relay contacts (A/C and B/D).

   PCIS de-energizes to isolate (except HPCI/RCIC)

   Obj. V.B.1
   Obj. V.C.1

   a. This arrangement creates trip subchannels A1/A2 and B1/B2.

   b. A trip of either sensor relay within a trip channel will cause opening of the associated contact and <u>de-energization</u> of the associated relay. This condition will create a "half isolation" signal within <u>both</u> logic channels but <u>NO VALVE MOVEMENT</u>.

      HPCI/RCIC are <u>energize</u> to actuate

      Obj. V.B.3
      Obj. V.C.3

   c. Should a trip of either sensor relay in the other trip channel occur, conditions will exist to de-energize the valve actuation relays in <u>each</u> logic channel, causing <u>both</u> isolation valves to close.

      PCIS logic is arranged as follows:

A1 **OR** A2 **<u>AND</u>** B1 **OR** B2	= **Inboard AND Outboard** valve closure

      Note: <u>Most</u> PCIS logic is assembled as above. The MSL drains however are an exception.

      The MSL drain logic is as follows:

      A1 **AND** B1 = I/B valve closure

      A2 **AND** B2 = O/B valve closure

2. The Channel A logic is powered from RPS Bus A, and contains the valve actuation relay associated with the Inboard Valve.

   A1 A2

INSTRUCTOR NOTES

3. The Channel B logic is powered from RPS Bus B, and contains the valve actuation relay associated with the Outboard Valve. — B1 B2

4. It is noteworthy to point out that while RPS A &B supplies power to their respective logic channels, a loss of "A" RPS would result in the closure of the I/B MSL drain valve FCV-55, and a loss of "B" RPS would result in the closure of the O/B MSL drain valve FCV-56. This is due to a loss of power to their respective relays rather than satisfying the logic.

D. Group 1 (MSIV) Isolation Logic

1. TP-2 provides a simplified diagram of the isolation logic for the "A" main steamline inboard isolation valve (FCV-1-14). — 2-730E927-10

2. The MSIV is provided with both an AC-powered pilot solenoid (FSV-1-14C) and a DC-powered pilot solenoid (FSV-1-14B). — Obj. V.B.2, Obj. V.C.2

   Both of these pilot solenoids must be de-energized to cause the MSIV to close.

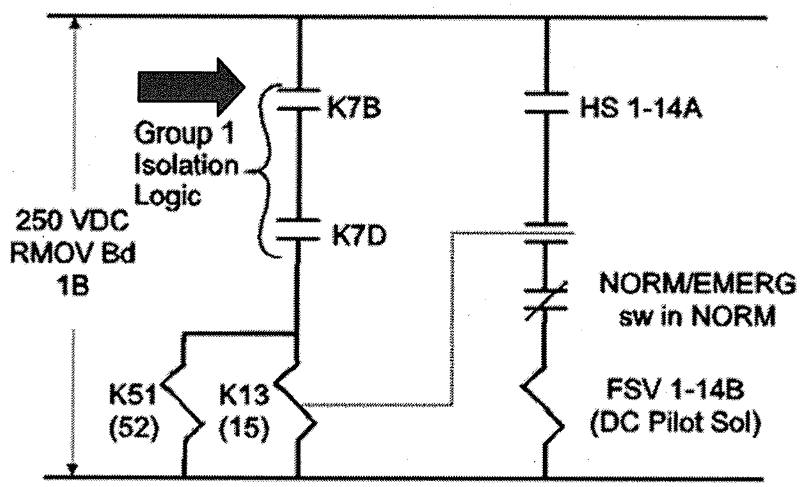
3. With the control handswitch in the AUTO/OPEN position, the associated HS-1-14A contacts will be closed.

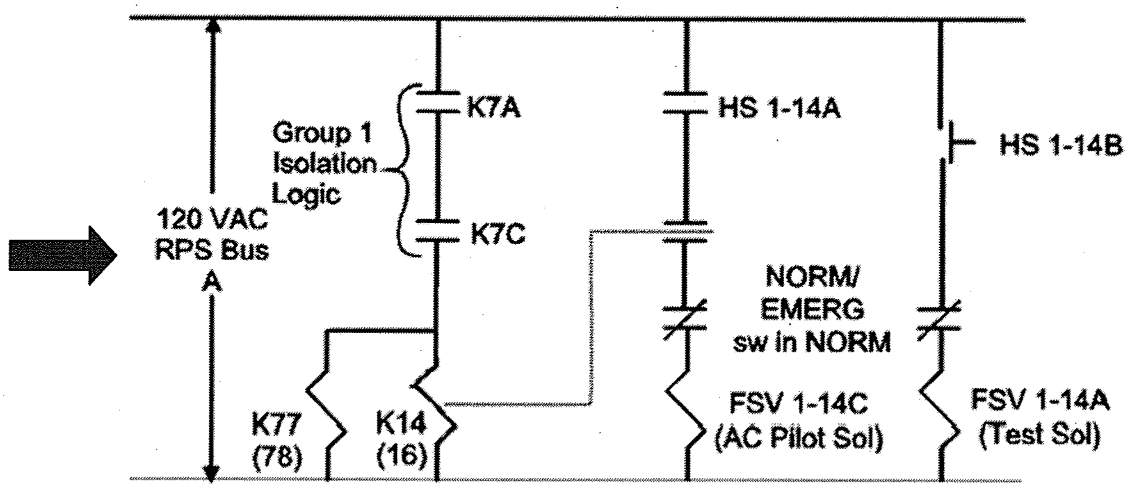
   a. Should a Group 1 isolation signal exist, the K7A,B,C,D relays will de-energize (see TP-3), causing the associated contacts to open.

   b. When these contacts open, the K13/K51 and K14/K77 relays de-energize, opening the associated contacts. This will cause the pilot solenoids to de-energize and the MSIV will close.

4. Further detail regarding the MSIV isolation and reset logic can be seen in TP-3. This is a simplified illustration of the A1 isolation Sub-channel (relay K7A) — 2-730E927-7

OPL171.017
Revision 15
Appendix C
Page 54 of 56



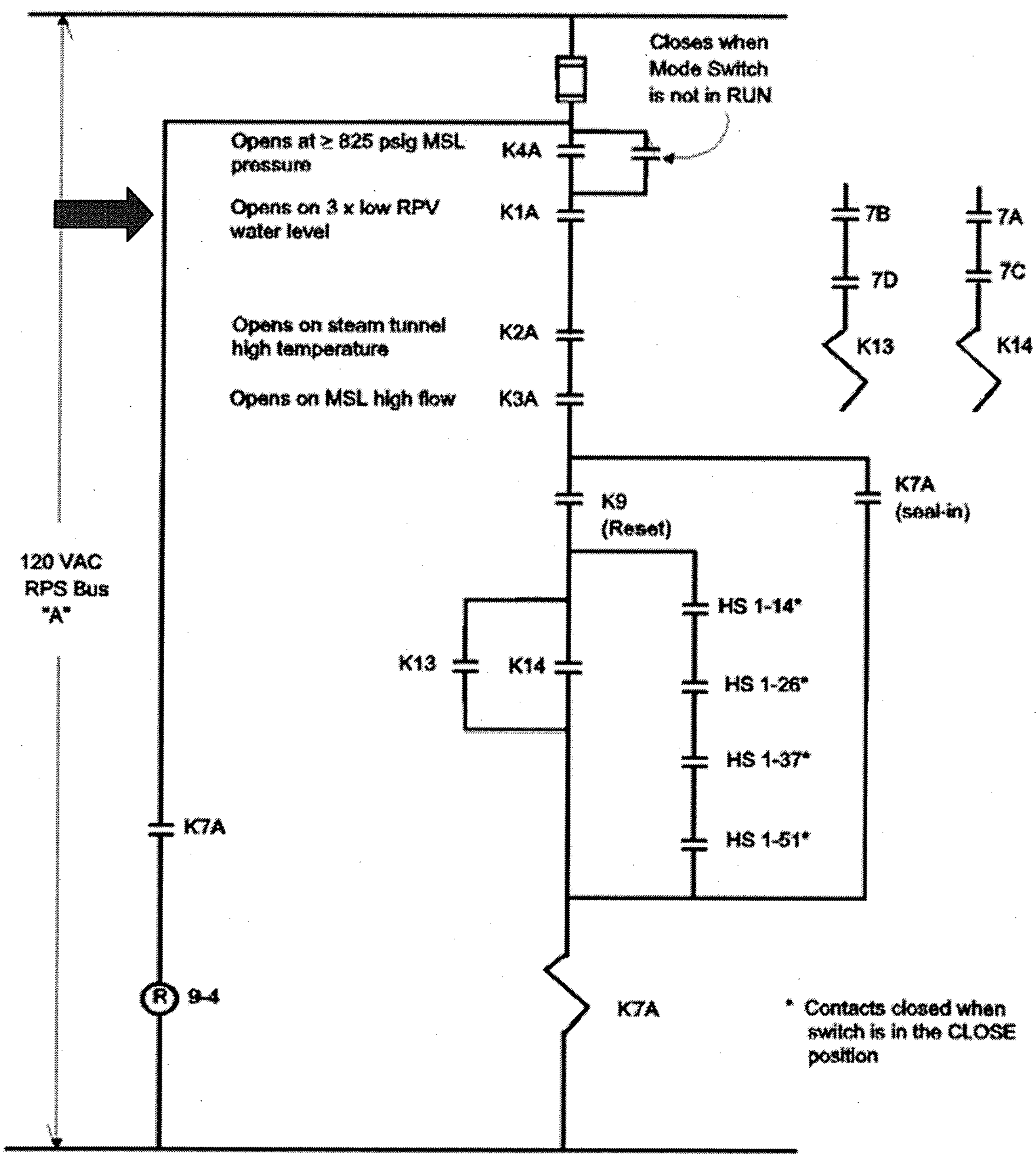


* HS 1-14 contacts shown with the control switch in the CLOSED position. Contacts will be closed with the control switch in the AUTO/OPEN position

(Reference Drawing: 2-730E927 sheet 10)

TP-2 TYPICAL MSIV CONTROL CIRCUIT (FCV-1-14)

OPL171.017
Revision 15
Appendix C
Page 55 of 56



(Reference: 2-730E927 SH7)

TP-3 GROUP 1 ISOLATION AND RESET LOGIC CHANNEL A1

BFN Unit 2	Main Steam System	2-OI-1 Rev. 0047 Page 9 of 64

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.

ES-401 Sample Written Examination Question Worksheet Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 K4.05	
	Importance Rating	2.9	

Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Single failures will not impair the function ability of the system .

Proposed Question: Common 17

RPS MG Set A has tripped. RPS Distribution Panel A has NOT yet been transferred to its alternate source.

The LL3 instrument providing input to PCIS Channel B2 fails downscale.

Which of the following describes the response of MSIVs and Steam Line Drains?

A. Only the Inboard Steam Line Drain valve and all MSIVs close.

B. Only the Outboard Steam Line Drain valve and all MSIVs close.

C. Inboard and Outboard Steam Line Drain valves and all MSIVs close.

D. Inboard and Outboard Steam Line Drain valves close, and all MSIVs remain open.

Proposed Answer: C

Explanation (Optional):

Channel B2 tripped would give a Group 1 logic B/D tripped, loss of RPS A would remove power from Group 1 logic A/C and result in a full MSIV isolation. A2 (Loss of RPS) and B2 closes outboard steam line drain. Loss of A logic power from RPS A will close the Inboard steam line drain.

See Figure 25.7 in SD-25

Technical Reference(s): SD-025 (Attach if not previously provided)

Examination Outline Cross-reference:	Level	RO	SRO
239002 SRVs	Tier #	2	------
K1.01 (10CFR 55.41.3)	Group #	1	------
Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following:	K/A #	239002K1.01	
• Nuclear boiler	Importance Rating	3.8	------

Proposed Question: **# 43**

During a transient on Unit 1, Reactor Pressure reached 1150 psig.

Which ONE of the following identifies how many SRVs opened?

A. Four

B. Eight

C. Nine

D. Thirteen

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausible in that this would be the correct answer if Reactor Pressure was between 1135 and 1145 psig.

B **CORRECT**: The first two groups open with Reactor Pressure > 1145 psig. Each of these groups has 4 valves.

C INCORRECT**:** Plausible in that this would be the correct answer if group 2 had 5 SRVs instead of group 3

D INCORRECT: Plausible in that this would be the correct answer if Reactor Pressure was > 1155 psig.

KA Justification:

The KA is met because the question tests the candidates' knowledge of the cause-effect relationship between the Nuclear Boiler and SRVs.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.009, Rev. 11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.009 V.B.2 (As available)

Question Source:
Bank # OPL171.009 #3
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:

(e) This 'relief mode' logic can be defeated by use of a switch on 9-3. This switch "MSRV AUTO ACTUATION LOGIC INHIBIT" (XS-1-202) also brings in an alarm on 9-3.

d. Valve setpoints for safety function — Obj. V.B.2, Obj. V.C.1, Obj. V.D.1, Obj. V.E.1

(1) 4 valves @ 1135 psig ± 3%

(2) 4 valves @ 1145 psig ± 3%

(3) 5 valves @ 1155 psig ± 3% — TP-3

e. Blowdown path — Obj. V.B.2, Obj. V.C.1

(1) Individually piped to the suppression pool via the T-Quenchers below the minimum water level. The T-Quenchers enhance thermal mixing in the Suppression Pool

(2) Each SRV has two vacuum breakers (one 10 inch and one 2 1/2 inch vacuum breaker on SRV tailpiece)in parallel. They are provided to allow entry of drywell air into the relief line to prevent water from the suppression pool being "pulled" up into the relief line upon completion of blowdown. Without the vacuum breaker the steam in the relief line condenses and forms a vacuum in the relief line drawing water from the pool into the line. Subsequent reopening of the valve with its relief line partially filled with water could over pressurize the relief line, with a potential for tail pipe damage. If the vacuum breakers were to fail open, steam could be discharged directly to the drywell during SRV operation. — Obj. V.B.5, Obj. V.C.2, Obj. V.C.3, Obj. V.D.2, Obj. V.E.2

BROWNS FERRY EXAM BANK

OPL171.009 3

During a transient, RPV pressure reached 1150 psig.

Assuming no operator action, how many SRVs opened?

3. A. Four

B. Eight

C. Nine

D. Thirteen

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
259002 Reactor Water Level Control System	Tier #	2	
A1.01 (10 CFR 55.41.5)	Group #	1	
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including:	K/A #	259002A1.05	
• Reactor water level	Importance Rating	3.8	

Proposed Question: **# 44**

Unit 2 Feedwater Level Control System (FWLCS) is operating in 3-Element Control with Narrow Range Level Instruments indicating as follows:

- 2-LT-3-53, LEVEL A, (+) 46 inches
- 2-LT-3-60, LEVEL B, (+) 32 inches
- 2-LT-3-206, LEVEL C, (+) 34 inches
- 2-LT-3-253, LEVEL D, (+) 33 inches

Which ONE of the following completes the statement?

If 2-LT-3-60, LEVEL B, is manually bypassed, the FWLCS will control Reactor Water Level based on ______.

A. **ONLY** the 2-LT-3-206 instrument

B. **LOWEST** of 2-LT-3-206 **OR** 2-LT-3-253 instruments

C. **AVERAGE** of 2-LT-3-206 **AND** 2-LT-3-253 instruments

D. **AVERAGE** of 2-LT-3-53, 2-LT-3-206, **AND** 2-LT-3-253 instruments

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that if FWLCS selected the middle of the 3 remaining channels when one channel is bypassed, this would be the correct answer.

B INCORRECT: Plausible in that if FWLCS selected the lower of the channels not manually or automatically bypassed, this would be the correct answer.

C **CORRECT:** The average level value is used for the three element control logic. The algorithm validates each level signal by comparing them to the average. Level signals that deviate from the average by more than 8 inches are declared invalid, and are discarded from the average. LT-3-53 deviation is > 8" and is bypassed and LT-3-60 is manually bypassed. 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

D INCORRECT: Plausible in that if candidate fails to recognize that 2-LT-3-53, LEVEL A is bypassed due to deviation >8 inches from average, this would be the correct answer.

KA Justification:

The KA is met because the question test candidates' ability to predict and monitor changes in Reactor water level associated with operating the Reactor Water Level Control System. Candidate must recognize that one level channel meets the criteria to be automatically bypassed. Then, when another channel is manually bypassed, candidate must predict how the level control logic will function to monitor for expected changes in Reactor Level.

Question Cognitive Level:

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): OPL171.012 Rev 14 (Attach if not previously provided)
2-OI-3 Rev 136

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.012 V.B.5 (As available)

Question Source:
Bank # BFN 1006 Audit #44
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: Although this question has been modified from its original bank form to meet the KA, it does not meet the criteria for a significantly modified question and is therefore designated as a Bank Question.

BFN Unit 2	Reactor Feedwater System	2-OI-3 Rev. 0136 Page 200 of 216

Illustration 8
(Page 1 of 7)

RFWCS Instrumentation

1.0 NARROW RANGE REACTOR WATER LEVEL

1.1 Components

2-LI-3-53

2-LI-3-60

2-LI-3-206

2-LI-3-253

1.2 Description

The instruments are located on Panel 2-9-5 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, 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.

4. Additional process measurements include: Obj. V.B.1

   a. Four feedwater line temperature RTD outputs.

   b. Three reactor feedwater pump turbine speed signals output from the three Woodward Governors. Each has 2 MPUs but only one needed for speed indication to pnl. 9-6. Third MPU is for zero speed.

B. Component Description

1. Reactor Water Level Obj. V.B.1 Obj. V.D.5

   a. Four independent narrow range level transmitters (LT-3-53,-60,-206 and 253). They are differential pressure transmitters connected to water reference condensing chambers. Digital readouts are spanned for a reactor level of -10 to + 70 inches but, analog range is still 0" to 60".

   b. The control algorithm checks the signal quality. If BAD (failed or out-of-range high or low), the signal is discarded. If GOOD, the signal is further processed.

   c. Each level signal is pressure compensated for density differences by the algorithm and the four signals are averaged.

   d. The algorithm validates each level signal by comparing them to the average. Level signals that deviate from the average by more than 8 inches are declared invalid, and are discarded from the average.

OPL171.012
Revision 14
Page 16 of 85

e. The algorithm applies a second validation process to prevent an individual GOOD, but faulty level signal from degrading the average. The average is compared to the median level signal. Where there is an even number of signals, the median will select the higher of the two middle values. If the average and median values deviate by more than 4 inches, the algorithm will validate the individual level signals to the median, instead of the average. In this case any individual level signal that deviates from the median by more than 8 inches is declared invalid and is discarded from the calculations.

GOOD in this case means on scale but faulty. **Example:** variable leg leak which causes a low level signal on two LIs.. 23",24",34",& 35"= ave. of 29". Median value of 34" > 29" by 5". The 23" & 24" signals are discarded causing median to go back to an average of 34.5".

f. The average level value is used for the single element and three element control logic's.

(1) The individual density compensated levels are output to Control Room indicators. Obj. V.B.1

(2) The average level is output to one pen of a two-pen recorder.

(3) Reactor Vessel high and low level alarms are generated by comparing the average level to high (>39") and low (<27") setpoints.

(4) The average level is also used in the Recirculation pump runback level interlock logic within the algorithm. Obj. V.B.7 Obj. V.C.6

g. If one level signal is BAD or invalid, the algorithm will calculate the average of the three remaining level signals and will control on that value. Obj. V.B.6 Obj. V.C.5

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 ≈ 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.

Examination Outline Cross-reference:	Level	RO	SRO
261000 SGTS	Tier #	2	-------
K4.05 (10CFR 55.41.7)	Group #	1	-------
Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: • Fission product gas removal	K/A #	261000K4.05	
	Importance Rating	2.6	-------

Proposed Question: **# 45**

Which ONE of the following completes the statement?

Standby Gas Treatment System __**(1)**__ are designed to remove a **MAXIMUM** of __**(2)**__ of elemental iodine.

A. **(1)** HEPA Filters
(2) 70%

B. **(1)** Carbon Beds
(2) 70%

C. **(1)** HEPA Filters
(2) 99.9%

D. **(1)** Carbon Beds
(2) 99.9%

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that HEPA filters function to remove fine particulate matter. Part 2 incorrect – Plausible in that this is a recognizable value associated with the performance of SGTS filter trains. The electric heaters reduce Relative Humidity down to 70% which is part of the criteria for Carbon Bed iodine removal capability.

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:** Parts 1 and 2 correct - Carbon Beds are designed to remove at least 99.9% of elemental iodine upon entering conditions of 70% relative humidity at 190°F.

KA Justification:

The KA is met because the question tests candidates' knowledge of Standby Gas Treatment System Carbon Bed design criteria which provide for fission product gas removal.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.018 Rev. 10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.018 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

k. Decay heat removal crosstie valves

Crosstie solenoid valves and control switches have been removed because of inability to meet EQ requirements. Line size has been increased and a manual, locked damper has been placed in the lines to ensure sufficient flow. — DCN W10416A

l. Cross-tie valve for Trains A and B (22) — TP-1

(1) Normally closed — Review H.O.2 (PIP-95-32)

(2) No automatic actions

(3) Used for full cross-tie capability between Trains A and B.

(4) Normally powered from Dsl Aux Bd A, automatically transfers to Dsl Aux Bd B on loss of power to Board A. — Obj. V.E.5

3. Moisture Separator — Obj. V.B.6.a, Obj. V.C.4.a, Obj. V.D.4.a, Obj. V.E.2

a. Reduces moisture content of incoming air

b. Woven nylon mesh, traps water droplets

c. Moisture drains by gravity to SBGT sump and is then pumped to Radwaste.

4. Electric Heater — Obj. V.B.6.b, Obj. V.C.4.b, Obj. V.D.4.b, Obj. V.E.2

a. The relative humidity heater reduces relative humidity to < 70%.

b. 40kW heaters for relative humidity control. SGT A and B powered from A and B 480v Dsl Aux Bds respectively. SGT C from the SGT board.

INSTRUCTOR NOTES

c. 15kW charcoal bed heaters formerly maintained a 125°F charcoal bed temperature when SBGT was out of service. Heater control switches were spring-return-to-neutral and required resetting after SBGT operation.
Due to the Technical Specification requirement of 10 hours' monthly operation with the relative humidity heaters in service, the charcoal bed heaters are no longer needed

LER 85-029-01

d. The relative humidity heater is energized automatically on startup by the fan breaker closure and is de-energized on shutdown by fan breaker opening.

e. The heater will also trip if ambient temperature reaches 180°F.

f. If 480 Volt Load Shed logic is initiated, the Train A and B relative humidity heaters will automatically trip. They will restart after 40 seconds. Train C is not affected by the 480 volt load shed logic.

g. Relative humidity heater control switches (12, 34, 60) in ON or OFF cause annunciation.

5. Prefilter

Used to remove large particles (dust, dirt, lint) and to protect HEPA filter

Obj. V.B.6.c
Obj. V.C.4.c
Obj. V.D.4.c
Obj. V.E.2

6. HEPA Filter

Removes 99.9% of 0.3 micron particles

Obj. V.B.6.d
Obj. V.C.4.d
Obj. V.D.4.d
Obj. V.E.2

7. Carbon Bed (Adsorber Type)

a. Designed to remove at least 95% of iodine in the form of methyl iodine (CH_3I) and 99.9% of elemental iodine upon entering conditions of 70% relative humidity at 190°F

b. Made up of individual rectangular canisters of charcoal

Obj. V.B.6.e
Obj. V.C.4.e
Obj. V.D.4.e
Obj. V.E.2

Examination Outline Cross-reference:	Level	RO	SRO
262001 A.C. Electrical Distribution	Tier #	2	------
K3.04 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following:	K/A #	262001K3.04	
• Uninterruptible power supply	Importance Rating	3.1	------

Proposed Question: **# 46**

The Unit 1 Unit Preferred Inverter is operating in a normal lineup, when a loss of off-site power **AND** a failure of DG "A" to start occurs.

Based **ONLY** on the above plant conditions, which ONE of the responses will identify the power source for Battery Board 1, panel 10?

The Unit Preferred Inverter is powered from ___________________ .

A. 480V RMOV BD 1A

B. 250 VDC Battery Board 4

C. 250 VDC Battery Board 5

D. the Unit Preferred Transformer

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: The UPS Rectifier/Inverter is normally powered from the 480V RMOV BD 1A, but it is NOT energized based on the conditions given. Plausible because the candidate may believe that 480V RMOV BD supplied by auto transfer to DG "B".

B INCORRECT: Battery Board 4 is the alternate DC supply to the inverter and would have to be manually shifted to supply it. Plausible because easily confused with Battery Board 5 and it is the normal supply to one of the MMG's. MMG's are also a Unit Preferred System.

C **CORRECT:** Loss of off-site power and a failure of DG "A" to start would result in no power to 4kV SD BD 1A, 480V SD BD 1A, and 480V RMOV BD 1A, which is the Normal supply to the Unit Preferred Rectifier/Inverter. The UPS would automatically shift to 250 VDC Battery Board 5 supplying the inverter, when the diode in the inverter is no longer reversed biased by the rectifier output.

D INCORRECT: The Unit Preferred Transformer is supplied by 480V RMOV BD 1A, which is also the normal supply to the Rectifier/Inverter. This RMOV Board has no power based on the given conditions. IF it were powered, it would have to be manually shifted to supply the static inverter. Plausible because candidate may believe it is powered from 480V RMOV Bd "B".

KA Justification:

The KA is met because the question tests knowledge of the effects of loss of offsite power and failure of EDG A has on the Unit 1 Unit Preferred Inverter which is an uninterruptible power supply.

Question Cognitive Level:

This question is low cognitive or memory question.

Technical Reference(s): OPL171.102 Rev 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.2.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 X

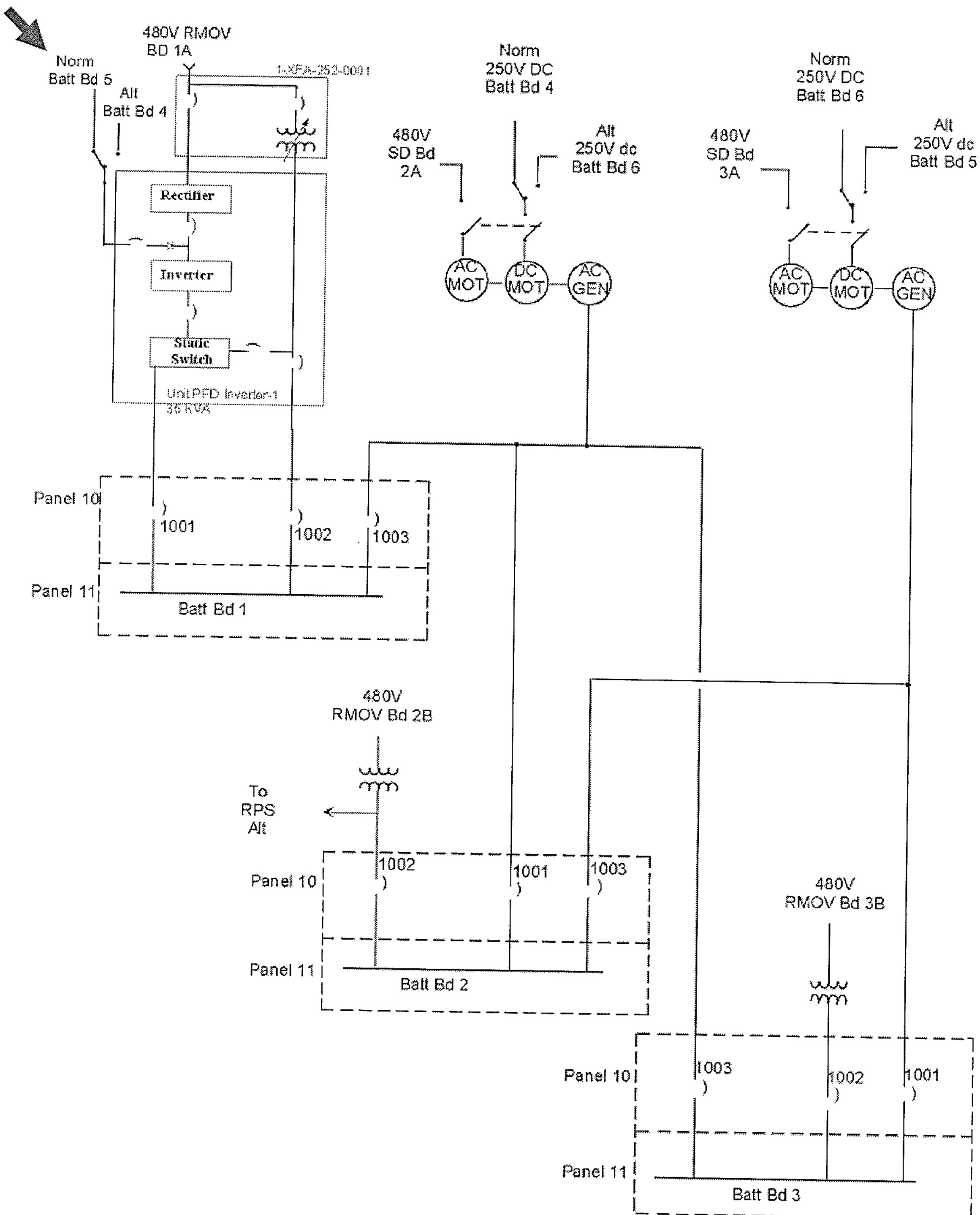
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

OPL171.102
Revision 7
Page 56 of 67



TP-2: POWER SUPPLIES TO UPS BATTERY BOARD CABINETS

Instructor Notes

TP-2

(b) The INVERTER Unit (Unit 1)

Unit 1 is powered by an uninterruptible power supply (inverter unit).Normal supply is 480VAC to the rectifier/ inverter unit itself where it is converted to DC volts then back to a 'smooth' 120/ 240VAC signal fed to Batt. Bd 1 Panel 10/11. There is a backup 250VDC backup power supply fed to the inverter unit for a bumpless transfer in case of loss of AC power. Additionally there is a regulated AC alternate power to the inverter static switch for a continuation of power in case of inverter failure.

(c) Unit Preferred Transformer

The alternate power source is the unit preferred transformer. This transformer receives power from the 480V portion of the standby AC power system. Unit 1 transformer is from the 480 V RMOV Bd 1A Unit 2 & 3 transformers are powered from 480V RMOV Board 2B & 3B. Transfers to this source are done manually at battery board 2 panel 11.

Note that the UPS transformer is also the alternate RPS power supply for U2

Examination Outline Cross-reference:	Level	RO	SRO
262002 UPS (AC/DC)	Tier #	2	-------
A2.02 (10CFR 55.41.5)	Group #	1	-------
Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: • Over voltage	K/A #	262002A2.02	
	Importance Rating	2.5	-------

Proposed Question: **# 47**

Which ONE of the following completes the statements?

The 1001 **AND** 1003 breaker from Unit 2 Unit Preferred System (UPS) Motor-Motor-Generator (MMG) set will trip on __**(1)**__ at the output of the MMG.

In accordance with 2-AOI-57-4, "Loss of Unit Preferred," if UPS is lost, the crew must __**(2)**__.

A. **(1)** under frequency **ONLY**
(2) take manual control of Master Feedwater Level Controller

B. **(1)** under frequency **ONLY**
(2) verify Reactor Feedwater Control System is maintaining Reactor Water Level

C. **(1)** under frequency **OR** overvoltage
(2) take manual control of Master Feedwater Level Controller

D. **(1)** under frequency **OR** overvoltage
(2) verify Reactor Feedwater Control System is maintaining Reactor Water Level

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that under frequency ONLY at the generator output will trip the DC Motor of the MMG set. Part 2 incorrect – Plausible in that loss of UPS does impact Feedwater Level Control System. RFW Control System Panel Display Stations on Panel 2-9-5 is disabled. PDS Controls are inoperative and displays become blank. The RFW Control System continues to control system parameters according to water level setpoint.

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

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

D **CORRECT:** Part 1 correct – The 1001 and 1003 breakers from an MMG set will trip on overvoltage or under frequency at the output of the MMG. Part 2 correct – Per 2-AOI-57-4, Subsequent action 4.2[1], verify RFW Control System is maintaining Reactor Water Level. The RFW Control System continues to control system parameters according to water level setpoint.

KA Justification:

The KA is met because the question tests the Candidates' ability to predict the impacts of Over voltage on the Unit 2 Unit Preferred System MMG which is an uninterruptable power supply. Then, assess impact of loss of UPS on FWLC to determine correct actions in accordance with 2-AOI-57-4.

Question Cognitive Level:

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. Candidate must predict impact of loss of UPS on FWLC to determine appropriate action to take.

Technical Reference(s): OPL171.102 Rev. 7 (Attach if not previously provided)

2-AOI-57-4 Rev. 47

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.102 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
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

Instructor Notes

(d) Another Unit's MMG set

The second alternate is from another unit's MMG set output. Unit 2 MMG is the second alternate for either Unit 1 or Unit 3; Unit 3 is the second alternate for Unit 2. Transfers to this source are done manually at Battery Board 2 panel 11.

b. MMG Sets (Unit 2&3)

Obj. V.B.2.b
TP-11
Obj.V.D.2.c
Obj.V.D.2.d/j
Obj V.E.2.c
Obj.V.E.2.d/i
Obj V.B.2.h
Obj.V.C.3.e
Obj.V.D.2.j
Obj.V.E.2.i

(1) The MMG is normally driven by the AC motor, powered from 480V Shutdown Board A. Should this supply fail, the AC motor is automatically disconnected and the DC motor starts, powered from 250V Battery Board. The DC motor has an alternate power supply from another 250V Battery Board. Transfer to the alternate DC source is manual. Under-frequency on the generator output will trip the DC motor. Transfer of the MMG set back to the AC motor is manual.

(2) The 1001 and 1003 breakers from an MMG set will trip on overvoltage or underfrequency at the output of the MMG. Also Unit 2 MMG breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.

Instructor Notes

(3) When an under frequency or overvoltage condition exists at the Generator Output the following occurs:

Obj. V.B.2.h
Obj. V.C.3.e
Obj. V.D.2.j
Obj. V.E.2.i

(a) BB panel 10 breakers from the MMG set trip.

U2	1001 (U2)	1003 (U1&3)
U3	1001 (U3)	1003 (U2)

(b) Excitation is lost and the MMG Set continues to run. (The Hold to build up voltage switch must be depressed to restore voltage.)

(4) The starting sequence for the MMG is as follows:

(a) Start the AC motor first. This is a larger motor and has enough power to compensate for the flywheel load.

Instructor: Emphasis procedural adherence

(b) Transfer to DC motor by stopping the AC motor. This automatically starts the DC motor allows the speed to be controlled for paralleling.

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0041 Page 5 of 32

2.0 **SYMPTOMS (continued)**

F. Loss of RPIS. REFER TO 2-AOI-85-4.

G. RFW Control System Panel Display Stations on Panel 2-9-5 disabled. PDS Controls are inoperative and displays become blank. The RFW Control System continues to control system parameters according to water level setpoint.

H. The following RFW Control System annunciators in alarm on Panel 2-9-6:

1. RFPT GOVERNOR POWER FAILURE OR GOVERNOR ABNORMAL (2-XA-55-6C, Window 12).

2. RFWCS TROUBLE (2-XA-55-6C, Window 28).

I. The following EHC Control System annunciators in alarm on Panel 2-9-6:

1. EHC POWER ABNORMAL (2-XA-55-7B Window 5)

2. EHC/TSI SYSTEM TROUBLE (2-XA-55-7B Window 6)

J. EHC Control System PLU 1 (power load unbalance) can bypass with a sustained loss of power to Panel 9-9 Cabinet 5. An uninterruptible power supply will keep the PLU energized for approximately 15 minutes after normal power is lost.

K. EHC Control System HMI on Panel 2-9-31 may become blank if power is lost to Panel 9-9 Cabinet 6. An uninterruptible power supply will keep this component energized for approximately 15 minutes after normal power is lost.

L. RECIRC FLOW SYSTEM TROUBLE ALARM (2-XA-55-4A, Window 23).

M. Loss of power to CRD Select Modules.

N. ANN: PNL 2-9-21 SYS LEAK DETECTION POWER FAILURE (2-XA-55-3D, Window 31) on loss of power to Panel 2-9-21 Steam Leak Detection Panel.

O. TIP isolation signal when Cabinet 5 (Breaker 503) is de-energized.

BFN Unit 2	Loss of Unit Preferred	2-AOI-57-4 Rev. 0041 Page 8 of 32

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

NOTE

The blanks to the side of steps contained in Section 4.0 Operator Actions are intended for place keeping only. Initials are **NOT** required. If necessary, place keeping marks may be made directly in the Control Room copy of this instruction. **CONTACT** Management Services for a replacement copy when time permits.

4.2 Subsequent Actions

[1] **VERIFY** the following:

- RFW Control System is maintaining Reactor Water Level. □
- Recirc Flow Control System maintaining Recirc pump speeds. □
- EHC Control System maintaining Reactor Pressure and Turbine control parameters. □
- **VERIFY** TIP ISOLATION. □

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

ENTER the appropriate EOI(s). (Otherwise N/A) □

CAUTION

While RPIS and the process computer are inoperable, control rod movement may only be performed by manual reactor scram.

[3] **IF** control rod movement is required while RPIS and the process computer are inoperable, **THEN**

INSERT a MANUAL SCRAM. **REFER TO** 2-AOI-100-1. (Otherwise N/A) □

DISTRACTOR PLAUSIBILITY SUPPORT

Instructor Notes

(d) Another Unit's MMG set

The second alternate is from another unit's MMG set output. Unit 2 MMG is the second alternate for either Unit 1 or Unit 3; Unit 3 is the second alternate for Unit 2. Transfers to this source are done manually at Battery Board 2 panel 11.

b. MMG Sets (Unit 2&3)

Obj. V.B.2.b
TP-11
Obj.V.D.2.c
Obj.V.D.2.d/j
Obj V.E.2.c
Obj.V.E.2.d/i
Obj V.B.2.h
Obj.V.C.3.e
Obj.V.D.2.j
Obj.V.E.2.i

(1) The MMG is normally driven by the AC motor, powered from 480V Shutdown Board A. Should this supply fail, the AC motor is automatically disconnected and the DC motor starts, powered from 250V Battery Board. The DC motor has an alternate power supply from another 250V Battery Board. Transfer to the alternate DC source is manual. Under-frequency on the generator output will trip the DC motor. Transfer of the MMG set back to the AC motor is manual.

(2) The 1001 and 1003 breakers from an MMG set will trip on overvoltage or underfrequency at the output of the MMG. Also Unit 2 MMG breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.

Examination Outline Cross-reference:	Level	RO	SRO
263000 DC Electrical Distribution	Tier #	2	------
K6.02 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the D.C. ELECTRICAL DISTRIBUTION : • Battery ventilation	K/A #	263000K6.02	
	Importance Rating	2.5	------

Proposed Question: **# 48**

Which ONE of the following is a concern to plant operation if the Battery and Board Room Exhaust Fans are not operating properly?

A. The lead-calcium batteries tend to release toxic gas into the atmosphere above 90 ^{O}F.

B. The design limit for hydrogen concentration in the rooms may be reached when the batteries are being charged.

C. Electrical Maintenance will not be able to obtain accurate cell specific gravity readings if temperature is above 90 ^{O}F.

D. The quarterly battery SR frequency is lowered to weekly when temperatures are outside the 70 ^{O}F to 90 ^{O}F temperature range.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Lead-calcium batteries suffer degraded performance at high temperatures but do not release toxic gas as a result.

B **CORRECT**: Battery Room ventilation is required to prevent buildup of explosive hydrogen concentration.

C INCORRECT: This would be correct for low temperatures.

D INCORRECT: Quarterly battery SR frequency is lowered if temperatures were below the temperature range, not above it.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the impacts of a loss / malfunction of battery ventilation on the DC Electrical Distribution System.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 0-OI-31, Rev. 136 (Attach if not previously provided)

OPL171.037 Rev. 12

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.037 V.B.10 (As available)

Question Source: Bank # HLT 0707 # 23

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 0707

(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 129 of 285

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. ☐

INSTRUCTOR NOTES

7. Battery Room Ventilation Systems

   a. Purpose

      Obj. V.B.10
      Obj. V.C.10
      Obj. V.D.8

      The various battery room ventilation systems provide adequate room ventilation to prevent an explosive atmosphere due to hydrogen buildup from the batteries.

   b. The Unit Battery Rooms 1, 2 and 3 and the Communications Battery Room are supplied air through the door ventilators. Air is exhausted with Battery and Board Room Exhaust Fans 1A and 1B (Battery Room 1 and 2, and communications battery room), and Unit 3 Battery and Board Room Exhaust Fans 3A and 3B (Battery Room 3).

      Plant/Station Battery Rooms are supplied air via an HVAC unit located outside the rooms to maintain an optimum temperature between 70 and 80 degrees F. A small exhaust fan is located in the ceiling with a off and on switch located on the wall. (speed is variable) The purpose of the exhaust fan is to keep hydrogen concentration below 2%. With the exhaust fan off it will take over 8 hours to reach the design limit of 2% hydrogen. Upon loss of the exhaust fan, a "system abnormal" will alarm in the control room. The ceiling also has vent pipes to exhaust the flow of air. Battery Room 4 also required the installation of a new bypass damper with the existing ventilation fan to maintain hydrogen concentration below the design limit. (Damper located by EHC Unit). The existing grille and damper between battery room 4 and the adjacent board room was blocked.

   c. The 250V DC Shutdown Board Battery Rooms are supplied with supply and exhaust fans for each unit.

   d. Each DG 125V DC battery has an exhaust fan that provides adequate ventilation in the battery area.

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 °F.

C. The lead-calcium batteries tend to release toxic gas into the atmosphere above 90 °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 °F to 90 °F temperature range.

Examination Outline Cross-reference:	Level	RO	SRO
264000 Emergency Generators (Diesel/Jet)	Tier #	2	-------
K5.05 (10CFR 55.41.5)	Group #	1	-------
Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) :	K/A #	264000K5.05	
• Paralleling A.C. power sources	Importance Rating	3.4	-------

Proposed Question: **# 49**

Diesel Generator (D/G) 'A' is synchronized to 4KV Shutdown Board 'A'. The instrumentation readings for the D/G are as follows:

- Voltage = 4160 VAC
- Frequency = 60 Hz
- Current = 280 amps
- Vars = 2200 Kvars
- Watts = 2600 Kw

Which ONE of the following is the correct action to obtain a 0.8 lagging power factor?

Take the ______________________________ .

[REFERENCE PROVIDED]

A. Governor control switch to **RAISE**.

B. Governor control switch to **LOWER**.

C. Voltage Regulator control switch to **RAISE**.

D. Voltage Regulator control switch to **LOWER**.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: The governor controls KW not KVAR. Candidate misunderstanding of governor controlling speed and real load or KW.

B INCORRECT: The governor controls KW not KVAR. Candidate misunderstanding of governor controlling speed and real load or KW.

C INCORRECT: Taking the voltage regulator control switch to raise will increase generator excitation and raise KVAR. This will place the generator operating point farther away from the 0.8 power factor line. Candidate error in determining where the generator is operating in relationship to the 0.8 pf line.

D **CORRECT:** Need to lower KVARs by lowering generator excitation to lower reactive load. Desired operation at 2600 KW = a 1950 KVAR with a 0.8 lagging power factor.

KA Justification:

The KA is met because it tests knowledge of operational implications of paralleled AC sources design and how KW and KVAR are controlled to obtain optimum power factor on a DG

Question Cognitive Level:

This question is rated as C/A due to the requirement to solve a problem using references. This requires mentally using this knowledge and its meaning to predict the correct outcome.

Technical Reference(s): 0-OI-82 Rev 112 (Attach if not previously provided)
OPL171.038 Rev17

Proposed references to be provided to applicants during examination: **0-OI-82 Illustration -1**

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

Question Source:
Bank # LXR TEST OPL171.038 #3 Last used BFN 1006 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
Comprehension or Analysis X

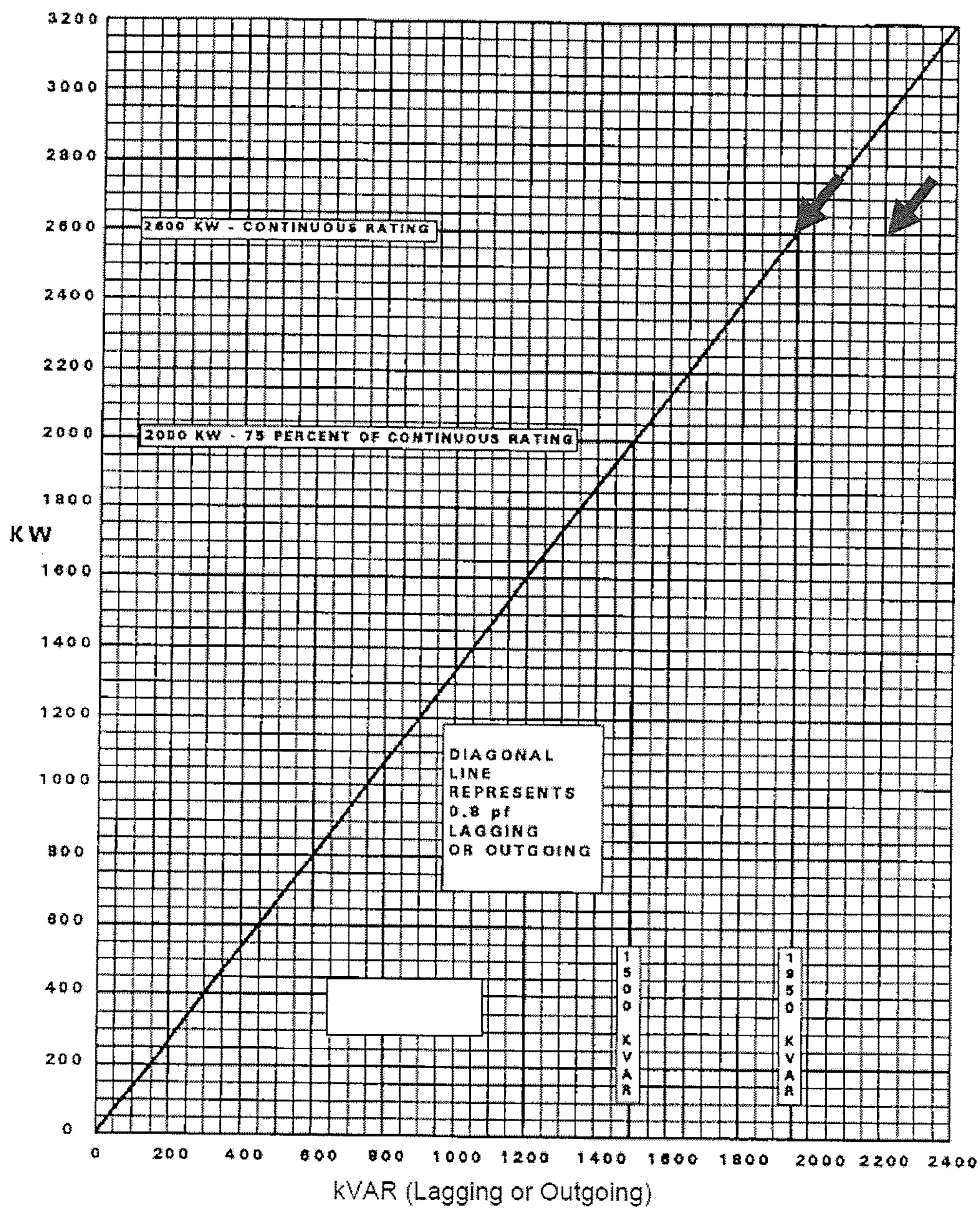
10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BFN Unit 0	Standby Diesel Generator System	0-OI-82 Rev. 0112 Page 171 of 174

Illustration 1
(Page 1 of 1)

DG kW vs. kVAR Loading



kVAR

INSTRUCTOR NOTES

I. 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

Examination Outline Cross-reference:	Level	RO	SRO
300000 Instrument Air System (IAS)	Tier #	2	-----
K6.07 (10CFR 55.41.7)	Group #	1	-----
Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: • Valves	K/A #	300000K6.07	
	Importance Rating	2.5	-----

Proposed Question: **# 50**

“G” Control Air Compressor’s microcontroller fails, causing the Compressor Inlet Flow Valve to **throttle** open and the Compressor Bypass Control Valve to fail **fully** open.

Which ONE of following completes the statement below?

“G” Air Compressor’s discharge pressure will _______.

A. decrease to **LESS THAN** 100 psig

B. stabilize at 100 to 105 psig

C. increase to120 psig

D. increase to 132 psig

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Pressure will decrease to the point of the compressor not supplying compressed air (compressor running). Any air entering the compressor will be discharged through the Bypass Control Valve, to the Air Silencer, and back to atmosphere. The two selected lead air compressors start at 98 psig; the first lag at 96 psig and the second lag at 94 psig.

B INCORRECT: Pressure will decrease to the point of the compressor not supplying air (Unloaded with the compressor running). The two selected lead air compressors start at 98 psig. Plausible in that the normal pressure control band is 100-105 psig. The header will be at this pressure but the discharge of the G Compressor will be less than 100 psig.

C INCORRECT: Plausible if the candidate doesn’t know what the Bypass Control Valve does. IF he/she believes the valve bypasses the normal pressure control. 120 psig is a recognizable value in that it is the rated pressure of “G” Control Air Compressor.

D INCORRECT: Compressor discharge pressure lowers. Plausible if the candidate doesn’t know what the Bypass Control Valve does. IF he/she believes the valve bypasses the normal pressure control. Compressor Relief Valve setpoint is 132 psig.

KA Justification:

K/A asks for effect of a malfunction of a control air system valve. Question asks about the effect of failure of the Bypass Control Valve on the 'G' Air Compressor and Control Air System.

Question Cognitive Level:

Answering the question involves the multi-part mental process of assembling, sorting, or integrating the parts, which also requires the candidate to predict an outcome from the valves failure.

Technical Reference(s): OPL171.054, Rev 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.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:

3. Control Air System Component Description

   a. Four Rotary Screw Air Compressors **A-D** (2-stage and rotary screw type) are located El 565, U-1 Turbine Building. — Ingersoll-Rand H125W, TP-28, DCN 66433

      (1) Supply air to the control air receivers at ≈ 524 scfm each at a normal operating pressure of 94 - 108 psig.

      (2) 480V, 60 Hz, 3-phase, drive motors

      (c) The primary controller normally controls the loading sequence for each control air compressor **A (B, C, D)** through the SEQUENCE SELECTOR switch positions (automatic control positions 1, 2, 3, and 5)

      (d) The controller contains the logic to load and unload the compressors automatically according to control air header pressure.

         i. The compressors (2) in **LEAD** position has a pressure range of 98 to 108 psig — TP-5

         ii. As air pressure lowers to 98 psig, the compressors will go to full load.

         iii. As air pressure rises to 108 psig, the first compressors will go to unload.

         iv. First LAG compressor operating range is 96 to 106 psig.

         v. Second LAG compressor operating range is 94 to 104 psig.

(c) Relief valves on the compressors discharge set at 132 psig protects the compressor and piping.

G Air Compressor - centrifugal type, two stage

(a) Located 565' EL Turbine Bldg., Unit 1 end. Control Air Compressor **G** is the primary control air compressor and provides most of the control air needed for normal plant operation.

(b) Rated at 1445 SCFM @ 120 psig.

OPL171.054
Revision 15
Page 14 of 69

i. In the **UNLOAD** position, the compressor will run but not supply compressed air. The Compressor Inlet Flow Valve is throttled open and the Compressor Bypass Control Valve opens fully.

(a) Air is drawn through the Inlet Filter and through the Inlet Valve to the first stage impeller of the compressor.

(b) The compressed air discharged from the first stage impeller passes through internally finned copper tubes inside the intercooler, and then through a moisture separator.

(c) The second stage impeller takes suction from the intercooler, raises the pressure, and discharges to the after-cooler (similar to intercooler). Moisture is again removed from the compressed air by a moisture separator.

(d) The compressed air is then passed to the Control Air header with some of the air being recirculated through the silencer via the bypass valve.

(e) Both the Inlet Valve and the Bypass Valve are positioned by the microcontroller to maintain the compressor discharge air at the desired pressure (≈ 100-105 psig).

(3) Control air receiver pressures should be between 90 and 105 psig. **G** air compressor will normally maintain receiver pressure >100 psig. When **G** air compressor is not operating, then the primary/backup controllers will maintain 90 to 101 psig.

in pipe corrosion which led to failure of components.

(4) Relief valves on the receivers set at 115 psig.

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0039 Page 5 of 34

3.0 AUTOMATIC ACTIONS

- Service Air crosstie to Control Air valve, 0-FCV-33-1, opens at control air header pressure less than or equal to 85 psig.
- Control Air Compressors A,B,C,D will start as their on-line pressure setpoints are reached.
- The Emergency Control Bay Air Compressor will start at Control Air Header pressure less than or equal to 73 psig.
- Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901, will close when Control Air Header pressure reaches equal to or less than to 65 psig at the valve.
- Unit 1 to Unit 2 Control Air Crosstie, 1-PCV-032-3901, will close when Control Air Header pressure reaches equal to or less than to 65 psig at the valve.

Examination Outline Cross-reference:	Level	RO	SRO
300000 Instrument Air System (IAS)	Tier #	2	------
K6.12 (10CFR 55.41.7)	Group #	1	------
Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM:	K/A #	300000K6.12	
• Breakers, relays and disconnects	Importance Rating	2.9	------

Proposed Question: **# 51**

Control Air Compressors 'A' **AND** 'C' are in service. A momentary loss of power to 480V Shutdown Board 1B occurs. Three seconds later, normal voltage is restored.

Which ONE of the following describes the impact of this board loss on the Air System?

Control Air Compressor __(1)__ will trip **AND** __(2)__ automatically re-start when normal voltage is restored.

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

B. **(1)** C
(2) will

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

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

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: 'A' compressor is powered from 480V SD Bd 1B, and will therefore trip. The compressor will not auto start when normal voltage is restored. Plausible in that Control Air Compressor G does restart if voltage restored within 4 seconds.

B INCORRECT: 'C' is powered from 480v Common Bd 1, which is **not** affected by this event. Plausible in that candidates could confuse 480V SD Bd 1B which does supply A with 480 V Common Bd 1 which does not. If C power supply had been momentarily interrupted, the second part would NOT be true with voltage restored within 4 seconds.

C **CORRECT: '**A' compressor is powered from 480V SD Bd 1B, which **is** affected by this event. It does **NOT** have auto restart capability for ≤ 4 sec power loss, like Control Air Compressor 'G'.

D INCORRECT: C is powered from 480v Common Bd 1, which is **not** affected by this event. The 'G' compressor power loss logic is set @ ≤ 4 seconds on a loss of 480V RMOV Bd 2A.

KA Justification:

The effect of a breaker failure resulting in momentary loss of 480V Shutdown Board 1B to the instrument air system (Control Air at BFN) agrees with the stated K/A. .

Question Cognitive Level:

This question is high comprehension because the examinee must evaluate the situation and predict the effect on the instrument/control air system. This involves a multi-part mental process of assembling, sorting, and integrating the parts of the system.

Technical Reference(s): OPL171.054 Rev 15 (Attach if not previously provided)
0-OI-32 Rev 127

Proposed references to be provided to applicants during examination: NONE

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

Question Source: Bank #
Modified Bank # BFN 0801 #52 (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 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:

OPL171.054 r15

3. Control Air System Component Description	
a. Four Reciprocating Air Compressors **A-D** (2-stage, double acting, Y-type) are located El 565, U-1 Turbine Building.	
(1) Supply air to the control air receivers at 610 scfm each at a normal operating pressure of 90 - 101 psig.	
(2) 480V, 60 Hz, 3-phase, drive motors	
(3) Power supplies	
A from **480V Shutdown Board 1B**	DCN 17780
D from **480V Shutdown Board 2A**	
B from **480V Common Board 1**	
C from **480V Common Board 1**	
(a) Control air compressors which are powered from the 480 VAC shutdown boards are tripped automatically due to:	Obj. V.B.1. Obj. V.C.1.
i. under voltage on the shutdown board.	
ii. load shed logic during an accident signal concurrent with a loss of offsite power.	
<u>NOTE</u>: The compressors must be restarted manually after power is restored to the board.	
(b) Units powered from common boards also trip due to under voltage.	

iv. The primary controller power is auctioneered from one of three sources:	
• **480VAC Shutdown Board 1A.** • **480VAC Shutdown Board 2A** • **480VAC Common Board 1**	Same power supply as for air compressors
(c) Each power supply has a 480VAC to 120VAC transformer.	0-45E769-5
(d) The backup controller 0-PIC-032-0002 loads and unloads the compressors at the same control air header pressure setpoints as the primary controller.	TP-4
(e) When the backup is in control compressor **A** will run at full load, **B**, **C**, & **D** will load and unload at 1.5 psig increments, in alphabetical order as pressure falls and rises.	Independent of selector switch position
(f) The backup controller is powered from 250 VDC control power on **480 VAC Common Board 1**	
e. <u>G Air Compressor</u> - centrifugal type, two stage	
(1) Located 565' EL Turbine Bldg., Unit 1 end. Control Air Compressor **G** is the primary control air compressor and provides most of the control air needed for normal plant operation.	
(2) Rated at 1445 SCFM @ 120 psig.	
(3) Power Supply	
(a) **4 kV Shutdown Board B** supplies power to the compressor motor.	
(b) **480 V RMOV Bd. 2A** Supplies the following:	
• Pre lube pump	
• Oil reservoir heater	
• Cooling water pumps	
• Panel(s) control power	
• Auto Restart circuit	
(c) Except for short power interruptions on the **480v RMOV Bd**, Loss of <u>either</u> of these two power supplies will result in a shutdown of the **G** air compressor.	
(d) With the **G** air compressor **AUTO START** selector switch in **ON** the compressor will automatically restart if there is a momentary interruption of power (< 4 seconds) of the **480v RMOV board 2A**. (see 0-OI-32)	DCN F41321A Power interruptions > 4 seconds will lock out the Auto Restart circuit and trip the compressor.
i. This feature was designed to maintain Control Air Compressor **G** operation during board transfers and momentary interruptions in power involving **480V RMOV Board 2A**.	The feed from **4KV Shutdown Board B** to the compressor motor is <u>not</u> affected by the auto restart circuit.
ii. If power is <u>NOT</u> restored within 4 seconds, the compressor will trip and must be manually restarted when power is restored.	TP-7

3. Component Description	Obj. V.E.6
a. Compressors **E** and **F** (EL 565, U-3 Turbine Building) are designated for service air.	Obj. V.D.4
b. The **F** air compressor is rated for approximately 630 SCFM @ 105 psig, centrifugal type, 2 stages	
c. The power supply for both compressors is **480VAC Common Board 3**.	

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 0	Control Air System	0-OI-32 Rev. 0127 Page 9 of 113

3.0 **PRECAUTIONS AND LIMITATIONS (continued)**

E. Control Air Compressor G will automatically trip and remain tripped on any of the following conditions:

1. Vibration high, Stage 1 - 1.00 mil
2. Vibration high, Stage 2 - 0.94 mil
3. Lube Oil Pressure low - 16 psig
4. Lube Oil Temperature high - 125°F
5. Lube Oil Temperature low - 65°F
6. Air Temperature high, Stage 1 - 125°F
7. Discharge Air Temperature high - 125°F
8. Seal Air Pressure low - 6 psig

F. A loss of power to 4KV Shutdown Board B OR a sustained loss of power (greater than 4 seconds) to 480V RMOV Board 2A will result in a trip of Control Air Compressor G.

G. Control Air Compressor G has an auto restart circuit which will restart the compressor after a momentary power loss (up to 4 seconds) from 480V RMOV Board 2A. This feature was designed to maintain Control Air Compressor G operation during board transfers and momentary interruptions in power involving 480V RMOV Board 2A. The restart circuit is in place when COMPR G AUTO-RESTART ON-OFF SELECTOR switch, 0-HS-032-3087, is in the ON position AND the compressor is running. The auto restart circuit will reset automatically after each restart attempt, thus enabling multiple restart attempts.

H. During a surge condition, Control Air Compressor G will alarm and automatically unload. The compressor will automatically reload after 6 seconds for the first 3 surges in ten minutes. If a fourth surge occurs within the 10 minute period, the compressor will remain unloaded until being acknowledged at the Microcontroller. Section 6.1 provides additional instruction on compressor surge.

I. Control Air Compressor G shall **NOT** be manually restarted until it has come to a complete rest.

J. When blowing down Control Air Compressor G raw cooling water strainer open RCW STRAINER BLOWDOWN VLV, 0-DRV-024-1510 only half way (45°) open.[PER 207496]

Examination Outline Cross-reference:	Level	RO	SRO
300000 Instrument Air	Tier #	2	------
K2.01 (10CFR 55.41.7)	Group #	1	------
Knowledge of electrical power supplies to the following: • Instrument air compressor	K/A #	300000K2.01	
	Importance Rating	2.8	------

Proposed Question: **# 52**

All Units are operating at 100% Reactor Power, when a momentary undervoltage condition occurs on 480V RMOV Board 2A. Five seconds later, normal voltage is restored.

Which ONE of the following describes the impact of this board loss on the Air System?

A. Control Air Compressors 'B' **AND** 'C' will trip. **BOTH** will need to be re-started locally.

B. Service Air Compressors 'E' **AND** 'F' will trip. **BOTH** will need to be re-started locally.

C. Control Air Compressor 'A' will trip, **AND** will automatically re-start when normal voltage is sensed.

D. Control Air Compressor 'G' will trip, **AND** will **NOT** automatically re-start when normal voltage is sensed.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: B & C compressors are powered from 480V Common Bd 1, which is not affected by this event. Air Compressor G is powered from 4kV SD Bd B / 480V RMOV Bd 2A, which **is** affected by this event.

B INCORRECT: E & F compressors are powered from 480V Common Bd 3, which is not affected by this event. Air Compressor G is powered from 4kV SD Bd B / 480V RMOV Bd 2A, which **is** affected by this event.

C INCORRECT: A compressor is powered from 480V SD Bd 1B, which is not affected by this event. Air Compressor G is powered from 4kV SD Bd B / 480V RMOV Bd 2A, which **is** affected by this event.

D **CORRECT:** Air Compressor G is powered from 4kV SD Bd B / 480V RMOV Bd 2A, which **is** affected by this event. The 'G' compressor power loss logic is set @ 4 seconds on a loss of 480V RMOV Bd 2A. Transfer of the RMOV Bd is a Manual action, thus > 4 seconds will have transpired prior to the transfer.

Technical Reference(s):	OPL171.054 Rev 13	(Attach if not previously provided)
	0-OI-57B Rev 181	(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: 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	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments: Reviewed Rev 181 of 0-OI-57B. The latest revision of this procedure has no impact on the question.

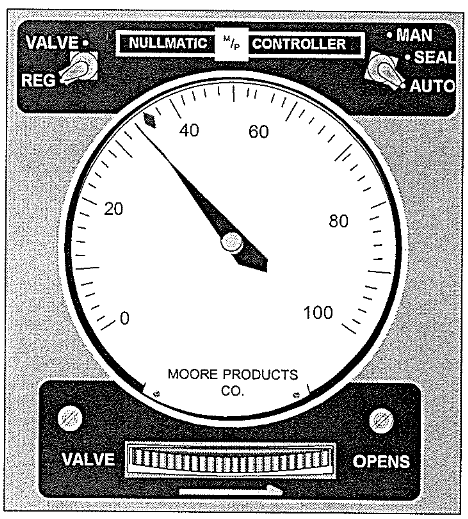
Examination Outline Cross-reference:	Level	RO	SRO
400000 Component Cooling Water	Tier #	2	-------
A1.01 (10CFR 55.41.5)	Group #	1	-------
Ability to predict and/or monitor changes in parameters associated with operating the COMPONENT COOLING WATER SYSTEM controls including:	K/A #	400000A1.01	
• CCW flow rate	Importance Rating	2.8	-------

Proposed Question: **# 52**

Which ONE of the following completes the statement below?

The Unit 2 Reactor Building Closed Cooling Water (RBCCW) Temperature Controller, 2-TIC-24-80, is located in Unit 2 Reactor Building at __**(1)**__ .

If the controller is placed in AUTO with the indications shown below, the Temperature Control Valve will modulate to a more __**(2)**__ .



A. **(1)** Panel 2-25-196, Elevation 565'
(2) closed position

B. **(1)** RBCCW Heat Exchanger area, Elevation 593'
(2) close position

C. **(1)** Panel 2-25-196, Elevation 565'
(2) open position

D. **(1)** RBCCW Heat Exchanger area, Elevation 593'
(2) open position

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Part 1 correct - RBCCW Temp Controller, 2-TIC-24-80, is located in Unit 2 Reactor Building at Panel 2-25-196, Elevation 565'. Part 2 correct - with the RED indicator (Set Point) higher than the BLACK needle, indicates that actual temperature is cooler than desired. The TCV will modulate CLOSED.

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

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

D INCORRECT: Part 1 incorrect - Plausible in that several RCW valves associated with RBCCW are located at the RBCCW Heat Exchanger area, Reactor Building Elevation 593'. Part 2 incorrect - Plausibility based on misconception that with the feedback signal less than the control set point that the TCV would modulate Open to remove the deviation or that the controller is bypassing flow rather than controlling cooling water flow through the heat exchanger.

KA Justification:

The KA is met because the question tests the ability to predict and monitor changes in CCW Heat Exchanger flow in response to operating CCW Temperature control valve from Auto to Manual with a deviation between set point and feedback signal.

Question Cognitive Level:

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-24 Rev 77 (Attach if not previously provided)
OPL171.048 Rev 14

Proposed references to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # BFN 0801 #53 (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:

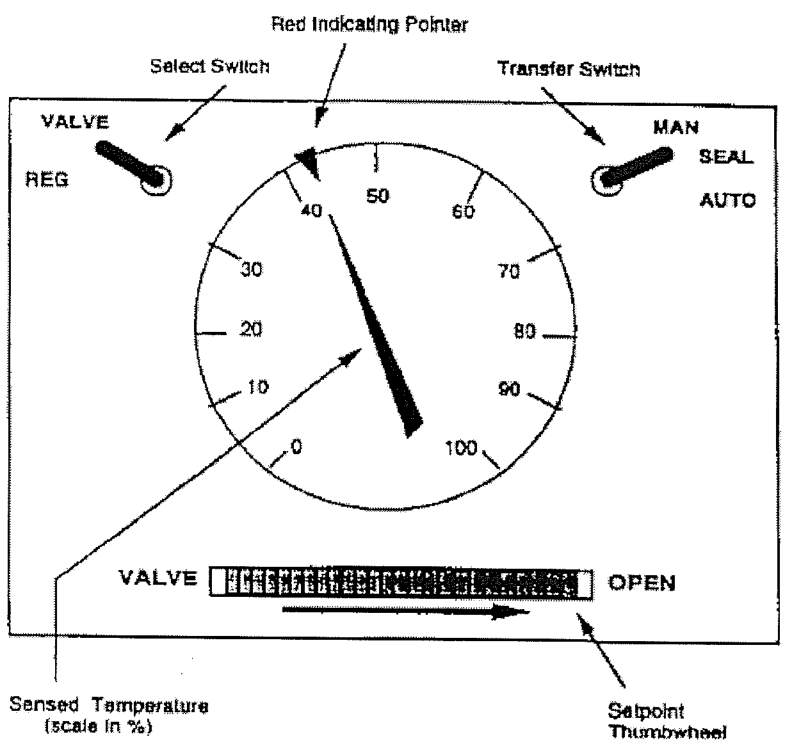
BFN Unit 2	Raw Cooling Water System	2-OI-24 Rev. 0077 Page 58 of 58

Illustration 1
(Page 4 of 4)

Operation of 2 TIC 24 80(85) and 1 TIC 24 90 for RBCCW Temperature Control

3.0 RBCCW TEMPERATURE CONTROLLERS - NULLMATIC M/P CONTROLLER

NOTE
Setpoint: 27% scale = 80°F
Maximum Allowed Setpoint: 36% scale = 95°F.
Range: 40°F to 190°F
Valve is fully closed at 0% and fully open at 100%.



INSTRUCTOR NOTES

(2) Output from the temperature sensor is changed by the temperature modifier from a milivolt signal to an air signal which is proportional to the temperature.

3-15 psig typical

(3) The temperature indicating controller (TIC) compares the temperature from the TM to a desired temperature (or setpoint.)

Some electronic TICs process the milivolt signal from the thermocouple directly or from a TM.

(4) The TIC sends a signal to the valve positioner and TCV to throttle open the valve if temperature is higher than the setpoint, or throttle it closed if below the desired temperature.

Electronic TICs send an electrical output signal to a TM which develops the air signal to the valve positioner.

b. most TCVs are Control Air operated valves which throttle flow through individual loads for temperature control.

c. Nullmatic Temperature Controllers

TP-5

(1) This is the type used on RBCCW heat exchangers.

(2) The BLACK (center) pointer always indicates RBCCW temp. at outlet of RBCCW Heat Exchanger.

See OI-24 Illustration 1 for detailed information

(3) The RED (peripheral) pointer is controlled by the Select Switch. In VALVE position, it senses regulating air pressure to control valve air operator and displays in % valve is closed. In REG position, it senses controlling press. from air regulator (controlled by thumb wheel) and displays desired control point in same units as the BLACK pointer.

BFN Unit 2	Reactor Building Closed Cooling Water System	2-OI-70 Rev. 0061 Page 29 of 63

8.3 **Placing Spare Heat Exchanger in Service (continued)**

[4] **VENT** the RCW System side of the spare RBCCW Heat Exchanger **UNTIL** a solid stream of water is discharged, **THEN CLOSE**, using the following: □

- RBCCW CLR C RCW VENT, 1-VTV-024-1089 (south end RBCCW Heat Exchanger area, El 596'). □
- RBCCW CLR C RCW VENT, 1-VTV-024-1090 (north end RBCCW Heat Exchanger area, El 593'). □

[5] **VERIFY OPEN** the following (south end Drywell equipment hatch area, El 565'):

- TCV-24-90A INLET, 1-SHV-024-0724C. □
- TCV-24-90B INLET, 1-SHV-024-0722C. □
- TCV-24-90A OUTLET, 1-SHV-024-0725C. □
- TCV-24-90B OUTLET, 1-SHV-024-0723C. □

[6] **DETERMINE** position of temperature control valve on RBCCW Heat Exchanger to be taken out of service, RBCCW HX A(B) TEMP CONT, 2-TIC-24-80(85), located at Panel 2-25-196, El 565'. □

[7] **PLACE** RBCCW SECTIONALIZING VLV TRANSFER, 2-XS-70-48, in EMERG (480V Reactor MOV Board 2B, compartment 5A). □

NOTE
The following action will lower RCW System pressure.

[8] **PERFORM** the following at Panel 1-25-196, El 565' and **REFER TO** Illustration 1:

- **VERIFY** in service, RBCCW SPARE HX TEMP CONT, 1-TIC-24-90. □
- **PLACE** in MANUAL **AND OPEN** temperature control valve to same position as temperature control valve on RBCCW Heater Exchanger to be taken out of service. □

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Reactor Building Closed Cooling Water System	2-OI-70 Rev. 0061 Page 28 of 63

8.3 **Placing Spare Heat Exchanger in Service**

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

- Unit 2 RBCCW System is in operation. ☐
- Raw Cooling Water available to supply Spare RBCCW Heat Exchanger. REFER TO 2-OI-24. ☐
- Spare RBCCW Heat Exchanger available for use on Unit 2. ☐

[1.1] [NRC/C] **WHEN** the RCW or EECW supplied to any RBCCW heat exchanger is put into service or taken out of service, **THEN**

NOTIFY Chemistry Shift Supervisor so any required sampling can be initiated or stopped. [NRC LER 259/88010] ☐

NOTE

All operations are performed locally unless noted otherwise.

[2] **NOTIFY** Unit 1 and Unit 3 that Unit 2 will be placing the spare heat exchanger in operation on Unit 2. ☐

CAUTION

Filling and venting the Raw Cooling Water side of the spare heat exchanger may cause a lowering in RCW System pressure, resulting in an ESF actuation. Slow and cautious performance of any actions that may cause system pressure to lower will minimize this problem.

[3] **VERIFY OPEN**, RCW TO RBCCW HTX C, 1-SHV-024-0720C (north end RBCCW Heat Exchanger area, El 593', Chain operated valve in overhead). ☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 2	Reactor Building Closed Cooling Water System	2-OI-70 Rev. 0061 Page 29 of 63

8.3 **Placing Spare Heat Exchanger in Service (continued)**

[4] **VENT** the RCW System side of the spare RBCCW Heat Exchanger **UNTIL** a solid stream of water is discharged, **THEN CLOSE**, using the following: ☐

- RBCCW CLR C RCW VENT, 1-VTV-024-1089 (south end RBCCW Heat Exchanger area, El 596'). ☐
- RBCCW CLR C RCW VENT, 1-VTV-024-1090 (north end RBCCW Heat Exchanger area, El 593'). ☐

[5] **VERIFY OPEN** the following (south end Drywell equipment hatch area, El 565'):

- TCV-24-90A INLET, 1-SHV-024-0724C. ☐
- TCV-24-90B INLET, 1-SHV-024-0722C. ☐
- TCV-24-90A OUTLET, 1-SHV-024-0725C. ☐
- TCV-24-90B OUTLET, 1-SHV-024-0723C. ☐

[6] **DETERMINE** position of temperature control valve on RBCCW Heat Exchanger to be taken out of service, RBCCW HX A(B) TEMP CONT, 2-TIC-24-80(85), located at Panel 2-25-196, El 565'. ☐

[7] **PLACE** RBCCW SECTIONALIZING VLV TRANSFER, 2-XS-70-48, in EMERG (480V Reactor MOV Board 2B, compartment 5A). ☐

NOTE
The following action will lower RCW System pressure.

[8] **PERFORM** the following at Panel 1-25-196, El 565' and **REFER TO** Illustration 1:

- **VERIFY** in service, RBCCW SPARE HX TEMP CONT, 1-TIC-24-90. ☐
- **PLACE** in MANUAL **AND OPEN** temperature control valve to same position as temperature control valve on RBCCW Heater Exchanger to be taken out of service. ☐

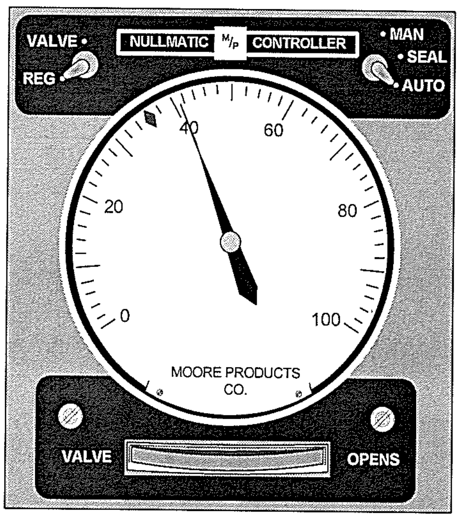
53. At 10:00 a.m., the Unit 2 RBCCW Temperature Control Valve (TCV), (2-TIC-24-80(85)) was place in Manual as follows:

- REG was selected
- MAN was selected

Before transferring from Manual back to Automatic, in order to NULL the controller, the RBAUO i: required to place the __(1)__**AND** adjust the thumbwheel until the RED pointer lines up with the BLACK pointer.

At 11:00 a.m., the controller was transferred to auto. If the RBAUO observes the following indicati after the controller was transferred back to auto, this means that the TCV will modulate to a more __(2)__ .

After Transfer:



A. **(1)** Transfer Switch to SEAL.
(2) open position.

B. **(1)** Selector Switch to VALVE.
(2) closed position.

C. **(1)** Transfer Switch to SEAL.
(2) closed position.

D. **(1)** Selector Switch to VALVE.
(2) open position.

ANSWER: A

BROWNS FERRY 0801

Examination Outline Cross-reference:	Level	RO	SRO
400000 Component Cooling Water	Tier #	2	------
A1.04 (10CFR 55.41.5)	Group #	1	------
Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: • Surge Tank Level	K/A #	400000A1.04	
	Importance Rating	2.8	------

Proposed Question: **# 53**

Unit 2 RBCCW Heat Exchanger 2A is being filled and vented per 2-OI-70, "Reactor Building Closed Cooling Water System."

Which ONE of the following completes the statement?

While filling the Heat Exchanger, RBCCW Surge Tank Level will lower until RBCCW SYS SURGE TANK FILL VALVE, 2-FCV-70-1, __________.

A. is manually opened from the Control Room

B. is manually opened locally at the Surge Tank

C. automatically opens at 4 inches below the Surge Tank centerline

D. automatically opens at 4 inches above the Surge Tank centerline

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: RBCCW SYS SURGE TANK FILL VALVE, 2-FCV-70-1, is operated remotely from Control Room Panel 2-9-4.

B INCORRECT: Plausible in that manual BYPASS VLV, 2-FCV-70-1, is LOCALLY operated at the surge tank.

C INCORRECT: Plausible in that it is logical to have automatic make up capability to the RBCCW Surge Tank and 4 inches below the Surge Tank centerline is the set point for the Surge Tank Level Low Alarm. Additionally other plant head tanks automatically fill on low level. Examples: Demin Water Head Tank / PSC Surge Tank.

D INCORRECT**:** Plausible in that it is logical to have automatic make up capability to the RBCCW Surge Tank and 4 inches above the Surge Tank centerline is a recognizable value as the set point for the Surge Tank Level High Alarm. Additionally other plant head tanks automatically fill on low level. Examples: Demin Water Head Tank / PSC Surge Tank.

KA Justification:

The KA is met because the question tests candidates' ability to predict and monitor changes in Surge Tank Level associated with operating RBCCW controls to fill a Heat Exchanger.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 2-OI-70 Rev. 61 (Attach if not previously provided)

2-ARP-9-4C Rev. 30

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 2	Reactor Building Closed Cooling Water System	2-OI-70 Rev. 0061 Page 40 of 63

8.7 Filling And Venting Of An Individual Heat Exchanger.

NOTE

Filling of RBCCW SYS SURGE TANK may be performed from Control Room (Step 8.7[2]) or locally at surge tank (Step 8.7[3]).

CAUTION

If RBCCW SURGE TANK HIGH LEVEL (XA-55-4C, window 6) Annunciator alarms, and the fill valve is open, it should be closed immediately to prevent tank overflow.

[1] **STATION** personnel to monitor RBCCW surge tank level as the heat exchanger is filled. ☐

[2] **IF** System Fill is to be performed from the Control Room **THEN** (Otherwise N/A)

[2.1] **ESTABLISH** direct communications between the Personnel at the RBCCW System Surge Tank, the heat exchanger to be filled and vented and the Control Room Operator. ☐

[2.2] **OPEN** RBCCW SYS SURGE TANK FILL VALVE, 2-FCV-70-1, using 2-HS-70-1 (Panel 2-9-4). ☐

[2.3] **FILL** system until RBCCW Surge Tank level is normal (4 inches below tank centerline to 4 inches above tank centerline), **THEN** ☐

[2.3.1] **CLOSE** RBCCW SYS SURGE TANK FILL VALVE, 2-FCV-70-1, (Panel 2-9-4). ☐

[2.3.2] **MAINTAIN** this range during fill and vent. ☐

[3] **IF** System Fill is to be performed locally at the RBCCW Surge Tank, **THEN** (Otherwise N/A)

[3.1] **ESTABLISH** direct communications between the Personnel at the RBCCW System Surge Tank, the heat exchanger to be filled and vented and the Control Room Operator. ☐

[3.2] **OPEN** FCV-70-1 BYPASS VLV 2-BYV-002-1369 (locally at surge tank). ☐

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0030 Page 20 of 44

RBCCW
SURGE TANK
LEVEL LOW
2-LA-70-2B
13

(Page 1 of 1)

Sensor/Trip Point:

2-LS-70-2B 4 inches below center line of tank

Sensor Location: On the RBCCW surge tank in the MG Set Room, El 639'

Probable Cause:
A. Normal leakage.
B. Drain valves open.
C. Abnormal leakage.

Automatic Action: None

Operator Action:

A. **ADD** water to the RBCCW Surge Tank for approximately one minute or until low level alarm resets using the following: ☐
- RBCCW SYS SURGE TANK FILL VLV, 2-FCV-70-1 (Panel 2-9-4) OR ☐
- FCV-70-1 BYPASS VLV, 2-HCV-2-1369 (locally). ☐

B. **IF** alarm does **NOT** resets, **THEN** **CHECK** tank locally. ☐

C. **IF** unable to maintain RBCCW Surge Tank level, **THEN** **REFER TO** 2-AOI-70-1. ☐

D. **IF** necessary to add water more than once per shift, **THEN** **CHECK** Drywell floor drain system for excessive operation AND **INSPECT** system outside Drywell for leakage. ☐

References: 45N614-4 2-47E610-70-1 45N620-4
FSAR Sections 10.6.4 and 13.6.2

BFN Unit 2	Panel 9-4 2-XA-55-4C	2-ARP-9-4C Rev. 0030 Page 12 of 44

RBCCW SURGE TANK LEVEL HIGH 2-LA-70-2A 6	Sensor/Trip Point: 2-LS-70-2A 4 inches above center line of tank

(Page 1 of 2)

Sensor Location: RBCCW surge tank in the MG set room El 639'.

Probable Cause:
A. Makeup valve, 2-FCV-70-1, open.
B. Bypass valve 2-2-1369 leaking.
C. Leak into the system.

Automatic Action: None

Operator Action:
A. **CHECK** make-up valve 2-FCV-70-1, 2-HS-70-1, CLOSED on Panel 2-9-4. ☐
B. **CHECK** RBCCW system water leaving the RBCCW system heat exchangers is 100°F or less on 2-TI-70-3, Panel 2-9-4. ☐
C. **DISPATCH** personnel to verify high level and to ensure bypass valve, 2-HCV-2-1369, for 2-FCV-70-1 is CLOSED. **OBSERVE** sight glass level. ☐
D. **OPEN** surge tank drain valve, 2-70-609. **CLOSE** valve when desired level is obtained. ☐
E. **IF** level continues to rise, **THEN REQUEST** Chemistry to pull and analyze a sample for total gamma activity and attempt to qualify source of leak. ☐
F. **CHECK** activity reading on 2-RM-90-131 (2-RR-90-134 Ch 3). ☐

Continued on Next Page

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 1	Panel 9-20 1-XA-55-20B	1-ARP-9-20B Rev. 0028 Page 17 of 39

DEMIN WATER HEAD TANK LEVEL ABN 0-LA-2-159 (14)	Sensor/Trip Point:	
		High level:
	LS-2-159A	9 ft (Elevation 727.25) rising
		Low level:
	LS-2-159C	7 ft 4 in (Elevation 725.58) lowering

(Page 1 of 1)

Sensor Location: Reactor Building Roof

Probable Cause:

A. Excessive demineralized water usage.
B. System leakage.
C. Level switch malfunction.
D. Demineralized water transfer pump malfunction.

Automatic Action:

A. Both demineralized water transfer pumps secure on high level.
B. Second demineralized water transfer pumps starts on low level.

Operator Action:

A. **CHECK** Demineralized Water Transfer Pumps 0-HS-2-154A on Panel 1-9-22, and 0-HS-2-155A on Panel 1-9-20, in AUTO and operating status lights illuminated. ☐
B. **CHECK** 0-HS-2-159, DEMIN WTR HD TNKS 1 INLET VLV, on Panel 1-9-22, in OPEN and light illuminated. ☐
C. **DISPATCH** personnel to the demin water head tank to determine condition. ☐
D. **IF** level is high, **THEN** **VERIFY** both Demineralized Water Transfer Pumps are off. ☐
E. **IF** level is low, **THEN** **PERFORM** the following:
   - **CHECK** demineralized water storage tank level, on Panel 1-9-20, greater than 26 ft with LI-2-153 and **START** both transfer pumps. ☐
   - **OPEN** FCV-2-159 BYPASS VLV, 0-BYV-002-0926, to restore level (Elev. 565, T-4 M-line). ☐
   - **CHECK** system for leaks. ☐

References: 0-47W491-3 1-45E620-12-2

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0040 Page 39 of 52

PSC HEAD TANK LEVEL LOW 1-LA-75-79 26	Sensor/Trip Point: 1-LS-075-0078D	BETWEEN 644'-11" & 645'-2 1/4"

(Page 1 of 1)

Sensor Location: Rx Bldg, El. 639', R-1 T-Line.

Probable Cause:

A. Both pumps **NOT** running.
   1. Level switch malfunctioned.
   2. Thermal overloads **NOT** reset, 480V Reactor MOV Board 1C and 1B, compartments 11C and 7A respectively.
   3. FCV-75-57 and -58 closed (PCIS Group II isolation).
   4. Strainer ΔP push buttons **NOT** reset.

B. Both pumps running.

   Pump discharge pressure < 60 psig. A minimum pressure of 55 psig is required to reach EL. 645'.

Automatic Action: Low level switch starts both pumps.

Operator Action:

A. **VERIFY** both pumps are running. ☐
B. **VERIFY** power available to pumps. ☐
C. **CHECK** PSC PUMP SUCTION INBD and OUTBD ISOL VALVES, 1-FCV-75-57 and 58 open. ☐
D. **IF** the alarm does **NOT** reset within a few minutes, **THEN DISPATCH** personnel to CHECK pumps locally. ☐
E. **IF** the PSC Head Tank Pumps will **NOT** maintain the RHR and Core Spray Systems charged above TRM Limits, **THEN LINE UP** the condensate transfer system to each loop. **REFER TO** 1-OI-75. ☐
F. **REFER TO** Tech Spec 3.5.1, 3.5.2,TRM Sections 3.3.3.1 & 3.5.4. ☐

References:

1-47W610-75-1	1-45E751-3 and -5	1-45E620-3
TRM 3.5.4	Tech Spec 3.5.1,3.5.2	TRM 3.3.3.1, 3.5.4

Examination Outline Cross-reference:	Level	RO	SRO
201001 CRD Hydraulic	Tier #	2	------
K5.05 (10CFR 55.41.5)	Group #	2	------
Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD DRIVE HYDRAULIC SYSTEM :	K/A #	201002K5.05	
• Indications of pump runout: Plant-Specific	Importance Rating	2.7	------

Proposed Question: **# 54**

Unit 1 Control Rod Drive System has ruptured on the Charging Water Header resulting in CRD Pump 1A operating at pump runout.

Which ONE of the following completes the statement?

This condition is indicated by CRD Pump 1A motor amps __**(1)**__ than normal **AND** CRD Flow Control Valve FULL __**(2)**__.

A. **(1)** LOWER
(2) OPEN

B. **(1)** HIGHER
(2) OPEN

C. **(1)** LOWER
(2) CLOSED

D. **(1)** HIGHER
(2) CLOSED

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 Correct – Plausibility based on misconception that pumping against backpressure of atmospheric as opposed to above Reactor Pressure would result in lower motor amps. Part 2 Correct – Plausible in that if CRD flow elements providing feedback to CRD FCV were downstream of where Charging Water Header ties in, the TCV would see low flow and go full open.

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 – The increase in Pump flow associated with going from normal flow to runout conditions would result in CRD Pump 1A motor amps higher than normal. Part 2 correct - CRD flow elements providing feedback to CRD FCV are upstream of where Charging Water Header ties in resulting in high flow sensed by the controller. The TCV would go full closed in response to the high flow condition.

KA Justification:

The KA is met because the question tests knowledge of indication and operational implications of CRD Pump 1A at runout due to a break in the system on the Charging Water Header.

Question Cognitive Level:

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): OPL171.005 Rev. 17 (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
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 X
55.43

Comments:

ii. The CRD pump will try to recharge all the accumulators at once. Flow through the charging header will cause the flow control valves to close.

iii. To prevent pump runout and probable tripping of the pump motor on over-current, a restricting orifice is provided to limit the maximum rate of recharging to 179 gpm. (Maximum flow is with the reactor at atmospheric pressure).

Q: What is a good indication of pump runout?

A: Indication of pump runout is high current on the CRDH pumps

SER-3-05

iv. A throttle valve downstream of the restricting orifice is provided to provide additional throttling if required.

v. The accumulators cannot be recharged until the scram is reset (with the scram inlet and outlet valves closed) due to drive seal leakage being greater than pump capacity.

SER 3-05

Charging pressure will be approximately equal to reactor pressure

(c) Charging water pressure is independent of reactor vessel pressure. and is set by manually positioning the pump discharge throttling valve between 1475-1500 psig per OI-85.

Examination Outline Cross-reference:	Level	RO	SRO
201003 Control Rod and Drive Mechanism	Tier #	2	------
K1.01 (10CFR 55.41.2 to 41.9)	Group #	2	------
Knowledge of the physical connections and/or cause effect relationships between CONTROL ROD AND DRIVE MECHANISM and the following: • Control rod drive hydraulic system	K/A #	201003K1.01	
	Importance Rating	3.2	------

Proposed Question: **# 55**

During a UNIT 1 startup, a control rod drive mechanism is difficult to withdraw (will not move after several attempts to notch the rod out) and is stuck at position 00.

HCU hydraulic lines were vented and the problem is **NOT** believed to be air in the hydraulic system.

Which ONE of the actions, listed below, is the correct 1-OI-85, "Control Rod Drive System" set of actions to be taken to address difficult to withdraw control rods, and to get the control rod to move?

GO TO ______________________________ .

A. ROD IN, **then** ROD OUT NOTCH with the CRD CONTROL SWITCH, release if rod moves

B. ROD OUT NOTCH with the CRD CONTROL SWITCH, **then** NOTCH OVERRIDE with the CRD NOTCH OVERRIDE SWITCH, release switches if rod moves

C. EMERGENCY IN with the CRD NOTCH OVERRIDE SWITCH, then simultaneously place the CRD CONTROL SWITCH in ROD OUT NOTCH, release switches if rod moves

D. EMERGENCY IN, **then** NOTCH OVERRIDE with the CRD NOTCH OVERRIDE SWITCH, **and then** simultaneously place CRD CONTROL SWITCH in ROD OUT NOTCH, release switches if rod moves

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: This method may be used to vent some air from the CRDH lines but stem gives NOT believed to be air. RMCS settle time will be enforced between in and out signals. This method does give a withdrawal signal. Candidate may believe this will unstick the rod because it does give a withdrawal signal.

B INCORRECT: Would still ONLY get a single rod out notch signal. IF rod wouldn't move with single rod out notch signal, it won't move now. IF went to notch override first, then rod out, at least you would get a continuous withdrawal signal and vent any air from the withdrawal header/lines. Candidate misconception that notch override is giving a signal continuous withdrawal signal in this condition.

C INCORRECT: Drives rod in ONLY. Rod won't move out. It already has a continuous insert signal. May chose because of rod out notch signal. Candidate confusion that this is giving a continuous withdrawal signal.

D **CORRECT:** This is procedurally correct per 1-OI-85. The double clutch method is described.

KA Justification:

Question asks if relationship is understood between CRDH and CRDM and control room controls. It tests knowledge of double clutching a stuck rod to get it unstuck from the full in position. Candidate must understand RMCS, CRDH, and CRDM systems to determine how controls may be operated to unstick the control rod.

Question Cognitive Level:

This question has high cognitive value because; the candidate must recognize interaction between systems, including consequences and implications.

Technical Reference(s): 1-OI-85 Rev 23 (Attach if not previously provided)

OPL171.005 Rev 17

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.26 (As available)

Question Source: Bank # FERMI 2

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 1	Control Rod Drive System	1-OI-85 Rev. 0023 Page 136 of 221

8.15 Control Rod Difficult to Withdraw

[1] **VERIFY** the control rod will **NOT** notch out and **REFER** Section 6.6. ☐

[2] **REVIEW** all Precautions and Limitations in Section 3.0. ☐

CAUTION

[NER/C] Never pull control rods except in a deliberate, carefully controlled manner, while closely monitoring the Reactor response. [INPO SOER-96-001]

[3] [NRC/C] **IF** RWM is enforcing, **THEN**

VERIFY RWM is operable and latched into the correct ROD GROUP. [NRC-IR 84-02] ☐

NOTES

1) Steps 8.15[4] through 8.15[6] should be used when the control rod is at position 00 while Step 8.15[7] should be used when the control rod is at or between positions 02 and 46.
2) Double clutching of a control rod at position 00 will place the rod at the "overtravel in" stop, independent of the RMCS timer, allowing maximum available time to establish over-piston pressure required to maintain the collet open and prevent the collet fingers from engaging the 00 notch.
3) Step 8.15[4] may be repeated as necessary until it is determined that this method will **NOT** free the control rod.

[4] **IF** the control rod problem is **NOT** believed to be air in the hydraulic system, **THEN**

PERFORM the following to double clutch the control rod at position 00:

[4.1] **PLACE AND HOLD** CRD NOTCH OVERRIDE, 1-HS-85-47, in EMERG ROD IN, for several seconds. ☐

[4.2] **CHECK** the control rod full in indication (double green dashes) on the Full Core Display for the associated control rod. ☐

[4.3] **SIMULTANEOUSLY PLACE AND HOLD** CRD NOTCH OVERRIDE, 1-HS-85-47, in NOTCH OVERRIDE AND CRD CONTROL SWITCH, 1-HS-85-48, in ROD OUT NOTCH. ☐

BFN Unit 1	Control Rod Drive System	1-OI-85 Rev. 0023 Page 137 of 221

8.15 Control Rod Difficult to Withdraw (continued)

[4.4] **WHEN** EITHER of the following occur:

- Control rod begins to move, **OR** ☐
- It is determined the rod will **NOT** move, **THEN**

RELEASE 1-HS-85-47 and 1-HS-85-48. ☐

[4.5] **IF** the control rod successfully notches out, **THEN**

PROCEED to Section 6.6 and **WITHDRAW** the control rod to the appropriate position. ☐

[4.6] **IF** desired, **THEN**

REPEAT Steps 8.15[4.1] through 8.15[4.5] several times prior to raising drive water pressure in Step 8.15[5]. ☐

[5] **IF** double clutching the control rod was unsuccessful, **THEN**

PERFORM the following to withdraw the control rod using elevated drive water pressure:

[5.1] **RAISE** drive water differential pressure to 300 psid as indicated on CRD DRIVE WTR HDR DP, 1-PDI-85-17A using CRD DRIVE WATER PRESS CONTROL VLV, 1-HS-85-23A. ☐

[5.2] **PERFORM** the following to double clutch the control rod at position 00 using elevated Control Rod Drive pressure:

[5.2.1] **PLACE AND HOLD** CRD NOTCH OVERRIDE, 1-HS-85-47, in EMERG ROD IN, for several seconds. ☐

[5.2.2] **CHECK** the control rod full in indication (double green dashes) on the Full Core Display for the associated control rod. ☐

[5.2.3] **SIMULTANEOUSLY PLACE AND HOLD** CRD NOTCH OVERRIDE, 1-HS-85-47, in NOTCH OVERRIDE and CRD CONTROL SWITCH, 1-HS-85-48, in ROD OUT NOTCH. ☐

Examination Outline Cross-reference:	Level	RO	SRO
215001 Traversing In-core Probe	Tier #	2	-------
K4.01 (10CFR 55.41.7)	Group #	2	-------
Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following:	K/A #	215001K4.01	
• Primary containment isolation: Mark-I&II(Not-BWR1)	Importance Rating	3.4	-------

Proposed Question: **# 56**

Unit 1 is operating at 100% Reactor Power with the "A" Traversing In-Core Probe (TIP) inserted in the core. A transient occurs resulting in the following plant conditions:

- Reactor Level is (-) 20 inches
- Drywell pressure is 1.5 psig

Which ONE of the following completes the statement?

The "A" TIP will withdraw to the __**(1)**__ position **AND** the Ball Valve position will be __**(2)**__.

A. **(1)** 'PARKED'
(2) open

B. **(1)** 'PARKED'
(2) closed

C. **(1)** 'IN-SHIELD'
(2) open

D. **(1)** 'IN-SHIELD'
(2) closed

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - The TIP is withdrawn to the 'in-shield'. For the ball valve to close, it must be in the 'in-shield' position. Plausible in that there are TIP interlocks associated with the 'PARKED' position. Part 2 incorrect, the Ball Valve will close. Plausible in that shear valve will not close.

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

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

D **CORRECT:** Per 1-AOI-64-2E, on a Group 8 signal, an AUTO withdraw signal is actuated. The TIP is withdrawn to the 'in-shield' position. Part 2 = Once in the 'in shield position, the Ball Valve will automatically close

KA Justification:

The KA is met because the question tests knowledge of TIP design feature and interlocks which provide for Primary containment isolation.

Question Cognitive Level:

Candidate must recognize Reactor Level is less than the set point for a Group 8 isolation and predict the impact on the TIP System.

Technical Reference(s): OPL171.17 Rev 15, OPL171.023 Rev 6 (Attach if not previously provided)

1-AOI-64-2E Rev 1 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.023 V.B.5 (As available)

Question Source:

Bank #	Hatch 09 #12	
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

7. Group 8

Obj. V.B.2
Obj. V.C.2

This group provides for isolation of the five Traversing Incore Probe (TIP) Guide Tubes via solenoid operated ball valves. Signals which initiate a Group 8 isolation are:

Gives TIP auto withdraw signal then closes the ball valve
Blue light on normal

- RPV Low Level (+2" or Level 3)
- Drywell High Pressure (2.45 psig)

G. Instrumentation

1. Sensors and logic are arranged such that no single failure will either initiate nor prevent an isolation.

Obj. V.B.3
Obj. V.C.3

2. The sensor arrangements used for the various isolation signals are as follows:

a. RPV low level

Obj. V.B.2
Obj. V.C.2

- Eight dp transmitters are used to produce low RPV level isolation signals. Four transmitters are used for the +2" (or Level 3) isolation; the other four are used for the -122" (Level 1) isolation.
- The (+2" or Level 3) transmitters are LIS-3-203A-D, while the (-122" or Level 1) transmitters are LIS-3-56A-D.

b. Main Steam Line Area High Temperature

Obj. V.B.2
Obj. V.C.2

- High temperature in the vicinity of the main steam lines is detected by 16 bimetallic temperature switches located along the main steam line between the drywell wall and the main turbine.

4 for each area total
1 from each area in A1, A2, B1, B2

BFN Unit 1	Traversing Incore Probe Isolation	1-AOI-64-2e Rev. 0001 Page 3 of 7

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 8, Traversing Incore Probe (TIP) Isolation and detection of a reactor coolant leak in a TIP guide tube.

2.0 SYMPTOMS

NOTE

A PCIS Group 8 isolation is initiated by either of the following:

- Reactor Vessel Water Level Low
- Drywell High Pressure

[1] Any one or more of the following annunciators in alarm:

- RX VESSEL WTR LEVEL LOW HALF SCRAM (1-XA-55-4A, Window 2) in alarm (Group 8 Isolation).
- DRYWELL PRESSURE HIGH HALF SCRAM (1-XA-55-4A, Window 8) in alarm (Group 8 Isolation).
- AIR PARTICULATE MONITOR RADIATION HIGH 1-RA-90-50A (1-XA-55-3A, Window 2) in alarm (indicative of TIP guide tube leak).
- RX BLDG AREA RADIATION HIGH 1-RA-90-1D (1-XA-55-3A, Window 22) in alarm (indicative of TIP guide tube leak).

3.0 AUTOMATIC ACTIONS

[1] **IF** a Group 8 isolation occurred, **THEN** the following are automatic actions:

- **IF** TIP probes are outside their shields, **THEN** TIP withdrawal initiated to IN-SHIELD position.
- TIP Ball Valves receive a close signal, or close after TIP probes are withdrawn to their IN-SHIELD position.
- TIP Purge Valves closes (no indications provided).

INSTRUCTOR NOTES

(iv) For Automatic TIP operation from 3-D Monicore (and local ATCU), the TIP must be in the 'PARKED' position prior to starting (Manually performed at NUMAC unit in Control room)

(v) All TIP machines can be configured to run simultaneously or one or more channels may be excluded. (via 3-D Monicore or locally at the ATCU)

(vi) Once the scanning selection is made and the TIP is at the Parked position, the ATCU controls the actual scanning function and interfacing with 3-D to download data.

(vii) Once initiated AUTO -TIP scan can be aborted at each individual NUMAC ATCU by pressing the ABORT AUTO-TIP soft key or via the 3-D Monicore program

(b) Manual Operation: Obj.V.B.5/V.C.5

(i) Location operation at the ATCU performed for:

- Exercising TIP drives for Rx startup
- Exercising Ball valves
- Selecting specific areas of the core to be monitored
- Used in conjunction with hand crank to Determine core top and bottom positions, parked position , in shield position for setting the travel limits.
- Obtain torque data

HLT 4 NRC Exam

12. 215001A3.03 001

Unit 1 is operating at 100% power with the "A" Traversing In-Core Probe (TIP) inserted in the core to perform 57CP-C51-010-0, "TIP Flux Probing Monitor".

A transient occurs on **Unit 1** with the following plant conditions:

Reactor pressure .... 900 psig and stable
Reactor level (lowest) .... -20 inches and slowly increasing
Drywell pressure .... 1.5 psig and stable
Drywell temperature .... 129°F

Which ONE of the following completes the statement below?

The "A" TIP will withdraw to the ______________ and the Ball Valve position will be ____________.

A. Indexer (Parked) position;
open

B. Indexer (Parked) position;
closed

C. In Shield position;
open

D.✓ In Shield position;
closed

Description:

The TIP receives a signal to withdraw to the "in-shield" position upon receipt of a group 2 signal (1.85 psig DW press or +3" RPV water level. The ball valve auto closes when the probe is fully withdrawn.
The normal position for the TIP is in the Indexer with the ball valve open.

Examination Outline Cross-reference:	Level	RO	SRO
230000 RHR/LPCI: Torus/Suppression Pool Cooling Mode	Tier #	2	------
G2.4.31 (10CFR 55.41.10)	Group #	2	------
Knowledge of annunciator alarms, indications, or response procedures.	K/A #	219000G2.4.31	
	Importance Rating	4.2	------

Proposed Question: **# 57**

Unit 2 is at 100% Reactor Power with Residual Heat Removal (RHR) Loop II in Suppression Pool Cooling mode. The following alarms are received on **Unit 1**:

- DRYWELL PRESSURE HIGH HALF SCRAM, (1-9-4A, Window 8)
- RX PRESS LOW CORE SPRAY/RHR PERMISSIVE, (1-9-3C, Window 35)

Which ONE of the following describes the current status of **Unit 2** RHR system **AND** what actions, if any, must be taken to restore Suppression Pool Cooling on Unit 2?

A. **ALL** four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling IMMEDIATELY.

B. 2A **AND** 2C RHR Pumps are tripped. 2B **AND** 2D pumps are unaffected. **NO** additional action is required.

C. **ALL** four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling after a 60-second time delay.

D. 2B **AND** 2D RHR Pumps are tripped. 2A **AND** 2C pumps are unaffected. Place RHR Loop I in Suppression Pool Cooling IMMEDIATELY.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: This is plausible because all four RHR pumps on Unit 2 will trip, but they are locked out from manual start for 60 seconds based on Diesel Generator and/or Shutdown Board loading concerns.

B INCORRECT: This is plausible based on RHR Loop II being the preferred pumps for Unit 2.

C **CORRECT:** Candidate must determine that the combination of Unit 1 annunciators indicates a CAS initiation and the response of Unit 2 RHR pumps in Suppression Pool Cooling. Then, must recognize that Preferred and Non-preferred Emergency Core Cooling System (ECCS) Pumps do NOT apply with the given conditions. Unit 1 Preferred RHR pumps are 1A and 1C. Unit 2 Preferred RHR pumps are 2B and 2D. LOCA signals are divided into two separate signals, one referred to as a Pre Accident Signal (PAS) and the other referred to as a Common Accident Signal (CAS). If a unit receives a CAS, then all its respective RHR and Core Spray pumps will sequence on based upon power source to the SD Boards. All RHR and Core Spray pumps on the non-affected unit will trip (if running) and will be blocked from manual starting for 60 seconds. After 60 seconds all RHR pumps on the non-affected unit may be manually started.

D INCORRECT: This is plausible if taken from the perspective of Unit 1 operation, NOT Unit 2 operation.

KA Justification:

This question satisfies the KIA statement by requiring the candidate to use knowledge of annunciators for specific plant conditions to determine which RHR pumps can be used for Suppression Pool Cooling.

Question Cognitive Level:

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-9-3C Rev. 22 / OPL171.044 R. 17 (Attach if not previously provided)
1-ARP-9-4A Rev. 18 / 2-OI-74 Rev. 152

Proposed references to be provided to applicants during examination: ____

Learning Objective: OPL171.044 V.B.9/13 (As available)

Question Source: Bank # BFN 0610 #32
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 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
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X
55.43

Comments: Question stem has been modified from original to meet KA. However, changes do not meet requirement of significantly modified question and is therefore identified as a Bank Question. Original attached.

BFN Unit 1	Panel 1-9-3 1-XA-55-3C	1-ARP-9-3C Rev. 0022 Page 41 of 41

RX PRESS LOW
CORE SPRAY/RHR
PERMISSIVE
1-PA-3-74
35

(Page 1 of 1)

Sensor/Trip Point:	
1-PIS-003-0074A	450 psi
1-PIS-003-0074B	450 psi
1-PIS-068-0095	450 psi
1-PIS-068-0096	450 psi

Sensor Location:	1-PIS-003-0074A	1-PIS-003-0074B	1-PIS-068-0095	1-PIS-068-0096
	1-PNLA-009-0081	1-PNLA-009-0082	1-PNLA-009-0081	1-PNLA-009-0082
	AUX. INST. Rm.	AUX. INST. Rm.	AUX. INST. Rm.	AUX. INST. Rm.
	EL 593'	EL 593'	EL 593'	EL 593'

Probable Cause:

A. Reactor Pressure ≤ 450 psig.
B. Sensor Malfunction.

Automatic Action:

A. Switch No. 2 permits opening of Inboard Injection Valves for Core Spray (1-FCV-75-25) and RHR (1-FCV-74-67).
B. In conjunction with High Drywell Pressure (> 2.45 psig) provides Auto Start Signal to Core Spray and RHR (LPCI).

NOTE

Switch No. 1 (setpoint 230 psig) auto closes the Recirc Pump A Disch. Valve, 1-FCV-68-3.

Operator Action:

A. **VERIFY** RPV pressure by multiple indications. ☐
B. **MONITOR** drywell pressure. ☐

References: 1-45E620-2-1 1-47E610-3-1 1-47W600-58 GE 0-730E930-3 and -9 Tech Spec 3.3.5.1

BFN Unit 1	Panel 9-4 1-XA-55-4A	1-ARP-9-4A Rev. 0018 Page 11 of 47

DRYWELL PRESSURE HIGH HALF SCRAM (8)	Sensor/Trip Point:	
	1-PIS-064-0056A 1-PIS-064-0056B 1-PIS-064-0056C 1-PIS-064-0056D	2.45 psig positive pressure in the drywell.

(Page 1 of 1)

Sensor Location:
1-PIS-064-0056A, 1-PNLA-009-0083, Auxiliary Instrument Room
1-PIS-064-0056B, 1-PNLA-009-0084, Auxiliary Instrument Room
1-PIS-064-0056C, 1-PNLA-009-0085, Auxiliary Instrument Room
1-PIS-064-0056D, 1-PNLA-009-0086, Auxiliary Instrument Room

Probable Cause:
A. ≥ 2.45 psig in the drywell.
B. SI/SR in progress.

Automatic Action:
A. Half scram if one sensor actuates.
B. Reactor scram if one sensor per channel actuates and group 2, 6 and 8 PCIS.

Operator Action:
A. **VERIFY** alarm by multiple indications. ☐
B. **IF** drywell pressure is ≥ 2.45 psig **AND** reactor has **NOT** scrammed, **THEN**
MANUALLY SCRAM the reactor. **ENTER** 1-EOI-1 & 1-EOI-2 FLOWCHARTS. ☐
C. **DISPATCH** personnel to the pressure switches to check for abnormal condition. ☐
D. **IF alarm is NOT valid**, **OR** initiating condition is corrected, **THEN** with SRO permission, **RESET** Half Scram. **REFER TO** 1-OI-99. ☐

References:
1-45E620-5-1 1-730E915-1
FSAR Sections 7.2.3.1, 7.2.3.5, 13.6.2

INSTRUCTOR NOTES

- **Common Accident Signal**

  -122" Rx water level (Level 1)

  OR

  2.45 psig DW pressure

  AND

  <450 psig Rx pressure

anticipation of a CAS injection requirement.
PAS and CAS are initiated by any unit Core Spray logic.

2. If a unit receives an accident signal, then all its respective RHR and Core Spray pumps will sequence on based upon power source to the SD Boards.

Level 1
OR
2.45# **AND** < 450 psig RPV

3. **Affected**, **non-affected** and **preferred pump logic** applies to Units 1 & 2 because they share DGs and SD boards. Unit 3 pumps are not affected by Unit 1/2 signals.

All 8 DGs are started by any unit PAS signal.

   a. All RHR and Core Spray pumps on the **non-affected** unit will trip (if running) and will be blocked from **manual** starting for 60 seconds.

   b. After 60 seconds all RHR pumps on the **non-affected** unit may be **manually** started.

   Operator diligence required to prevent overloading SD boards/DG's

   c. The **non-preferred** pumps on the **non-affected** unit are also prevented from automatically starting until the affected unit's accident signal is clear.

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0152 Page 393 of 442

Appendix A
(Page 2 of 7)

Unit 1 & 2 Core Spray/RHR Logic Discussion

2.2 **ECCS Preferred Pump Logic**

Concurrent Accident Signals On Unit 1 and Unit 2

With normal power available, the starting and running of RHR pumps on a 4KV Shutdown Board already loaded by the opposite unit's Core Spray, RHR pumps, and RHRSW pumps could overload the affected 4KV Shutdown Boards and trip the normal feeder breaker. This would result in a temporary loss of power to the affected 4KV Shutdown Boards while the boards are being transferred to their diesels. To prevent this undesirable transient, Unit 2 RHR Pumps 2A and 2C are load shed on a Unit 1 accident signal and Unit 1 Pumps 1B and 1D will be load shed on a Unit 2 accident signal. Unit 2 Core Spray Pumps 2A and 2C are load shed on a Unit 1 accident signal and Unit 1 Core Spray Pumps 1B and 1D will be load shed on a Unit 2 accident signal. This makes the Preferred ECCS pumps Unit 1 Division I Core Spray and RHR Pumps and Unit 2 Division 2 Core Spray and RHR Pumps. Conversely, the Non-preferred ECCS pumps are Unit 1 Division 2 Core Spray and RHR Pumps and Unit 2 Division 1 Core Spray and RHR Pumps.

The preferred and non-preferred ECCS pumps are as follows:

UNIT 1 & 2

PREFERRED ECCS Pumps

CS 1A, CS 1C, RHR 1A, RHR 1C
CS 2B, CS 2D, RHR 2B, RHR 2D

NON-PREFERRED ECCS Pumps

CS 1B, CS 1D, RHR 1B, RHR 1D
CS 2A, CS 2C, RHR 2A, RHR 2C

UNIT 3

Unit 3 does not have ECCS Preferred/Non-Preferred Pump Logic.

Accident Signal On One Unit

With an accident on one unit, ECCS Preferred pump logic trips all running RHR and Core Spray pumps on the non-accident unit.

0610 NRC RO EXAM

32. RO 219000K2.02 001/C/A/T2G2/OI-74//219000K2.02//RO/SRO/BANK

Given the following plant conditions:

- Unit 2 is at 100% rated power with Residual Heat Removal (RHR) Loop II in Suppression Pool Cooling mode to support a High Pressure Coolant Injection (HPCI) Full Flow Test surveillance.
- Unit 1 experiences a LOCA which results in a Common Accident Signal (CAS) initiation on Unit 1.

Which ONE of the following describes the current status of Unit 2 RHR system and what actions must be taken to restore Suppression Pool Cooling on Unit 2?

A. ALL four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling IMMEDIATELY.

B. '2A' and '2C' RHR Pumps are tripped. '2B' and '2D' pumps are unaffected. NO additional action is required.

C✓ ALL four RHR pumps receive a trip signal. Place RHR Loop II in Suppression Pool Cooling after a 60-second time delay.

D. '2B' and '2D' RHR Pumps are tripped. '2A' and '2C' pumps are unaffected. Place RHR Loop I in Suppression Pool Cooling IMMEDIATELY.

K/A Statement:

219000 RHR/LPCI: Torus/Pool Cooling Mode
K2.02 - Knowledge of electrical power supplies to the following: Pumps

K/A Justification: This question satisfies the K/A statement by requiring the candidate to use specific plant conditions and times to determine which RHR pumps can be used for Suppression Pool Cooling.

References: 2-OI-74, OPL171.044

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.

0610 NRC Exam

Examination Outline Cross-reference:	Level	RO	SRO
230000 RHR/LPCI: Torus/Pool Spray Mode	Tier #	2	------
K2.02 (10CFR 55.41.7)	Group #	2	------
Knowledge of electrical power supplies to the following: • Pumps	K/A #	230000K2.02	
	Importance Rating	2.8	------

Proposed Question: **# 58**

Unit 3 is operating at 100% Reactor Power with the Alternate Supply Breaker 1528 to 4 kV Unit Board 3B tagged out of service. An accident results in the following conditions:

- Unit Station Service Transformer 3B locks out
- Suppression Chamber Pressure reaches 3 psig
- 3A **AND** 3B RHR pumps are running in Suppression Chamber Spray Mode.

Which ONE of the following completes the statement?

The power supply for the 4 kV Shutdown Board to RHR Pump 3A is __**(1)**__ **AND** RHR Pump 3B is__**(2)**__ .

A. **(1)** Common Station Service Transformer A
(2) Common Station Service Transformer A

B. **(1)** Common Station Service Transformer A
(2) its associated Emergency Diesel Generator

C. **(1)** its associated Emergency Diesel Generator
(2) Common Station Service Transformer A

D. **(1)** its associated Emergency Diesel Generator
(2) its associated Emergency Diesel Generator

Proposed Answer: **B**

Explanation (Optional):

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

B **CORRECT**: 500 kV through USSTs is the normal supply to all U3 Unit Boards which in turn supply the 4kV Shutdown Boards. CSSTs are the alternate supply to the Unit Boards. EDGs are the emergency supply in case there is a loss of both normal and alternate supplies. Ordinarily the Unit Boards automatically transfer to alternate, however in this case the Unit Board 3B Alt is tagged out. So, when USST is lost, the 3C D/G will start and supply the 3EC 4 kV Shutdown Board which feeds RHR Pump 3B. Unit Board 3A will transfer and be supplied power via the CSST A. Unit Board 3A feeds 4 kV Shutdown Board 3EA which feeds RHR Pump 3A.

C INCORRECT: Part 1 and 2 incorrect - Plausible since the examinee must know which Unit Boards Supply which Shutdown Boards then RHR Pumps to eliminate these distractors.

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

KA Justification:

The KA is met because it tests knowledge of electric power supplies to RHR Pumps.

Question Cognitive Level:

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)
OPL171.036 Rev. 12
3-ARP-9-8B Rev. 14

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.036 V.B.8 (As available)

Question Source: Bank #
Modified Bank # Hatch 09 #22 (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:

b. Condensate pumps (3) 900hp (Unit 3),1250 hp (for U-1 and Unit 2)

c. Condensate booster pumps (3) 1750hp (Unit 3), 3000 hp (each for Unit 1and Unit 2)

d. Raw Cooling Water pumps (3) 300hp each

e. Control Rod Drive Water Pump A , 250 hp.(Units 1, 2, and 3; Board C)

f. 480V Unit Board transformers (2) (Boards 1A and 1B)

g. 480V Water Supply Board transformers (4kV B boards)

2. There are nine 4kV Unit Boards - three per unit. They are located in the turbine building on Elev. 604 (A and C Boards) and Elev. 586 (B Boards). The USSTs are the normal supply and start buses are the alternate.

Refer to prints 15E-500 series.

a. USST A is the normal supply to 4kV Unit Board C and USST B is the normal power supply to 4kV Unit Boards A and B. (All Units)

Obj. V.B.6.d
Obj. V.C.1.d
Obj. V.D.6.d

b. 4kV Start Bus 1A is the alternate power supply to 4kV Unit Boards 1A, 2A, 2C, 3A, and 3C.

c. 4kV Start Bus 1B is the alternate power supply to 4kV Unit Boards 1B, 1C, 2B, and 3B.

3. U1 and U2 4kV Unit Boards A and B supply power to 4kV Shutdown Buses 1 and 2 thereby providing off-site power to the Standby AC Power System. 3A, 3B 4kV Unit Boards supply power directly to the U3 4kV Shutdown Boards.

4. <u>Control Room Indications</u>

Obj. V.B.6.a
Obj. V.B.6.c
Obj. V.B.7
Obj. V.C.1.a
Obj. V.C.1.c
Obj. V.D.6.a
Obj. V.D.6.c
Obj. V.D.7

a. Voltmeter, 2 ammeters (one on each supply) on panel 9-8 from each 4kV Unit Board.

b. Ammeters - in the Control Room for each of the boards' pump motors.

No Amp Meters for CRD Pumps

5. Indication of the 4kV Unit Boards' voltages and

amperages are available on panel 9-8. In addition, each boards pump motor amps is also available. (except CRD pumps)

Monitor redundant indications

6. Transfer Schemes

a. General Operation

0-45E763-1, 2

The 4kV Unit Boards are normally fed from the Unit Station Service Transformers with an alternate feed from the 4kV Start Buses.

Obj. V.C.2.d
Obj. V.B.8.d
Obj.V.D.8.d
Illustration 1

0-OI-57A

Transfer to the Start Buses may be manual or automatic but transfer back to the USST is manual only. All manual transfers and transformer trip-actuated transfers are fast transfers. Undervoltage relay-actuated transfer is delayed until bus voltage has decreased to 30% normal. A voltage relay prevents automatic transfer to a dead bus. The breakers are electrically interlocked to prevent paralleling the Unit and Common transformers.

Only 1C, 2C, 3A/B/C Unit Board have 30% slow transfer. Removed from 1A/B & 2A/B Unit Board.

b. Automatic Fast transfer of Unit boards occur on Gen protective relaying or USST relaying.

To automatically fast-transfer from normal to alternate

(1) normal feed breaker tripped

(2) 43 selector switch in AUTO

(3) Alternate feed line-side voltage available 27SUX

(4) Alternate feeder breaker closes, provided no lock-outs are present.

c. To automatically transfer from normal to alternate from undervoltage

UV transfer only on 1C, 2C, 3A, 3B, 3C

(1) 43 transfer switch in AUTO

(2) Alternate voltage available

BFN Unit 3	Panel 9-8 3-XA-55-8B	3-ARP-9-8B Rev. 0014 Page 13 of 38

4KV UNIT BD 3A AUTO XFR 3-XA-57-4 (10)	Sensor/Trip Point:	
	43 Switch in AUTO (XS-57-4) and	Generator/Transformer protective relays
	Alt feeder brk 1432 (52a contacts) closed	or undervoltage relay

(Page 1 of 1)

Sensor Location: Unit Bd 3A
El 604', T-16 C-LINE
Turb Bldg

Probable Cause:

A. Protective relay operation.
- 86TX, 86TF, 86C (any relay causes high speed transfer).
- 27TUAX (time delayed transfer).

B. Fuse failure (metering potential transformer).
C. Relay malfunction.

Automatic Action: Transfer to alternate feeder (Start Bus 1A).

Operator Action:

A. **VERIFY** Unit in stable condition by checking:
- Condensate Pump 3A ☐
- Condensate Booster Pump 3A ☐
- RCW Pump 3A ☐
- CCW Pump 3A ☐

B. ON Panel 3-9-8, **CHECK**:
1. Alternate bkr to Unit Bd 3A closed (red light illuminated). ☐
2. Normal bkr Unit Bd 3A open (green light illuminated). ☐
3. **SELECT** Unit Bd 3A with volt switch and **CHECK** voltage on meter (3-EI-57-28). ☐

C. **CHECK** Unit Bd 3A for abnormal conditions: relay targets, smoke, burned paint, bkr position, etc. ☐
D. **REFER TO** 0-OI-57A for board transfer. ☐

References: 3-45E721 0-45N763-1 3-45N620-11

(7) CASx (CASA or CASB) accident signal (after 5 second delay via BBRX relay) — -122" RxVL OR 2.45 DWP AND < 450# RPV

I. 4kV Shutdown Boards (Normal Power Seeking)

Refer to prints 15E-500 series Key Diagram of STDBY Aux. Power System
Obj. V.B.6.c
Obj. V.C.1.c
Obj. V.D.6.c

1. Power sources

a. 4kV supplies to each U1/2 Shutdown Board: are as follows:

Board	NORMAL Supply
A	Shutdown Bus 1
B	Shutdown Bus 1
C	Shutdown Bus 2
D	Shutdown Bus 2

The first alternate is from the other Shutdown Bus. The second alternate is from the diesel generator. The third alternate is from the U3 diesel generators via a U3 Shutdown Board.

SBO
3 ½ via bustie board
½ ½ via other SD Bus

b. There are two possible 4kV supplies to each U3 Shutdown Board:

Board	NORMAL Supply
3EA	Unit Board 3A
3EB	Unit Board 3A
3EC	Unit Board 3B
3ED	Unit Board 3B

(1) The first alternate is from the diesel generators. The U1/2 diesel generators cannot supply power to the U3 Shutdown Boards alone. They may, however, be paralleled with the U3 diesel generators for backfeed operation. The tie breaker off the unit 3 Shutdown Board is interlocked as follows:

U1/2 RHR B	II	Shutdown Board C	1/2 C	pumps come off the C and B shutdown boards respectively.
U1/2 RHR D	II	Shutdown Board D	1/2 D	
U3 RHR A	1	Shutdown Bd 3EA	3A	BFN events have occurred due to racking out the wrong breaker which resulted in LCO 3.0.3 BFPER-997296 See OPL171.045 for details. TP-10, 11 and 12
U3 RHR C	1	Shutdown Bd 3EB	3B	
U3 RHR B	II	Shutdown Bd 3EC	3C	
U3 RHR D	II	Shutdown Bd 3ED	3D	

c. Pump cooling — Obj. V.B.5

(1) Pump bearings cooled by RHR pump discharge from seal heat exchanger

(2) Seal heat exchanger normally by EECW — North or South headers

(3) EECW also cools the RHR room coolers

d. Check valves located on the discharge of the pumps — TP-1 and 2

(1) Prevents backflow through the pumps

(2) Maintains a water leg in the discharge piping

(3) Water legs kept filled up to the injection valves by the keep fill system. — Obj. V.B.6 Obj. V.D.2

(4) This prevents water hammer on pump start and possible pipe/valve damage.

(5) Also enables water to reach the core in the shortest possible time in the event of a LOCA

(6) Discharge piping kept pressurized (Tech. Requirements Manual limit) — TRM 3.5.4

3. RHR Heat Exchangers — Obj. V.B.7 Obj. V.E.5

a. Four vertical, shell and tube per unit — Baffled at top. RHRSW vents on top head (2)

b. Located in separate portions of Rx Bldg.

c. Design data/conditions

(1) Shell side fluid - Rx water or S/P water @ 10000 gpm

HLT 4 NRC Exam

22. 226001K2.02 001

Unit 1 was operating at 100% power with the Alternate Supply Breaker to 4160VAC bus "1E" tagged out.

A loss of Startup Transformer (SAT) "1D" occurred.

- o Torus Pressure reaches 3 psig during the transient.
- o "1A" and "1B" RHR pumps are running in the Torus Spray Mode.

The power supply for the 4160 VAC bus to the "1A" RHR Pump is ___(1)___ and to the "1B" RHR Pump is ___(2)___.

A. (1) SAT "1C"
(2) SAT "1C"

B.✓ (1) its associated EDG
(2) SAT "1C"

C. (1) SAT "1C"
(2) its associated EDG

D. (1) its associated EDG
(2) its associated EDG

Examination Outline Cross-reference:	Level	RO	SRO
234000 Fuel Handling Equipment	Tier #	2	------
A4.02 (10CFR 55.41.7)	Group #	2	------
Ability to manually operate and/or monitor in the control room: • Control rod drive system	K/A #	234000A4.02	
	Importance Rating	3.4	------

Proposed Question: **# 59**

Given the following:

- Unit 1 is in Mode 5
- The Refuel Platform is over the Spent Fuel Pool
- The Reactor Mode Switch is in START & HOT STBY for testing

Which ONE of the following identifies when a rod block will occur?

A. When the Refuel Platform Fuel Grapple is lowered.

B. When a load is placed on the Refuel Platform Fuel Grapple.

C. When the Refuel Platform is driven near or over the core.

D. When the Refuel Platform starts moving towards the core.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Plausible in that this would be the correct answer if the Mode Switch was in Refuel and Platform near or over the core.

B INCORRECT: Plausible in that this is true if the service platform hoist is loaded.

C **CORRECT:** As the Refuel Platform is driven near the core with the Mode Switch in Startup, a rod block will occur.

D INCORRECT: The refuel platform can move towards the core but will be stopped when the platform starts to move over the core

KA Justification:

The KA is met because the question tests the ability to monitor Control Rod Drive system in the control room as it applies to Fuel Handling Equipment.

Question Cognitive Level:

This question is rated as Fundamental Knowledge

Technical Reference(s): 0-GOI-100-3A Rev. 53 (Attach if not previously provided)

OPL171.053 Rev. 18

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.053 V.B.5 (As available)

Question Source: Bank # Cooper 08 #59

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Cooper 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:

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.)

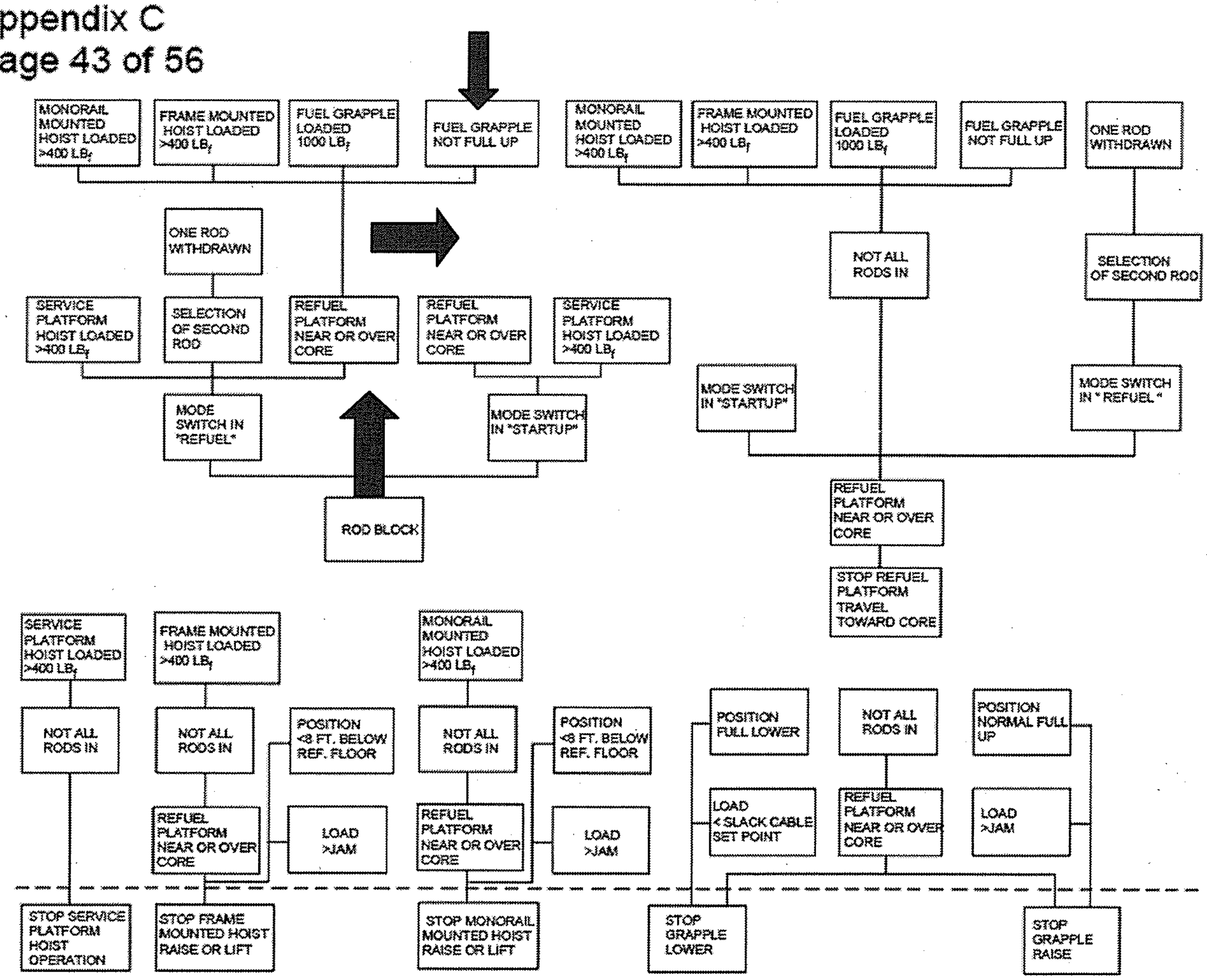
E. The Associated Hoist operation will be stopped if any of the following exist.

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.

OPL171.053
Revision 18
Appendix C
Page 43 of 56



TP-3: Refueling Rod Blocks and Refueling Interlocks

COOPER 2008

Question Number	New, Modified or Bank	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
NRC RO 59	Bank 1477	00	07/28/1999	01/30/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	1	1	4	Multiple Choice	

Topic Area	Description
Systems	COR0012102001100B Refueling

Related Lessons
COR0012102 Refueling

Related Objectives
COR0012102001100B Given conditions associated with refueling activities, determine if the following should occur: Refueling mast restrictions

Related References
10CFR55.41(b)7

Related Skills (K/A)
234000.K4.02 Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and /or interlocks which provide for the following: (CFR: 41.7) Prevention of control rod movement during core alterations (3.3 / 4.1)

QUESTION: NRC RO 59

Given the following:

- Core offload is in progress.
- The refuel platform is over the fuel pool.
- The Reactor Mode Switch is placed in START & HOT STBY for testing.

If core off-load activities continue, **WHEN** will a rod block occur?

a. When the refuel platform starts moving towards the core.

b. When the refuel platform is driven near or over the core.

c. When the Fuel Grapple is lowered.

d. When a load is placed on the Fuel Grapple.

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ANSWER: NRC RO 59

b. When the refuel platform is driven near or over the core.

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EXPLANATION OF ANSWER: b. correct. As a bundle is moved from the fuel pool to the core the rod block will occur when the refuel bridge is driven over the core. a. The refuel platform can move towards the core but will be stopped when the platform starts to move over the core. c & d. the rod block would occur before this point.

Examination Outline Cross-reference:	Level	RO	SRO
259001 Reactor Feedwater System	Tier #	2	------
K5.03 (10CFR 55.41.5)	Group #	2	------
Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM :	K/A #	259001K5.03	
• Turbine operation: TDRFP's-Only	Importance Rating	2.8	------

Proposed Question: **# 60**

RFPT 1A OVERSPEED TEST TRIP LOCKOUT, 1-HS-3-109A, has just been placed in the 'ELEC' position per 1-OI-3, "Reactor Feedwater System," Section 8.10, "Overspeed Trip Exerciser Test," when RFPT 1A experiences an **ACTUAL** over-speed condition.

Which ONE of the following describes the **AUTOMATIC** response of RFPT 1A?

A. Trips as a result of the electrical trip solenoid.

B. Trips as a result of the mechanical trip mechanism.

C. Will **ONLY** trip when 1-HS-3-109A is restored to the 'NORM' position.

D. Ramps up due to the overspeed condition **AND** locks at a high speed stop.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: The mechanical trip solenoid is still active, and will actuate, causing a trip of the RFPT.

B **CORRECT:** The test blocks the electrical device trip but leaves the mechanical trip system active.

C INCORRECT: Yes the RFPT will trip when restored to NORM; however, the mechanical trip system remains active even in ELEC.

D INCORRECT: Even though it will ramp up, there is no protective function short of the mechanical overspeed trip device.

KA Justification:

The KA is met because the question tests the candidate's knowledge of the operational implications of Turbine operation as it applies to the Reactor Feedwater System.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.026 Rev. 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.026 V.B.5 (As available)

Question Source: Bank # BFN 1006 Audit #63

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:

i. The Emergency Governor Lockout Valve provides a method to periodically test and exercise the Trip Dump Valve and mechanical overspeed mechanism without tripping the turbine. To accomplish this it is moved up into position to block the pressure holding the Pressure Relay Valve open from being dumped by the Trip Dump Valve.

Review trip system failure events described in INPO O and MR 399.

j. Placing the OVERSPEED TEST TRIP LOCKOUT Switch in the MECH position on panel 9-6 energizes the Lockout Solenoid Valve providing the oil pressure to move the Emergency Governor Lockout Valve to block trips. Emergency Trip Governor Valve position indication changes from Green (normal) to amber (lockout). Electrical overspeed will deenergize the Lockout Solenoid Valve.

OI-3 section 8.10

k. To provide continuous trip protection for the RFPT during testing, the lockout oil pressure is also ported to the Electrical Trip Solenoid Valve which will dump lockout pressure should a trip condition occur while testing the Trip Dump Valve and overspeed mechanism. (The 1/8" orifice in the oil supply cannot maintain pressure with a trip dump.) Electrical overspeed will also deenergize the lockout solenoid yielding earlier response to an actual overspeed condition.

Note: the lockout valve blocks ALL trips. Only removal of the lockout will restore trips. Electrical overspeed removal of the lockout occurs before the actual trip setpoint is reached. See OI-3 8.9.49.

l. The OVERSPEED TEST pushbutton supplies oil pressure to move the overspeed plunger which trips the Trip Dump Valve to exercise the overspeed mechanism and Trip Dump Valve. The green normal indication extinguishes and the white trip light lights.

m. The OVERSPEED TEST RESET pushbutton performs the Trip Dump Valve reset for this test. The normal Trip Reset will not function because the Stop and Control Valves must be closed for a normal reset. The white trip light must be extinguished and the green reset light must come on before returning the OVERSPEED TEST TRIP LOCKOUT switch to normal to preclude an actual turbine trip.

n. 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

o. '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

p. 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.

q. 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.

r. 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

Examination Outline Cross-reference:	Level	RO	SRO
271000 Offgas System	Tier #	2	------
K3.02 (10CFR 55.41.5)	Group #	2	------
Knowledge of the effect that a loss or malfunction of the OFFGAS SYSTEM will have on following:	K/A #	271000K3.02	
• †Off-site radioactive release rate	Importance Rating	3.3	------

Proposed Question: **# 61**

Unit 2 Offgas Post Treat Radiation Monitor, 2-RM-90-265A, has failed downscale.

Which ONE of the following identifies the impact of this failure?

If Offgas Post Treat Radiation Monitor, 2-RM-90-266A, reaches the High-High-High setpoint, Off-Gas System Isolation Valve, 2-FCV-66-28, __**(1)**__ close.

If Offgas Post Treat Radiation Monitor, 2-RM-90-266A, fails downscale, Off-Gas System Isolation Valve, 2-FCV-66-28, __**(2)**__ close.

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

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

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

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

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Parts 1 and 2 correct - OG POST TREATMENT RAD MONITOR DOWNSCALE (55-4C-32) alarms when signal is < 1 cps and sends a trip signal to the Off-Gas isolation logic. OG POST-TREATMENT OFF-GAS HI-HI-HI/INOP (55-4C-35) alarms at 6.2X105 cps sends a trip signal to the Off-Gas isolation logic. Off-Gas isolation is a two-out-of-two logic. Downscale, Hi-Hi-Hi or INOP on RM-90-265A AND Downscale, Hi-Hi-Hi or INOP on RM-90-266A will automatically isolate the Off-Gas system after a 5 second time delay. (FCV-66-28 closes).

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

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

D INCORRECT: Part 1 incorrect – Plausible in that two channels are required for an isolation signal to 2-FCV-66-28 to be generated. Some process radiation monitors do not combine downscale with high radiation to generate the trips signal. Example: this combination would not result in a actuation of trip logic for Rx Zone Rad Monitors. Part 2 incorrect – Plausibility based on the misconception that the downscale does not result in a trip condition which is true of some process rad monitors. Example: Downscale on MSL Rad Monitors does not result in actuation of associated trip logic.

KA Justification:

The KA is met because the question tests candidates' knowledge of the effect that a malfunction of the OFFGAS SYSTEM Post Treatment Radiation Monitor will have on Offgas Isolation Valve 2-FCV-66-28 and therefore Off-site radioactive release rate.

Question Cognitive Level:

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): OPL171.033 Rev. 13 (Attach if not previously provided)

2-OI-90 Rev. 79

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033 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 2	Radiation Monitoring System	2-OI-90 Rev. 0079 Page 10 of 70

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 - opens Adsorber Inlet Valve, 2-FCV-66-113A, and closes Adsorber Bypass Valve, 2-FCV-66-113B, if 2-HS-66-113 is in AUTO.

   b. High-High - Alarms only.

   c. High-High-High - sends a close signal to Off-Gas System Isolation Valve, 2-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 or 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/SRO.

BFN Unit 2	Radiation Monitoring System	2-OI-90 Rev. 0079 Page 40 of 70

Illustration 1
(Page 2 of 4)

Radiation Monitoring System Operational Summary

NOTE
Only the noble gas detectors are required by Technical Specifications.

Stack Gas Radiation Monitors 0-RE-90-147&148	Two radiation detectors monitor activity release rates from the Off-Gas stack. PNL 0-25-39
Off-Gas Pretreatment Radiation Monitors 2-RE-90-157&160	Two radiation detectors monitor radiation at the inlet of the 6-hour holdup volume. PNL 2-25-38
Off-Gas Post-treatment Radiation Monitors 2-RE-90-265&266	Two radiation detectors monitor radiation downstream of the charcoal beds (adsorbers). If adsorber control switch is in AUTO, the detector High trip ensures Off-Gas flow is directed through the adsorbers by inserting a CLOSE signal to the adsorber bypass valve and an OPEN signal to the adsorber inlet valve. High-High gives alarm signal. When the High-High-High trip is actuated, the Off-Gas System isolation valve closes after a 5-second time delay. PNL 2-25-94
Main Steam Line Radiation Monitors 2-RE-90-136,137	Two detectors monitor the Main Steam Lines for high radiation.
Process Liquid Radiation Monitors	Radiation detectors monitor radiation in the following systems:
2-RE-90-131A	Reactor Building Closed Cooling Water (off-line), Pnl 2-25-339
2-RE-90-130	Radwaste Effluent Discharge (in-line only)
2-RE-90-133A & 134A	RHR Service Water (off-line), Pnl 2-25-337 & 338
2-RE-90-132A	Raw Cooling Water (off-line), Pnl 2-25-336

INSTRUCTOR NOTES

(5) Off-Gas isolation is a two-out-of-two logic — Obj. V.B.4.b; Obj. V.C.4.a

(a) Downscale, Hi-Hi-Hi or INOP on RM-90-265A

AND

Downscale, Hi-Hi-Hi or INOP on RM-90-266A

will automatically isolate the Off-Gas system after a 5 second time delay. (FCV-66-28 closes)

3. Stack-Gas Radiation Monitoring System (RM-90-147 & 148) — Obj. V.D.7; Obj. V.B.3.b; Obj. V.C.3.b

a. Purpose

(1) Used to indicate and record release rates from the stack during normal operation and to alarm whenever limits are reached

(2) To monitor the stack gas effluent, a sample is drawn through an isokinetic probe which is located two-thirds of the way up the stack — Note: isokinetic probe explained in section 9 of this lesson

b. The stack receives exhaust gases from following:

(1) Steam Jet Air Ejector (SJAE)

(2) Steam Packing Exhauster (SPE)

(3) Mechanical vacuum pump

(4) Standby Gas Treatment (SGT)

(5) Stack Gas Analyzer Room Vent

PLAUSIBILITY SUPPORT

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

PLAUSIBILITY SUPPORT

INSTRUCTOR NOTES

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

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

-OR-

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

-OR-

One monitor INOP in each channel (division) — One-out-of-two, twice; Obj. V.B.3.f; Obj. V.C.3.f

-OR-

Loss of RPS power to either channel

(2) Automatic actions — Obj. V.B.1,3.e; Obj. V.C.1,3.e; Obj V.D.6

(a) Refuel Zone Trip

(i) Isolate Refuel Zone

(ii) Starts Standby Gas Treatment System

(iii) PCIS Group 6 isolation — Obj. V.B.1,3.f, 3.g; Obj. V.C.1,3.f, 3.g

(iv) Starts CREVs

(b) Reactor Zone Trip

(i) Isolate Control Room, Reactor Zone, and Refueling Zone ventilation

(ii) Starts Standby Gas Treatment System

(iii) Start CREVs

(iv) PCIS Group 6 isolation

Examination Outline Cross-reference:	Level	RO	SRO
288000 Plant Ventilation Systems	Tier #	2	------
A3.01 (10CFR 55.41.7)	Group #	2	------
Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: • Isolation/initiation signals	K/A #	288000A3.01	
	Importance Rating	3.8	------

Proposed Question: **# 62**

Given the following Control Room Emergency Ventilation (CREV) system conditions:

- CREV Train A was started to prove operability following maintenance on the charcoal trays using the STOP-AUTO-START switch on Panel 9-22.
- The SYSTEM PRIORITY SELECTOR SWITCH is selected for "TRAIN-B".

Which ONE of the following describes the CREV system response should a valid CREV initiation signal be received?

CREV Train B would__**(1)**__ **AND** CREV Train A would __**(2)**__.

A. **(1)** initiate
(2) shutdown

B. **(1) NOT** initiate
(2) shutdown

C. **(1)** initiate
(2) NOT shutdown

D. **(1) NOT** initiate
(2) NOT shutdown

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT Part 1 correct – See explanation C. Part 2 incorrect – See Explanation B.

B INCORRECT: Part 1 incorrect - Normally, when an auto initiation signal is received, the TRAIN selected for "secondary" begins its start sequence but will not finish if the Primary CREV train is running. This is sensed by looking at the ΔP across the HEPA filter. Since Train B was selected as the Primary CREV unit, the start sequence does not look at the ΔP. Part 2 incorrect - This would be correct if CREV Train A was started using the AUTO-INITIATE TEST switch, as would be the case during the periodic surveillance test.

C **CORRECT:** Part 1 correct - CREV Train B will initiate without a time delay since the CREV UNIT PRIMARY SELECTOR SWITCH is selected for "TRAIN-B". Part 2 correct - CREV will not automatically shutdown with a valid initiation signal present.

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

KA Justification:

The KA is met because the question tests the ability to monitor automatic operation of Control Room Emergency Ventilation including system initiation signals for the given conditions.

Question Cognitive Level:

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-31 Rev. 136 (Attach if not previously provided)
1-EOI-3 Rev 12, OPL171.067 Rev 16

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.2.g (As available)

Question Source: Bank # 0707 #38
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Browns Ferry 0707

(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 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.

7.20 Shutdown of Control Room Emergency Ventilation (CREV) Fans to Standby Readiness

CAUTION

In the event the pressurization units have initiated automatically on a Group Six Isolation signal or control room ventilation inlet duct high radiation, the initiating condition should be removed or corrected prior to shutting down the units.

NOTES

1) [NRC/C] After an automatic initiation, the CREV System is required to be manually SHUTDOWN from the control room by placing the CREV Train handswitches to STOP, which also resets the initiation logic. A local shutdown will **NOT** reset the seal-in logic. [LER 88-035]
2) Normally, the train selected by the CREV PRIMARY UNIT SELECTOR as the lead train starts on auto initiation and the other train remains idle, unless the lead train trips. Upon restoring the system to standby, the handswitch for the idle train is required to be turned to STOP first to prevent it from starting when the Lead train is stopped.
3) The charcoal adsorber resistance heaters will be automatically placed in operation to maintain the charcoal beds at 10 degrees F greater than ambient temperature, provided that fan A(B) power supply breakers 14C (13C2) on 480V Reactor MOV Board 1A(3B) are closed.
4) If a CREV train is in service for testing, and an actuation signal is received, both trains will be running. In this case, ONLY the train under test will be required to be shutdown.

[1] **IF** CREV was manually or automatically initiated,

AND conditions requiring the initiation are cleared, **THEN**

STOP CREV train A(B) as follows:

[1.1] **VERIFY** CREV TRAIN A INIT/CB ISOL, 0-HS-31-150A, and CREV TRAIN B INIT/CB ISOL, 0-HS-31-150B, are in the AUTO position at Panel 2-9-22. ☐

[1.2] For the CREV TRAIN that is NOT running, **PLACE** CREV TRAIN A, 0-HS-31-7214A, or CREV TRAIN B, 0-HS-31-7213A, momentarily in STOP at Panel 2-9-22. ☐

BROWNS FERRY 0707 #38

Examination Outline Cross-reference:	Level	RO	SRO
290003K1.04	Tier #	2	
Knowledge of the physical connections and/or cause-effect relationships between Control Room HVAC system and the following: Nuclear Steam Supply Shut off System (NSSSS/PCIS).	Group #	2	
	K/A #	290003K1.04	
	Importance Rating	3.2	3.3

Proposed Question: **RO # 38**

Given the following Control Room Emergency Ventilation (CREV) system conditions:

- CREV Train "A" was started to prove operability following maintenance on the charcoal trays using the STOP-AUTO-START switch on Panel 9-22.
- The SYSTEM PRIORITY SELECTOR SWITCH is selected for "TRAIN-B".

Which ONE of the following describes the CREV system response should a valid CREV initiation signal be received?

On a valid initiation, CREV Train "B" would ________(1)________ and CREV Train "A" would __________(2)__________.

	(1)	(2)
A.	initiate	shutdown.
B.	initiate	NOT shutdown.
C.	NOT initiate	shutdown.
D.	NOT initiate	NOT shutdown.

Examination Outline Cross-reference:	Level	RO	SRO
290001 Secondary Containment	Tier #	2	------
A1.01 (10CFR 55.41.5)	Group #	2	------
Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including:	K/A #	290001A1.01	
• System lineups	Importance Rating	3.1	------

Proposed Question: **# 63**

On Unit 1, the Standby Gas Treatment System (SGTS) A Control Switch, 1-HS-65-18A, on Panel 1-9-25 has been placed in the pull-to-lock position.

Which one of the following conditions would still cause SGTS A to start?

A. Unit 2 drywell pressure rises to 2.5 psig.

B. Unit 3 SGTS A start pushbutton is depressed.

C. The local (SGTS Building) SGTS A start pushbutton is depressed.

D. SGT TRAIN "A" INBD ISOL TEST SIG Keylock switch (HS-65-48A) is placed in the TEST position.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: With the SGTS A Control Switch in Pull to Lock, the system will not auto start on 2.5 psig. Plausible in that this condition will normally cause SGTS A to start.

B INCORRECT: With the SGTS A Control Switch in Pull to Lock, the system will not start with the Unit 3 SGTS A Start Pushbutton. Plausibility based misconception that Unit Control Switch will not affect operation from Unit 3.

C **CORRECT:** With control switch in pulled-out (STOP) position, the blower can still be started locally.

D INCORRECT: With the SGTS A Control Switch in Pull to Lock, the system will not auto start with SGT TRAIN "A" INBD ISOL TEST SIG Keylock switch (HS-65-48A) placed in the TEST. Plausible in that this condition will normally cause SGTS A to start.

KA Justification:

The KA is met because the question tests the candidate's ability to predict changes in the SGTS associated with operating the SGTS Control Switch.

Question Cognitive Level:

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. Candidate must be able to predict the effect of changing the Control Switch position from its normal line up on the operation of the system.

Technical Reference(s):	OPL171.018 Rev 10	(Attach if not previously provided)
	0-OI-65 Rev 53	

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:	Bank #	OPL171.018 #13	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

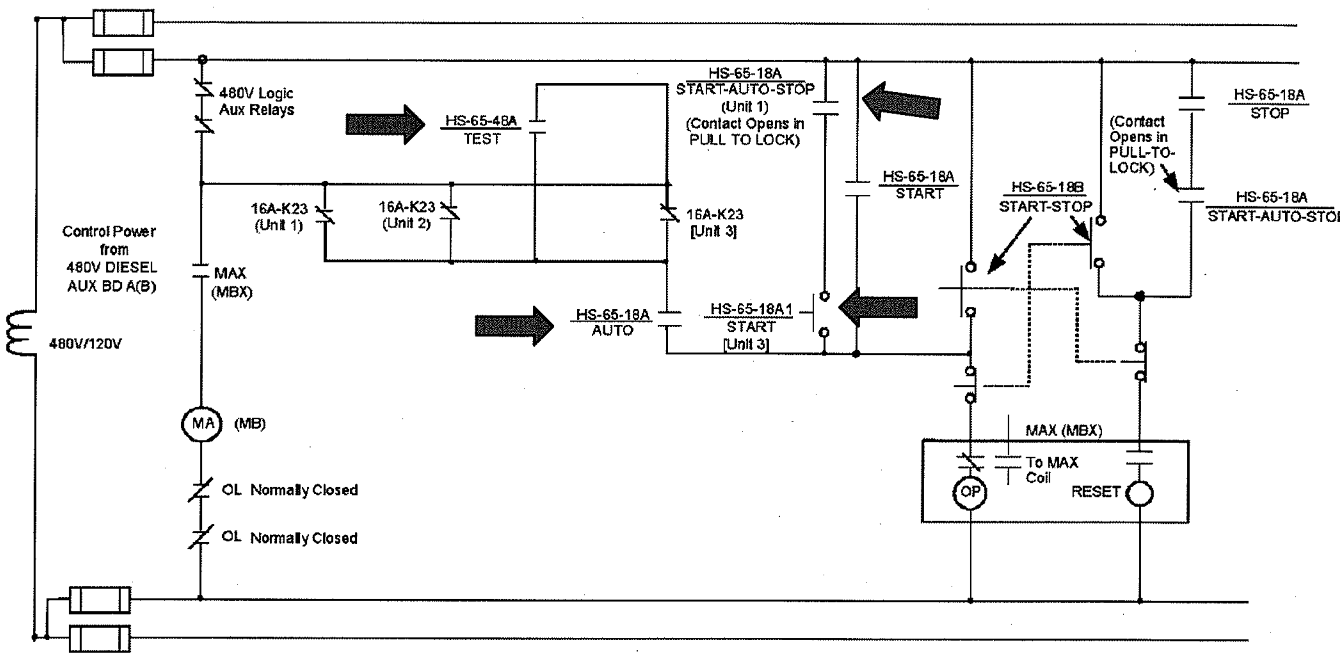
(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

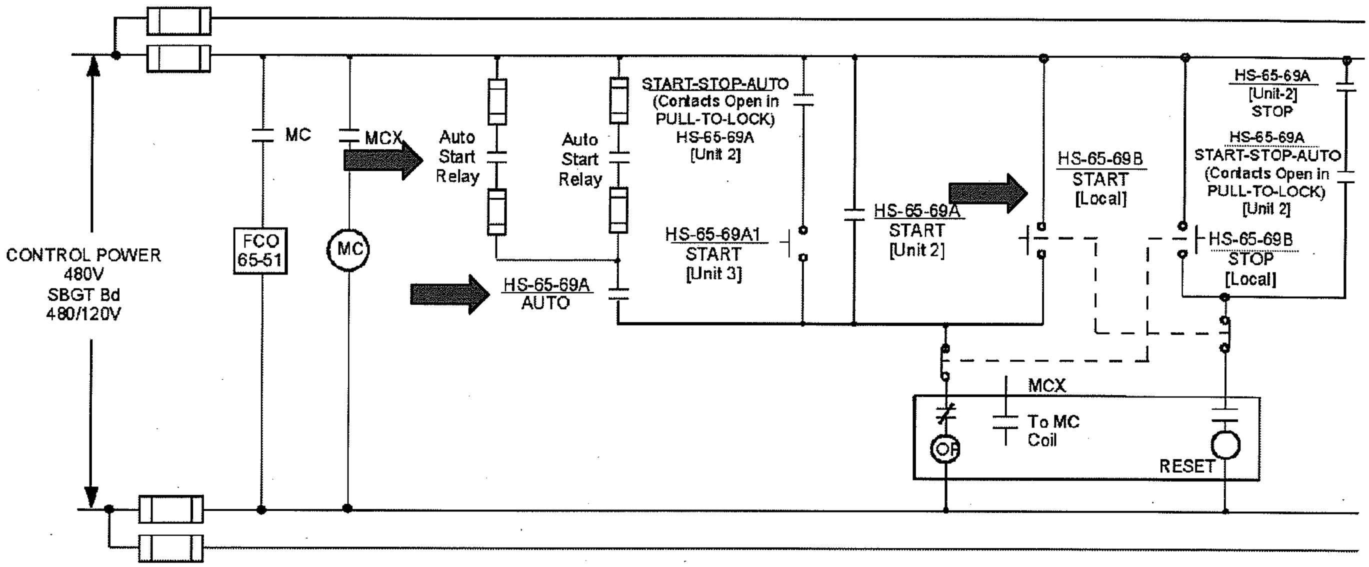
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

OPL171.018
Revision 10
Appendix C
Page 36 of 37



TP-3: SGT A (B) CONTROL CIRCUIT



TP-3: SGT A (B) CONTROL CIRCUIT

INSTRUCTOR NOTES

3. Control Logic

a. Control switch must be in AUTO for auto-start signal to start train.

b. With control switch in pulled-out (STOP) position, the blower can still be started locally.

c. CS is spring-return-to-AUTO, unless pulled out in STOP.

d. Switch can be pulled out in the STOP position only.

e. Inlet damper will auto-open when fan motor coil is energized, if in AUTO. — TP4; LER 88-017

f. SGT A and B will trip on initiation of the 480V load-shed logic, but will auto-restart after forty seconds if initiation signal is present. SGT C is not affected by 480 volt load shed logic initiation.

g. LER 88-017 covers an event that occurred at BFNP. With the supply breaker (480VAC) open, an engineer directed Maintenance to change the state of the latching relay (MCX) to the "operate state." The control switch was in the LOCKOUT position. When the supply breaker was closed, the fan started since the MCX contact in series with the MC coil was closed. — LER 88-017; Review 0-OI-65 P&L for Relay Information

(1) Be aware that the pull-to-lock logic will not always prevent equipment start. MCX relay has two states:

RESET (blue PB out)
OR
ACTUATED (PB depressed),

(2) If not RESET SGT may start when power is restored.

4. Emergency Operation

a. CAD System operation after a LOCA

BFN Unit 0	Standby Gas Treatment System	0-OI-65 Rev. 0053 Page 10 of 41

3.0 **PRECAUTIONS AND LIMITATIONS (continued)**

T. [NRC/C] If any relays are ACTUATED, Site Engineering SHALL be contacted prior to energizing the circuit. The pull-to-lock logic will **NOT** inhibit the SGT Blower from starting when the SGT Blower breaker is racked in and the MCX relay is actuated (blue contact position indicator retracted). [NRC LER 88-017]

U. Start relays, MAX and MBX for Standby Gas Treatment trains "A" and "B" respectively, are of a different type than the MCX for train "C". However, the same problem exist for these relays as does for the MCX relay. If the contacts are closed (pulled up) prior to the breaker being closed, the standby gas treatment train will start when the breaker is closed. FAILURE to have the contacts **open** (dropped down position) will result in the associated Standby Gas Treatment train starting when the breaker is closed.

V. The following signals on any unit will start all three SGT trains when the respective control switches are in AUTO:

1. High drywell pressure (2.45 psig).
2. Low Reactor Water Level (LEVEL 3).
3. High Rx Zone Ventilation Radiation (72 MR/hr).
4. High Refuel Zone Ventilation Radiation (72 MR/hr).
5. One out of two taken twice trip logic for Reactor Zone Ventilation Radiation downscale.
6. One out of two taken twice trip logic for Refuel Zone Ventilation Radiation downscale.

W. When the control room handswitch for an SGT Fan is in PULL-TO-LOCK, the fan may still be operated locally.

X. The following system valves fail open upon a loss of power (all other system valves fail closed):

1. SGT FILTER BANK C OUTLET DAMPER, 0-DMP-065-0067
2. SGT FAN A INLET DAMPER, 0-DMP-065-0017
3. SGT FAN B INLET DAMPER, 0-DMP-065-0039

Y. The SGT FILTER BANK A & B BYPASS DAMPER, 0-DMP-065-0022, is normally fed power from 480V Diesel Aux Bd A. Power to 0-DMP-065-0022 is automatically transferred to 480V Diesel Aux Bd B upon a loss of power from Aux Bd A.

Examination Outline Cross-reference:	Level	RO	SRO
290002 Reactor Vessel Internals	Tier #	2	
A2.01 (10CFR 55.41.5)	Group #	2	
Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: • LOCA	K/A #	290002A2.01	
	Importance Rating	3.7	

Proposed Question: **# 64**

Which ONE of the following completes the statement?

Jet Pumps are designed such that following a DBA LOCA, a re-floodable core volume **NO** lower than __**(1)**__ is assured. Following a DBA LOCA with **ALL** ECCS available, Severe Accident Management Guidelines __**(2)**__ be required to be entered.

A. **(1)** (-) 180 inches
(2) will

B. **(1)** (-) 180 inches
(2) will **NOT**

C. **(1)** (-) 215 inches
(2) will

D. **(1)** (-) 215 inches
(2) will **NOT**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect – Plausible in that (-) 180 inches is a recognizable value associated with Low Reactor Water Level accident conditions and criteria for adequate core cooling. This is the minimum zero injection water level limit. Part 2 incorrect – Plausible in that a severe accident has occurred in a DBA LOCA and candidate may have the misconception that under these conditions SAMG entry is required regardless of whether adequate core cooling is met or not.

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

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

D **CORRECT**: Part 1 correct - Jet Pumps are designed such that following a DBA LOCA a re-floodable core volume **NO** lower than two thirds core height is assured. Two thirds core height corresponds to (-) 215 inches. Part 2 correct - ECCS is designed such that adequate core cooling will be met following a LOCA, assuming the worst case single active component failure in the ECCS. With all ECCS available, adequate core cooling is assure. Therefore, SAMGs are not required to be entered.

KA Justification:

The KA is met because the question tests the candidates' ability to predict the impacts of a LOCA on the Reactor Vessel Internals and based on those predictions, use procedures to control or mitigate the consequences of those abnormal conditions or operations in that the candidate must utilize the applicable sections and steps of EOI-1, "RPV Control," and EOI-C1, "Alternate Level Control" to determine that these procedures will not be exited for the SAMGs based on current plant conditions and predicted impact.

Question Cognitive Level:

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): OPL171.212, Rev. 4 (Attach if not previously provided)

OPL171.201 Rev. 7 / OPL171.002 Rev. 9

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.212 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

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

b. Flows out:

(1) Steam flow:
14.15 x E+6 lbs/hr

(2) Flow to Cleanup System:
0.13 x E+6 lbs/hr

(3) Total flow out:
14.28 x E+6 lbs/hr

c. Flows in:

(1) Feedwater flow:
14.10 x E+6 lbs/hr

(2) Control Rod Drive System:
0.05 x E+6 lbs/hr

(3) RWCU System return water flow:
0.13 x E+6 lbs/hr

(4) Total flow in:
14.28 x E+6 lbs/hr

2. Core Floodability — TP-26

a. Applicability — Obj. V.B.7, Obj. V.C.7

(1) Applicable to a loss of coolant accident.

(2) The worst case loss of coolant accident is a 28" recirculation suction line break with the reactor at full power, steady state.

(3) In this case the core will become completely uncovered.

(4) This will be discussed in detail during the Emergency Core Cooling System presentation.

b. Design features

(1) The emergency core cooling systems and the reactor vessel design must be compatible so that following a loss of coolant accident the core can be adequately cooled.

(2) There are several systems that will provide water to the reactor following a loss of coolant accident.

Procedure Use:

EOI's

(3) One of these systems is the low Pressure Coolant Injection System (LPCI) mode of RHR.

(4) For simplification only the LPCI system will be discussed here.

(5) The LPCI system injects water into the reactor vessel using the RHR pumps via both recirculation inlet lines and down the 20 jet pumps.

(6) This flooding water then increases the water level in the reactor starting at the bottom of the vessel and working its way up into the core. Calculations in the FSAR show that leakage through slip fit (and unit 1 bolted accesses) into the downcomer will not exceed 964 gpm (unit 1) or 807 gpm (units 2 & 3) while level is being restored.

UNIT DIFFERENCE

(7) When the water level reaches the top of the jet pump mixing sections, water will begin spilling out into the downcomer area and out of the vessel through the broken recirculation line.

(8) This elevation where water begins to spill out of the jet pumps is 2/3 of the height of the active fuel.

(9) Calculations show that if flooding of the reactor vessel is accomplished within a specified time frame & the level maintained at the 2/3 point, the core will be adequately cooled indefinitely and the integrity of the fuel cladding maintained.

NOTE: These Calculations are based on FSAR and not BFNP EOI Program Manual.

(a) Lower 2/3 of the core cooled because it is flooded with water.

(b) Upper 1/3 of the core

(i) Vigorous boiling in the lower 2/3 of the core provides a mixture of steam and water which, upon flowing upward, cools the upper 1/3 of the core.

Fundamental:

What are the 3 types of heat transfer and which is prevalent during this condition?

1. Radiation
2. Conduction
3. Convection

(ii) Long term (after fuel decay heat has lowered) there will be less boiling in the lower 2/3 of the core to provide the flow of steam and water to cool the upper 1/3 of the core.

(iii) Fuel clad temperature would raise with time. However, it would still remain acceptable under these conditions.

(10) Under the assumed conditions, water would have to be continually made up to the vessel to accommodate for the following cooling losses:

(a) Boil off AND,

(b) The aforementioned leakages.

A. Key Words and Terms — Obj. V.B.10

1. Section I-C to the Program Manual (see Attachment 1) provides definitions for terms, phrases, and acronyms used in the EOIs. The following terms/phrases are to be highlighted in this lesson:

   a. Adequate Core Cooling — Obj. V.B.10.a

   Any of the following conditions (1-4):

   (1) Submergence: Reactor water level is verified at or above TAF, and based on present and past trends and plant conditions, is expected to remain above TAF.

   (2) Spray Cooling: During the execution of C1, the following conditions are met:

   - The reactor can be determined to be shutdown without boron (note 1)

     AND

   - One Core Spray subsystem is injecting at or above 6250 gpm. — One spray ring for design pattern

     AND

   - RPV water level can be determined to be above -215 inches (2/3 core height)

   (3) Steam Cooling With Injection:

   - During execution of C5 and C1, RPV water level can be maintained above the lower water level band allowed by the procedure, [Minimum Steam Cooling Water Level (MSCWL) -180 inches]. — This will maintain PCT < 1500 °F

     OR

   - Reactor pressure can be maintained above MARFP following reactor depressurization.

X. Lesson Body

A. EOI Transition into SAMG - Loss of Coolable Geometry

1. The SAMGs are entered, then the core geometry is assumed to be changed and NOT coolable. The EOI strategies are employed for accidents inside BFN design basis. When accidents progress to a point where BFN design basis is exceeded, SAMG entry will be required.

2. SAMG entry is required, i.e., core geometry assumed to be lost. These are the specific EOI contingency points: Obj. V.B.1

   a. In EOI Step C1-25, ALTERNATE LEVEL CONTROL, when primary containment flooding is required and either one Core Spray loop is not injecting at >6250 gpm, or RPV water level cannot be determined to be above -215 inches.
   b. In EOI Step C4-14, RPV FLOODING, when the reactor is NOT assured of remaining sub critical under all conditions and the RPV pressure due to injection will not remain above MARFP with at least four MSRVs open.
   c. In EOI Step C4-24 and 25, RPV FLOODING, when the reactor will remain sub critical under all conditions and the RPV pressure due to injection will not remain 70 psig over suppression chamber pressure with at least four MSRVs open.
   d. In EOI C5-26, LEVEL/POWER CONTROL, with control rods out and unable to restore and maintain RPV water level above -180 inches.

3. At each of these specific points, we cannot assume a coolable geometry exists and SAMG entry is required.

4. Once the SAMGs are entered, the EOI flowcharts no longer apply because the configuration of the core may no longer be amenable to adequate cooling. All EOI flowcharts will be exited and will not be referred to again. Any subsequent EOI entry condition which is received will NOT result in EOI entry. Obj. V.B.2.c

DISTRACTOR PLAUSIBILITY SUPPORT
OPL171.201
Revision 7
Page 27 of 8

B. Key Words and Terms — Obj. V.B.10

1. Section I-C to the Program Manual (see Attachment 1) provides definitions for terms, phrases, and acronyms used in the EOIs. The following terms/phrases are to be highlighted in this lesson:

   a. Adequate Core Cooling — Obj. V.B.10.a

      Any of the following conditions (1-4):

      (1) Submergence: Reactor water level is verified at or above TAF, and based on present and past trends and plant conditions, is expected to remain above TAF.

      (2) Spray Cooling: During the execution of C1, the following conditions are met:

      - The reactor can be determined to be shutdown without boron (note 1)

        AND

      - One Core Spray subsystem is injecting at or above 6250 gpm. — One spray ring for design pattern

        AND

      - RPV water level can be determined to be above -215 inches (2/3 core height)

      (3) Steam Cooling With Injection:

      - During execution of C5 and C1, RPV water level can be maintained above the lower water level band allowed by the procedure, [Minimum Steam Cooling Water Level (MSCWL) - 180 inches]. — This will maintain PCT < 1500 °F

        OR

      - Reactor pressure can be maintained above MARFP following reactor depressurization.

Examination Outline Cross-reference:	Level	RO	SRO
290003 Control Room HVAC	Tier #	2	------
K6.01 (10CFR 55.41.7)	Group #	2	------
Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC :	K/A #	290003K6.01	
• Electrical power	Importance Rating	2.7	------

Proposed Question: **# 65**

Which ONE of the following combinations of electrical board losses would result in **BOTH** Control Room Emergency Ventilation Fans being de-energized? (Assume normal alignment)

A. 480V Shutdown Board 1B; 4kV Shutdown Board 3EC

B. 480V Shutdown Board 1A; 480V Shutdown Board 2B

C. 480V Shutdown Board 3B; 4kV Shutdown Board A

D. 4kV Shutdown Board B; 4kV Shutdown Board 3EA

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: These do not meet the combination of power supplies for the CREV trains.

B INCORRECT: These do not meet the combination of power supplies for the CREV trains.

C **CORRECT**: Correct since the power supplies are 480 VAC RMOV Board 3B for fan B and 480 VAC RMOV Board 1A for fan A which is supplied by 4KV Shutdown Board A

D INCORRECT: These do not meet the combination of power supplies for the CREV trains.

KA Justification:

The KA is met because the question tests whether the candidate has knowledge of the effect that a loss or malfunction of Electrical power will have on Control Room Emergency Ventilation.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.067, Rev. 16 (Attach if not previously provided)

0-OI-31 Att 3 Rev. 133

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.067 V.B.2 (As available)

Question Source:

Bank #	BFN 2004-301 #42	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam BFN 2004-301

(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	Attachment 3 Electrical Lineup Checklist	0-OI-31/ATT-3 Rev. 0133 Page 6 of 22

4.0 ATTACHMENT DATA

Performed On: ____________

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
Control Bay - 4160V Shutdown Board B - El 593'			
18	0-BKR-031-2100 4KV SUPPLY FOR 1&2 CONTROL BAY CHILLER A	CLOSED	____ ____
Control Bay - 4160V Shutdown Board D - El 593'			
12	0-BKR-031-2200 4KV SUPPLY FOR 1&2 CONTROL BAY CHILLER B	CLOSED	____ ____
Control Bay - 480V Reactor MOV Board 1A - El 621'			
1A	SHUTDOWN BOARD ROOMS EXHAUST FAN 1A	ON	____ ____
9A	1-BKR-031-2300 ELECT BD RM AHU 1A	ON	____ ____
R9A	250V SHUTDOWN BD BATTERY ROOM EXHAUST FAN 1A	ON	____ ____
R9B	250V SHUTDOWN BD BATTERY ROOM SUPPLY FAN 1A	ON	____ ____
R9D1	250V SHUTDOWN BD BATTERY ROOM DUCT HEATER	ON	____ ____
14D	AUXILIARY PRESSURIZATION FAN A	OFF(1)	____ ____
14C	0-BKR-31-7214 CREVS FILTRATION UNIT A	ON	____ ____

(1) Leads are lifted at breaker per DCN W17527. Fan is inoperable.

BFN Unit 0	Attachment 3 Electrical Lineup Checklist	0-OI-31/ATT-3 Rev. 0133 Page 10 of 22

4.0 ATTACHMENT DATA (continued)

Performed On: ________________

Panel/Breaker Number	Component Description	Required Position	Initials 1st/IV
	Control Bay - 480V Reactor MOV Board 3B - El 593'		
10B	3-BKR-031-7206 ELEC BD RM ACU 3B	ON	____ ____
13C2	0-BKR-031-7213 CREVS FILTRATION UNIT B	ON	____ ____
17A	3-BKR-031-0139 UNIT 3 CONTROL BAY SUPPLY FAN 3B	ON	____ ____
R9A	3-BKR-031-0651 SDBR CHILLER 3A-2	ON	____ ____
R9B1	3-BKR-031-0667 SDBR CHILLER 3B-2	ON	____ ____
R9B2	3-BKR-031-0608 DIVISION II DUCT HEATERS SHUTDOWN BOARD ROOMS UNIT 3	ON	____ ____
R9C	3-BKR-031-0645 CHILLED WATER CIRC PUMP 3A-2 SHUTDOWN BOARD ROOMS UNIT 3	ON	____ ____
R9D	3-BKR-031-0661 CHILLED WATER CIRC PUMP 3B-2 SHUTDOWN BOARD ROOMS UNIT 3	ON	____ ____
R9E	3-BKR-031-0611 AIR HANDLING UNIT 3A-2 SHUTDOWN BOARD ROOMS UNIT 3	ON	____ ____
R9F	3-BKR-031-0612 AIR HANDLING UNIT 3B-2 SHUTDOWN BOARD ROOMS UNIT 3	ON	____ ____

OPL171.067
Revision 16
Page 5 of 6

1. Control Room Emergency Ventilation (CREV) is designed to supply and process the outdoor air needed for pressurization during isolated conditions. There are 2 CREV units rated at 3000 cfm each. A CREV unit consists of Motor-driven fan, (power supply is from 480V RMOV Bd 1A for CREV Fan A; RMOV Bd 3B for CREV Fan B), HEPA filter (common), charcoal filter assemblies located in the CREVS Equipment Room, charcoal heater, and inlet isolation damper and a backflow check outlet damper. They are designed to maintain a positive pressurization to 1/8" w.g. minimum to the control room.

   Tech. Spec. 3.7.3
   Obj.V.B.2/ V.C.6 /V.C.7
   (Old CREV Units abandoned in place as Auxiliary Pressurization Systems)
   TP-4
   2-47E2865-4

   a. A CREV may be started manually from control room Panel 2-9-22 if local control switch is in AUTO position via a 3 position, spring-return to center switch. (STOP-AUTO-START). Actuates only the CREVS unit & associated damper, not the isolation dampers.

      Red indicating lights on panel 3-9-21 to provide indication of CREV Fan A and/or B running on Unit 3. Annunciators are on panel 9-6 for all units.

   b. There is also a 2 position maintained contact, one per train, AUTO-INITIATE/ TEST switch which is used to perform system level actions for that train (primarily testing). It provides the same response as auto start.

   c. Local start at local control station in Relay Room is done using a 2 position maintained, one per train, AUTO-TEST switch. Isolation dampers do not operate automatically if started from local panel.

   d. Automatic start signals are:

      Obj. V.B.1/V.B.2

      (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.

         Obj. V.C.1
         Obj. V.C.17

         T. S. 3.3.7.1

      (2) Reactor zone ventilation systems radiation high $\geq$72 MR/hr

Browns Ferry Nuclear Plant 2004-301
SRO Inital Exam

42. 288000K6.01 001/T2G2//VENTILATION/MEM 2 7/2 7/B/BF04301/R/TCK

Which ONE of the following Combinations of electrical board losses would result in both CREV units being inoperable? (Assume normal alignment and no board transfers)

A. 480V Shutdown Board 1B; 4kV Shutdown Board 3EC

B. 480V Shutdown Board 1A; 480V Shutdown Board 28

C.✓ 480V Shutdown Board 3B; 4kV Shutdown Board A

5. 4kV Shutdown Board B; 4kV Shutdown Beard 3EA

K/A 288000 K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the PLANT VENTILATION SYSTEMS: A.C. electrical. (2.7/2.7)

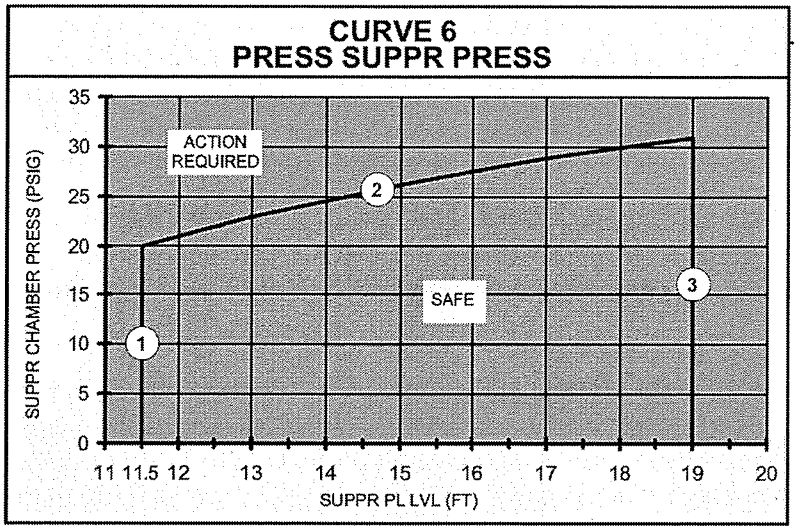
References: OPL171.067, Rev.11, Pg 28 of 60
Learning Objective #B2

A, B, and D. Incorrect since these do not meet the combination of power supplies for the CREV trains.

C. Correct since the power supplies are 480 VAC RMOV Board 3B for fan B and 480 VAC RMOV Board 1A for fan A which is supplied by 4KV Shutdown Board A.

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.25 (10CFR 55.41.10)	Tier #	3	------
Ability to interpret reference materials, such as graphs, curves, tables, etc.	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.

KA Justification:

The KA is met because the question tests the candidate's ability to interpret Pressure Suppression Pressure Curve bounding limitations on Suppression Chamber Pressure versus Suppression Pool Level.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

RO Level Justification: Tests

Technical Reference(s): OPL171.201, Rev. 7 (Attach if not previously provided)

EOI Program Manual Sect. 2-VI-H, Rev. 10

0-TI-394, Rev. 4

Proposed references to be provided to applicants during examination: Embedded EOI Curve 6 - PSP

Learning Objective: OPL171.201 V.B.12 (As available)

Question Source: Bank # BFN 1006 #66

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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.

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.

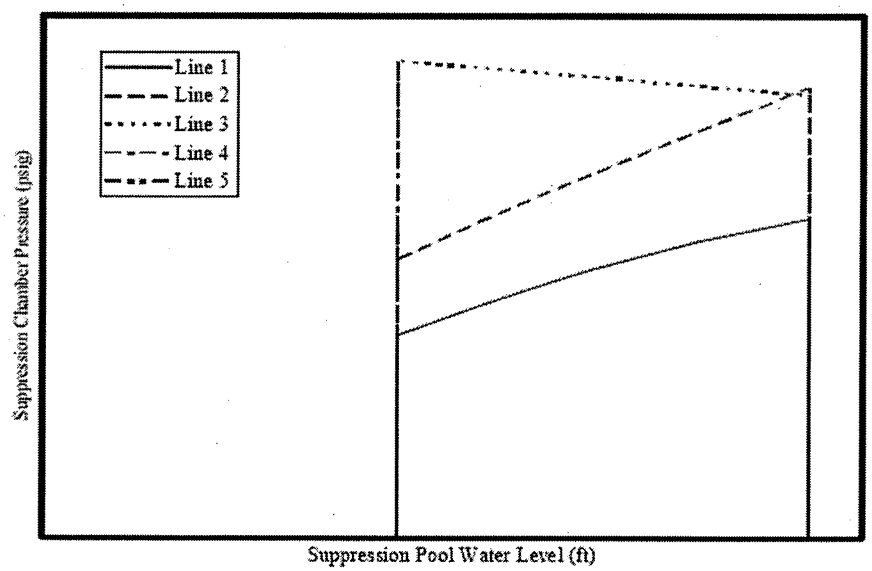
Line 5 is the suppression pool water level corresponding to the Maximum Pressure Suppression Primary Containment Water Level. Above this elevation, the pressure suppression function of the containment cannot be assured. The PSP is therefore vertical at this elevation.

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 suppression pool water levels considered.

A personal computer running Microsoft Excel™ is used to compute the results of this calculation as one might use a hand calculator. Three or four significant figures are sufficient to obtain reasonable EPG/SAG Appendix C calculation results. The personal computer carries more significant figures and hence is more accurate. Since the results given in this calculation are based on the precision resident in the computer, any hand calculations using the as-displayed precision of the data shown herein may yield results which are less precise.

Tables 1 contains a list of abbreviations employed for parameter notation. Table 2 identifies the notation used for the properties of water.

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

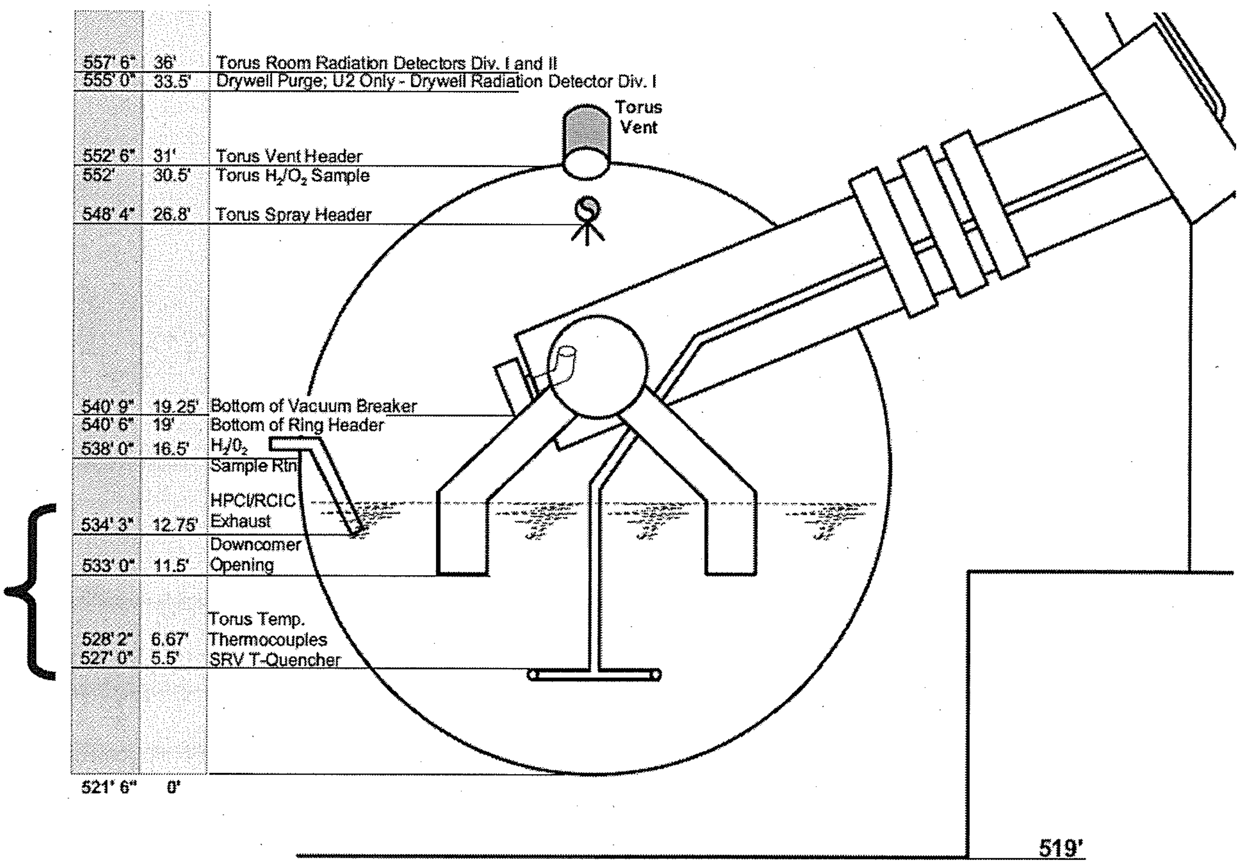
Technical Instruction

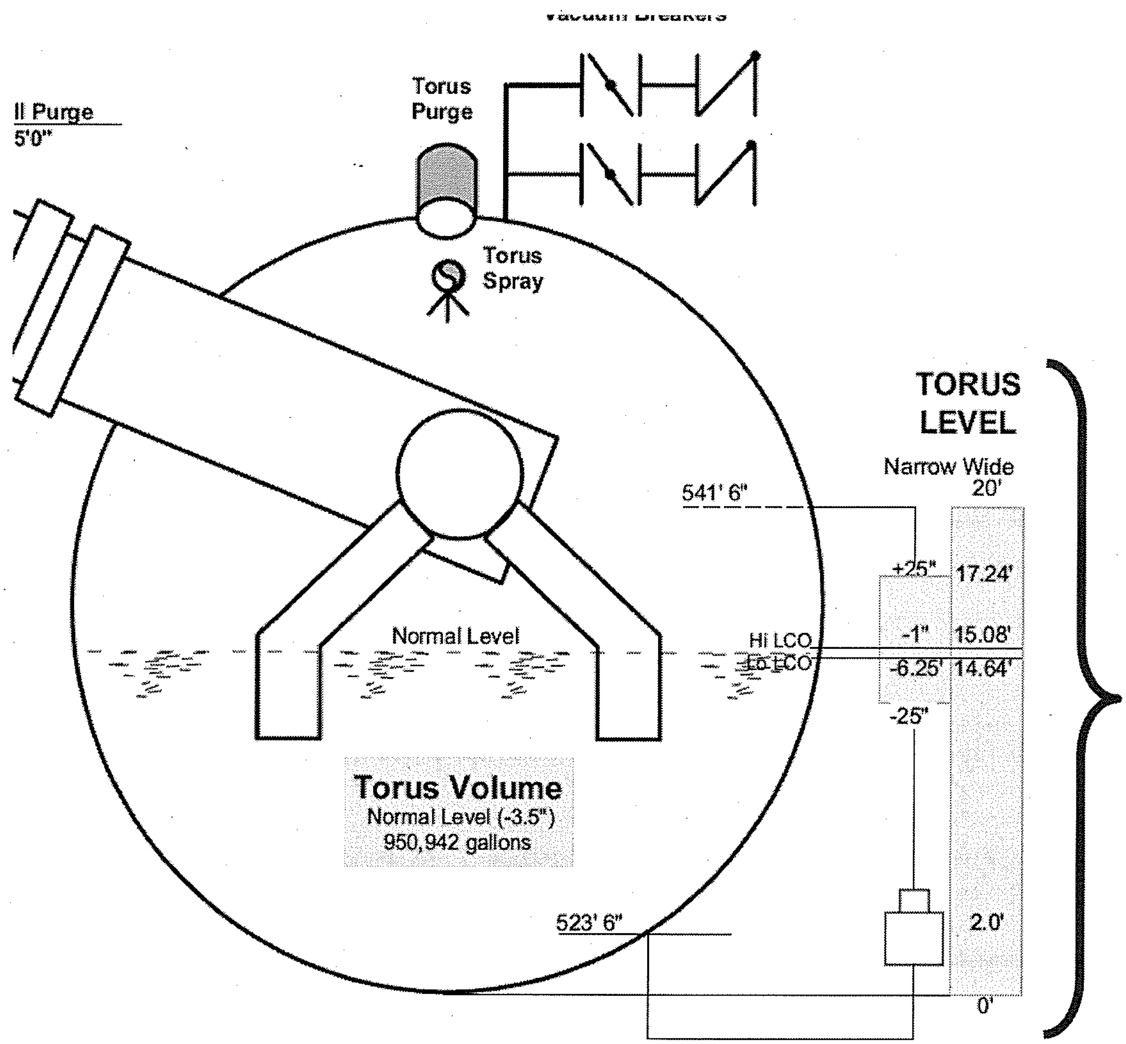
0-TI-394

Technical Support for Severe Accident Management Guidelines (SAMG)

Revision 0004

ILLUSTRATION 1 EXCERPT





Vacuum Breakers
Torus Purge
ll Purge
5'0"
Torus Spray
TORUS LEVEL
Narrow Wide
20'
541' 6"
+25"
17.24'
Hi LCO
Lo LCO
-1"
15.08'
-6.25'
14.64'
-25"
Normal Level
Torus Volume
Normal Level (-3.5")
950,942 gallons
523' 6"
2.0'
0'

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.27 (CFR: 41.7)	Tier #	3	------
Knowledge of system purpose and/or function.	Group #	------	------
	K/A #	G2.1.27	
	Importance Rating	3.9	------

Proposed Question: **# 67**

Which ONE of the following is a Design Basis of HPCI?

A. Maintain sufficient reactor water inventory so the fuel won't overheat when a reactor isolation **AND** loss of feedwater occurs.

B. Make up water to the vessel in the event of a loss of coolant situation that does **NOT** result in rapid vessel depressurization.

C. Assures that the reactor core is adequately cooled to limit fuel clad temperature to < 1800 °F in the event of a large break in the reactor coolant system.

D. Assures that the reactor core is adequately cooled to limit primary containment pressure in the event of a small break in the reactor coolant system.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Maintains reactor water inventory so the fuel won't overheat is true, but this statement is the design basis for RCIC. Candidate may confuse the basis for HPCI and RCIC because they are similar in many respects. HPCI can also supply water to the reactor when a MSIV isolation and a loss of feedwater occur.

B **CORRECT:** Provides Adequate Core Cooling (ACC) for all break sizes that do NOT result in rapid depressurization of the reactor vessel. Correct design basis statement.

C INCORRECT: ECCS general design criteria is to limit fuel clad temperatures < 2200 °F. 1800 °F is EOI MZIRWL fuel clad temperature. Candidate may confuse EOI zero injection water level fuel clad temperature with ECCS design value.

D INCORRECT: HPCI design basis isn't about limiting primary containment pressure. Candidate may confuse primary containment design criteria with HPCI.

KA Justification:

The question meets the K/A by asking the design basis of HPCI.

Question Cognitive Level:

This is a low cognitive question. It asks for recall of the basis of the system or discrete bits of information.

Technical Reference(s): OPL171.042 Rev 20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

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

Question Source:		
Bank #	Quad Cities 98	
Modified Bank #		(Note changes or attach parent)
New		

Question History: Last NRC Exam Quad Cities 1998

(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. General Description

The High Pressure Coolant Injection System (HPCI) consists of a steam turbine-driven system driving a constant-flow pump assembly to inject either Condensate Storage Tank (CST) water or Suppression Pool (Torus) water into the reactor under emergency conditions at the rate of at least 5000 gpm over an 1174 -150 psi reactor pressure range. — Obj. V.B.1, Obj. V.C.1

1. System Design Basis

   a. To provide adequate core cooling for all break sizes which do not result in rapid depressurization of the reactor vessel — Obj. V.E.1, SER 3-05

   b. To function independent of off-site power sources and diesel generators — SER 3-05

2. Components — Obj. V.D.1, Obj. V.E.2

   a. Turbine

   b. Main and booster pumps

   c. Turbine auxiliaries

3. Flow Path — TP-1, Obj. V.C.1, Obj. V.B.1, Obj. V.E.10

   a. One 100% system

   b. Steam path

      (1) From B Main Steam line upstream of the flow restrictor

      (2) Through isolation valves

      (3) Through stop valve and control valves

DISTRACTOR PLAUSIBILITY SUPPORT

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

DISTRACTOR PLAUSIBILITY SUPPORT

- During C4 execution it has been verified that based on present and past trends and plant conditions, either Minimum Reactor Flooding Pressure (all rods in) or Minimum Alternate Reactor Flooding Pressure (all rods not in) can be maintained.

(1) Steam Cooling Without Injection: During C1 execution RPV water level has not yet lowered to [Minimum Zero Injection Water Level (MZIWL) -200] and reactor pressure is either at the lifting point of the MSRVs or is stabilized and not rising. — This will maintain PCT < 1800 °F

b. Augment — Obj. V.B.10.b

(1) To supplement the systems that are currently in use.

(2) "**Augment** RPV water level with the following systems:"

c. Verify — Obj. V.B.10.c

(1) To observe an expected characteristic or condition and, if not as expected, to take action to place it in the expected condition. Usually applied for response to automatic actions, but is not limited to only those actions.

(2) "**Verify** recirc flow runback to minimum."

d. Injection Subsystem — Obj. V.B.10.d

Any of the following independent flow paths capable of delivering coolant to the RPV:

(1) Condensate System, with at least one Condensate pump and one Condensate Booster pump capable of delivering Coolant to the RPV.

(2) LPCI System I, or II, with at least one operable pump capable of delivering Coolant to the RPV is one Injection Subsystem.

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.28 (10CFR 55.41.7)	Tier #	3	------
Knowledge of the purpose and function of major system components and controls.	Group #	------	------
	K/A #	G2.1.28	
	Importance Rating	4.1	------

Proposed Question: **# 68**

Which ONE of the following defines the purpose of the Rod Worth Minimizer (RWM) in accordance with Technical Specifications?

A. Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when Reactor Power is < 10%.

B. Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when Reactor Power is > 27%.

C. Ensures that the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when Reactor Power is < 10%.

D. Ensures that the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when Reactor Power is > 27%.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: The purpose of the RWM system is to limit control rod worth such that the fuel enthalpy limit of 280 cal/gm will not be exceeded during a Control Rod Drop Accident (CRDA). TS Table 3.3.2.1-1 requires the RWM to be operable in modes 1 and 2 with thermal power <10% RTP.

B INCORRECT: 1st part correct. 2nd part incorrect - Plausible in that ≥ 27% is the TS requirement for the RBM, and the candidate may confuse the requirements between the RBM and RWM.

C INCORRECT: 1st part is incorrect. Plausible because the RBM does provide rod blocks to prevent MCPR from being exceeded due to additional rod withdrawal. 2nd part is correct.

D INCORRECT: 1st part is incorrect. Plausible because the RBM does provide rod blocks to prevent MCPR from being exceeded. 2nd part is incorrect. Plausible because ≥ 27% is the TS requirement for the RBM, and the candidate may confuse the requirements between the RBM and RWM.

KA Justification:

The KA is met because the question tests knowledge of the purpose of the Rod Worth Minimizer.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.024 Rev. 14 (Attach if not previously provided)

TS 3.1-20 Amm 253

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.024 V.B.1 / 3 (As available)

Question Source: Bank # Hatch 09 #66

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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1	-------------NOTE------------ Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." -------------------------------	
		Move associated control rod(s) to correct position.	8 hours
	OR		
	A.2	Declare associated control rod(s) inoperable.	8 hours

(continued)

INSTRUCTOR NOTES

X. Lesson Body

A. Banked Position Withdrawal Sequence (BPWS) — TP-1

1. Basis - The BPWS is designed to ensure under all operating conditions that control rod worths are limited such that a control rod drop accident would result in peak fuel energy deposition of less than 280 cal/gram (safety basis). Restrictions on control rod patterns while at low power are required during both startup and shutdown. — Obj. V.B.1; SER-03-05

2. Review Tech Spec and Bases Section 3.1.6.1 - Note: This is required for both ROs and SROs.

3. Fuels group in Chattanooga designs the Control Rod Withdrawal Sequence such that no single control rod notch will cause less than a 60 second reactor period. — QUESTIONING ATTITUDE

B. Sequence Types — Obj. V.B.2

1. Sequence A - This sequence results in the center control rod (30-31) being fully inserted when 50 percent control rod density (black and white pattern) is reached. A1 versus A2 sequences differ only in which control rod groups beyond the black and white pattern will ultimately be deep and which will be shallow or fully withdrawn. — SR-3.1.3.5(A)

2. Sequence B - This sequence results in the center control rod being fully withdrawn when 50 percent control rod density is reached. B1 versus B2 sequences differ as described above for Sequence A.

INSTRUCTOR NOTES

5. When all the control rods in Group 1 have been withdrawn to the Group 1 withdraw limit, the operator proceeds to Group 2.

6. After Group 1 control rods have been withdrawn, a given control rod or set of control rods may comprise more than one group.

Example: TP-2

Gps 2 - 6 same rods
Gps 7 -12 same rods
Gps 13 - 17 same rods

7. In this case the withdraw limit for a control rod in a given group will be the same as the insert limit for the next higher group in which the control rod appears.

D. Reduced Notch Worth Procedure (RNWP) — TP-1

1. Basis - The RNWP is a conservative extension of the BPWS, and is designed to further lower notch worth in order to reduce the chance of a scram on short period during startup. Since this is not a concern during shutdown, RNWP procedures need not be utilized except for startup pull sheets.

NOTE: Single notch withdrawals should never result in a reactor period faster than 60 sec.

2. A high notch worth control rod is designated by an asterisk on the control rod withdrawal sequence sheet. The designated high worth control rod must be withdrawn a single notch at a time within the indicated range of high notch worth.

2-MINUTE SITUATIONAL AWARENESS

E. ** Rod Worth Minimizer Purpose and Terms — SOER 84-2 Recommendation 7d

1. The RWM system design is based on Banked Position Withdrawal (BPWS) system design requirements.

2. The RWM, in conjunction with the control rod velocity limiter, limits the amount of fuel damage that could occur during a control rod drop accident.

OBJ. V.B.3
OBJ. V.C.1

a. The RWM acts to enforce of the programmed control rod patterns and generates a rod block if significant deviation from the programmed sequence is detected.

TP-3

DISTRACTOR PLAUSIBILITY SUPPORT

Table 3.3.2.1-1 (page 1 of 1)

Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range - Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range - Upscale	(f), (g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. Inop	(g),(h)	2	SR 3.3.2.1.1	NA
e. Downscale	(g),(h)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(i)
2. Rod Worth Minimizer	$1^{(c)}, 2^{(c)}$	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch - Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 27% and ≤ 62% RTP and MCPR less than the value specified in the COLR.

(b) THERMAL POWER > 62% and ≤ 82% RTP and MCPR less than the value specified in the COLR.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

(f) THERMAL POWER > 82% and < 90% RTP and MCPR less than the value specified in the COLR.

(g) THERMAL POWER ≥ 90% RTP and MCPR less than the value specified in the COLR.

(h) THERMAL POWER ≥ 27% and < 90% RTP and MCPR less than the value specified in the COLR.

(i) Greater than or equal to the Allowable Value specified in the COLR.

DISTRACTOR PLAUSIBILITY SUPPORT

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch - Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn.

(continued)

HLT 4 NRC Exam

66. G2.1.27 001

Which ONE of the following defines the purpose of the Rod Worth Minimizer (RWM) IAW Technical Specifications.

A. Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when power is $\geq$ 29%.

B.✓ Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when reactor power is < 10%.

C. Ensures that the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when power is $\geq$ 29%.

D. Ensures that the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when reactor power is < 10%.

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.2 (10CFR 55.41.6) Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	Tier #	3	-------
	Group #	-------	-------
	K/A #	G2.2.2	
	Importance Rating	4.6	-------

Proposed Question: **# 69**

Unit 1 Plant Startup is in progress.

Which ONE of the following identifies the criteria specified in 1-GOI-100-1A,"Unit Startup," for Control Rod single notch withdrawal?

Control Rod withdrawal is limited to single notch when the __**(1)**__ SRM count rate doubling is reached **AND** must continue until __**(2)**__.

A. **(1)** fourth
(2) the Reactor is Critical

B. **(1)** fifth
(2) the Reactor is Critical

C. **(1)** fourth
(2) Reactor Power is in the heating range

D. **(1)** fifth
(2) Reactor Power is in the heating range

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 Calculations have shown that when the initial SRM count rate has doubled 5 times that the reactor is very near criticality. Part 2 incorrect - Plausible in that 1-GOI-100-1A contains several cautions regarding the careful and controlled approach to criticality and the point of criticality is the trigger for several actions in the GOI.

C **CORRECT:** Part 1 correct – In accordance with 1-GOI-100-1A, A review of startup data has revealed that when count rate doubles five times, criticality is imminent. As an added precaution, the fourth count rate doubling has been chosen as a starting point to limit rod withdrawal to single notch movement. Part 2 correct – In accordance with 1-GOI-100-1A, once required, Control rod withdrawal is limited to single-notch withdrawal until Reactor power is in the heating range.

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

KA Justification:

The KA is met because the question tests the candidates' ability to manipulate Control Rod console controls as required based on SRM response to operate the facility between shutdown and designated power levels.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): 1-GOI-100-1A Rev. 23 (Attach if not previously provided)

OPL171.059 Rev. 11

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.059 V.B.3 / 4 (As available)

Question Source: Bank #

Modified Bank # Nine Mile 2 08 #70 (Note changes or attach parent)

New

Question History: Last NRC Exam Nine Mile 2 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:

INSTRUCTOR NOTES

2. Review instruction steps from 5.2 through 5.29.

a. SRM reading are recorded prior to start up to determine the count rate at which single notch withdrawal should begin. Calculations have shown that when the initial SRM count rate has doubled 5 times that the reactor is very near criticality, so when the initial count rate on any SRM has increased by a factor of 16 (four doublings) single notch withdrawal shall begin. ***Criticality should be expected at all times.***
   - Obj. V.B.4
   - Obj. V.C.4
   - Expect the unexpected.

b. IRM downscale functions are bypassed on Range 1, so to verify the downscale function operable the IRMs will have to be ranged to Range 2 or 3.
   - Obj. V.B.3
   - Obj. V.C.3

c. For control rods that are difficult to move from the full in position, increased drive water pressure is allowed by OI-85, should be referred to in this situation.
   - Always return the drive water DP to normal after the rod is moved. This will prevent double notching.

d. Surveillance requirements for RWM are completed prior to withdrawing control rods for the purpose of making the reactor critical.
   - Operations Management Expectation. This is also required by the GOI

e. Control rods shall not be pulled for startup if the Plant Control Air is supplying the Drywell Control Air System.
   - Obj. V.B.4
   - Obj. V.C.4

f. The Unit Operator is responsible for controlling reactivity and should be alert for any conditions that might affect reactivity. Any activity that could affect reactivity should be coordinated with the operator. These activities would include recirc control changes, addition of feedwater, use of nuclear steam. It is vital that good communications are exercised during these evolutions. The operator should be aware that a startup following operation at high power and peak Xenon could result in extremely high notch worth.
   - Obj. V.B.4
   - Obj. V.C.4
   - Conservative Decision Making and Follow Procedures.

g. All activities that can distract the operator and supervisors during the approach to criticality should be avoided. These activities could include shift turnover, surveillance's, and excessive personnel in the control room.
   - Obj. V.B.4
   - Obj. V.C.4

h. Verify moderator temperature is greater than the temperature required by TS 3.4.9-1 Curve 3 within 15 minutes prior to withdrawing control rods to achieve critical.
   - SR 3.4.9.2 Ex/2-SR-3.4.9.1(1) (which is also the heatup monitoring SR).

INSTRUCTOR NOTES

i. Performance of the heautp and cooldown rate monitoring surveillance is required 15 minutes prior to heatup and pressurization. — 2-SR-3.4.9.1(1); Use *"HUR"* on ICS

3. Review instruction steps 5.29 through 5.42 — SRO in CR

a. If a single notch withdrawal results in a reactor period of less than 60 seconds, the last control rod pulled will be reinserted until a period of greater than 60 seconds is obtained, the Reactor Engineer, Reactivity Manager, and SM approval is required to resume rod withdrawal. — Obj. V.B.5.b; Obj. V.C.5.b

b. If a reactor period of less than 30 seconds is observed, control rods shall be inserted until the reactor is subcritical, and obtain the Reactor Engineer, Reactivity Manager, and SM approval to resume rod withdrawal. — Obj. V.B.5.c; Obj. V.C.5.c

c. If a reactor period of less than 5 seconds is observed, the reactor shall be shut down and cannot be restarted until an assessment has been performed. — Obj. V.B.5.d; Obj. V.C.5.d

d. Near end of core life, criticality may occur before five doublings due to a stronger top peak flux and the buildup of plutonium. — Obj. V.B.4; Obj. V.C.4

e. Single notch withdrawal must begin when the SRM count rate has increased by a factor of 16 (four doublings), and may be stopped after reaching the heat range. — Obj. V.B.3; Obj. V.C.3 (e through g)

f. The operator should expect the reactor to go critical at ANY TIME while pulling control rods for startup. — Withdraw CR to maintain ≥100 second period *as indicated on the period meter.*

g. Inadvertent criticality could result from extended operation close to the point of criticality.

h. GE SIL 316 cautions when rod movement is restricted to single notch withdrawal failure to stop at each notch position may result in high notch worth. — Obj. V.B.4; Obj. V.C.4

i. When the reactor is critical and the desired period is obtained, the time, rod group, rod number, rod notch, and Reactor water temperature shall be recorded on data sheet and in the NOMS Narrative Log. — Obj. V.B.6

j. Reactor periods may be calculated by: — Obj. V.B.5.a; Obj. V.C.5.a

(1) multiplying the time for a 10% power rise by 10.5

DISTRACTOR PLAUSIBILITY SUPPORT

5.0 **INSTRUCTION STEPS (continued)**

NOTE

[NER/C] A review of startup data has revealed that when count rate doubles five times, criticality is imminent. As an added precaution, the fourth count rate doubling has been chosen as a starting point to limit rod withdrawal to single notch movement. This requirement along with close monitoring of neutron monitoring instrumentation should assure a slow controlled approach to criticality. Criticality should be expected at all times. [SOER 88-002]

[24.2] **CALCULATE** SRM count rate at which notch withdrawal limitations shall be imposed by multiplying pre-startup count rate, recorded in Step 5.0[24.1], by a factor of 16. **RECORD** results below and at Step 5.0[26]:

CHANNEL A LEVEL ____________ cps

CHANNEL C LEVEL ____________ cps

CHANNEL B LEVEL ____________ cps

CHANNEL D LEVEL ____________ cps

(R) ________ Initials ________ Date ________ Time
1st

(R) ________ Initials ________ Date ________ Time
Reactor Engineer

[24.3] **RECORD** SRM BYPASS, 1-HS-92-7A/S3 joystick position. (N/A if **NOT** bypassed.)

CHANNEL ____________ BYPASSED

(R) ________ Initials ________ Date ________ Time

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0023 Page 81 of 171

5.0 INSTRUCTION STEPS (continued)

CAUTIONS

1) Near end of core life, criticality may occur before five doublings due to a stronger top peak flux and buildup of plutonium.
2) [NER/C] When rod movement is restricted to notch withdrawal, failure to stop at each notch position may result in high notch worth. [GE SIL 316]

NOTE

Once required, Control rod withdrawal is limited to single-notch withdrawal until Reactor power is in the heating range.

[26] **WHEN** SRMs indicate the calculated values recorded below:

CHANNEL A LEVEL ____________ cps

CHANNEL C LEVEL ____________ cps

CHANNEL B LEVEL ____________ cps

CHANNEL D LEVEL ____________ cps

THEN

START single-notch withdrawal of control rods.

(R) ________ Initials ________ Date ________ Time
1st

(R) ________ Initials ________ Date ________ Time
Reactor Engineer

Nine Mile Point Unit 2 Reactor Operator Written Examination
Draft Submittal

RO 70	Tier 3	K/A Number Generic	Statement 2.2.2	IR 4.6	Origin B	Source Question NMP-2 Bank SYSID 22775
LOK F	Grp NA	10 CFR 55.41(b) 10	LOD (1-5)	Reference Documents N2-OP-101A Rev 14		
Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.						

QUESTION 70

Plant startup is in progress with the following:

- Mode switch is in Start/Hot Standby.
- RSCS Group 2 rods are being withdrawn using Continuous Withdrawal
- Reactor is Subcritical.

Which one of the following describes the criteria for using SINGLE NOTCH WITHDRAWAL per N2-OP-101A, Plant Startup?

A. Starting with RSCS Group 4 until criticality is achieved.

B. Starting with RSCS with Group 5 after the Reactor is critical.

C. When TWO SRMs approach 3 count rate doublings in RSCS group 4.

D. When TWO SRMs approach 3 count rate doublings prior to RSCS group 3.

Correct Answer: D When TWO SRMs approach 3 count rate doublings prior to RSCS group 3, SINGLE NOTCH WITHDRAWAL is required per N2-OP-101A, Plant Startup.
Plausible Distractors: A through C are plausible; and ALL of these answer choices invoke Single Rod Withdrawal requirements too late in the startup process to meet the requirements of N2-OP-101A

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.39 (10CFR 55.41.7)	Tier #	3	-------
Knowledge of less than or equal to one hour Technical Specification action statements for systems.	Group #	-------	-------
	K/A #	G2.2.39	
	Importance Rating		-------

Proposed Question: **# 70**

Which ONE of the following completes the statement?

In accordance with Unit 2 Tech Spec 3.4.10, "Reactor Steam Dome Pressure," if the **MAXIMUM** Reactor Steam Dome Pressure of __**(1)**__ is exceeded, it must be restored within a **MAXIMUM** completion time of __**(2)**__ .

A. **(1)** 1050 psig
(2) 15 minutes

B. **(1)** 1050 psig
(2) 1 hour

C. **(1)** 1073 psig
(2) 15 minutes

D. **(1)** 1073 psig
(2) 1 hour

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 correct – In accordance with Unit 2 Tech Spec 3.4.10, the reactor steam dome pressure shall be ≤ 1050 psig. Part 2 correct – In accordance with Unit 2 Tech Spec 3.4.10 Condition A, if Reactor steam dome pressure not within limit, it must be restored with completion time of 15 minutes.

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

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

D INCORRECT: Part 1 is incorrect – Plausible in that this is a recognizable value associated with Reactor Pressure, i.e. EOI entry. Part 2 incorrect – Plausible in that 1 hour is common completion time in Tech Specs.

KA Justification:

The KA is met because the question tests knowledge of less than or equal to one hour Technical Specification action statements for TS 3.4.10, Reactor Steam Dome Pressure.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): U2 TS 3.4-30 Amm 254 (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:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

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

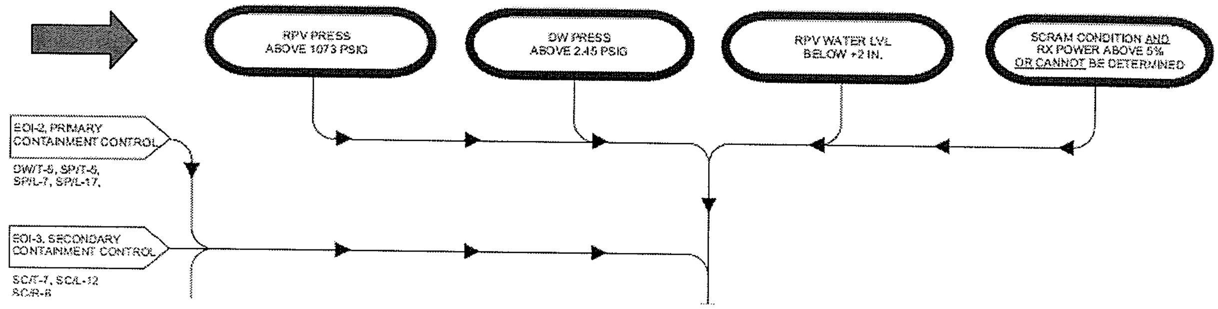
APPLICABILITY: MODES 1 and 2.

ACTIONS

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

PLAUSIBILITY SUPPORT

RPV CONTROL



2-EOI-1	PAGE 1 OF 1
RPV CONTROL	
UNIT 2 BROWNS FERRY NUCLEAR PLANT	
REV: 12	

PLAUSIBILITY SUPPORT

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

--------------------NOTES--------------------

1. Both required RHR shutdown cooling subsystems and recirculation pumps may not be in operation for up to 2 hours per 8 hour period.
2. One required RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

--------------------NOTE--------------------

Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1	Verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.43 (10CFR 55.41.10)	Tier #	3	-------
Knowledge of the process used to track inoperable alarms.	Group #	-------	-------
	K/A #	G2.4.43	
	Importance Rating	3.0	-------

Proposed Question: **# 71**

Which ONE of the following describes the meaning of a BLUE 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 "NOT ABNORMAL" for current plant conditions

C. is associated with ongoing testing **OR** maintenance

D. window is being relocated to a different window location

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** In accordance with “Annunciator Disablement,” OPDP-4, a blue magnetic border indicates that an alarm is out of service.

B INCORRECT: In accordance with “Annunciator System,” 0-OI-55, a hot pink border indicates that an alarm is "NOT ABNORMAL" for current plant conditions.

C INCORRECT: In accordance with “Annunciator Disablement,” OPDP-4, a white magnetic border indicates that an alarm is out of service for TESTING or MAINTENANCE.

D INCORRECT**:** In accordance with “Annunciator System,” 0-OI-55, section 8.5, a yellow border is used to signify that an annunciator window is being relocated.

KA Justification:

The KA is met because the question tests knowledge of “Annunciator Disablement,” OPDP-4, process for tracking inoperable alarms.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

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 # BFN 1006 # 75 (Note changes or attach parent)

New

Question History: Last NRC Exam Browns Ferry 1006

(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:

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 11 of 21

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.

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 0	Annunciator System	0-OI-55 Rev. 0046 Page 21 of 46

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. ☐

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 0	Annunciator System	0-OI-55 Rev. 0044 Page 22 of 45

8.5 Identification of Nuisance Alarms

8.5.1 Short Term Nuisance Alarms

- REFERENCE OPDP-4, Annunciator Disablement

NOTES

1) This section applies to annunciators which are modified (or being modified) to the new annunciator system.
2) All annunciator relocation performed by this procedure is temporary and is performed in accordance with the Work Order Process.
3) The "Yellow" borders for identification of relocated windows communicate to personnel the correct annunciator response procedure for relocated annunciators and are required to meet the following criteria:
   - Yellow in color,
   - The temporary location is delineated on the top border,
   - The correct ARP is referenced for response on the bottom border.
4) The new window annunciator location(s) are updated to reflect the same description as used in the original annunciator window location(s).

DISTRACTOR PLAUSIBILITY SUPPORT

NPG Standard Department Procedure	Annunciator Disablement	OPDP-4 Rev. 0004 Page 11 of 21

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.

BROWNS FERRY 1006

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. window is being relocated to a different window location

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: In accordance with "Annunciator Disablement," OPDP-4, a blue magnetic border indicates that an alarm is out of service.

B **CORRECT**: In accordance with "Annunciator Disablement," OPDP-4, a white magnetic border 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 border indicates that an alarm is "NOT ABNORMAL" for current plant conditions.

D INCORRECT**:** In accordance with "Annunciator System," 0-OI-55, section 8.5, a yellow border is used to signify that an annunciator window is being relocated.

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.13 (10CFR 55.41.12) Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	Tier #	3	------
	Group #	------	------
	K/A #	G2.3.13	
	Importance Rating	3.4	------

Proposed Question: **# 72**

A valve lineup is to be performed on valves with the following conditions:

- Area temperature is 105° F
- Area radiation is 40 mr/hr
- The valves are located 15' off the floor

Independent Verification of this valve lineup is expected to take 0.5 hour.

Which one of the following choices completes the statement below in accordance with SPP-10.3, "Verification Program?"

Based on the above conditions, Independent Verification of this lineup _______.

A. **CANNOT** be exempted

B. may be exempted due to elevation

C. may be exempted due to excessive dose

D. may be exempted due extreme temperature

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT**:** Plausible in that candidate may believe dose levels are not high enough to warrant waiving IV. If the criteria for waiving IV was based on valve located in a High Radiation Area, this would be the correct answer.

B INCORRECT: Plausible in that there are multiple criteria in SPP-10.3 for waiving Independent Verification. However, valve in a hazardous location due to elevation is not

C **CORRECT:** Activities involving significant radiation exposure can be waived in accordance with SPP 10.3. As a guideline, an exposure greater than 10 mrem TEDE to perform verification would be considered excessive. This verification would result in dose of 20 mrem.

D INCORRECT: Plausible in that there are multiple criteria in SPP-10.3 for waiving Independent Verification. However, extreme temperature is not one.

KA Justification:

The KA is met because the question tests knowledge of radiological safety procedural requirements pertaining to licensed operator duties. Specifically, when the requirements for Independent Verification may be waived based on excessive dose.

Question Cognitive Level:

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. Candidate must determine dose to be accumulated during the verification. Then, compare that to SPP-10.3 criteria for waiving IV to determine the correct answer.

Technical Reference(s): SPP-10.3 Rev. 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # Brunswick 08 # 72 (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

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

3.4.1 Clearance Activities

A. Verification is required for all clearance (hold order) activities (except when verification during clearance release is waived as allowed by Section 3.4.3B). IV or CV shall be used as specified in Section 3.4.4 or 3.4.5.

B. If authorized by the Ops Manager, CV may be used in lieu of IV for clearances which were prepared and reviewed prior to the issue date of Revision 2 of this SPP.

3.4.2 **Verification Requirements for Other Activities**

For activities other than clearance activities, IV or CV is required for those systems listed in Appendix A and shall include the following as a minimum:

A. All valves, breakers, and other components in safety-related systems where an inappropriate positioning could adversely affect system/plant operation or containment integrity.

B. All valves, breakers, and other components in fire protection system major flow paths, including fire fighting water supply and storage, carbon dioxide storage systems, fire protection systems, and all components necessary for the system to function and supply extinguishing media to the fire.

C. All valves, breakers, and other components in gaseous and liquid radioactive waste handling and processing systems where an inappropriate positioning could result in radioactive material release to the environment.

3.4.3 **Activities Exempt From Independent and Concurrent Verification Requirements**

The following items may be exempted from verification requirements.

A. Calculations performed by qualified computer software.

B. Activities for which verifications would be required and one or more of the following conditions exist. These exemptions shall NOT be applied during hold order placement.

1. Out-of-service systems/channels/components for which configuration control will **not** be maintained and will be verified to be in the proper configuration during the return to operable status.

2. Activities involving significant radiation exposure. As a guideline, an exposure greater than 10 mrem TEDE to perform verification would be considered excessive.

3. Activities occurring during emergency conditions (imminent danger to plant or personnel) requiring rapid personnel action.

4. Components located within locked/covered/controlled access areas provided access to the area has not occurred since the last documented verification.

For these instances, the decision not to perform a verification is to be documented on the procedure/instruction or work document.

74. A valve lineup is to be performed in an area that has the following conditions:

Area temperature	115° F
Area radiation	40 mr/hr

Independent verification of this valve lineup is expected to take 0.5 hour.

Which one of the following choices completes the statement below in accordance with OPS-NGGC-1303, Independent Verification?

Independent verification of this lineup, based on the above conditions, may be waived because of ___________.

A. both extreme temperature and excessive dose

B.✓ excessive dose only

C. extreme temperature only

D. either extreme temperature or excessive dose

REFERENCE:
NGGC-1303

EXPLANATION:
IV may be waived if the dose will be excessive (as a guideline 10 mrem is excessive) or if personnel safety issues exists (e.g. temperature is above 120° F). IV of this lineup would result in a dose of 20 mrem.

CHOICE "A" Incorrect. Would be allowed to be waived based on dose only.

CHOICE "B" Correct answer.

CHOICE "C" Incorrect. Would be allowed to be waived based on dose not temperature.

CHOICE "D" Incorrect. Would be allowed to be waived based on dose only.

2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
(CFR: 41.12 / 45.9 / 45.10)

IMPORTANCE RO 3.2 SRO 3.7

SOURCE: Bank

LESSON PLAN/OBJECTIVE:
CLS-LP-201C, Obj. 10b. Describe the following regarding OPS-NGGC-1303: Exemptions from Independent Verification.

COG LEVEL: High

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.5 (10CFR 55.41.11/ 12)	Tier #	3	-------
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	Group #	-------	-------
	K/A #	G2.3.5	
	Importance Rating	2.9	-------

Proposed Question: **# 73**

Which ONE of the following completes the statement?

The Wide Range Gaseous Effluent Radiation Monitor System (WRGERMS) consists of __**(1)**__ ranges, **AND** has __**(2)**__ .

A. **(1)** TWO
(2) monitors in **ALL** three Units Control Rooms

B. **(1)** THREE
(2) monitors in **ALL** three Units Control Rooms

C. **(1)** TWO
(2) a monitor in Unit 2 Control Room **ONLY**

D. **(1)** THREE
(2) a monitor in Unit 2 Control Room **ONLY**

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 = incorrect, Normal, Intermediate and high ranges are supplied. Part 2 = incorrect, The only remote monitoring is from Unit 2. Plausible in that Units 1 & 3 receive WRGRM alarms. 1/3-9-3A windows 6 & 13.

B INCORRECT: Part 1 = correct, Normal, Intermediate and high ranges are supplied. Part 2 = incorrect, The only remote monitoring is from Unit 2. Plausible in that Units 1 & 3 receive WRGRM alarms. 1/3-9-3A windows 6 & 13.

C INCORRECT: Part 1 = incorrect, Normal, Intermediate and high ranges are supplied. Part 2 = correct, Units 1 & 3 only receive common alarms. 1/3-9-3A windows 6 & 13. The only remote monitoring is from Unit 2.

D **CORRECT:** Part 1 = correct, Normal, Intermediate and high ranges are supplied. Part 2 = correct, Units 1 & 3 only receive common alarms. 1/3-9-3A windows 6 & 13. The only remote monitoring is from Unit 2.

KA Justification:

The KA is met because the question tests the ability to use the Wide Range Gaseous Effluent Radiation Monitor System which is a fixed radiation monitor.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): OPL171.033 Rev 13 (Attach if not previously provided)

2-OI-90 Rev 79

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.033 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:

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INSTRUCTOR NOTES

(2) The objectives of this stack-gas radiation monitoring system are twofold:

(a) Indicate and record release rates from the stack during normal operation and alarm when limits are reached

Obj. V.B.1,3
Obj. V.C.1,3
Obj.V.D.8

(b) Indicate and record release rates from the stack during accident conditions which could result in gross radiation release

b. Wide Range Gaseous Effluent Radiation Monitor (0-RM-90-306)consists of the following:

(1) 0-RE-90-093 Normal Range noble gas detector

Normal range particulate and lodine filters are abandoned in place

(2) 0-RE-90-98A Intermediate Range noble gas detector

(3) 0-RE-90-98B High range noble gas detector

e. Remote Computer

(1) Located in the unit 2 control room panel 2-9-10

(2) Displays stack release rate, status, stack flow rate, and detector virtual instruments, and annunciators.

Detailed operation is contained in 2-OI-90 Illustration 2

2-OI-90 Illustration 2

g. The WRGERM System changes state and resets at the following values:

(1) Normal to Mid Range = 1.00E-02 (.01) uCi/sec

(2) Mid to High Range = 1.00E+01 (10) uCi/sec

(3) High to Mid Range = 5.00E+00 (5) uCi/sec

(4) Mid to Normal Range = 1.00E-03 (.001) uCi/sec

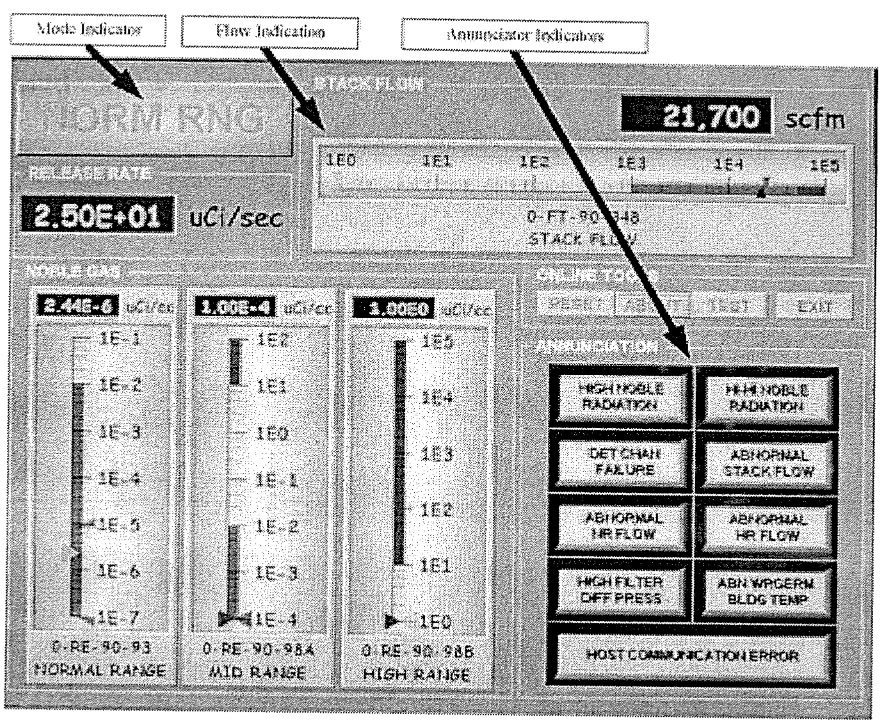
****

BFN Unit 2	Radiation Monitoring System	2-OI-90 Rev. 0079 Page 43 of 70

Illustration 2
(Page 1 of 11)
Wide Range Gaseous Effluent Radiation Monitor Operation

PANEL LAYOUT

Remote Main Screen (Control Room Screen)



DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0040 Page 3 of 52

Annunciator Window Legend

FUEL POOL FLOOR AREA RADIATION HIGH 1-RA-90-1A (1)	AIR PARTICULATE MONITOR RADIATION HIGH 1-RA-90-50A (2)	RHRSW/RCW EFFLUENT RADIATION HIGH 1-RA-90-132 (3)	RX BLDG,TURB BLDG, RF ZONE EXH RADIATION HIGH 1-RA-90-250A (4)	OG PRETREATMENT RAD[illegible] H[illegible] 1-RA-[illegible] (5)	STACK GAS RADIATION HIGH-HIGH 1-RA-90-147A (6)	MAIN STEAM LINE RADIATION HIGH 1-RA-90-135A (7)
AREA RADIATION MONITOR DOWNSCALE 1-RA-90-1C (8)	SPARE (9)	PSC PUMP SUCTION STNR DIFF PRESS HIGH 1-PDA-75-74 (10)	RX BLDG, TURB BLDG, RF ZONE EXH RAD MON DNSC 1-RA-90-250B (11)	OG PRETREATMENT RADIATION MONITOR DOW[illegible] 1-RA-[illegible] (12)	STACK GAS RADIATION HIGH 1-RA-90-147B (13)	MAIN STEAM LINE RADIATION MONITOR DOWNSCALE 1-RA-90-135B (14)
RADWASTE BLDG AREA RADIATION HIGH 1-RA-90-1F (15)	RADWASTE EFFLUENT RADIATION HIGH 0-RA-90-130A (16)	RBCCW EFFLUENT RADIATION HIGH 1-RA-90-131A (17)	TURB BLDG ROOF EXH VENT RADIATION HIGH 1-RA-90-251A (18)	PSC HEAD TANK LE[illegible] 1-LA-75-78 (19)	STACK GAS RADIATION MONITOR NSC/INOP 1-RA-90-147C (20)	REACTOR ZONE EXHAUST RADIATION HIGH 1-RA-90-142A (21)
RX BLDG AREA RADIATION HIGH 1-RA-90-1D (22)	RADWASTE EFFL RADIATION MONITOR DOWNSCALE 0-RA-90-130C (23)	RBCCW/RCW/RHRSW EFFL RADIATION MONITOR DNSC/INOP 1-RA-90-131B (24)	TURB BLDG ROOF EXH VENT RADIATION MONITOR DOWNSCALE 1-RA-90-251B (25)	PSC HEAD TANK LEVEL LOW 1-LA-75-79 (26)	MAIN STEAM LINE RADIATION HIGH-HIGH 1-RA-90-135C (27)	REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE 1-RA-90-140B (28)
TURBINE BLDG AREA RADIATION HIGH 1-RA-90-1E (29)	SPARE (30)	SPARE (31)	SPARE (32)	OG PRETREATMENT SAMPLE FLOW ABNORMAL 1-FA-90-165 (33)	REFUELING ZONE EXHAUST RADIATION HIGH 1-RA-90-140A (34)	RX ZONE EXH RADIATION MONITOR DNSC 1-RA-90-142B (35)

DISTRACTOR PLAUSIBILITY SUPPORT

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0043 Page 3 of 51

Annunciator Window Legend

FUEL POOL FLOOR AREA RADIATION HIGH 3-RA-90-1A (1)	AIR PARTICULATE MONITOR RADIATION HIGH 3-RA-90-50A (2)	RHRSW/RCW EFFLUENT RADIATION HIGH 3-RA-90-132 (3)	RX BLDG, TURB BLDG, RF ZONE EXH RADIATION HIGH 3-RA-90-250A (4)	OG PRETREATMENT RADI... HI... 3-RA-90-157A (5)	STACK GAS RADIATION HIGH-HIGH 3-RA-90-147A (6)	MAIN STEAM LINE RADIATION HIGH 3-RA-90-135A (7)
AREA RADIATION MONITOR DOWNSCALE 3-RA-90-1C (8)	AIR PARTICULATE MONITOR MALFUNCTION 3-RA-90-50B (9)	PSC PUMP SUCTION STNR DIFF PRESS HIGH 3-PDA-75-74 (10)	RX BLDG, TURB BLDG, RF ZONE EXH RAD MON DNSC 3-RA-90-250B (11)	OG PRETREATMENT RADIATION MONITOR DOWN... 3-RA-... (12)	STACK GAS RADIATION HIGH 3-RA-90-147B (13)	MAIN STEAM LINE RADIATION MONITOR DOWNSCALE 3-RA-90-135B (14)
RADWASTE BLDG AREA RADIATION 3-RA-90-1F (15)	RADWASTE EFFLUENT RADIATION HIGH 0-RA-90-130A (16)	RBCCW EFFLUENT RADIATION HIGH 3-RA-90-131A (17)	TURB BLDG ROOF EXH VENT RADIATION HIGH 3-RA-90-251A (18)	PSC HEAD TAN... 3-LA-75-78 (19)	STACK GAS RADIATION MONITOR DNSC/INOP 3-RA-90-147C (20)	REACTOR ZONE EXHAUST RADIATION HIGH 3-RA-90-142A (21)
RX BLDG AREA RADIATION HIGH 3-RA-90-1D (22)	RADWASTE EFFL RADIATION MONITOR DOWNSCALE 0-RA-90-130C (23)	RBCCW/RCW/RHRSW EFFL RADIATION MONITOR DNSC/INOP 3-RA-90-131B (24)	TURB BLDG ROOF EXH VENT RADIATION MONITOR DOWNSCALE 3-RA-90-251B (25)	PSC HEAD TANK LEVEL LOW 3-LA-75-79 (26)	MAIN STEAM LINE RADIATION HIGH-HIGH 3-RA-90-135C (27)	REFUELING ZONE EXHAUST RADIATION MONITOR DOWNSCALE 3-RA-90-140B (28)
TURBINE BLDG AREA RADIATION HIGH 3-RA-90-1E (29)	(SPARE) (30)	(SPARE) (31)	(SPARE) (32)	OG PRETREATMENT SAMPLE FLOW ABNORMAL 3-FA-90-165 (33)	REFUELING ZONE EXHAUST RADIATION HIGH 3-RA-90-140A (34)	RX ZONE EXH RADIATION MONITOR DNSC 3-RA-90-142B (35)

Examination Outline Cross-reference:

G2.4.42 (10CFR 55.41.10)

Knowledge of emergency response facilities.

	RO	SRO
Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.4.42	
Importance Rating	2.6	------

Proposed Question: **# 74**

A plant emergency is in progress that requires a declaration in accordance with EPIP-1, "Emergency Plan Implementing Procedure." The plant emergency in progress is **NOT** a security threat to facility protection.

Which ONE of the following is the **LOWEST** classification level that requires the Technical Support Center (TSC) **AND** Operations Support Center (OSC) to be activated?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Plausibility based on misconception that any declaration of an event requires activation of OSC and TSC

B **CORRECT**: The TSC and OSC are required to be activated at the Alert or higher emergency classification.

C INCORRECT: Plausible because some actions are first initiated at the Site Area Emergency level (e.g., State headquarters are established at the Morgan County Courthouse and Joint Information Center at Calhoun Community College is staffed.)

D INCORRECT: Plausible because some actions are first initiated at the General Emergency level (e.g., PARs are issued to the state).

KA Justification:

The KA is met because the question tests knowledge of what Emergency Action Level Emergency Response Facilities, OSC and TSC, are required to be activated.

Question Cognitive Level:

This question is rated as Fundamental Knowledge.

Technical Reference(s): EPIP-6 Rev. 30 / EPIP-7 Rev. 27 (Attach if not previously provided)
OPL171.075 Rev. 25

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.075 V.B.10 (As available)

Question Source:
Bank # Quad Cities 09 #75
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Quad Cities 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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X
55.43

Comments:

BROWNS FERRY	ACTIVATION AND OPERATION OF THE OPERATIONS SUPPORT CENTER	EPIP-7

1.0 INTRODUCTION

1.1 Purpose

The purpose of this procedure is to describe the process for activation of the OSC as well as define the activities and responsibilities of OSC team members.

2.0 REFERENCES

2.1 Industry Documents

A. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"
B. 10 CFR 50.47, "Code of Federal Regulations"

2.2 Plant Instructions

A. TVA Radiological Emergency Plan
B. BFN EPIP - 1, "Emergency Classification Procedure"
C. BFN EPIP - 2, "Notification of Unusual Event"
D. BFN EPIP - 3, "Alert"
E. BFN EPIP - 4, "Site Area Emergency"
F. BFN EPIP - 5, "General Emergency"
G. BFN EPIP - 16, "Termination and Recovery"
H. BFN Business Practice (BP) 319, "Emergency Preparedness Guidelines"

3.0 INSTRUCTIONS

3.1 Activation

The OSC is required to be activated at the Alert or higher emergency classification, however, activation may occur at the discretion of the Shift Manager. Once an emergency classification has been declared, the Shift Manager (SM) becomes the Site Emergency Director (SED). Depending upon the emergency classification declared, steps to activate the OSC are specified in the applicable EPIP for that emergency classification. Activation time for the OSC is defined in the Radiological Emergency Plan.

3.2 Methods of Notification of Emergency Response Organization (ERO)

Notification of the OSC personnel can be accomplished by one or more of the following methods:

- Activation of the Emergency Paging System (EPS) is the primary method.
- Manual call-out through utilization of the call-out list.
- Plant Public Address (PA) announcement.
- Activation of the Assembly and Accountability siren.

3.3 ERO Information

SPP 1.9, "Emergency Preparedness" provides the ERO with information regarding duty assignments and response to emergency call-outs.

BROWNS FERRY	ACTIVATION AND OPERATION OF THE TECHNICAL SUPPORT CENTER	EPIP-6

1.0 INTRODUCTION

1.1 Purpose

The purpose of this procedure is to describe activation of the Technical Support Center (TSC), define the TSC organization and provide for TSC operations by defining staff responsibilities.

2.0 REFERENCES

2.1 Industry Documents

A. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"
B. 10 CFR 50.47, Code of Federal Regulations

2.2 Plant Instructions

A. TVA Radiological Emergency Plan
B. Emergency Plan Implementing Procedure (EPIP) - 1, "Emergency Classification Procedure"
C. EPIP - 2, "Notification of Unusual Event"
D. EPIP - 3, "Alert"
E. EPIP - 4, "Site Area Emergency"
F. EPIP - 5, "General Emergency"
G. EPIP - 16, "Termination and Recovery"
H. EPIP-15, "Emergency Exposures"
I. EPIP-11, "Security and Access Control"

3.0 INSTRUCTIONS

3.1 Activation

The TSC is required to be activated at the Alert or higher emergency classification, however, activation can occur at the discretion of the Shift Manager (SM). Once an emergency classification has been declared, the SM becomes the Site Emergency Director (SED). Depending upon the emergency classification declared, steps to activate the TSC are specified in the applicable EPIP for the emergency classification. When the TSC is activated, the on-call SED will obtain a turnover from the SM/SED, ensure that minimum staffing is met for the emergency center, and assume the responsibilities of the SED from the SM/SED. Once the responsibilities of the SM/SED have been assumed by the on-call SED, command and control of the emergency response transfers to the TSC. TSC activation time is defined in the Radiological Emergency Plan.

3.2 Methods of Notification of Emergency Response Organization (ERO)

Notification of the TSC personnel can be accomplished by one or more of the following methods:

- Activation of the Emergency Paging System (EPS) is the primary method.
- Manual call-out through utilization of the call-out list.
- Plant Public Address (PA) announcement
- Activation of the Assembly and Accountability Siren

3.3 ERO Information

SPP 1.9, "Emergency Preparedness" provides the Emergency Response Organization (ERO) with information regarding duty assignments and response to emergency call-outs.

INSTRUCTORS NOTES

E. Alert, EPIP-3 — Refer to EPIP-3

1. Lowest classification during which emergency centers are required to be manned.
2. This classification assures that emergency personnel are readily available to respond if the situation becomes more serious. EPIP-3 contains the direction for activating the emergency response organization for the Alert.
3. Upon declaration of this class, the following actions are performed: — Obj. V. B.8
   a. Notification Requirements based on Emergency Centers staffed or not staffed. — Required for EPIP-3, 4, 5
   b. The Operations Duty Specialist (ODS) should be notified by the SM within five minutes of the event classification. The ODS relays the information to the EDO, the State of Alabama, and the CECC Director. The EDO keeps the CECC Director informed of the situation as necessary.
   c. SM/SED completes Appendix A.
   d. Site emergency response personnel, including the Plant Manager, are notified by the Unit 1 operator using Appendix B.
   e. Fax a copy of Attachment A to the ODS
   f. A plant PA announcement is made.
   g. The SM/SED notifies the NRC as soon as possible and within one hour of the event classification.

      Note: Any notification may be delegated to other individuals.
   h. If the situation warrants accountability, activate the Accountability Alarm in accordance with EPIP-8.
   i. The CECC is staffed by the ODS.
   j. The TSC and OSC are activated.

DISTRACTOR PLAUSIBILITY SUPPORT

INSTRUCTORS NOTES

F. Site Area Emergency, EPIP-4. — Refer to EPIP-4

1. Notification Requirements based on Emergency Centers Staffed or Not Staffed.

2. Upon declaration of this class, the actions described in E.3. are performed. In addition: — Obj. V.B.8

   a. A precautionary Accountability is initiated (if not already performed) and **then** an evacuation of non-emergency responders is initiated in accordance with EPIP-8. — EPIP-4 makes this mandatory. Accountability then Evacuation

   b. If appropriate, protective actions for the public are recommended to State agencies by the CECC (not required for SAE).

   c. Also of interest at Site Area Emergency: State headquarters are established at the Morgan County Courthouse and Joint Information Center at Calhoun Community College is staffed.

3. The initiating conditions and emergency action levels which require the Site Area Emergency are explained in the Technical Basis. EPIP-4 directs a continuous mode of evaluation and reevaluation of changing conditions for the event using EPIP-1. When those changes are recognized, they are to be communicated to offsite agencies. — Review Appendix C

4. Discuss all sections of EPIP-4 and stress the following: 3.4 & Appendix A.

G. General Emergency, EPIP-5 — Refer to EPIP-5

1. Notification Requirements based on Emergency Centers Staffed or Not Staffed.

2. This classification initiates predetermined protective action for the public, provides continuous assessment of information, and initiates additional measures as required by releases of radioactivity. — Conservative decision making

DISTRACTOR PLAUSIBILITY SUPPORT

INSTRUCTORS NOTES

3. EPIP-5 contains the directions for activating the emergency response for the General Emergency and the guidance for making protective action recommendations. — Review: EPIP-5 Attachment C for PARs

4. The Site Emergency Director must make any required recommendations until the CECC is staffed. This responsibility cannot be delegated until CECC is in operation. Recommendations are required at General Emergency. — Obj. V.B.7

5. If this is the initial classification, the SM notifies the ODS within 5 minutes, and the ODS notifies the local governmental agencies within 15 minutes, and recommends protective actions. If in a General Emergency and ODS cannot be contacted use phone numbers at bottom of page 2 of EPIP-5 to contact local counties directly and State of Alabama Rad Health Duty Officer. — SM has 5 min; ODS has 15 min

6. The initiating conditions and emergency action levels which require the General Emergency are explained in the Technical Basis. EPIP-5 directs a continuous mode of evaluation and reevaluation of changing conditions for the event using EPIP. When those changes are recognized, they are to be communicated to offsite agencies. — Review Appendix C

7. A plant evacuation of non-emergency responders, must be conducted in accordance with EPIP-8.

8. Discuss all sections of EPIP-5 and stress Protective Action Recommendations (Appendix C). — Obj. V.B. 9

H. Emergency Organizations — EPIP-6 & 7

1. The onsite organization is composed of the Site Emergency Director and technical staff located in the Technical Support Center, the on-shift Operations personnel, and additional support personnel in the Operations Support Center — Obj. V.B.10; NP REP Plan Appendix "A"

2. The Technical Support Center (TSC) is staffed during an ALERT, SITE AREA EMERGENCY, or GENERAL EMERGENCY. — EPIP-6; TP-1

EXAMINATION ANSWER KEY

U.S. Nuclear Regulatory Commission 2009 SRO Written Exam (Quad Cities)

75 ID: QDC.ILT.15550 Points: 1.00

A plant emergency is in progress that requires a declaration in accordance with the Exelon Nuclear Emergency Plan (E-Plan).

- The plant emergency in progress is NOT a security threat to facility protection.

Which one of the following states the lowest classification level that REQUIRES the Technical Support Center (TSC) and Operations Support Center (OSC) to be ACTIVATED?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

Answer Explanation:

Answer: The TSC and OSC must be activated at an ALERT classification or higher when NOT a security event.

Distractor 1 is incorrect: Plausible if the candidate assumes TSC is always activated at an Unusual Event (events other than a security event).
Distractor 2 is incorrect: Plausible because some actions are first initiated at the Site Area Emergency level (e.g., Assembly/Accountability).
Distractor 3 is incorrect: Plausible because some actions are first initiated at the General Emergency level (e.g., PARs are issued to the state).

Reference: G-1 / EP Overview Rev 7
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 3 Group: N/A

Question Source: Catawba ILT Bank # 581
Question History: 2008 Catawba ILT NRC Exam

10 CFR Part 55 Content: 41.10

Comments: Changed answer location (response to NRC comment).

Associated objective(s):

NGET Objective link (Refer to Non-Acredited Project for NGET/RWT objectives)

2.4.42 Knowledge of emergency response facilities. (RO=2.6 / SRO=3.8)

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.47	Tier #	3	-------
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	Group #	N/A	-------
	K/A #	G2.4.47	
	Importance Rating	4.2	-------

Proposed Question: **# 75**

ALL High Pressure Injection has been lost on Unit 2.

- At 16:00:00, Reactor Water Level is (-) 110 inches
- At 16:02:00, Reactor Water Level is (-) 118 inches

If level continues to lower at the same rate, which ONE of the following completes the statement?

A Common Accident Signal will be initiated by __(1)__ Range level instruments **AND** the **EARLIEST** time that **ALL** Core Spray Pumps will have auto started is ___(2)___ .

A. (1) Emergency
(2) 16:03:07

B. (1) Post Accident
(2) 16:03:07

C. (1) Emergency
(2) 16:03:21

D. (1) Post Accident
(2) 16:03:21

Proposed Answer: **C**

Explanation (Optional):

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

B INCORRECT: (1) Incorrect, this instrument indicates (-)268 to (+)58 inches and initiates the Containment Spray Interlock. Candidate may select because instrument indication is within the desired range of Level 1. (2) Time is incorrect. Plausible in that this would be the correct answer for D/G Voltage Available (DGVA) sequence. Since there is no loss of offsite power, a Normal Voltage Available (NVA) sequence will occur.

C **CORRECT:** 1) Correct instrument. Emergency Range is (-)155 to (+)60 inches. Initiates HPCI, RCIC, RHR, CS and ADS. (2) Time is correct, level trend is 4 inches/min. Three minutes to Level 1, and with Normal Voltage Available (NVA), the last Core Spray Pump will sequence on 21 seconds after the accident signal is received.

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

KA Justification:

The KA is met because the candidate must diagnose and determine trend and know correct control room instrument (range and function).

Question Cognitive Level:

This is higher cognitive because the examinee must know at what level Core Spray auto starts, calculate the time to the level, know the Core Spray sequence times based on the given plant conditions, and calculate the total time. The examinee must also know which type of instrumentation initiates the signal. He/she must use a multi-part mental process of assembling, sorting, or integrating parts of multiple systems to predict the outcome.

Technical Reference(s): OPL171.038 Rev 17 (Attach if not previously provided)

OPL171.003 Rev 19

Proposed references to be provided to applicants during examination: NONE

Learning Objective: OPL171.038 V.B.9, 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:

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)

b. Capable of fast starting and being ready to load within 10 seconds.

[illegible]

INST. NUM.	FUNCTION
LT 3-184 & 185	ADS CONFIRMATORY
LT 3-203 A,B,C & D	Rx SCRAM, PCIS 2,3,6 & 8
LT 3-206 A,B,C & D	HPCI, RCIC, Main & RFP TRIPS
LT 3-53, 60, 206, & 253	FEEDWATER LEVEL CONTROL
PT 3-54, 3-61 & 207	FEEDWATER LEVEL CONTROL
PT 3-204 A,B,C, & D	RECIRC PUMP TRIP

[illegible]

INST. NUM.	FUNCTION
LT 3-56 A,B,C & D	RECIRC PUMP TRIP, PCIS GP 1
LT 3-58 A,B,C & D	HPCI,RCIC,RHR,CS & ADS INIT.
LT 3-46 A & B	PNL 25-32 INDICATION
PT 3-74 A & B	RHR & CS LOGIC
PT 3-22 AA,BB,C & D	HIGH PRES SCRAM
PIS 3-22A	MECH VAC PUMP TRIP
PT 3-58	RECORDER PNL 9-5
PT 3-79	PNL 25-32 INDICATION

POST ACCIDENT RANGE INSTRUMENTS (+32 TO -268")

INST. NUM.	FUNCTION
LT 3-52 & 62	CONTAINMENT SPRAY INTERLOCK

[illegible]

INST. NUM.	FUNCTION
LT 3-55	PANEL 9-3 INDICATION

TP-6 Reactor Vessel Level/Pressure Instrumentation