QUESTION 1

Which ONE of the following completes both statements below?

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

Thermal Limits __(2)__ required to be adjusted in accordance with Technical Specification 3.4.1, Reactor Coolant System, for continued power operation.

(Assume operation greater than 24 hours)

A. (1) lower
(2) are

B. (1) lower
(2) are NOT

C. (1) rise
(2) are

D. (1) rise
(2) are NOT

Answer: A

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

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

Explanation: **A CORRECT** – The value of Critical Power will lower for a recirculation pump trip , and the. The single loop limit(s) for APLHGR and MCPR must be applied in accordance with Technical Specification 3.4.1, Reactor Coolant System (RCS).

B Incorrect –First Part: Correct. Second Part: Incorrect since Tech Specs requires both APLHGR and MCPR to be adjusted for power. Plausible if the examinee is not aware of the requirements of TS 3.4.1 or is not familiar with the time requirement for the applicable action statement.

C Incorrect –First Part: Incorrect. Plausible because CPR will rise. The candidate may confuse Critical Power with Critical Power Ratio. Second Part: Correct.

D Incorrect – First Part: Incorrect. Plausible because CPR will rise. Second Part: Incorrect since Tech Specs requires both APLHGR and MCPR to be adjusted for power. Plausible if the examinee is not aware of the requirements of TS 3.4.1 or is not familiar with the time requirement for the applicable action statement.

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

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: Brunswick 2010 #37

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

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

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

OR

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

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

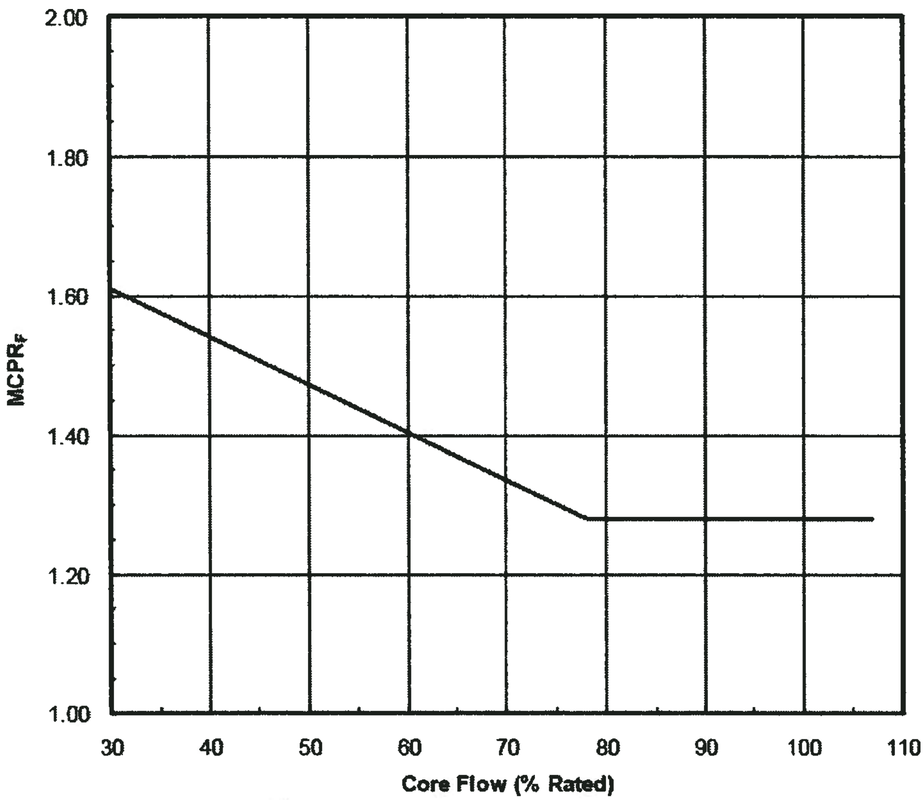
b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

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

APPLICABILITY: MODES 1 and 2.

ACTIONS

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



Core Flow (%Rated)	MCPR$_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 MCPR$_F$ for ATRIUM-10 Fuel
(Values bound all EOOS conditions)

Brunswick 2010 #37

37. 295001 K1.03 002

Which one of the following completes the statement below?

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

A. (1) rise
   (2) are

B. (1) rise
   (2) are not

C.✓ (1) lower
   (2) are

D. (1) lower
   (2) are not

QUESTION 2

Unit 3 was operating at 100% Reactor Power with all equipment in a normal lineup when the following occurred:

- A total loss of all off-site power in conjunction with a large break LOCA.

Subsequently,

- RPV level drops below (-) 122 inches.

Which ONE of the following completes the statement below?

When the Diesel Generator output breakers close, RHR pumps will start __(1)__ and the B SGT fan auto starts __(2)__.

A. (1) in 7 seconds
(2) in 40 seconds

B. (1) in 7 seconds
(2) ONLY if A SGT fan fails to start

C. (1) immediately
(2) in 40 seconds

D. (1) immediately
(2) ONLY if A SGT fan fails to start

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295003 AA2.04	
	Importance Rating	3.5	

295003 AA2.04 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C.POWER: System lineups

Explanation: **Answer C - CORRECT:** RHR pumps will start immediately when on DGVA.

A – Incorrect – First Part: Incorrect. Plausible because Core Spray pumps start 7 seconds after an accident signal with DGVA. Part (2) is Correct

B– Incorrect – First Part: Incorrect. Plausible because Core Spray pumps start 7 seconds after an accident signal with DGVA. Second Part: Incorrect. The B SGT fan will auto start in 40 seconds regardless of whether A SGT fan starts.

D– Incorrect – First Part: Correct. Second Part: Incorrect. The B SGT fan will auto start in 40 seconds regardless of whether A SGT fan starts.

Technical Reference(s) 0-AOI-57-1A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: BFN 0707 #24

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

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

3.0 AUTOMATIC ACTIONS (continued)

V. Unit 1/2 480V Load Shed occurs on a loss of offsite power in conjunction with a LOCA signal:

1. One RBCCW pump auto restarts (after 40 seconds on U1 and U2).

2. Drywell Blowers auto restart on non-accident unit (after 40 seconds). Drywell Blowers with their respective auto restart inhibit switches in the INHIBIT position will not auto restart.

3. Drywell coolers are manually restarted on the accident unit. A Drywell Blower with its auto restart inhibit switch in the INHIBIT position can be manually restarted after a ten minute time delay.

4. SGT TRAINS A & B trip, but will AUTO RESTART in 40 seconds when an initiation signal is present.

5. Loss of Control Bay Chilled Water Pumps A & B. (may be restarted after 10 minutes with use of bypass switch).

W. Unit 3 480V load shedding occurs as follows:

1. Division I 480V load shedding will occur when an accident signal is present and diesel generator voltage is available on the 4160V shutdown board supplying the 480V shutdown board 3A as follows:

   a. RBCCW pump 3A trips

   b. Drywell blowers 3A1, 3A2, 3A3, and 3A4 trip

   c. After a 40 second time delay, with the control switch in Normal After Start, RBCCW pump 3A restarts

   d. After a 40 second time delay, Drywell blowers 3A1 and 3A2 can be manually restarted

   e. Drywell blowers 3A3 and 3A4 cannot be restarted until the load shed signal is corrected

Lesson Plan Content

Outline of Instruction | **Ins a**

a) Accident signal received (CAS_x)

(1) Signals diesel generators to start.

(2) Opens diesel output breakers if shut. — Obj. I

b) If normal voltage is available (NVA), load will sequence on as follows: — Obj. I Obj. I

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				

Obj. I Obj. I

c) **If normal voltage is NOT available: i.e. on (DGVA)***

(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)

4. OI-82 cautions the operator to avoid routine fast starts between 15 minutes and 3 hours after engine shutdown. See OI-82 P&L for applicability to specific DGs. — Proc Adhe & ME DCN DCN OPE

a. During these conditions, the soakback subsystem and the immersion heater in the cooling water system maintain the oil system ready to

24. Given the following Unit 3 plant conditions:

- Operating at 100% rated power with all equipment in a normal lineup.
- A total loss of all off-site power occurs in conjunction with a large break LOCA.
- Drywell pressure peaks at 22 psig and is subsequently lowered to 2.3 psig using Drywell Sprays.
- RPV pressure lowered to 400 psig and is stable.
- RPV level drops below (-) 122 inches.
- Assume no operator actions.

Which ONE of the following describes the expected response of the RHR pumps and SGT system as a result of these conditions?

When the DG output breakers close, RHR pumps will start _____(1)_____. The B SGT fan _____(2)_____.

	(1)	(2)
A.	in 7 seconds	auto starts ONLY if A SGT fan fails to start
B.	in 7 seconds	auto starts in 40 seconds
C.	immediately	auto starts ONLY if A SGT fan fails to start
D.	immediately	auto starts in 40 seconds

QUESTION 3

Unit 2 was operating at 100% Reactor Power when the following occurred:

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

Which ONE of the following completes the statements below?

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

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

A. (1) energized
   (2) will

B. (1) energized
   (2) will NOT

C. (1) de-energized
   (2) will

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

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295004AA1.03	
	Importance Rating	3.4	

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

Explanation: **Answer D - CORRECT:** First Part - 480v Shutdown Board 2A is de-energized with the loss of 4kV Shutdown Board B. It is the normal feeder to the 480v S/D Board 2A and the transfer to alternate power is manual. Second Part - 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.

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

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

Technical Reference(s)- OPL171.036, OPL171.037, 0-OI-57B

Proposed references to be provided to applicants during examination: None

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

Question Source:	Bank: X Modified Bank: New
Question History:	Previous NRC: BFN 1006 #3
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis: X

10 CFR Part 55 Content: 55. 41 7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

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	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.

OPL171.037
Revision 11

d. Distribution

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

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		

BFN 1006 NRC Exam #3

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

Proposed Question: # 3

Unit 2 was operating at 100% Reactor Power.

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

Which ONE of the following completes the statements?

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

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

A. (1) energized
(2) will

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

C. (1) energized
(2) will NOT

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

QUESTION 4

Given the following conditions on Unit 1:

- The Reactor is at 40% power
- The RPT EOC Recirc Pump Trip logic is enabled in accordance with 1-OI-68, Reactor Recirculation System

Which ONE of the following conditions will result in an automatic trip of the Reactor Recirculation Pumps?

A. Turbine trip or load reject condition

B. Low Reactor Water Level (+) 2 inches

C. Reactor Dome High Pressure 1073 psig

D. Reactor Feedpump trip coincident with Reactor water level (+) 27 inches

Answer: **A**

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

Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : Recirculation system: Plant-Specific

Explanation: **A CORRECT** – The RPT Breakers Open (Trips the feeder breakers) on a Turbine Stop Valve Closure, Power > 30% by Turbine 1st Stage Pressure. (EOC/RPT)

B- Incorrect –Plausible because ATWS RPT trips the reactor recirc pumps on RPV Low Low Level (Level 2) and the setpoint is (-) 45 inches.

C- Incorrect –Plausible because High Reactor Dome Pressure will trip the reactor recirc pumps, however the setpoint is ≥1148 psig. 1073 psig is the Reactor Scram setpoint.

D- Incorrect –Plausible because Any individual RFP trip (i.e. flow is < 19 percent) and Reactor water level ≤ 27 inches, causes a Reactor Recirc pumps to run back.

Technical Reference(s): 1-OI-68

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New
Question History:	Previous NRC: Perry 2010 #14
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X

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

3.8 Electrical

A. The power supplies to the MMR and DFR relays are listed below.

VFD 1A	
I&C BUS A (BKR 215)	1-RLY-068-MMR3/A & DFR3/A
ICS PNL 532 (BKR 30)	1-RLY-068-MMR2/A & DFR2/A
UNIT PFD (BKR 615)	1-RLY-068-MMR1/A & DFR1/A
VFD 1B	
I&C BUS B (BKR 315)	1-RLY-068-MMR3/B & DFR3/B
ICS PNL 532 (BKR 26)	1-RLY-068-MMR2/B & DFR2/B
UNIT PFD (BKR 616)	1-RLY-068-MMR1/B & DFR1/B

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

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

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

3. Turbine trip or load reject condition, when ≥ 30% power by turbine first stage pressure (EOC/RPT).

8.4 Resetting Recirc Pump Runback

NOTES

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

- Total feedwater flow ≤ 19 percent (15 second time delay). (Indicated by annunciators RECIRC LOOP A FLOW LIMITER ENFORCING and RECIRC LOOP B FLOW LIMITER ENFORCING)
- Any individual RFP flow is < 19 percent and Reactor water level ≤ 27 inches. (Indicated by annunciators and amber light above 1-HS-68-32 and 1-HS-68-41.)

Perry 2010 NRC #14

NRC Exam - 2010

QUESTION RO 14

With the reactor at 100% power, which of the following conditions will result in a reactor scram and a direct automatic transfer of the Recirculation Pumps from fast speed to slow speed?

A. Main turbine trip

B. Reactor feedwater pump trip

C. Drywell pressure high - 1.68 psig

D. Reactor water level high - Level 8

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

K&A: Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Recirculation system:

Main Turbine Generator Trip / 3

Explanation: **Answer A** – A main turbine trip from >38% power will initiate a reactor scram and EOC-RPT logic will initiate a downshift of Recirc pumps. This is based on the MT Stop valve position (direct)

B – incorrect – RFPT trip will cause a FCV runback when RPV hits L4

C – incorrect – DW pressure high will cause a Rx scram, but not a direct down shift of Recirc pumps – the subsequent lowering of feedwater flow will cause a RR Pump downshift after a time delay

D – incorrect – RPV water level high will cause a Rx scram, but not a direct down shift of Recirc pumps – the subsequent lowering of feedwater flow will cause a RR Pump downshift after a time delay

Technical Reference(s): ONI-N32 rev 9 & ARI-H13-P680-004-A3 rev 14	Reference Attached: ONI-N32 p 4 & ARI-H13-P680-004-A3 p 9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OT-COMBINED-B33-E.3

Question Source:	Bank #	INL-1294
	Modified Bank #	
	New	
Question History:	Previous NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	x
	55.43	

Comments: Level of Difficulty = x

QUESTION 5

An RPS failure to scram condition occurred on Unit 1.

In accordance with 1-AOI-100-1, Reactor Scram, the RO ATC inserts control rods by arming and depressing BOTH of the following:

- ARI MANUAL INITIATE, 1-HS-68-119A
- ARI MANUAL INITIATE, 1-HS-68-119B

Which ONE of the following completes both statements below?

The response, as a result of the Alternate Rod Insertion (ARI) manual initiation, is that the ATWS/ARI/RPT valves will __(1)__.

The failure of a SINGLE channel of ARI to initiate __(2)__ prevent the depressurization of the scram air header via ARI.

A. (1) energize
(2) will

B. (1) energize
(2) will NOT

C. (1) de-energize
(2) will

D. (1) de-energize
(2) will NOT

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295006.K1.03	
	Importance Rating	3.7	

Knowledge of the operational implications of the following concepts as they apply to SCRAM: Reactivity control

Explanation: **B CORRECT** – Part 1: CORRECT- ATWS/ARI/RPT valves are energize to open valves. Part 2: CORRECT- There are identical sets of ARI valves from each channel of ARI which reposition to block and vent off the scram air header.

A- Incorrect – Part 1: Correct- See B. Part 2: Incorrect. Plausible because the backup scram valves will not vent if both channels of RPS do not actuate

C- Incorrect – Part 1: Incorrect. Plausible because RPS scram valves are de-energize to actuate. Part 2: Incorrect- See A.

D- Incorrect – Part 1: Incorrect- See C. Part 2: Correct- See B.

Technical Reference(s): OPL171.005, 1-AOI-100-1,1-OI-85

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New : X

Question History: Previous NRC:

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

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

3.2 ATWS/ARI/RPT

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

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

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

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

AND

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

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

4.2 Subsequent Actions

NOTES

1) The steps in this section are written in general order of importance for most anticipated events; however, they are not required to be performed in order, but as required to maintain stable conditions. Once a step is entered, all associated substeps are required to be completed in order, except those in Step 4.2[33](Return to Service). Steps which are not applicable for this scram should be N/A'd.

2) For Scram Response logic to initiate, all of the following conditions must be met:
   - Scram Response Logic is not inhibited (amber light at SCRAM RESPONSE INHIBIT/RESET switch, 1-HS-46-5 on Panel 1-9-5, is extinguished).
   - REACTOR WATER LEVEL CONTROL PDS, 1-LIC-46-5 on Panel 1-9-5, is in AUTO and at least one individual RFPT Speed Control PDS in AUTO.
   - Either RPS A or B Backup Scram channel activates.
   - Reactor Level (narrow range) falls below 0 inches within 60 seconds of first Backup Scram channel activating.

3) If Programmed Scram Response is initiated, the logic is reset by ANY of the following conditions:
   1. Placing REACTOR WATER LEVEL CONTROL PDS, 1-LIC-46-5 on Panel 1-9-5 in MANUAL.
   2. Reactor level (narrow range) exceeding level setpoint.
   3. Five minutes expire from the time the Scram Response logic was activated.
   4. Depressing SCRAM RESPONSE INHIBIT/RESET Switch, 1-HS-46-5, on Panel 1-9-5.

[1] **ANNOUNCE** Reactor SCRAM over PA system. ☐

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

PERFORM the following (otherwise N/A):

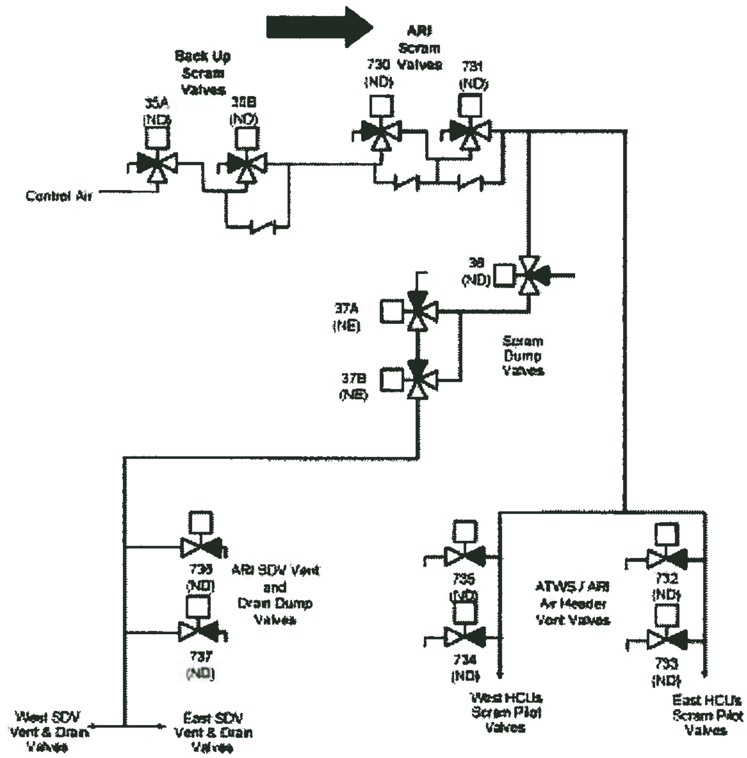
[2.1] **INITIATE** ARI by arming and depressing BOTH of the following:

- ARI MANUAL INITIATE, 1-HS-68-119A ☐
- ARI MANUAL INITIATE, 1-HS-68-119B ☐

[2.2] **VERIFY** the Reactor Recirc Pumps (if running) at minimum speed at Panel 1-9-4. ☐

[2.3] **REPORT** "ATWS Actions Complete" and power level. ☐

OPL171.005, CONTROL ROD DRIVE (CRD) HYDRAULICS, Rev. 18



TP-11 ATWS/ARI Valves

QUESTION 6

Which ONE of the following completes the statement below?

The immediate actions of 3-AOI-100-2, Control Room Abandonment, direct the control room operators to establish pressure control by ______ PRIOR to proceeding to the Backup Control Panel 3-25-32.

A. using the Bypass Jack

B. tripping the Main Turbine

C. closing ALL of the MSIVs

D. opening ONE of the SRVs

Answer is: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295016 G2.4.49	
	Importance Rating	4.6	

295016 Control Room Abandonment. G2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Explanation: **Answer B – CORRECT:** One of the immediate actions of 3-AOI-100-2 is to verify the Main Turbine tripped, thus ensuring that reactor pressure control is being automatically controlled on the Main Turbine Bypass Valves with steam going to the condenser.

A – incorrect – None of the actions in 3-AOI-100-2 direct using the Main Turbine Bypass Jack. This answer is plausible because using the Main Turbine Bypass jack will open bypass valves when in the pressure control mode of EHC and is used in EOI appendixes to rapidly depressurize the RPV.

C – incorrect – None of the actions in 3-AOI-100-2 direct using the Main Seam Relief Valves prior to abandoning the Control Room. This answer is plausible because Main Steam Relief valves are used from the backup Control Panel 25-32 to maintain pressure control. After the MSIVs are closed

D– incorrect – plausible. None of the actions in 3-AOI-100-2 direct using the Main Seam Relief Valves prior to abandoning the Control Room. This answer is plausible because Main Steam Relief valves are used from the backup Control Panel 25-32 to maintain pressure control.

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

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.208 V.B.8

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

4.0 OPERATOR ACTIONS

4.1 Immediate Action

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

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

LOWER core flow to between 50-60%. ☐

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

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

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

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

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

[6] **TRIP** Reactor Recirc Pumps. ☐

[7] **ISOLATE** RWCU. ☐

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

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

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

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

QUESTION 7

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

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

Which ONE of the following system loads is still being cooled by RBCCW?

A. Drywell Coolers

B. Fuel Pool Cooling heat exchangers

C. Reactor Water Cleanup Non-regenerative heat exchangers

D. Reactor Water Cleanup seal water and bearing oil coolers

Answer is: A

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

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

Explanation: **Answer A – CORRECT:** 1-FCV-70-48 automatically closes on low RBCCW supply header pressure <57 psig. This will isolate the non-essential loads and leave cooling to the essential loads only.

B – Incorrect –This is a non-essential load. Plausible because this is a load on RBCCW.

C – Incorrect – This is a non-essential load. Plausible because this is a load on RBCCW.

D– Incorrect – This is a non-essential load. Plausible because this is a load on RBCCW.

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

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.047 V.B.2, V.B.4

Question Source:	Bank: X Modified Bank: New
Question History:	Previous NRC: BFN ILT 0801 #7
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis:

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

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:

(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)

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

(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)

2. RBCCW Heat Loads

a. Essential loop loads

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

b. Non-essential loop loads

- 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)

1-AOI-70-1 Loss Of RBCCW

3.0 AUTOMATIC ACTIONS

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

7. 295018 AA1.02

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

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

Which ONE of the following system loads is still being cooled by RBCCW?

A. Drywell Coolers.

B. Fuel Pool Cooling heat exchangers.

C. RWCU non-regenerative heat exchangers.

D. RWCU Pump seal water and bearing oil coolers.

QUESTION 8

In accordance with 0-AOI-32-1, Loss of Control and Service Air Compressors, which ONE of the following is the HIGHEST Control Air Pressure, as indicated by 3-PI-32-88, which requires the reactor to be manually scrammed?

A. < 85 psig

B. < 73 psig

C. < 55 psig

D. < 45 psig

Answer is: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295019G2.4.1	
	Importance Rating	4.6	

295019 Partial or Complete Loss of Instrument Air. G2.4.1Knowledge of EOP entry conditions and immediate action steps.

Explanation: **Answer C–CORRECT:** When Control Air pressure lowers to below 55 psig, the reactor is required to be manually scrammed.

A– Incorrect –Service Air crosstie to Control Air valve, 0-FCV-33-1, opens at control air header pressure less than or equal to 85 psig.

B – Incorrect –The Emergency Control Bay Air Compressor will start at Control Air Header pressure less than or equal to 73 psig

D– Incorrect –When control air pressure instantaneouslydrops to < 45 psig, the MSIV accumulator air will be routed to close the outboard MSIVs.

Technical Reference(s): 0-AOI -32-1 Rev 41, Loss of Control and Service Air Compressors; 3-AOI-32-2, Loss of Control Air Rev 22

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.054 Obj.V.B.8

Question Source: Bank:
Modified Bank:
New X

Question History: None

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

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

0-AOI-32-1 Rev 41

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0041 Page 6 of 35

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

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.

4.2 Subsequent Actions

[1] **VERIFY** automatic actions. ☐

[2] **PERFORM** automatic actions that failed to occur. (Otherwise N/A) ☐

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

ENTER the appropriate EOI(s) (otherwise N/A). ☐

[4] **IF** CONTROL AIR PRESSURE is continuing to lower as indicated by 1-PI-32-20 on Panel 1-9-20 or 2(3)-PI-32-88 on Panel 2(3)-9-20, **AND** CONTROL AIR PRESSURE lowers below 55 psig, **THEN** (Otherwise N/A)

MANUALLY SCRAM the reactor. Refer to 1(2)(3)-AOI-100-1 and 1(2)(3)-AOI-32-2. ☐

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0041 Page 5 of 35

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.

Attachment 1
(Page 1 of 6)

Expected System Responses

1.0 MAIN STEAM

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

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

QUESTION 9

The following conditions exist on Unit 2:

- The Reactor is shutdown in MODE 4
- 2-AOI-74-1, Loss of Shutdown Cooling, has been entered due to a trip of the ONLY running RHR pump

Subsequently,

- The tripped RHR pump is restarted.

Which ONE of the following completes the statement below?

In accordance with 2-AOI-74-1, Loss of Shutdown Cooling, RHR flow should be re-established and maintained at _______ gpm.

A. 3,000 to 4,000

B. 6,000 to 6,500

C. 7,000 to 10,000

D. 14,000 to 20,000

Answer is: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295021 AA2.02	
	Importance Rating	3.4	

295021 Loss of Shutdown Cooling AA2.02 Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : RHR/shutdown cooling system flow

Explanation: **Answer C – CORRECT:** RHR flow should be re-established at 7-10k GPM for single pump operation.

A– Incorrect –this is the flowrate for RHSW operation IAW OI-74 when SDC is being established.

B – Incorrect – This is the flowrate for single loop operation with more than one fuel cell removed.

D– Incorrect – This is the flowrate for 2 pump operation.

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

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.044 Rev 18 ILT Objective 12

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

4.2 Subsequent Actions (continued)

NOTE
EQV 70933 requires 2-FCV-074-0002, 0013, 0025 and 0036 requires that due to limitations on the valves actuator, the valves shall not be stroked OPEN if the differential pressure across the disc exceeds 82 psid.

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

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

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

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

[14.9] THROTTLE RHR SYS I(II) LPCI OUTBD INJECTION VALVE, 2-FCV-74-52(66), to establish and maintain RHR flow as indicated by 2-FI-74-50(64), RHR SYS I(II) FLOW, as follows: ☐

RHR Pumps in Operation	1	2
Loop Flow	7,000 to 10,000	14,000 to 20,000
Loop Flow (1 or more fuel bundles removed from core)	6,000 to 6,500	N/A

QUESTION 10

Given the following conditions:

- Refueling is in progress on Unit 3
- An irradiated fuel bundle is dropped onto the top of the Unit 3 reactor core
- REFUELING ZONE EXHAUST RADIATION HIGH (3-9-3A, Window 34) is in alarm

Which ONE of the following completes both statements below?

This will result in the isolation of the __(1)__.

The reason for the resulting isolation is to ensure the release is ___(2)___.

A. (1) Refuel Zone ventilation ONLY
(2) controlled, filtered, and elevated

B. (1) Refuel Zone ventilation ONLY
(2) contained within secondary containment

C. (1) Refuel Zone and Unit 3 Reactor zone ventilation
(2) controlled, filtered, and elevated

D. (1) Refuel Zone and Unit 3 Reactor zone ventilation
(2) contained within secondary containment

Answer is: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295023 AK3.03	
	Importance Rating	3.3	

295023 Refueling Accidents AK3.03 Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS : Ventilation isolation

Explanation: **Answer A – CORRECT:** Part 1: CORRECT- At >72 mr/hr in the Refuel floor exhaust (Refuel Zone Exhaust Radiation High 3-XA-55-3A, window 34) a partial PCIS Group 6 isolation is actuated, which trips and isolates the supply and exhaust refuel floor fans for all three units, starts SGT, and aligns SGT to the Refuel zone ONLY. Part 2: CORRECT- The reason for the resulting ventilation isolation is to ensure that air normally discharged from all three units refuel floor ventilation will be treated by SGT prior to discharge to the environment, ensuring boundary dose from the fuel handling accident is within the limits of 10CFR100 by a controlled filtered elevated release of secondary containment building atmosphere.

B – Incorrect – Part 1- CORRECT: see A. Part 2- Incorrect: This is plausible as this is similar to the purpose of primary containment isolation.

C – Incorrect – Part 1- Incorrect: Plausible as the Unit 3 Reactor zone ventilation would isolate on a Reactor zone exhaust radiation at72 mr/hr, but would not isolate on a Refuel zone exhaust radiation high. Part 2- Incorrect: See- B.

D– Incorrect – Part 1- Incorrect: See C. Part 2: Correct- See A.

Technical Reference(s): 3-AOI-64-2d Rev 0016, 3-ARP-9-3A Rev 0045, Unit 3 TS Bases 3.6.4.2 (SCIV), FSAR Chapter 5.3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.033 Obj. V.B.4.a

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

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

3-AOI-64-2d

BFN Unit 3	Group 6 Ventilation System Isolation	3-AOI-64-2d Rev. 0016 Page 3 of 14

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 6 Ventilation System Isolation.

2.0 SYMPTOMS

NOTES
1) PCIS Group 6 Isolation is initiated by any one of the following signals: • Reactor vessel water level (LEVEL 3) • Drywell pressure at 2.45 psig • Reactor zone exhaust radiation at 72 mr/hr • Refuel zone exhaust radiation at 72 mr/hr
2) High Refuel Zone exhaust radiation causes only the automatic actions listed in Section 3.1.
3) Refuel Zone isolation due to Group 6 isolation initiated on Unit 1 or Unit 2.

3.0 AUTOMATIC ACTIONS

3.1 Refueling Zone Isolation

A. The following equipment TRIP and ISOLATE:

1. Refuel Zone Supply/Exhaust Fans/Dampers:

   a. REFUEL ZONE EXH FAN 3A DMPR, 3-FCO-064-0003A

   b. REFUEL ZONE SPLY FAN 3A DMPR, 3-FCO-064-0003B

   c. REFUEL ZONE EXH FAN 3B DMPR, 3-FCO-064-0004A

   d. REFUEL ZONE SPLY FAN 3B DMPR, 3-FCO-064-0004B

   e. REFUEL ZONE SPLY OUTBD ISOL DMPR, 3-FCO-064-0005

   f. REFUEL ZONE SPLY-INBD ISOL DMPR, 3-FCO-064-0006

   g. REFUEL ZONE EXH OUTBD ISOL DMPR, 3-FCO-064-0009

   h. REFUEL ZONE EXH INBD ISOL DMPR, 3-FCO-064-0010

C. Standby Gas Treatment System starts.

D. REFUEL ZONE EXH TO SGT CROSSTIE DMPR, 1 & 3 FCO-64-44, OPENS.

E. REFUEL ZONE EXH TO SGT CROSSTIE DMPR, 1-FCO-64-45, OPENS.

3-ARP-9-3A

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0045 Page 48 of 51

REFUELING ZONE EXHAUST RADIATION HIGH 3-RA-90-140A 34	Sensor/Trip Point:		
	3-RE-90-140A	72 MR/HR	
	3-RE-90-140B	72 MR/HR	
	3-RE-90-141A	72 MR/HR	Required setting of ≤ 100 MR/HR.
	3-RE-90-141B	72 MR/HR	

(Page 1 of 2)

Sensor Location: Rx Bldg, El 664' (Refuel Floor), R-17 P-LINE

Probable Cause:
A. Radiation levels have risen above alarm setpoint.
B. Refueling accident.
C. Dry Cask loading/unloading activities in progress.
D. Loss of power to NUMAC drawer.

Automatic Action:
A. Control Room and Refuel Zone ventilation isolates.
B. SGTS initiates.
C. Control Room Emergency Pressurization units start.

FSAR Chapter 5.3

5.3.3.2 Zone Ventilation System

The Reactor Building is divided into four ventilation zones which may be isolated independently. The refueling room which is common to the three units forms the refueling zone. The individual units below the refueling floor form the three reactor zones. The four-zone ventilation control system provides increased capability for localizing the consequences of an accident or radioactive release such that the effect may be localized in one zone while maintaining the ability to isolate the entire Reactor Building if necessary. With one or more zones isolated, normal operations may be continued in the unaffected zones. If radiation is detected in an unisolated zone, that zone too would isolate and the entire Reactor Building would still meet the requirements of secondary containment by assuring filtered elevated release. The zone system is not an engineered safeguard, and the failure of the zone system would not in any way prevent isolation or reduce the capacity of the Secondary Containment System.

A reactor zone is isolated upon isolation of the primary containment in that particular zone, by high radiation level in the ventilation exhaust duct leaving that particular zone, or by manual alignment. The refueling zone is always isolated when any reactor zone is isolated. The refueling zone only is isolated by a manual signal or by a high radiation signal from any of the six radiation monitors that serve the refueling zone (see FSAR Section 7.12.5). Upon isolation, all of the ventilation systems serving the isolated zone or zones are shut down, the ducts are isolated, and the Standby Gas Treatment System is started and begins exhausting from the isolated zone or zones.

5.3.2 <u>Safety Design Basis</u>

The Secondary Containment System provides secondary containment when the primary containments are intact. In the event of release of radioactivity to the Reactor Building atmosphere, the Secondary Containment System contains the necessary reliable, redundant components and subsystems to isolate, to contain, and to assure controlled filtered elevated release of Secondary Containment Building atmosphere.

Unit 3 TS Bases 3.6.4.2

BASES (continued)

APPLICABLE SAFETY ANALYSES	The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The accident for which secondary containment boundary is required is a loss of coolant accident (Ref. 1). The secondary containment performs no active function in response to this limiting event, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment. Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

QUESTION 11

Which ONE of the following completes the statements below?

3-XR-64-50, DRYWELL TEMPERATURE / PRESSURE RECORDER, at Panel 3-9-3 is identified as a "post accident monitoring instrument" by a/an __(1)__ label.

In accordance with 3-EOI-2, Primary Containment Control, 3-XR-64-50 indicating a Drywell Pressure of greater than __(2)__.

A. (1) orange
   (2) 12 psig requires initiating Drywell Sprays

B. (1) orange
   (2) 2.4 psig allows Suppression Chamber Sprays to be initiated

C. (1) black
   (2) 12 psig requires initiating Drywell Sprays

D. (1) black
   (2) 2.4 psig allows Suppression Chamber Sprays to be initiated

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295024 G2.4.3	
	Importance Rating	3.7	

High Drywell Pressure – Ability to identify post-accident instrumentation

Explanation: **D CORRECT** – First part PAM instruments are identified by Black Labels, Second Part When Drywell Pressure cannot be maintained below 2.4 psig the direction is to initiate Suppression Chamber Sprays prior to exceeding 12 psig.

A Incorrect –First Part: In-correct, plausible in that Orange Labels are used for EOI components. Second Part: Incorrect plausible in that a Suppression Chamber pressure exceeding 12 psig requires Drywell Sprays.

B Incorrect –First Part: In-correct see above. Second Part: Correct.

C Incorrect – First Part: Correct. Second Part: Incorrect. See Above

Technical Reference(s): EOI-2 Flowchart, TS 3.3.3.1 PAM Instrumentation

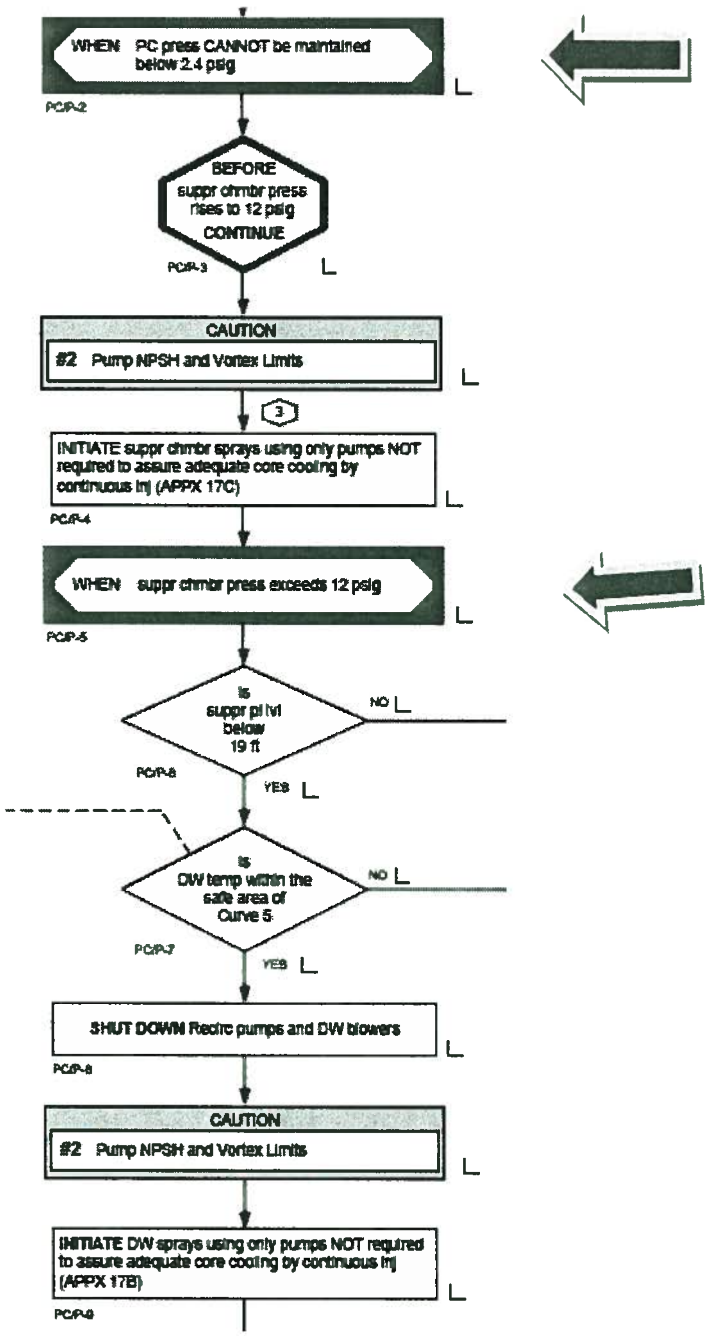
Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.017, Obj 2.o: OPL171.203 Obj 5

Question Source: Bank:
Modified Bank:
New X

Question History: Previous NRC:

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


WHEN PC press CANNOT be maintained below 2.4 psig
PC/P-2
BEFORE suppr chmbr press rises to 12 psig CONTINUE
PC/P-3
CAUTION
#2 Pump NPSH and Vortex Limits
3
INITIATE suppr chmbr sprays using only pumps NOT required to assure adequate core cooling by continuous inj (APPX 17C)
PC/P-4
WHEN suppr chmbr press exceeds 12 psig
PC/P-5
Is suppr pl lvl below 19 ft
PC/P-6
NO
YES
Is DW temp within the safe area of Curve 5
PC/P-7
NO
YES
SHUT DOWN Recirc pumps and DW blowers
PC/P-8
CAUTION
#2 Pump NPSH and Vortex Limits
INITIATE DW sprays using only pumps NOT required to assure adequate core cooling by continuous inj (APPX 17B)
PC/P-9

B 3.3 INSTRUMENTATION

B 3.3.3.1 Post Accident Monitoring (PAM) Instrumentation

<u>4. Drywell Pressure</u>
(PI-64-67B, XR-64-50, PI-64-160A, and XR-64-159)

Drywell pressure is a Category 1 variable provided to detect breach of the RCPB and to verify ECCS functions that operate to maintain RCS integrity. Two different ranges of drywell pressure channels (normal and wide range) receive signals that are transmitted from separate pressure transmitters and are continuously recorded and displayed on two control room recorders and two control room indicators. These recorders and indicators are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

QUESTION 12

Given the following conditions:

- Unit 1 has had a long period of full power operation
- An instantaneous loss of **ALL** AC power occurs and is **NOT** corrected
- The HPCI System failed to start
- Assume that the MINIMUM decay heat over the next hour is 173 MW/th

Which ONE of the following completes both statements below?

With **NO** operator action, over the next hour you would expect ___(1)___.

Over the next hour, SRV operation(s) will be at ___(2)___ Reactor pressure.

A. (1) SRVs to open and close periodically on overpressure
(2) 1145 psig

B. (1) SRVs to open and close periodically on overpressure
(2) 1135 psig

C. (1) SRV initial operation with RCIC operation precluding the need for further SRV operation
(2) 1145 psig

D. (1) SRV initial operation with RCIC operation precluding the need for further SRV operation
(2) 1135 psig

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295025EA2.05	
	Importance Rating	3.4	

295025EA2.05 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Decay heat generation

Explanation: **Answer B – CORRECT**-First Part: Candidate must recognize that RCIC steam demand is far less than decay heat generation. 173 MW/th is ~ 5% rated thermal power. Second Part: SRV's will continue to lift periodically at 1135psig reactor pressure.

A – Incorrect- First Part: Correct. Candidate must recognize that RCIC steam demand is far less than decay heat generation. 173 MW/th is ~ 5% rated thermal power. Second Part: Incorrect. Plausible because 4 SRV s have a safety lift setpoint of 1145psig.

C – incorrect –First part: Incorrect. Candidate must recognize that RCIC steam demand is far less than decay heat generation. 173 MW/th is ~ 5% rated thermal power. SRV's will continue to lift periodically on overpressure. Second Part: Correct.

D– incorrect - First part: Incorrect. Candidate must recognize that RCIC steam demand is far less than decay heat generation. 173 MW/th is ~ 5% rated thermal power. SRV's will continue to lift periodically on overpressure. Second Part: Incorrect. Plausible because 4 SRV s have a safety lift setpoint of 1145psig.

Technical Reference(s): OPL171.009

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New

Question History: Previous NRC: Fitzpatrick 2010 #81

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

10 CFR Part 55 Content: 55.41

(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

Fitzpatrick 2010 #81

QUESTION 81.

After a long period of full power operation, an instantaneous loss of **ALL** AC power occurs and is **NOT** corrected. The HPCI System failed to start.

- Assume that decay heat over the next hour is 6.2×10^{8} Btu/hr

With **NO** initial operator action, over the next hour you would expect ___(1)___ and, per AOP-49, "Station Blackout", operators should attempt to maintain RPV cooldown rate less than _(2)_?

A. (1) SRV's to open and close periodically on overpressure
(2) 80 °F/ hr

B. (1) SRV's to open and close periodically on overpressure
(2) 20 °F/ hr

C. (1) SRV initial operation with RCIC operation precluding the need for further SRV operation
(2) 80 °F/ hr

D. (1) SRV initial operation with RCIC operation precluding the need for further SRV operation
(2) 20 °F/ hr

QUESTION 13

Given the following conditions on Unit 1:

- Reactor power is 75%
- SRV 1-18 and SRV 1-23 are stuck open
- The immediate actions of 1-AOI-1-1, Relief Valve Stuck Open, have been completed
- Suppression pool temperature is 90° F and rising

The Unit Supervisor directs you to place RHR loop I and II in suppression pool cooling in service.

Which ONE of the following completes the statements below?

In accordance with 1-OI-74, Residual Heat Removal System, total RHR SYSTEM II flow rate should NOT exceed __(1)__ gpm.

In accordance with 1-AOI-1-1, Relief Valve Stuck Open, before Suppression Pool temperature exceeds __(2)__° F the reactor is required to be manually scrammed.

A. (1) 10,000
(2) 95

B. (1) 10,000
(2) 110

C. (1) 13,000
(2) 95

D. (1) 13,000
(2) 110

Correct Answer: **D**

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

K&A: Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling

Explanation: **Answer D – CORRECT**: First Part: IAW 1-OI-74, two RHR pumps per loop should be running when maximizing suppression pool cooling, and total RHR SYS II flowrate should not exceed 13,000 GPM with 2 RHR pumps in operation. Second Part: The Unit 1 RCIC turbine area Rx Bldg el 519' is Appendix R Fire Zone 01-01.

A – Incorrect – First Part: Incorrect. Plausible because 10,000 GPM is the maximum total flowrate for one RHR pump in operation. Second Part: Incorrect. Plausible because IAW 1-AOI-1-1, If any relief valve is stuck open and a fire exists in an appendix R fire area, then manually scram the reactor before suppression pool temperature exceeds 95 °F.

B – Incorrect – First Part: Incorrect. Plausible because 10,000 GPM is the maximum total flowrate for one RHR pump in operation. Second Part: Correct.

C– Incorrect – First Part: Correct. Second Part: Incorrect. Plausible because IAW 1-AOI-1-1, If any relief valve is stuck open and a fire exists in an appendix R fire area, then manually scram the reactor before suppression pool temperature exceeds 95 °F.

Technical Reference(s): 1-OI-74; 1-EOI-2;1-AOI-1-1;0-SSI-001

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.044 ILT Obj. 13

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

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

NOTES
1) Once initial transient of SRV opening has stabilized (pressure regulator compensation) the Heat Balance will indicate bad data.
2) The SRV TAILPIPE FLOW MONITOR may seal-in an OPEN position indication.

4.2 Subsequent Action

4.2.1 Action if a fire exists with SRV stuck open

[1] **IF** an SRV is open and a fire exists in ANY Appendix R fire area, **THEN** (Otherwise N/A):

INITIATE a manual scram before the Suppression Pool temperature exceeds 95°F. ☐

3.5 Suppression Pool

A. Suppression Pool level is required to be maintained in accordance with Technical Specification 3.6.2.2 limits.

B. All available Suppression Pool Cooling is required to be initiated whenever pool temperature exceeds 95°F. REFER TO 1-EOI-2.

C. During Suppression Pool Cooling, high RHR Cooling Water flows may cause the Drywell DP Compressor to run for extended periods. REFER TO 1-OI-64 if required to operate without the Drywell DP Compressor.

D. [NRC/C] PSA concerns with RHR in Suppression Pool Cooling Mode with a LOCA and a LOSP identify that severe water hammer may occur during the pump restart. Therefore, the following guidelines should be used to try and maintain the system below the PSA Risk Assessment goals:

1. RHR in Suppression Pool Cooling should be minimized.
2. Two Loops of RHR in Suppression Pool Cooling should be minimized.
3. Use two pumps per loop if needed to maximize Suppression Pool Cooling in order to minimize total time spent in Suppression Pool Cooling. [NRC IN 87-10]
4. Suppression Pool Cooling run times are tracked in 1-SR-2 to ensure risk assessment goals are not exceeded.

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

[7.3] **VERIFY CLOSED** RHR SYS I LPCI INBD INJECT VALVE, 1-FCV-74-53. ☐

[7.4] **IF NO** RHR PUMP (1A OR 1C) is operating in Suppression Pool Cooling, **THEN**

VERIFY CLOSED RHR SYS I SUPPR POOL CLG/TEST VLV, 1-FCV-74-59. ☐

[7.5] **VERIFY CLOSED** RHR SYS I SUPPR CHBR SPRAY VALVE, 1-FCV-74-58. ☐

[7.6] **VERIFY CLOSED** RHR SYS I DW SPRAY OUTBD VLV, 1-FCV-74-60. ☐

[7.7] **VERIFY OPEN** RHR SYS I SUPPR CHBR/POOL ISOL VLV, 1-FCV-74-57. ☐

CAUTION

1) To prevent excessive vibration, RHR Pumps should NOT be allowed to operate for more than 3 minutes at minimum flow.

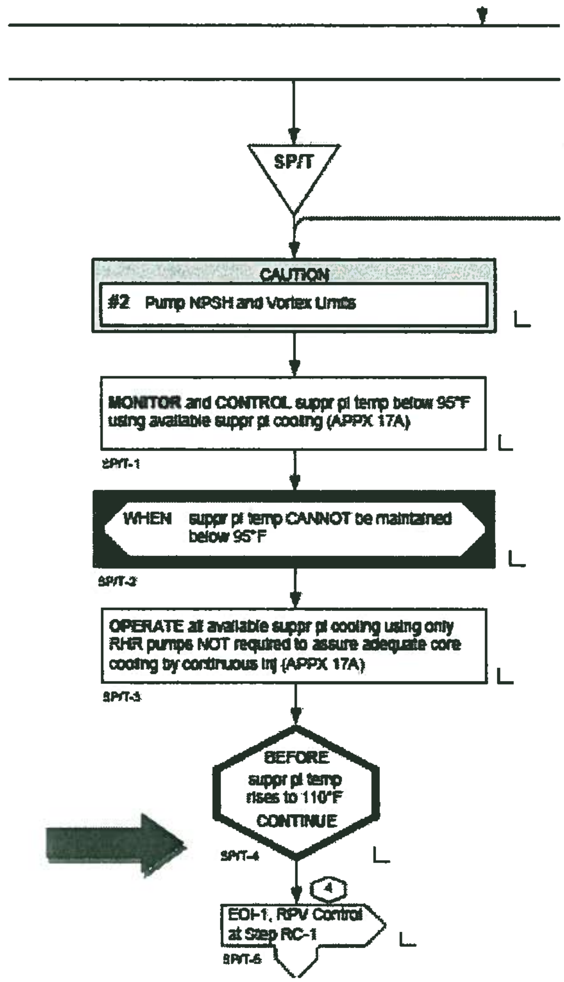
2) Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161kV source and large pumps are started. Unit Supervisors should evaluate placing the Capacitor Banks in Manual prior to starting RHR Pumps, as referenced in 0-OI-57A.

NOTES

1) RHR Flow should be monitored while in operation on 1-FI-74-50, RHR SYS I FLOW. RHR Flow should remain less than or equal to 10,000 gpm for 1-pump operation and is limited to less than 13,000 gpm, for two pump operation, due to the flow restricting orifice in the test return line.

2) During Suppression Pool Cooling, high RHR Cooling Water flows may cause the Drywell DP Compressor to run for extended periods.

1-EOI-2



SP/T
CAUTION
#2 Pump NPSH and Vortex Limits
MONITOR and CONTROL suppr pl temp below 95°F using available suppr pl cooling (APPX 17A)
SP/T-1
WHEN suppr pl temp CANNOT be maintained below 95°F
SP/T-2
OPERATE all available suppr pl cooling using only RHR pumps NOT required to assure adequate core cooling by continuous inj (APPX 17A)
SP/T-3
BEFORE suppr pl temp rises to 110°F CONTINUE
SP/T-4
4
EOI-1, RPV Control at Step RC-1
SP/T-5

QUESTION 14

Which ONE of the following completes the statement below?

EOI-2, Primary Containment Control, requires opening ADS valves on high Drywell temperature to minimize any continuing direct energy release to the drywell through a primary system break, and to ensure _____.

A. the MSRVs are opened while still operable

B. the RPV is depressurized prior to Reactor Recirc Pump damage

C. the RPV level instrumentation remains reliable while the RPV is depressurized

D. the RPV is depressurized prior to non-environmentally qualified nuclear instrumentation damage

Answer is: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295028 EK3.06	
	Importance Rating	3.4	

295028 EK3.06 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : ADS

Explanation: **Answer A – CORRECT:** If drywell temperature cannot be restored and maintained below the drywell design temperature (280 °F), emergency RPV depressurization is performed. This action minimizes any continuing direct energy release to the drywell through a primary system break and ensures that the MSRVs are opened while still operable.

B – Incorrect – Before drywell temperature reaches the maximum allowable drywell temperature while at power (200° F), EOI-1 RPV CONTROL is entered at step RC-1, because damage may occur to non-environmentally qualified (EQ) equipment such as recirculation pumps.

C– Incorrect – Boiling and loss of valid level indication can occur if the temperature of water in the instrument runs exceeds RPV saturation temperature (EOI Curve 8). RPV depressurization is not required when this curve is exceeded, and continued use of instrumentation is permitted until boiling is observed. If all level instrumentation was lost due to temperatures in the drywell and secondary containment such that RPV level could not be determined, then C4 RPV flooding would be entered and an emergency depressurization would be required. Emergency depressurization in and of itself would only serve to reduce RPV saturation temperature, making boiling more likely in the instrument runs in the drywell.

D – Incorrect – Before drywell temperature reaches the maximum allowable drywell temperature while at power (200° F), EOI-1 RPV CONTROL is entered at step RC-1, because damage may occur to non-environmentally qualified (EQ) instrumentation such as nuclear instrumentation components.

Technical Reference(s): EOIPM section 0-II-R rev 0000, 1-EOI-2 Primary Containment Control Rev 4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.203 ILT Obj. 4

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: Nine Mile Point 1, 2008 NRC question #47

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

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

1.0 PSTG/SATG PRIMARY CONTAINMENT CONTROL BASES (continued)

PSTG/SATG Step

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*DW/T-3 When drywell temperature cannot be restored and maintained below **A.34** (drywell design temperature), EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.*

Discussion

If drywell temperature cannot be restored and maintained below the drywell design temperature, emergency RPV depressurization is performed. This action minimizes any continuing direct energy release to the drywell through a primary system break and ensures that the MSRVs are opened while still operable.

1.0 PSTG/SATG PRIMARY CONTAINMENT CONTROL BASES (continued)

PSTG/SATG Step

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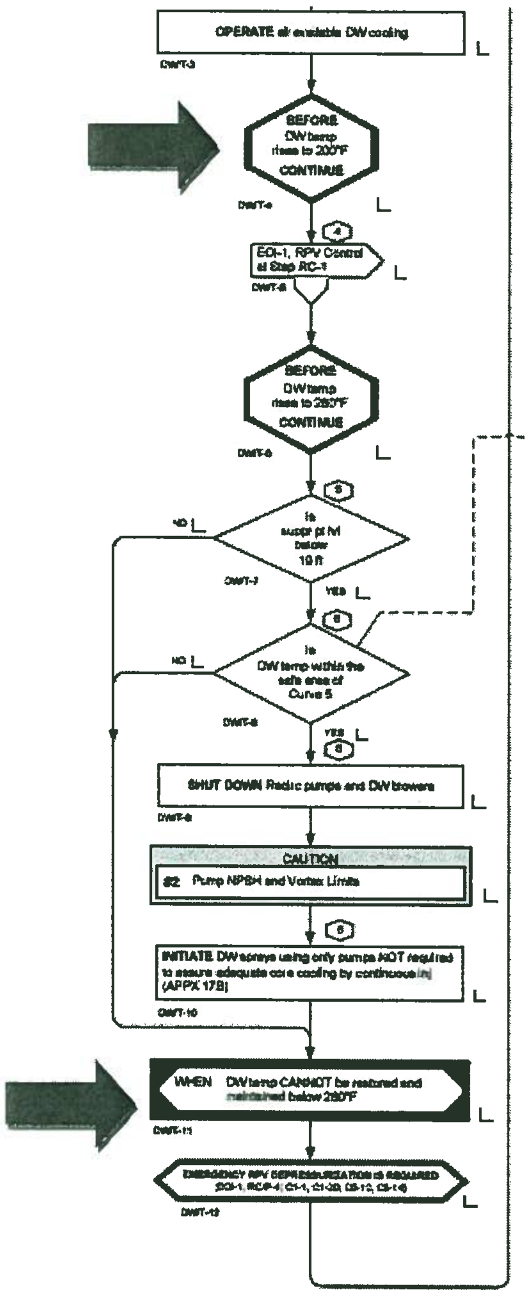
*DW/T-1.A [NRC/C] Before drywell temperature reaches **A.12** (maximum allowable drywell temperature while at power), enter the RPV Control guideline at Step RC-1 and execute it concurrently with this procedure.*

Discussion

If drywell temperature cannot be controlled by operation of all available drywell cooling, the RPV Control guideline is entered well before applicable component qualification and structural design temperature limits are reached. If drywell temperature reaches the maximum allowable drywell temperature while at power, damage may occur to non-environmentally qualified (EQ) equipment such as recirculation pumps and nuclear instrumentation components. Entering the RPV Control guideline at Step RC-1 ensures that, if possible, the reactor is scrammed before drywell sprays are initiated in Step DW/T-2 and in anticipation of possible RPV depressurization in Step DW/T-3. This helps ensure that actions are taken to limit the drywell temperature increase prior to substantially exceeding the temperature limits of non-EQ equipment.

1-EOI-2



OPERATE all available DW cooling
DWT-3
BEFORE DW temp rises to 200°F CONTINUE
DWT-4
4
EOI-1, RPV Control at Step RC-1
DWT-5
BEFORE DW temp rises to 280°F CONTINUE
DWT-6
5
Is suppr pl lvl below 19 ft
NO
YES
DWT-7
6
Is DW temp within the safe area of Curve 5
NO
YES
DWT-8
SHUT DOWN Recirc pumps and DW blowers
DWT-9
CAUTION
#2 Pump NPSH and Vortex Limits
INITIATE DW sprays using only pumps NOT required to assure adequate core cooling by continuous inj (APPX 17B)
DWT-10
WHEN DW temp CANNOT be restored and maintained below 280°F
DWT-11
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
DWT-12

Nine Mile Point 1, 2008 NRC question #47

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EK3.06	
	Importance Rating	3.4	

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : ADS

Proposed Question: Common 47

Which one of the following describes why a Blowdown is required when Drywell Temperature cannot be restored and maintained below 300°F?

To limit further release of energy into the Drywell and to ensure the...

A. RPV is depressurized while the ERVs are still operable.

B. RPV is depressurized prior to Recirc Pump seal damage.

C. reliability of all RPV level instrumentation as the RPV is depressurized.

D. reliability of ONLY the Fuel Zone RPV level instrumentation as the RPV is depressurized.

Proposed Answer: A

Explanation (Optional):

A. Correct – Per EOP Bases page 175 - If drywell temperature *cannot* be restored and maintained below 300°F, a blowdown is required. The blowdown is performed to limit further release of energy into the drywell and to ensure that the RPV is depressurized while the ERVs are still operable and before temperature rises high enough to damage the drywell.

B. Incorrect – Recirc Pump seal damage is not the concern
C. Incorrect – not the reason per EOP bases although at higher temperatures indicated vs actual level varies
D. Incorrect – not the reason per EOP bases although at higher temperatures indicated vs actual level varies

Technical Reference(s): EOP Bases (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

QUESTION 15

The following conditions exist for Unit 2:

At time 14:30

- Reactor is scrammed
- Reactor Water level is being maintained (+) 2 inches to (+) 51 inches with HPCI

At time 14:45

- Suppression Pool water level is 14.5 feet and lowering at 0.25 feet per minute

Which ONE of the following completes the statement below?

In accordance with BFN-ODM-4.20, Strategies for Successful Transient Mitigation, the decision to secure HPCI should FIRST be made at time ______.

A. 14:49

B. 14:52

C. 14:57

D. 15:03

Answer is: **A**

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

295030 Low Suppression Pool Water Level EK2.01 Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: HPCI: Plant-Specific

Explanation: **Answer A – CORRECT:** In accordance with ODM4.20, the decision to secure HPCI should be made when suppression pool water level is at 13.5 ft and lowering. At time: 14:49 suppression pool water level would be at 13.5 ft.

B – Incorrect – At time 14:52, the suppression pool water level would be at 12.75 ft. HPCI must be secured at 12.75 ft regardless of adequate core cooling in accordance with 2-EOI-2.

C – Incorrect –At time 14:57, the suppression pool water level would be 11.5 ft. In accordance with 2-EOI-2 before suppression pool water level drops to 11.5 ft, 2-EOI-1 should be entered at step RC-1. In addition, if suppression pool water level cannot be maintained >11.5 ft then emergency depressurization is required.

D– Incorrect – At time 15:03 the suppression pool water level would be at 10 ft. In accordance with EOI Caution #2, the vortex limit for Core Spray and RHR pumps is 10 ft.

Technical Reference(s): BFN-ODM-4.20, 2-EOI-2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.065 Obj. 19

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

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

4.7.4 Primary Containment Control (EOI-2) (continued)

When conditions are met that require securing drywell blowers, the time between securing the drywell blowers and initiation of drywell sprays should be minimized. A prolonged time frame with drywell blowers off and no sprays amplifies the containment pressure problem. If sprays cannot be established, consider restarting the drywell blowers.

C. SP/T Leg of flowchart

It is expected that the UO monitor Suppression Pool Temperature and notify the US of any adverse trend.

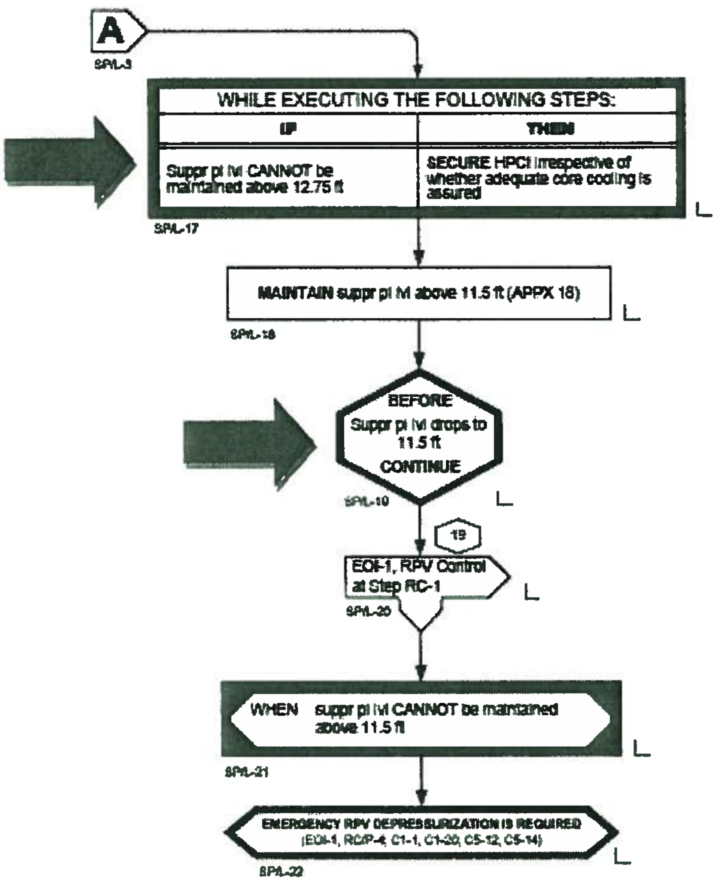
If Suppression Pool Temperature cannot to be maintained below 95°F then OPERATE ALL available loops of Suppression Pool Cooling not required for adequate core cooling.

D. SP/L Leg of flowchart

It is expected that the UO monitor Suppression Pool Level and notify the US of any adverse trend.

If Suppression Pool Level cannot be maintained above 12.75 feet, then Secure HPCI irrespective of adequate core cooling. The decision to secure HPCI should be made with Suppression Pool Level at 13.5 feet and lowering.

2-EOI-2



CAUTIONS

CAUTION # 2

Operating pumps with suction from the suppression pool above the NPSH Limit (Curve 1 or 2) or with suppression pool water level below 10 ft (Vortex limit) may cause equipment damage

CAUTION # 4

Reducing PC press will reduce the available NPSH for pumps taking suction from the suppr pl

QUESTION 16

The following conditions exist on Unit 2:

- A LOCA has occurred
- Reactor pressure is 400 psig and stable
- Reactor water level as indicated on Post-accident Flood Range, 2-LI-3-52 and 2-LI-3-62, is (-) 190 inches

Which ONE of the following completes the statements below?

[REFERENCE PROVIDED]

The reason correction curves are required to be used for 2-LI-3-52 and 2-LI-3-62 is because the level indicators are __(1)__. The top of active fuel __(2)__ submerged at this time.

A. (1) temperature compensated
(2) is

B. (1) temperature compensated
(2) is NOT

C. (1) calibrated at 0 psig
(2) is

D. (1) calibrated at 0 psig
(2) is NOT

Answer is: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295031 EK3.02	
	Importance Rating	4.4*	

295031 Reactor Water level EK3.02 Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL : Core coverage

Explanation: **Answer C – CORRECT:** Part 1-CORRECT- 2-LI-3-52 and 2-LI-3-62 are calibrated at a reactor pressure of 0 psig. Part 2-CORRECT- The indicated parameters place corrected water level above TAF.

A – Incorrect –Part 1- Incorrect- This is plausible because narrow range instrumentation is temperature compensated. Part 2- Correct-See C.

B– Incorrect – Part 1- Incorrect See A. Part 2- Incorrect- this is plausible because the chart can be misinterpreted a number of ways (see below for one example of using the chart backwards) rendering water level less than TAF.

D– Incorrect – Part 1-Correct-see C. Part 2-Incorrect- see B.

Technical Reference(s): PIP-95-64 rev 16 “2-LI-3-52 & 62 CORRECTION CURVES”

Proposed references to be provided to applicants during examination: PIP-95-64 rev 16 “2-LI-3-52 & 62 CORRECTION CURVES”

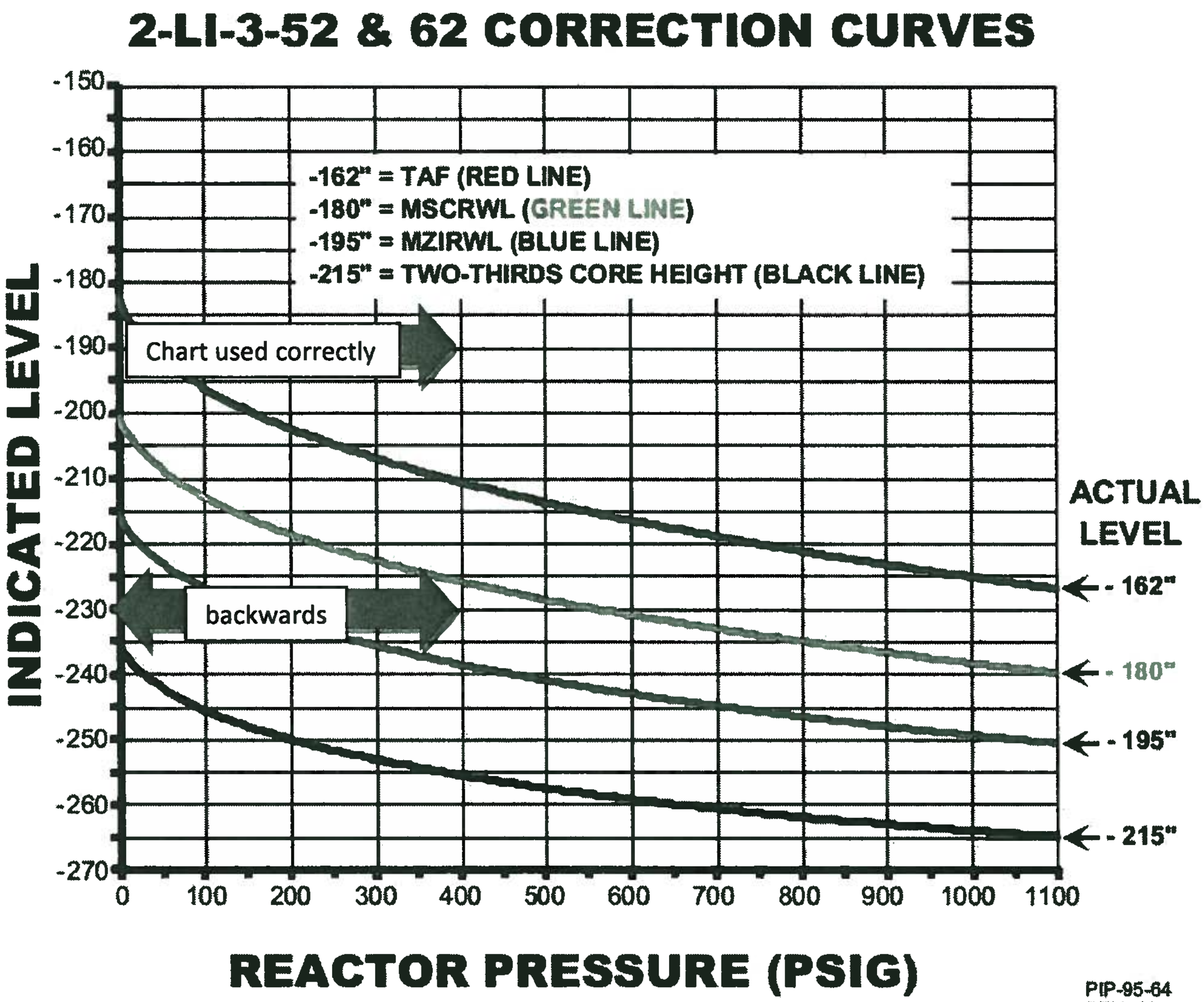
Learning Objective (As available): OPL171.201 OBJ. V.B.10

Question Source:	Bank:	
	Modified Bank:	X
	New	

Question History: Previous NRC: Cooper 2008 NRC Question #17

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

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


2-LI-3-52 & 62 CORRECTION CURVES
-162" = TAF (RED LINE)
-180" = MSCRWL (GREEN LINE)
-195" = MZIRWL (BLUE LINE)
-215" = TWO-THIRDS CORE HEIGHT (BLACK LINE)
Chart used correctly
backwards
INDICATED LEVEL
-150
-160
-170
-180
-190
-200
-210
-220
-230
-240
-250
-260
-270
ACTUAL LEVEL
- 162"
- 180"
- 195"
- 215"
0
100
200
300
400
500
600
700
800
900
1000
1100
REACTOR PRESSURE (PSIG)

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.

What, if anything, ensures Adequate Core Cooling at this time?

Adequate core cooling...

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.

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.

Question Number	New, Modified or Bank	Rev #	Revision Date	Last Used Date	Exam Bank	Applicability	
NRC RO 17	Modified 5340	01	02/02/2004	01/30/2008	NRC Style Question	RO: SRO: NLO:	Y Y N

Difficulty Level	Cognitive Level	Point Value	Response Time	Question Type	Inactive?
3	3	1	4	Multiple Choice	

Topic Area	Description
Emergency Operating Procedures	INT0080609, FLOWCHART 1A - RPV LEVEL

Related Lessons
INT0080609 OPS EOP FLOWCHART 1A - RPV CONTROL, RPV LEVEL

Related Objectives	
INT00806090010100	Describe the three mechanisms specified in the EOPs to assure adequate core cooling including the RPV water level band required and which is the preferred method.

Related References
10CFR55.41(b) 7

Related Skills (K/A)	
295031.EK3.02	Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.5 / 45/6) Core coverage (4.4 / 4.7)

QUESTION 17

An ATWS has occurred on Unit 2.

The following conditions exist:

- SLC pump 2A has been initiated
- SLC tank level indicates 65%
- Reactor water level is (-) 55 inches

Which ONE of the following completes the statement below?

The __(1)__ Shutdown Boron Weight has been injected at this point, and reactor __(2)__ while continuing with SLC addition.

A. (1) Cold
(2) cooldown can commence

B. (1) Cold
(2) water level can be raised

C. (1) Hot
(2) cooldown can commence

D. (1) Hot
(2) water level can be raised

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295037 EK1.04	
	Importance Rating	3.4	

295037 EK1.04 Knowledge of the operational implications of the following concepts as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Hot Shutdown boron weight: Plant-specific

Explanation: Answer **D – CORRECT** Engineering calculations have determined that when contents of the SLC tank have been injected into the RPV to a SLC tank level of 67% (Hot Shutdown Boron Weight), the reactor will remain subcritical irrespective of control rod position, when RPV water level is raised to uniformly mix injected boron. The Hot Shutdown Boron Weight (HSBW) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions. When an amount of boron sufficient to shutdown the reactor has been injected into the RPV, mixing is accomplished by raising RPV water level, thereby raising natural circulation flow through the vessel.

A – Incorrect – First Part: Incorrect. This is plausible if the candidate confuses the hot and cold shutdown boron weight percentages. Second Part: Incorrect. Plausible because a reactor cooldown can commence when cold shutdown boron weight has been injected.

B – Incorrect – First Part: Incorrect. This is plausible if the candidate confuses the hot and cold shutdown boron weight percentages. Second Part: Correct- see D.

C– Incorrect – First Part: Correct- see D. Second part Incorrect- see A.

Technical Reference(s): EOI 1-C-5, EOIPM SECTION 0-V-K

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.205 6.b

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

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

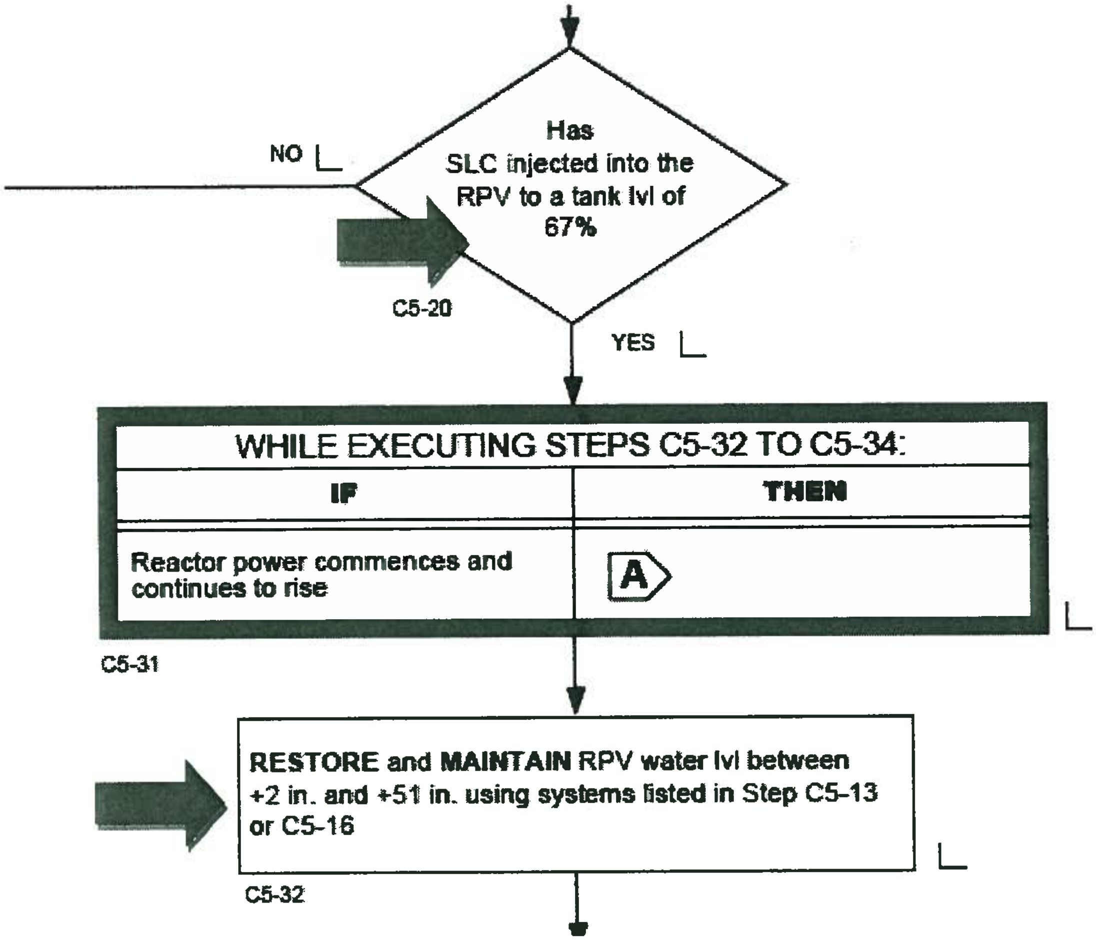
1.0 CONTINGENCY #5, LEVEL/POWER CONTROL BASES (continued)

DISCUSSION: C5-20

With boron injected into the lower plenum, little natural circulation and boron mixing occur if RPV water level is lowered to and maintained near the Minimum Steam Cooling RPV Water Level. Three-dimensional scale model tests indicate that the injected boron concentrates in the lower plenum and does not contribute to reactor shutdown until in-core distribution (mixing) is achieved. When an amount of boron sufficient to shut down the reactor has been injected into the RPV, mixing is accomplished by raising RPV water level in Step C5-32, thereby increasing natural circulation flow through the vessel.

The Hot Shutdown Boron Weight (HSBW, **A.72**) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions. The HSBW is utilized to assure the reactor will be shutdown irrespective of control rod position when RPV water level is raised to uniformly mix the injected boron. Refer to EOIPM Section 0-II-ZB for discussion of the HSBW.

When an amount of boron equivalent to the HSBW has been injected, RPV water level is restored to and maintained within the normal operating range. As RPV water level is increased, natural circulation flow is increased and the boron which has accumulated in the lower plenum is quickly mixed and distributed throughout the core region. This phenomenon is known as "boron remixing," thereby distinguishing it from any mixing which may have occurred in the early phase of the transient when some core flow was present.


Has SLC injected into the RPV to a tank lvl of 67%
NO
YES
C5-20
WHILE EXECUTING STEPS C5-32 TO C5-34:
IF
THEN
Reactor power commences and continues to rise
A
C5-31
RESTORE and MAINTAIN RPV water lvl between +2 in. and +51 in. using systems listed in Step C5-13 or C5-16
C5-32

OPL 171.205 Question #68

An ATWS has occurred on Unit 2, and SLC A has been initiated.
When 67% of the SLC tank is indicated, the Unit Operator restores and maintains water level +2 to +51 inches.

What is the significance of this step?

A. Hot Shutdown Boron Weight has been injected at this point and reactor water level is raised to aid in mixing.

B. SLC injection should be stopped until Rx level is raised to mix the boron already injected before cooldown may continue.

C. Cold Shutdown Boron Weight has been injected at this point, cooldown may commence.

D. Boron concentration in the botton head would reach the point where the boron could solidify with out additional RPV water.

ANSWER : A

QUESTION 18

Given the following conditions:

- A Liquid Effluent Discharge is in progress in accordance with 0-SI-4.8.A.1-1, Liquid Effluent Permit.

Which ONE of the following completes both statements below?

Consider each statement separately.

The 0-FCV-77-58A/B, RADWASTE LOW/HIGH FLOW RATE DISCHARGE ISOLATION VALVES will automatically close due to Radwaste Effluent Radiation __(1)__.

A HIGH liquid effluent release rate __(2)__ require entry into 0-EOI-4, Radioactivity Release Control.

A. (1) High-High ONLY
(2) does

B. (1) High-High ONLY
(2) does NOT

C. (1) High-High or Downscale
(2) does

D. (1) High-High or Downscale
(2) does NOT

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295038 EK2.06	
	Importance Rating	3.4	

295038 EK2.06 Knowledge of the interrelations between High Off-Site Release Rate and the following: Process Liquid radiation monitoring system

Explanation: **Answer D – CORRECT** First part correct HIGH-HIGH is correct but not the only signal from the radiation monitor that will isolate the FCV-77-58A/B Valves. Second Part correct EOI-4 entry is only required for an alert classification due to a gaseous release.

A – Incorrect – First Part: Incorrect. Plausible in that only a high-high radiation condition would isolate the discharge release path, especially if they know an inoperative condition is also an isolation of the path. Second Part: Incorrect. Plausible because the purpose of 0-EOI-4 is Radioactive Release Control.

B – Incorrect – First Part: Incorrect-See A. Second Part: Correct See D.

C – Incorrect – First Part: Correct-See D. Second Part: Incorrect See A.

Technical Reference(s): 0-EOI-4, ARP-9-3A, 0-SI-4.8.A.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.084 B.6, OPL171.033 B.4.b

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

10 CFR Part 55 Content: 55.41 (11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

The 0-RM-90-130 isolation logic closes 0-FCV-77-58A and B, 0-FCV-77-61 (on all units) and 0-FCV-77-279 on Hi-Hi Rad, Downscale and Inoperative signals.

(4) RADWASTE EFFLUENT RADIATION HIGH (55-3A-16) High and High-High setpoints are set by 0-SI-4.2.D.1 prior to each radwaste discharge.

(a) Alarm generated by radiation monitor (RM-90-130)

(b) High setpoint causes only causes alarm

(c) High-High setpoint or INOP causes the following valves to close:

1) 1/2/3-77-61
2) 0-77-58A/58B
3) 0-77-279

(5) RADWASTE EFFLUENT RADIATION MONITOR DNSCL / INOP (55-3A-23) Alarms when low detector (RM-90-130) output is sensed

(a) Alarm generated by radiation monitor (RM-90-130)

(i) FCV 77-61 - Radwaste discharge isolation auto closure on Radiation monitor $\geq$ upscale isolation setpoint, downscale, or Inop, Unit specific 1A gate is not full open, or two CCWP's are not operating

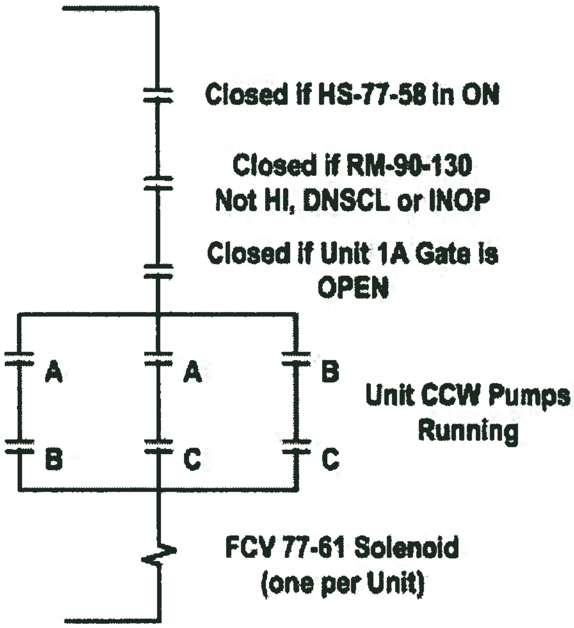
(ii) FCV 77-58B radwaste high flow rate discharge isolation valve auto closure on Radiation monitor $\geq$ upscale isolation setpoint, downscale, or Inop or all 3 unit 1A gates are closed.

(iii) FCV 77-58A radwaste low flow rate discharge isolation valve auto closure on Radiation monitor $\geq$ upscale isolation setpoint, downscale, or Inop or all 3 unit 1A gates closed

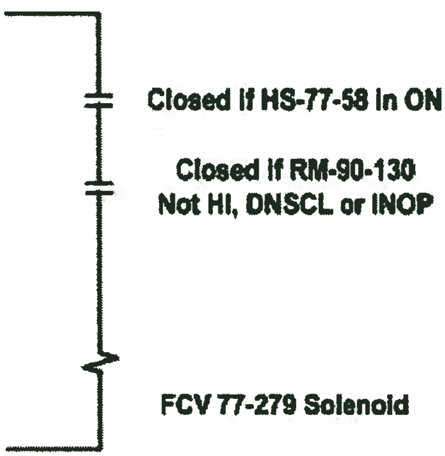
(iv) FCV 77-279 radwaste isolation valve to cooling tower blowdown auto closure on Radiation monitor $\geq$ upscale isolation setpoint, downscale or Inop.

0-EOI-4, Radioactive Release Control

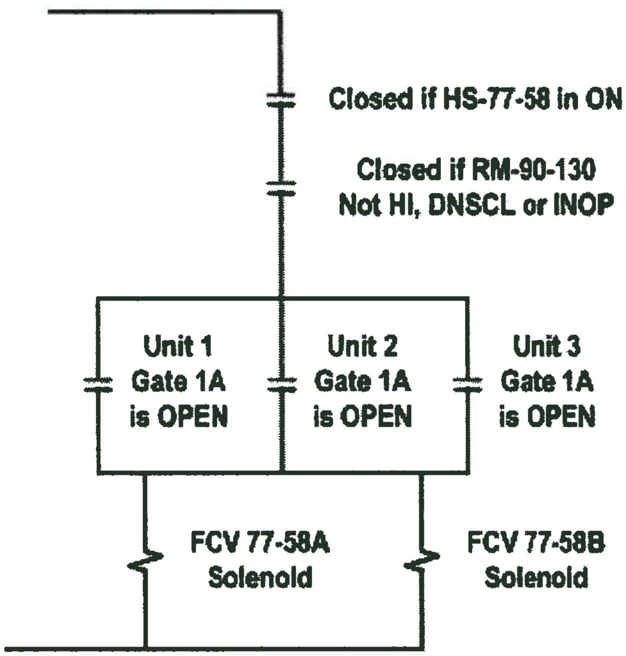
Gaseous offsite radioactivity release rate at or above that requiring an Alert (EPIP-1 Table 4.1-A)



FCV 77-61 LOGIC



FCV 77-279 LOGIC



FCV 77-58A and 77-58B LOGIC

RADWASTE
EFFLUENT
RADIATION
HIGH
0-RA-90-130A

16

(Page 1 of 1)

Sensor/Trip Point:

	Hi-Hi	Hi
0-RM-90-130	+ BKGD CPS (NOTE 1)	+ BKGD CPS (NOTE 1)

(1) CONTACT Instrument Shop for the most current setpoints per 0-SI-4.2.D.1, (Setpoints are listed in Radwaste on 0-RR-90-130).

The current isolation setpoint for effluent discharge will be listed on the Radwaste Panel recorder, 0-RR-90-130. Isolation occurs on Hi-Hi.

Sensor Location: 0-RE-090-0130, Radwaste discharge piping (an in-line Scintillation detector) to CCW discharge, El 565, Radwaste Bldg.

Probable Cause:

A. High radiation detected in effluent being discharged.
B. Contamination buildup in detector wetwell.
C. High background in area of detector.
D. Sensor malfunction.

Automatic Action:

A. Hi - Alarm 0-RA-90-130A only.
B. Hi-Hi - Isolation of valves, 1,2,3-FCV-77-61, and 0-FCV-77-58A, 58B and 279.

RADWASTE EFFL RADIATION MONITOR DOWNSCALE 0-RA-90-130C 23

(Page 1 of 1)

Sensor/Trip Point:

0-RM-90-130 Low detector output

Sensor Location: 0-RE-90-130, Radwaste discharge piping to river, El. 565', Radwaste Building.

Probable Cause:
A. Sensor malfunction.
B. Loss of power (24V DC Bus A) to detector.

Automatic Action: Isolation of valves 1,2,3-FCV-77-61 and 0-FCV-77-58A, 58B, and 279.

QUESTION 19

The following conditions exist:

- Electric Fire Pump "A" has automatically started due to a fire at the Common Station Service Transformer "A"
- FIRE PUMP SELECTOR SWITCH, 0-XS-26-43, on Panel 1-9-20 is in Position "A-B-C"
- Fire header pressure as sensed at 0-PS-26-44 is 95 psig

Which ONE of the following completes the statement below?

If fire header pressure as sensed at 0-PS-26-44 remains at 95 psig for 40 seconds, ______will be running.

A. ONLY the Electric Fire Pump "A"

B. ONLY the Electric Fire Pumps "A" and "B"

C. ONLY the Electric Fire Pumps "A", "B", and "C"

D. the Diesel Fire Pump and ALL Electric Fire Pumps

Answer is: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	600000 AK2.04	
	Importance Rating	2.5	

600000 Plant Fire On Site AK2.04 Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Breakers / relays / and disconnects

Explanation: **Answer C – CORRECT:** Only the Electric Fire Pumps A, B, and C will be running. EFP A will auto-start based on the transformer deluge actuation. Since header pressure remains at 95 psig, EFP B will automatically start based on time delay relay TD1 after 15 seconds, and EFP C will automatically start based on time delay relay TD2 after 30 seconds. The DFP is not running (TD3 will start the DFP at header pressure <120 psig for 45 seconds).

A – Incorrect –Since 95 psig is greater than the normal fire header pressure (approx 75 psig), it is plausible to believe that only the EFP A would be running due to its automatic start on deluge actuation.

B – Incorrect –If EFP auto starts were based on pressure only (and not the actuation of the deluge in this case for starting EFP A), it is plausible to believe that relays TD1, and TD2 would start EFP's A and B.

D – Incorrect – All of the EFP's would be running (see Answer C). Although header pressure less than 95 psig would eventually start the DFP, this happens based on TD3 after 45 seconds; therefore, the DFP is not running.

Technical Reference(s): 0-OI-26 Rev 95; DWG #0-45E644-1 Rev 26

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

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

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

Date ________

5.1 Automatic Start of a Fire Pump (continued)

D. Raw Service Water Pumps trip. ☐

E. Raw Service Water Storage Tank Isolation Valve 0-FCV-25-32, and Raw Head Tank Isolation Valve 0-FCV-25-70, close. (Panel 1-9-20). ☐

NOTES

1) 15 seconds after the initiating signal, if system header pressure is less than 120 psig the second selected Fire Pump starts.

2) 30 seconds after the initiating signal if, system header pressure is less than 120 psig the third selected Fire Pump starts.

3) 45 seconds after the initiating signal if, system header pressure is less than 120 psig the Diesel Driven Fire Pump starts.

[3] PERFORM the following Operator Actions:

[3.1] VERIFY all appropriate automatic actions have occurred. ☐

[3.2] NOTIFY the Intake AUO to perform Section 6.2 to flush Emergency Fire Pump Strainers. ☐

[3.3] LOG in Shift Turnover Checklist for the AUO to CONTINUE flushing of Fire Pump Strainers once per shift as long as the Fire Pumps are in operation. ☐

DWG #0-45E644-1

HIGH PRESSURE FIRE PROTECTION SYSTEM OPERATION

TEMPERATURE DETECTORS ARE PROVIDED TO ANNUNCIATE AND INITIATE FIRE PROTECTION UPON DETECTING AN ABNORMAL TEMPERATURE (SEE DWG 1-45N1625-1, 1-45N1625-2, 0-45N1625-3 & 4). THESE DETECTORS ARE ARRANGED IN CLUSTERS OF THREE IN THE REACTOR AND TURBINE BUILDINGS. IF ANY TWO OF THE THREE DETECTORS FOR THE FOG (ANY ONE FOR THE SPRAY SYSTEM) CLOSE, AN AUXILIARY RELAY WILL BE ENERGIZED. THIS RELAY WILL SEAL IN, OPEN THE SOLENOID VALVE TO RELEASE WATER (FOG OR SPRAY SEE NOTE 2) TO THE AFFECTED AREA, AND START THE FIRE PUMP. (SEE DWG 2-45E765-7 FOR THE FIRE PUMPS SCHEMATIC). THE PUMP STARTING SEQUENCE (ABC, BCA, OR CAB) IS CONTROLLED BY SELECTOR SWITCH 0-XS-26-43 ON PNL 9-20 IN THE UNIT 1 CONTROL ROOM. NORMAL PRESSURE CAUSED BY RAW WATER STORAGE TANK IS APPROXIMATELY 75 PSI AT THE FIRE PUMPS DISCHARGE. DESIGN PRESSURE WITH THE FIRE PUMP OR PUMPS RUNNING IS APPROXIMATELY 130 PSI AT THE FIRE PUMPS DISCHARGE. IF THE PROPER HEADER PRESSURE IS NOT MAINTAINED, TIME-DELAY RELAYS TD1 AND TD2 ARE ENERGIZED THROUGH THE FIRE PUMP DISCHARGE PRESSURE SWITCH PS-26-44 WHICH OPERATES AT APPROXIMATELY 100 PSI. AFTER 15 SECONDS, TD1 WILL START THE SECOND FIRE PUMP. AFTER 30 SECONDS, TD2 WILL START THE THIRD FIRE PUMP. IF ANY DIESEL GENERATOR IS OPERATING AND AN ACCIDENT SIGNAL IS RECEIVED, RELAY LOR1 OR LOR2 CONTACTS WILL PREVENT STARTING OF FIRE PUMPS A, B, AND C THROUGH RELAYS FPXA, FPXB, & FPXC. A DIESEL DRIVEN FIRE PUMP WILL START AFTER 45 SECONDS TIME DELAY THROUGH RELAY TD3. A CONTACT FROM TD3 OPERATES LATCHING RELAY DPS, WHICH MAINTAINS THE DIESEL FIRE PUMP CIRCUIT IN THE START POSITION UNTIL MANUALLY RESET BY 0-HS-26-106A2. THIS PREVENTS INADVERTENT MANUAL TRIPPING OF THE DIESEL FIRE PUMP BY REQUIRING TWO MANUAL OPERATIONS (SEE TVA SCHEMATIC OF DIESEL FIRE PUMP DWG 0-45E646).

PROTECTION SYSTEM

QUESTION 20

Given the following conditions:

- Unit 1 is operating at 80% with 100 VARS incoming
- The crew has entered 0-AOI-57-1E, Grid Instability
- System voltage is 508 kV
- System frequency is 60.03 hz

Which ONE of the following completes the statements below?

Under these conditions, the Main Generator Under-excited Reactive Ampere Limit (URAL) circuit helps prevent the possibility of __(1)__.

The operators will coordinate between the units and, as directed by 0-AOI-57-1E, Grid Instability, will __(2)__ power.

A. (1) the generator slipping a pole
(2) lower reactor

B. (1) the generator slipping a pole
(2) raise reactive

C. (1) exceeding heating limits on the rotor
(2) lower reactor

D. (1) exceeding heating limits on the rotor
(2) raise reactive

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	700000 AK1.03	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND GRID DISTURBANCES: Under-excitation

Explanation: **B** CORRECT –The Main Generator Under-excited Reactive Ampere Limit (URAL) circuit helps prevent the possibility of the generator slipping a pole. In the conditions given in the stem, voltage is low and frequency is within the normal range of 60$\pm$.05.With voltage low, 0-AOI-57-1E directs the operator to raise voltage (reactive power).

A Incorrect – First Part: Correct. Second Part: Incorrect. Plausible if the candidate confuses MVARS and MW adjustments: Power (speed) would be adjusted for a frequency change.

C Incorrect – First Part: Incorrect. Plausible because one of the causes of Generator Field overheating is excessive reactive load. Second Part: Incorrect-See A.

D Incorrect – First Part: Incorrect. Plausible because one of the causes of Generator Field overheating is excessive reactive load. Second Part: Incorrect. Plausible if the candidate confuses MVARS and MW adjustments: Power (speed) would be adjusted for a frequency change.

Technical Reference(s): 0-AOI-57-1E, 1-OI-47, OPL171.134, 1-ARP-9-8A window 2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New
Question History:	Previous NRC: BFN 1306 #20
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X

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

4.2 Subsequent Action (continued)

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

PERFORM the following steps:

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

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

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

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

PERFORM the following:

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

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

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

Illustration 7
(Page 1 of 1)

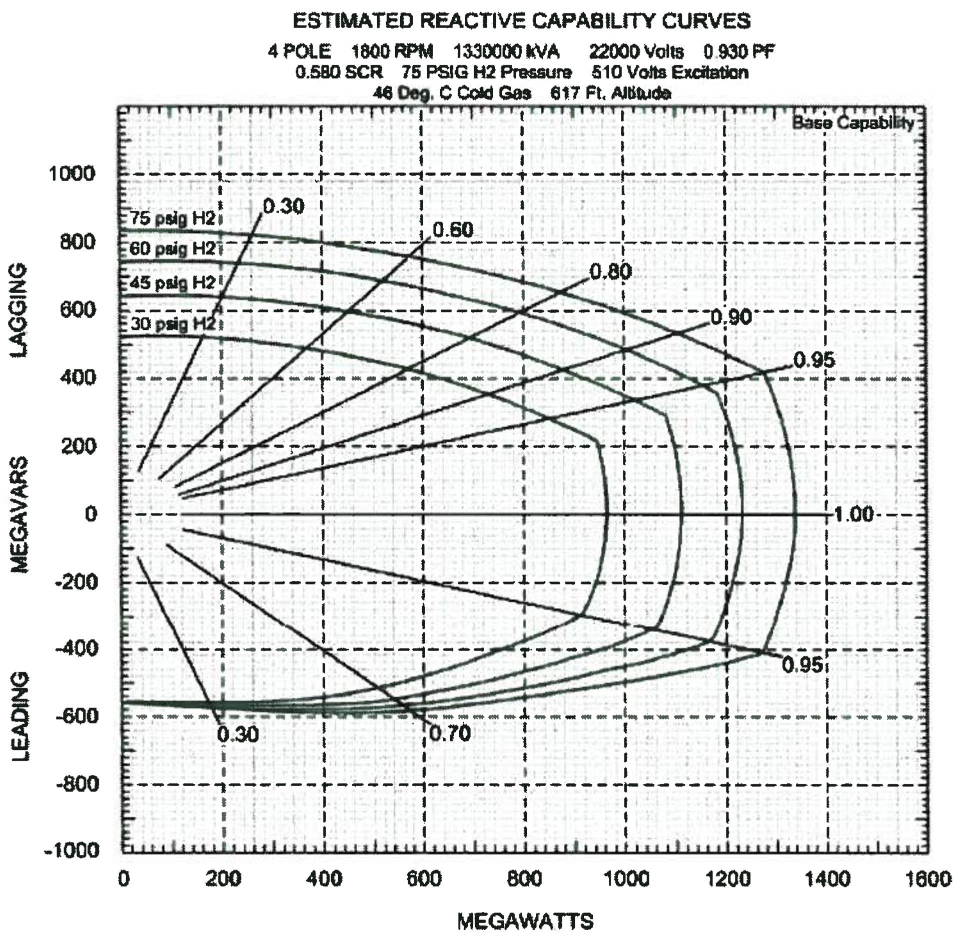
Generator Kilovar Limitations (Capability Curve)

NOTES

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

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

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



(a) It is equipped with an under-excited reactive ampere limit circuit (URAL) circuit. Its purpose is to prevent the field from weakening so much that the generator could commence slipping poles. This circuit senses Main Generator output voltage and current and develops an output when incoming VARS reach the limit. This output is applied to the AC regulator output signal to hold it and cause Main Generator field to remain at the limiting generator voltage and current.

Pole slips occur when the generator field becomes too weak to 'pull' all the turbine's power into the generator armature current. Such a weak field uncouples from the armature field slipping 1/4 revolution behind repeatedly.

GEN FIELD TEMPERATURE HIGH 1-TA-57-131 2

(Page 1 of 2)

Sensor/Trip Point:

.

1-TR-57-59   95°C

Sensor Location: GEN FIELD/STATOR AND EXCITER WINDING TEMP, 1-TR-57-59
Panel 1-9-8
Control Room

Probable Cause:

A. Excessive reactive load
B. H_2 coolers malfunction
C. Low H_2 pressure
D. Generator output phase imbalance
E. Generator exciter circuit malfunction (loss of field)
F. Failed sensor.
G. Generator Field breaker open (post trip).
H. Unit Startup in progress with the Gen Field Breaker Open.

Automatic Action: None

Operator Action:

NOTE
If unit startup is in progress with the Gen Field Breaker still open, then when the Generator Field Breaker is closed per 1-OI-47, the indication should come on scale and the alarm clear.

A. **CHECK** the following on Panel 1-9-8 to verify alarm or identify probable cause:
   1. GENERATOR FIELD TEMP, 1-TI-57-45. ☐
   2. GENERATOR MVARS, 1-EI-57-51 ☐
   3. EXCITER TEMPERATURE computer points, 35-83-2, 35-83-4 and 35-83-6. ☐
   4. GEN HYDROGEN PRESSURE, 1-PI-35-17A
   5. GEN HYDROGEN INLET TEMP, 1-TI-35-31 ☐
   6. GENERATOR FIELD TEMPERATURE computer point, TBD015 ☐

B. **DISPATCH** personnel to Stator Cooling Unit and Alterex air cooled housing unit to investigate the high temperature cause or status. ☐

C. **REFER TO** 1-OI-35. ☐

BFN ILT 1306 NRC #20

QUESTION 20

Given the following conditions:

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

Which ONE of the following completes the statements below?

The operators will coordinate between the units and, as directed by 0-AOI-57-1E, Grid Instability, lower___(1)___.

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

A. (1) reactor power approximately 1% per minute using the recirc master control buttons (LOWER SLOW 1, 2, 3-HS-96-33 or LOWER MEDIUM 1, 2, 3-HS-96-34)
(2) rise (closer to 1.0)

B. (1) reactor power approximately 1% per minute using the recirc master control buttons (LOWER SLOW 1, 2, 3-HS-96-33 or LOWER MEDIUM 1, 2, 3-HS-96-34)
(2) drop (further from 1.0)

C. (1) reactive power using VOLTAGE REGULATOR RAISE LOWER ADJUST (1, 2, 3 HS-57-26)
(2) rise (closer to 1.0)

D. (1) reactive power using VOLTAGE REGULATOR RAISE LOWER ADJUST (1, 2, 3 HS-57-26)
(2) drop (further from 1.0)

Answer: **C**

QUESTION 21

Unit 3 is at 26% power with the following conditions:

- The attached indications are observed on Reactor Water Level Narrow Range instruments

Which ONE of the following completes the statement below?

Reactor Feed Pump Turbines are __(1)__ and the Main Turbine is __(2)__.

See Attached 208 Instrument Indications on next page:

A. (1) operating
(2) operating

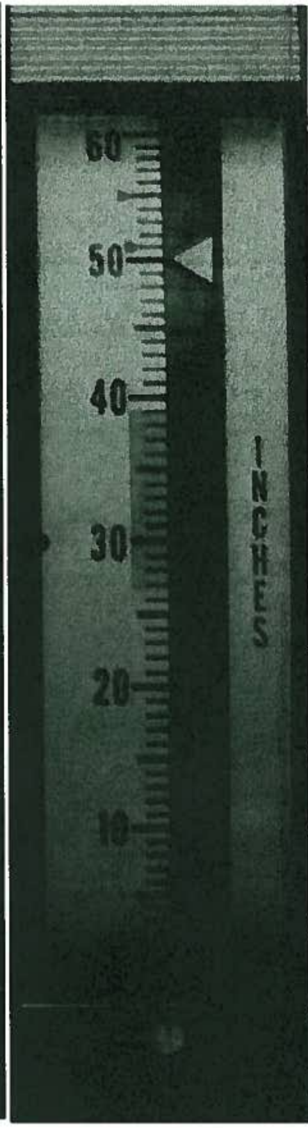
B. (1) operating
(2) tripped

C. (1) tripped
(2) operating

D. (1) tripped
(2) tripped

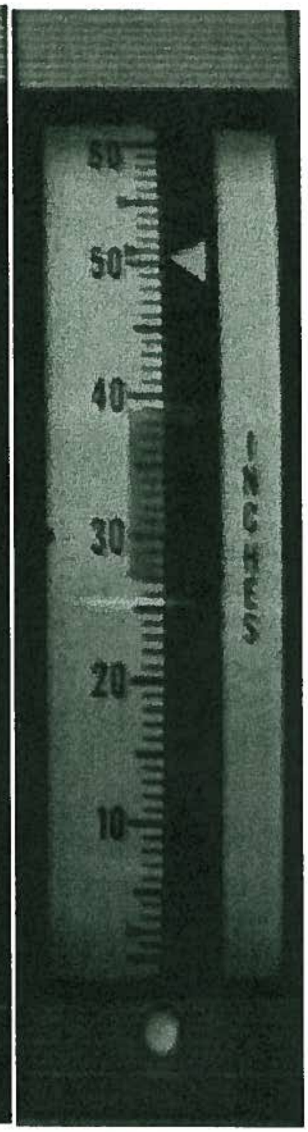
Answer: **D**

3-LI-3-208A



3-LI-3-208B

3-LI-3-208C



3-LI-3-208D

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<tr><td>Tier #</td><td>1</td><td></td></tr>
<tr><td>Group #</td><td>2</td><td></td></tr>
<tr><td>K/A#</td><td colspan="2">295008 AK2.02</td></tr>
<tr><td>Importance Rating</td><td>3.6</td><td></td></tr>
</table>

295008 AK2.02 Knowledge of the interrelations between High Reactor Water Level and the following: Reactor Feedwater System

Explanation: **Answer D – CORRECT** The 208A and 208C Level instruments greater than 55 inches the RFPTs and the Main Turbine Trip. The logic is 2 out of 2 taken once.

A – Incorrect – Plausible in that the 208A and 208C instrument also trip RCIC and the 208B and 208D instruments also trip HPCI. If they do not fully understand the instrument configuration and the logic may consider that the RFPTs or the main turbine may be operating or tripped

B – Incorrect – see A above

C – Incorrect – see A above

Technical Reference(s): 3-OI-3 and 3-OI-47

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.026 V.B.5

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

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

3.5.1 Automatic 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 $\geq$ +55 inches.

M. Reactor Feedwater Pump Turbines will trip on the following:

Reactor Vessel High Water Level (55" as sensed by LT-3-208A and 208C or 208B and 208D)

r. High Water Level Trip

1) High water level trip at 55" comes off of LS-3-208A, B, C, D.

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.

QUESTION 22

Given the following conditions for Unit 2:

- An ATWS has occurred
- Reactor power is currently 10%
- SLC tank level is 80%
- Emergency Depressurization is required due to Secondary Containment temperatures

In accordance with 2-EOI C-5, Level/Power Control, and 2-C-2, Emergency Depressurization, which ONE of the following completes the statements below?

Prior to Emergency Depressurization, __(1)__ are required to be secured.

Following Emergency Depressurization, RPV water level is INITIALLY restored using __(2)__.

A. (1) CS, LPCI, and HPCI
(2) ALL available injection sources

B. (1) CS, LPCI, and HPCI
(2) those injection sources which inject outside the shroud

C. (1) ONLY CS and LPCI
(2) ALL available injection sources

D. (1) ONLY CS and LPCI
(2) those injection sources which inject outside the shroud

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295014 G2.4.9	
	Importance Rating	3.8	

295014 Inadvertent Reactivity Addition. Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Explanation: **B** CORRECT – First part: In an ATWS prior to ED, all injection with the exception of Boron, CRD, and RCIC is stopped and prevented (step C5-22). Second part: In an ATWS following ED, RPV injection is initially restored slowly using the systems listed in step C5-25 (outside the shroud inj sources) in order to preclude core damage.

A Incorrect – First Part: Correct-See B. Second Part: Incorrect- This would be correct if an ATWS condition did not exist and level was being controlled IAW EOI-2-C-1 (Alternate Level Control) see-step C1-13

C Incorrect – First Part: Incorrect-This would be correct if an ATWS condition did not exist-see EOI-2-C-2 (Emergency Depressurization) step C2-4. Second Part: Incorrect See A.

D Incorrect – First Part: Incorrect See C. Second Part: Correct See B.

Technical Reference(s): EOI C-5,EOI C-1, EOI C-2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

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

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

2-EOI C-5 LEVEL/POWER CONTROL

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED (EOI-1, RC/P-4)

C5-21

STOP and PREVENT ALL inj into RPV EXCEPT from RCIC, CRD and SLC (APPX 4)

C5-22

Can at least two MSRVs be opened per C2, Emergency RPV Depressurization

NO

YES

C5-23

WHEN RPV press is below MSCP (Table 1A)

C5-24

CAUTION

#5 Rapid RPV inj may cause core damage

#2 Pump NPSH and Vortex limits

#3 Elevated suppr chmbr press may trip RCIC

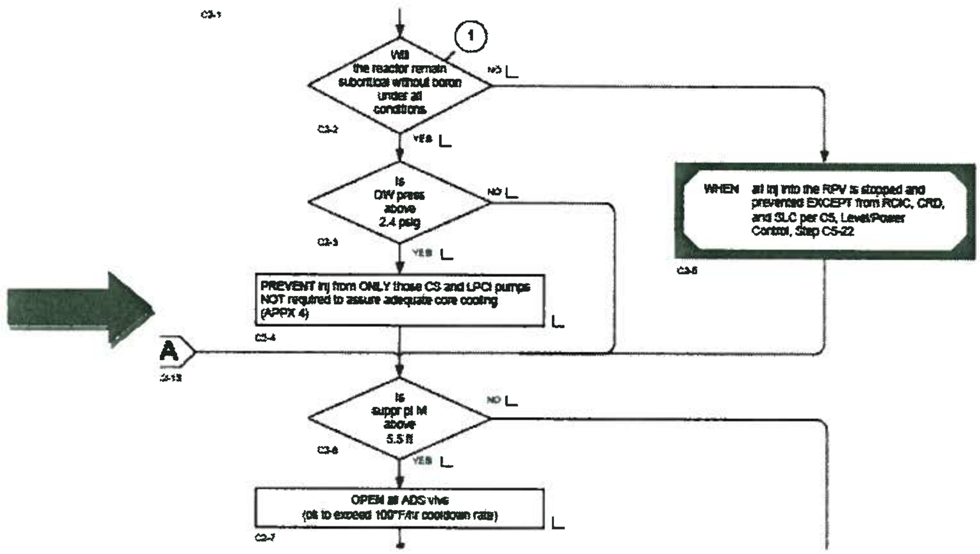
#6 HPCI or RCIC suction temp above 140°F

START and SLOWLY RAISE RPV inj with the following inj sources to restore and maintain RPV water lvl above -180 in.:

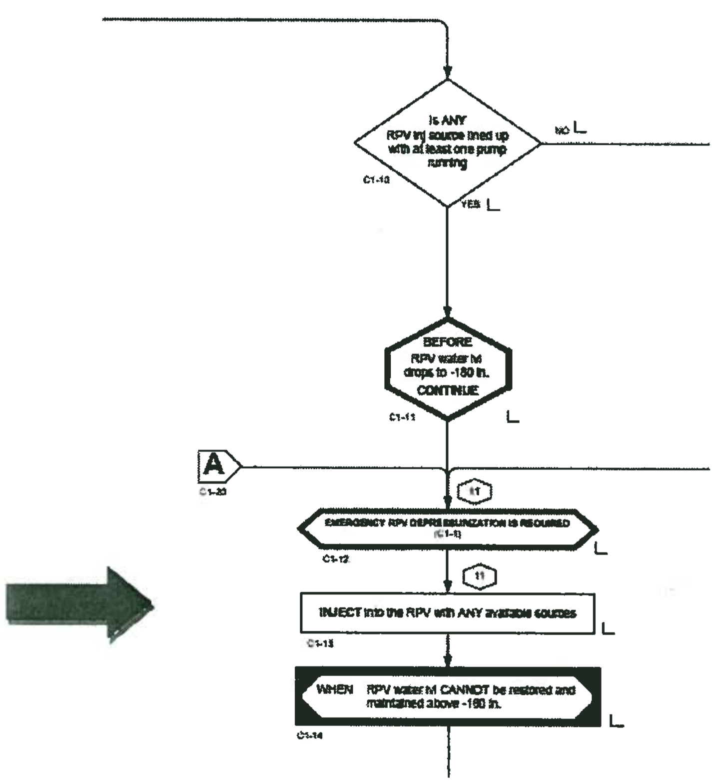
INJ SOURCE	APPX	INJ PRESS
CNDS and FW	5A	1210 psig
CRD	5B	1640 psig
RCIC with CST suction if available	5C	1200 psig
HPCI with CST suction if available	5D	1200 psig
CNDS	6A	480 psig
LPCI	6B, 6C	320 psig
SLC (boron tank)	7B	1450 psig

C5-25

2-EOI C2 EMERGENCY DEPRESSURIZATION



2-EOI C-1 ALTERNATE LEVEL CONTROL



QUESTION 23

An ATWS has occurred on Unit 2 with the following conditions:

- ATWS Actions are complete
- The MSIVs are open
- Five bypass valves are full open
- One bypass valve is partially open
- Reactor Pressure is 955 psig and stable
- Recirculation pumps are at 480 RPM
- Reactor water level is (+) 59 inches and lowering
- APRM's are unavailable

Which ONE of the following completes the statements below?

Reactor power is __(1)__ than 5%.

In accordance with 2-EOI-1, RPV Control, the recirc pumps are directed to __(2)__.

A. (1) less
(2) remain at 480 RPM

B. (1) less
(2) be tripped

C. (1) greater
(2) remain at 480 RPM

D. (1) greater
(2) be tripped

Answer is: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295015 AA2.01	
	Importance Rating	4.1	

295015 Incomplete SCRAM AA2.01: Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Reactor power

Explanation: **Answer D – CORRECT:** Reactor power is > 5% (APRM downscale) and the recirc pumps are required to be immediately tripped.
(1) Correct: Based on 5 bypass valves being full open and reactor pressure being maintained by the partially open bypass valve, reactor power is >5%. For Unit 2, nine Bypass valves are equivalent to approximately 25% reactor power. On a scram from 100% power, decay heat can initially provide as much as 6 % rated steam flow, therefore even under those conditions reactor power with 5 bypass valves full open, Power is >5%.
(2) Correct: with power above 5% EOI-1 directs immediately tripping the recirc pumps.

A – Incorrect – (1) Incorrect-The loss of electrical power to the APRMs by itself, however, does not mean that reactor power cannot be determined with respect to APRM downscale. See D (1). (2) Incorrect-If Reactor power is below 5% the Recirc pump can remain at 480 RPM. See D (2).

B – Incorrect – (1) Incorrect-see A (1). (2) Correct- see D (2).

C– Incorrect – (1) Correct-see D (1). (2) Incorrect- see A (2).

Technical Reference(s): 2-EOI-1, EOIPM 0-V-C

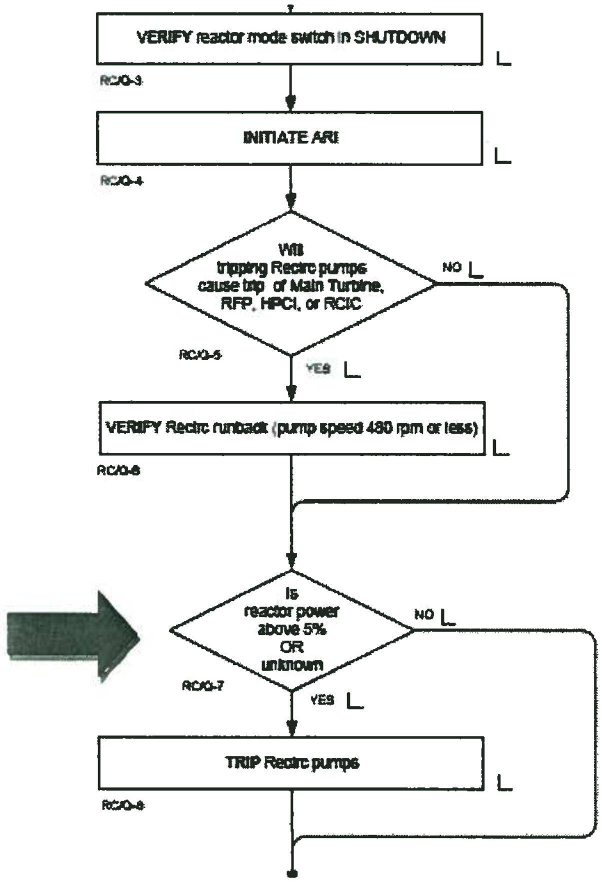
Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.202 Obj. V.B.12

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

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

2-EOI-1



VERIFY reactor mode switch in SHUTDOWN
RC/Q-3
INITIATE ARI
RC/Q-4
Will tripping Recirc pumps cause trip of Main Turbine, RFP, HPCI, or RCIC
NO
YES
RC/Q-5
VERIFY Recirc runback (pump speed 480 rpm or less)
RC/Q-6
Is reactor power above 5% OR unknown
NO
YES
RC/Q-7
TRIP Recirc pumps
RC/Q-8

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/Q-7 and RC/Q-8

If reactor power remains above the APRM downscale setpoint, the recirculation pumps are tripped to effect a prompt reduction in power. While tripping the pumps may place the plant in a high power-to-flow condition and thereby contribute to neutronic and thermal-hydraulic instabilities, continued recirculation pump operation may not be desirable or even possible:

- If RPV water level is lowered in accordance with Flowchart C5, the pumps will trip automatically when the low RPV water level trip setpoint is reached.
- Allowing reactor power to remain high would increase the flow demand on RPV injection systems and the heat load on the primary containment.

Tripping the recirculation pumps may also reduce the boron mixing efficiency if boron injection is required. However, three-dimensional scale model tests have demonstrated that natural circulation provides adequate mixing as long as RPV water level is above the elevation at which a natural circulation flowpath can be established.

If reactor power is below the APRM downscale trip setpoint, tripping the recirculation pumps results in little, if any, reduction in reactor power since power is already near the decay heat level. In this case, forced recirculation flow is continued, if possible, to enhance boron mixing if boron injection is later required.

If reactor power cannot be determined, it must be assumed to be above the APRM downscale setpoint and the recirculation pumps are tripped. The loss of electrical power to the APRMs by itself, however, does not mean that reactor power cannot be determined. The values of reactor period, steam flow, RPV pressure and pressure trend, suppression pool temperature and temperature trend, the number of open MSRVs and main turbine bypass valves, etc., may provide indications of reactor power with respect to the APRM downscale trip setpoint.

QUESTION 24

Unit 1 is at 100% power when, an inadvertent Group 1 isolation occurs.

Which ONE of the following completes the statement below?

Over the next two hours _____.

(Assuming NO Operator actions)

A. the drywell heat load is reduced, lowering drywell average temperature

B. the heat added to the suppression pool will migrate to the drywell, raising average drywell temperature

C. condensation of steam lowers the pressure in the suppression pool, opening the suppression chamber-drywell vacuum breakers

D. rising temperatures in the suppression pool will increase suppression pool pressure, opening the reactor building-suppression chamber vacuum breakers

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295020 AK3.03	
	Importance Rating	3.2	

295020 AK3.03 Knowledge of the reasons for the following responses as they apply to Inadvertent Containment Isolation: Drywell / Containment temperature response

Explanation: **Answer A – CORRECT-** (Following a reactor scram from a group I isolation, the heat load is significantly reduced in the drywell and the drywell cooling remains the same resulting in lowering drywell pressure

B – Incorrect – Plausible, as suppression pool water temperature will increase. However, Suppression pool water temperature does not directly affect drywell airspace temperature, and any temperature increase would be negated by drywell cooling.

C – Incorrect – Plausible as condensation of steam in the drywell is what causes drywell pressure to lower requiring the need for the suppression chamber-drywell vacuum breakers. However, that is condensing steam in the drywell not the suppression chamber. The reactor building-suppression chamber vacuum breakers provide protection in that case.

D – Incorrect – Plausible as the suppression chamber does relieve to the drywell in order to prevent a collapse of the drywell. However, the reactor building to suppression chamber breakers open on a vacuum in the suppression chamber.

Technical Reference(s): OPL171.016; FSAR Chapter 5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	X
	New	

Question History: Previous NRC: None

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

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

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
diameter vent pipes vent drywell to suppression chamber	
1) Inlet to vent pipes from drywell protected by jet deflectors to prevent damage to vent pipes from missiles and jet forces.	
2) Expansion joints are provided in vent pipes to allow relative motion between drywell and suppression chamber.	With dp established ~3ft. downcomer submerged has ~half its contents expelled. Operator Fundamentals SER-03-05
3) All eight vent pipes exhaust into a single 57-inch diameter vent ring header in the suppression chamber.	
4) 96 downcomer pipes extend from the vent ring header into the suppression pool below the water surface.	
e. Suppression chamber-drywell vacuum breakers	Obj. ILT-4
1) Purpose: To prevent exceeding design <u>external</u> pressures of the drywell (-2 psig). Vacuum breakers discharge from the torus (suppression chamber) to the drywell to equalize the pressure differential and to prevent backflow of water from the torus (suppression pool) into the vent header system via the downcomers.	Obj. LOR-3 Tech. Specs. 3.6.1.6 Obj. NLO-2.c, & 5 Obj. NLOR-2.c
2) Relieve from suppression chamber to drywell if there is a pressure differential greater than 0.5 psid between them.	Obj. NLO-7
3) The vacuum breakers are required when the steam in the drywell from a LOCA starts to condense.	Atmospheric collapse is amplified upon DW spray action.
4) The condensing steam could cause a vacuum to occur inside the drywell and the atmospheric pressure from the outside could collapse the containment vessel. (-2 psig external design) To prevent this collapse, the suppression chamber-drywell vacuum breakers vent air back into the drywell.	

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
pushbuttons provided.)	
a) Supplied solenoids and air supplies.	
b) Position indications on panel 9-3.	Obj. ILT-4 Obj. LOR-3
(1) Full closed - illuminated green check light on vertical board;	Describe different indications for each position for torus/DW and Rx building /torus vacuum breakers.
(2) Cracked open - green check light out.	
(3) >3° open - Red light by PB on.	One Vac. Bkr. Can be inop for closing if <3° open.
(4) >80° open - green light by PB out (red on)	
g. Reactor Building-Torus vacuum breakers	T.S.3.6.1.5
1) Purpose is to prevent exceeding the design external pressure of the suppression chamber (SC). and	Obj. ILT-5 Obj. LOR-4
2) Relieve from reactor building to suppression chamber if there is a pressure differential greater than 0.5 psid.	Obj. NLO-2.d, & 6 Obj. NLOR-2.d
3) Condensing steam in the suppression chamber will causes a vacuum to occur in the suppression chamber. (The same condition described earlier for the DW LOCA that condenses)	
4) To prevent exceeding the external pressure, air is admitted from the reactor building. (-2 psig external)	Actually 2 assemblies in parallel; One vac. bkr.
5) Two vacuum breakers are installed between the suppression chamber and the reactor building. Unid 64-20 & 21. Both valves have CAD N2 backup.	Assembly has an air operated damper & mechanical disc in series.
6) Two vacuum breakers are used for redundancy. The vacuum breakers consist of an air operated damper opening on sensed (.5 psi) delta-P or by electro mechanical demand by operator in control room via hand switches for each vac. bkr. and a check valve disc, which opens mechanically on delta-P in series with the damper.	Obj. NLO-7
7) Both sets of air operated valves are controlled (tested) by control switches on panel 9-3 in control room or open on increased dp.	

Question 89

The plant was operating at 80% power and has been on line for the last 6 months.
Maintenance is being performed on the Main Steam Line Flow Restrictor Transmitters.

- Transmitter B21N686B was placed in trip for maintenance.
- Transmitter B21N686C then failed upscale.

If all systems respond as designed and NO operator action is taken,

(1) How does Drywell pressure respond over the next 2 hours?
(2) Why does this response occur?
(3) What actions will be procedurally required?

a. (1) Drywell pressure will lower slowly.
(2) Following the Group 1 isolation, the drywell heat load is significantly reduced, lowering drywell average temperature.
(3) Suppression pool cooling will be required per EOP-6, Primary Containment Control.

b. (1) Drywell pressure will rise slowly.
(2) Following the Group 1 isolation, the heat added to the suppression pool migrates to the drywell, raising average drywell temperature.
(3) Suppression pool cooling will be required per EOP-6, Primary Containment Control.

c. (1) Drywell pressure will remain relatively constant.
(2) ONLY Main Steam Line A isolates.
(3) Adjust pressure set to maintain reactor pressure as close to full power steam dome pressure as possible.

d. (1) Drywell pressure is unchanged.
(2) Main Steam Lines are unaffected.
(3) Maintain at least one channel in a tripped condition or isolate the Main Steam Line within 12 hours.

Answer:

a. (1) Drywell pressure will lower slowly.
(2) Following the Group 1 isolation, the drywell heat load is significantly reduced, lowering drywell average temperature.
(3) Suppression pool cooling will be required per EOP-6, Primary Containment Control.

Explanation: With both B21N686B and B21N686C in a tripped condition a full Group 1 Isolation occurs.

Recirc pumps will shift to slow speed. The heat load on drywell cooling is significantly decreased. Drywell pressure was stable with the previous drywell heat load. The reduction in heat load results in lower drywell temperature and pressure after the scram. Suppression pool water temperature does not directly affect drywell airspace temperature.

Examination:	SRO 2	**Question No.:**	89	Rev 1	
Lesson Plan:	LP85407, LP85223	Primary Containment/ CRVIC			
Objective(s):	1.3	Recall the CRVIC System flowpaths for the following modes of operation while operating the system or on an exam in accordance with the identified figures and student text. .1 Group I			
	3.10	Given a Primary Containment System lineup and various plant conditions, evaluate the system indications/ responses. Determine if the indications/ responses are expected and normal while operating the system, or on an exam in accordance with station procedures and student text. .2 Cntmnt/Drywell Pressure			
Category:	Tier 1 / Group I				
Cognitive level:	H	**Difficulty Level:**	3	**Source:**	Contractor
K/A:	223002 A2.03 Ability to predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATON SYSTEM and based on those predictions use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) System Logic Failures	**ROI:**	3.0	**SROI:**	3.3
References:	Plant response post-scram.				

ITR'd: 6-13-06 OTPS Review Completed 6-29-06
Validated: 7-17-06

QUESTION 25

Given the following conditions for Unit 2:

- The Reactor is operating at 100% power
- An UNISOLABLE steam leak has occurred in the RWCU Heat Exchanger Room
- RWCU LEAK DETECTION TEMP HIGH, (2-9-3D, window 17) has alarmed

Which ONE of the following completes both statements below?

The rising temperature in the RWCU Heat Exchanger Room will affect the level indication on **panel 2-9-5**, Normal Range level instrument __(1)__.

When RWCU reaches the isolation setpoint, the MINIMUM indicated level for the affected instrument is __(2)__.

[REFERENCE PROVIDED]

A. (1) LI-3-206
(2) (+) 15 inches

B. (1) LI-3-206
(2) on scale

C. (1) LI-3-208B
(2) (+) 15 inches

D. (1) LI-3-208B
(2) on scale

Answer **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295032 EK1.03	
	Importance Rating	3.5	

295032 EK1.03 Knowledge of the operational implications of the following concepts as they apply to High Secondary Containment Area Temperature: Secondary containment leakage detection: Plant-specific

Explanation: **Answer B** – LI-3-206 is a panel 3-9-5 normal range level instrument. With the 9-3D at it alarm setpoint, temperature should be less than 150 °F so ON scale is the minimum indicated level.

A – Incorrect – first part correct, second part incorrect plausible if they read across caution 1 from LI-3-206 and the minimum indicated level would be +15 inches. In addition some RWCU HX temperature alarms do not alarm until 166 to 185 degrees which would then be +5 inches or if the they mis read and assume the MAX Safe RWCU room temperature of 220°F which then the correct response would be +15 inches.

C – Incorrect – First part incorrect 208B is on Panel 9-3. Plausible in that instruments 208A and 208D are 9-5 instruments. second part incorrect see above

D– Incorrect – First part incorrect see above, and second part correct

Technical Reference(s): Caution 1 (including Curve 8 and Table 6), Unit 2 Control Room Panel 2-9-5

Proposed references to be provided to applicants during examination: EOI Caution 1 (including Curve 8 and Table 6) ONLY

Learning Objective (As available): OPL171.017 A.2.b/n and A.4

Question Source: Bank:
Modified Bank: X
New

Question History: Previous NRC: BFN 1108 #85

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

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

RWCU LEAK DETECTION TEMP HIGH 2-TA-69-29 17

(Page 1 of 2)

Sensor/Trip Point:

Alarm	Alarm & Isolation
2-TS-69-29A 131°F	2-TIS-69-834A or C and B or D 185°F
2-TS-69-29B 118°F	2-TIS-69-835A or C and B or D 131°F
2-TS-69-29C 118°F	2-TIS-69-836A or C and B or D 148°F
2-TS-69-29D 135°F	2-TIS-69-837A or C and B or D 148°F
2-TS-69-29E 135°F	2-TIS-69-838A or C and B or D 139°F
2-TS-69-29F 135°F	2-TIS-69-839A or C and B or D 166°F
2-TS-69-29G 138°F	
2-TS-69-29H 138°F	

Sensor Location:

2-TE-69-29A & B - RWCU Demin Tanks 2A & 2B
2-TE-69-29C - RWCU Valve Room
2-TE-69-29D - RWCU Pump Room 2A
2-TE-69-29E - RWCU Pump Room 2B
2-TE-69-29F, G, & H - RWCU Heat Exchanger Room
2-TE-69-834A, B, C, & D - RWCU Piping in the Main Steam Tunnel
2-TE-69-835A, B, C, & D - RWCU Pipe Trench
2-TE-69-836A, B, C, & D - RWCU Pump Room 2A
2-TE-69-837A, B, C, & D - RWCU Pump Room 2B
2-TE-69-838A, B, C, & D - RWCU Heat Exchanger Room
2-TE-69-839A, B, C, & D - RWCU Heat Exchanger Room

RWCU ISOL LOGIC
CHANNEL A
TEMP HIGH
2-TA-69-834A

32

(Page 1 of 2)

Sensor/Trip Point:

Relay	Sensor	ATU	Setpoint
16A-K60A (9-15)	TE-69-834A	TIS-69-834A (9-83)	185°F
16A-K60C (9-15)	TE-69-834C	TIS-69-834C (9-85)	185°F
16A-K60A (9-15)	TE-69-835A	TIS-69-835A (9-83)	131°F
16A-K60C (9-15)	TE-69-835C	TIS-69-835C (9-85)	131°F
16A-K60A (9-15)	TE-69-836A	TIS-69-836A (9-83)	148°F
16A-K60C (9-15)	TE-69-836C	TIS-69-836C (9-85)	148°F
16A-K60A (9-15)	TE-69-837A	TIS-69-837A (9-83)	148°F
16A-K60C (9-15)	TE-69-837C	TIS-69-837C (9-85)	148°F
16A-K60A (9-15)	TE-69-838A	TIS-69-838A (9-83)	139°F
16A-K60C (9-15)	TE-69-838C	TIS-69-838C (9-85)	139°F
16A-K60A (9-15)	TE-69-839A	TIS-69-839A (9-83)	166°F
16A-K60C (9-15)	TE-69-839C	TIS-69-839C (9-85)	166°F

Sensor Location:

TE-69-834A(834C)	- RWCU Piping in the Main Steam Tunnel
TE-69-835A(835C)	- RWCU Pipe Trench
TE-69-836A(836C)	- RWCU Pump Room 2A
TE-69-837A(837C)	- RWCU Pump Room 2B
TE-69-838A(838C)	- RWCU Heat Exchanger Room
TE-69-839A(839C)	- RWCU Heat Exchanger Room

Probable Cause:

A. Pipe break.
   1. Steam leak.
   2. Water leak.

B. Sensor malfunction.
C. SI/SR in progress.

Automatic Action:

A. Alarm only, if sensors in Channel A actuate.
B. If sensors in Channel A and B actuate (1 out 2 twice logic), then 2-FCV-69-1, 2-FCV-69-2 and 2-FCV-69-12 close to isolate the RWCU system.

Operator Action:

A. **VERIFY** alarm by checking: ☐
   1. ATUs on Panel 2-9-83 and 2-9-85. ☐
   2. RWCU LEAK DETECTION TEMP HIGH annunciator in alarm (2-XA-55-3D, Window 17). ☐
   3. Area temperature indications on LEAK DETECTION SYSTEM TEMPERATURE, 2-TI-69-29, on Panel 2-9-21. ☐

CAUTIONS

CAUTION # 1

- An RPV water lvl instrument may be used to determine or trend lvl only when it reads above the Minimum Indicated Lvl associated with the highest max DW or SC run temp
- If DW temps or SC area temps (Table 6), as applicable, are outside the safe region of Curve 8, the associated instrument may be unreliable due to boiling in the run

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A/B	Emergency -155 to +60	on scale	N/A	below 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 +500	+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

Curve 8
RPV Saturation Temp

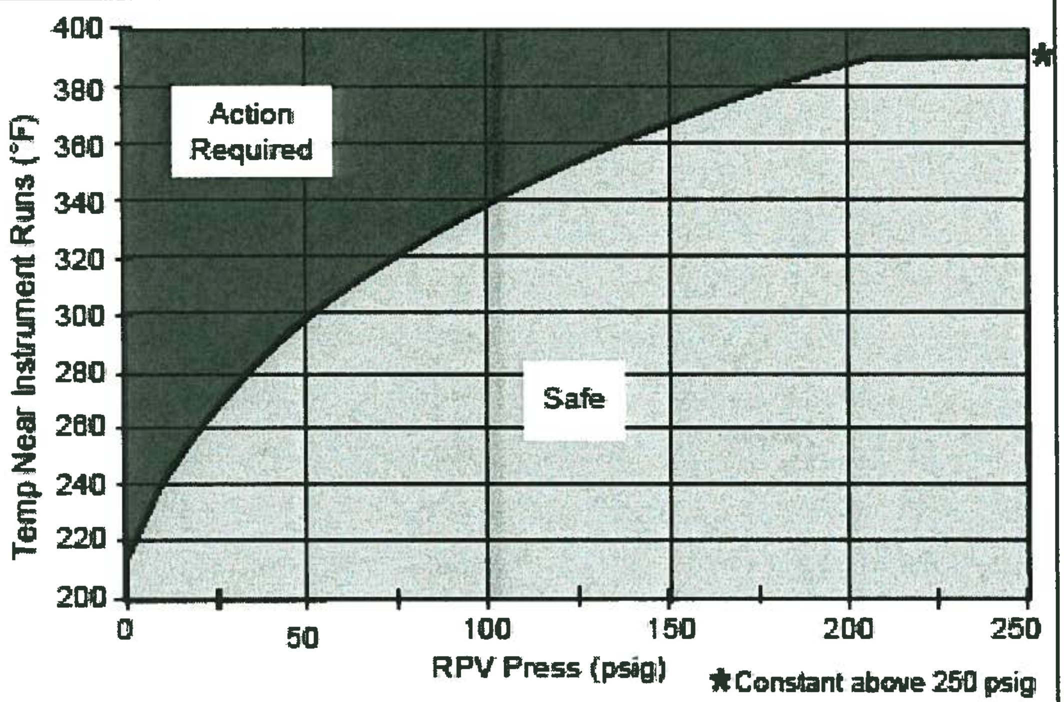


Table 6
Secondary Cntmt Instrument Runs

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

BFN 1108 #85

85. 295032EA2.02 001/1/2/SRO/NEW/H/3/BLC/MAB

Unit 2 is operating at 100% power.

A large unisolable steam leak has occurred in the RWCU Heat Exchanger Room and its maximum normal operating temperature has been exceeded.

Which ONE of the following identifies:

1) a Panel 9-5 Normal Range level instrument indication that is affected as the RWCU Heat Exchanger Room temperature approaches the maximum safe operating temperature

and

2) the procedure(s) required to be implemented before the room temperature exceeds the maximum safe operating temperature?

[REFERENCE PROVIDED]

A. LI-3-208B
GOI-100-12A

B. LI-3-208B
EOI-1

C. LI-3-53
GOI-100-12A

D✓ LI-3-53
EOI-1

Answer is D

QUESTION 26

Which ONE of the following combinations of Radiation Monitor signals would cause a Group 6 PCIS isolation?

	Radiation Monitor Detector	Indication
A.	Refuel Zone 2-RM-90-140A Refuel Zone 2-RM-90-141B	75 mR/hr 75 mR/hr
B.	Reactor Zone 2-RM-90-142A Reactor Zone 2-RM-90-142B	75 mR/hr 75 mR/hr
C.	Refuel Zone 2-RM-90-140B Refuel Zone 2-RM-90-141B	Downscale 75 mR/hr
D.	Reactor Zone 2-RM-90-143A Reactor Zone 2-RM-90-143B	Downscale Downscale

Answer is: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295033 EA1.02	
	Importance Rating	3.7	

295033 EA1.02 Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Process radiation monitoring system

Explanation: **Answer B – CORRECT:** Reactor Zone 2-RM-90-142A and Reactor Zone 2-RM-90-142B at 75mr/hr would satisfy the 2 out of 2 taken once logic for a group 6 PCIS isolation due to high radiation 72mr/hr.

A – Incorrect – Plausible misinterpretation of Reactor and Refuel Radiation monitor logic condition that would cause a group 6 isolation. 1 out of 2 taken twice high does not result in an isolation. 1 out of 2 downscale taken twice would result in an isolation.

C – Incorrect – Plausible misinterpretation of Reactor and Refuel Radiation monitor logic condition that would cause a group 6 isolation. 1 upscale and 1 downscale does not result in an isolation.

D– Incorrect – Plausible misinterpretations of Reactor and Refuel Radiation monitor logic condition that would cause a group 6 isolation. 2 out of 2 taken once downscale does not result in an isolation.

Technical Reference(s): 2-OI-90 rev 85

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.033 Obj. V.B.4

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

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

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. <u>Off-Gas Post-Treatment</u>

   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 <u>or</u> two channels downscale/INOP)

   a. Control Room Emergency Ventilation auto start. (Normal Control Room Ventilation isolates.)

Illustration 1
(Page 3 of 4)

Radiation Monitoring System Operational Summary

Subsystem	Operation
Off-Gas Pretreatment and Post Treatment Vial Samplers	Two portable sampling units not normally in service can be used to draw Off-Gas samples for laboratory analysis during operation or while shut down. Pretreatment samples can be drawn from the holdup volume inlet (or alternately from the SJAE second stage suction). Post Treatment samples can be drawn from the charcoal bed inlet and outlet. PNL 2-25-40 & 0-25-259
Reactor/Refueling Zone Exhaust Radiation Monitors 2-RM-90-140/142/ 2-RM-90-141/143	Each Control Room drawer contains all the monitors for that channel (both reactor and refuel zone). Each monitor has two detectors (i.e., 2-RM-90-141 Detector A and Detector B). The purpose of the second detector is to prevent spurious trips. For an upscale trip to be initiated, both detectors (i.e., 2-RM-90-141 Detector A and Detector B) in a channel (A <u>or</u> B) must reach the trip set-point. For a Downscale/INOP trip to be initiated, one detector for the associated zone in both channels must be in the Downscale/INOP state (i.e., 2-RM-90-140 Detector A <u>or</u> B and 2-RM-90-141 Detector A <u>or</u> B). Note that with this arrangement, one detector Downscale/INOP will render that channel INOP for a high trip. **Refer to Tech Spec 3.3.6.2, for Reactor Bldg Vent Radiation Monitoring system channel(s) which are inoperable for functional testing or inoperable for calibration or maintenance. The inoperable channel is to be placed in the tripped condition.** Note that placing the drawer in the inoperable state results in both radiation monitors (Reactor and Refuel Zone) being in the inoperable trip condition and is allowed by this note. Four in-line radiation detectors in each zone monitor the ventilation exhaust ducts. High radiation in either of these systems will isolate the respective ventilation, start Standby Gas Treatment, and Initiate Emergency Control Room Isolation and pressurization. In addition, High Radiation in the Reactor Zone Ventilation inserts a redundant Refuel Zone High Radiation trip and initiates a Group 6 Isolation. High Refuel Zone radiation in <u>any</u> unit will cause a Refuel Zone Isolation in <u>all</u> units.

QUESTION 27

Given the following conditions on Unit 2:

- Reactor Power is 100%
- REACTOR ZONE DIFFERENTIAL PRESSURE LOW (2-9-3D, window 32), 2-PDA-64-27 is intermittently in alarm
- Wind conditions on ICS are 25 mph
- No PCIS Group 6 Isolation signals exist

Which ONE of the following completes the statements below?

In accordance with 2-ARP-9-3D, entry into 2-EOI-3, Secondary Containment Control __(1)__ required.

A Reactor Building pressure of __(2)__ inch H_2O will cause the Reactor Zone ventilation to isolate.

A. (1) is
(2) (+) 0 .5

B. (1) is
(2) (-) 0.17

C. (1) is NOT
(2) (+) 0.5

D. (1) is NOT
(2) (-) 0.17

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	1	
	K/A#	295035 EA1.01	
	Importance Rating	3.6	

295035 E1.01 Ability to operate and / or monitor the following as they apply to Secondary Containment High Differential Pressure: Secondary Containment ventilation system

Explanation: **Answer C** – First Part: CORRECT- With wind speed on ICS > 20 mph an EOI entry is not required. Second Part: CORRECT- If reactor bldg d/p rises to + 0.5 inch H_20 the reactor zone ventilation system will isolate.

A – Incorrect – First Part: Incorrect- This is plausible as the given alarm comes in at -0.17 in H_20 which is an EOI entry condition. Second Part: Correct –See C.

B – Incorrect – First Part: Incorrect- See A. Second Part: Incorrect. This is plausible as -0.17 in H_20 is an EOI entry condition.

D– Incorrect – First Part: Correct- See C. Second Part: Incorrect- See B.

Technical Reference(s): 1-ARP-9-3D; 1-AOI-30B-1

Proposed references to be provided to applicants during examination: NONE

Learning Objective (As available): OPL171.067 2.b and h

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

REACTOR Z
DIFFERENTIAL
PRESSURE LOW
2-PDA-64-27

32

(Page 1 of 1)

Sensor/Trip Point:
2-PDIS-064-27 -0.17 in. of water

Sensor Location: Panel 2-25-213, Rx Bldg El. 639'

Probable Cause:
A. Securing/Alternating Refuel Zone Fans.
B. Trip of any Rx Bldg Zone Exh. Fan.
C. PCIS Group 6 Isolation.
D. Differential Pressure switches fail closed.
E. Rapidly changing Barometric pressure or high winds.
F. Normal ventilation in service with Standby Gas Treatment System running at t same time.
G. High energy line break in Secondary Containment.

Automatic Action: Annunciation only.

Operator Action:
A. IF the alarm is intermittent, THEN CHECK for high wind conditions (ex., >20 mph) on ICS. ☐
B. IF high wind conditions CANNOT be confirmed, THEN REQUEST personnel to check local Reactor Building differential pressure. ☐
C. IF alarm is due to high wind conditions, THEN EOI-3 entry is NOT required. ☐
D. IF alarm is valid, THEN INFORM Unit Supervisor of 2-EOI-3 entry condition. ☐
E. REQUEST personnel to check fans locally for any apparent problems. ☐
F. REFER TO 2-OI-30B and PLACE standby fan in service to restore normal differential pressure. ☐

References: 2-45E620-2 2-47E610-64-1

1-AOI-3B-1 Rev 13

1.0 PURPOSE

This Abnormal Operating Instruction provides symptoms, automatic action and operator action for the degradation or loss of Reactor Building or Refuel Zone ventilation for causes other than Group 6 Isolation. The Group 6 Isolation is addressed in another abnormal operating instruction.

2.0 SYMPTOMS

A. One or a combination of reactor or refuel zone supply or exhaust fans indicate shutdown on Panel 1-9-25.

B. One or more ventilation dampers in the ventilation flow path indicate closed on Panel 1-9-25.

C. Annunciator REAC BLDG VENTILATION ABNORMAL, XA-55-3D, Window 3, is in alarm.

D. Annunciator REACTOR ZONE DIFFERENTIAL PRESSURE LOW , XA-55-3D, Window 32, is in alarm.

E. REF ZONE STATIC DIFF PRESS CONT, 1-PDIC-064-0002, on Panel 25-213 indicates Reactor Building pressure is NOT within -0.25 to -0.40 inch H_2O.

F. REF ZONE STATIC DIFF PRESS CONT, 1-PDIC-064-0001, on Panel 25-219 indicates Refuel Zone pressure is NOT within -0.25 to -0.40 inch H_2O.

3.0 AUTOMATIC ACTION

A. If Reactor Building pressure has risen to +0.5 inch H_2O or lowered to -1.0 inch H_2O, the Reactor and Refuel Zone ventilation systems isolate.

QUESTION 28

Given the following conditions on Unit 3:

- Reactor Power is 100%
- Residual Heat Removal (RHR) is in the normal standby lineup
- The PSC head tank pumps are in their normal lineup
- PSC HEAD TANK LEVEL LOW (3-9-3A, Window 26) is in alarm

Which ONE of the following completes the statement below?

In accordance with the ARP for PSC HEAD TANK LEVEL LOW (3-9-3A, Window 26), ______.

A. BOTH PSC Head Tank Pumps A and B will automatically start

B. ONLY the PSC Head Tank Pump A will automatically start

C. ONLY the PSC Head Tank Pump B will automatically start

D. BOTH PSC Head Tank Pumps A and B will trip

ANSWER: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	203000 A4.03	
	Importance Rating	3.4	

203000 RHR/LPCI: Injection Mode (Plant Specific)A4.03Ability to manually operate and/or monitor in the control room: Keep fill system

Explanation: **Answer A**–Both PSC head tank pumps will automatically start at El 645' which is the PSC head tank level low alarm setpoint.

B– Incorrect – plausible one pump normally cycles to maintain head tank level. Plausible if you didn't know that pumps do this in alternating fashion.

C – Incorrect – plausible one pump normally cycles to maintain head tank level. Plausible if you didn't know that pumps do this in alternating fashion.

D– Incorrect – plausible to believe that pumps would trip on low level based on inadequate NPSH.

Technical Reference(s): 3-ARP-9-3A Rev 045

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):OPL171.044 Obj. ILT #13

Question Source: Bank:
Modified Bank:
New:X

Question History: Previous NRC: None

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

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

PSC HEAD TANK LEVEL LOW 3-LA-75-79 26	Sensor/Trip Point: 3-LS-75-78D Elevation 645'

(Page 1 of 1)

Sensor Location: Rx Bldg, El 639'

Probable Cause:

A. Both pumps are NOT running.
B. Level switch, 3-LS-75-78C & 78D, malfunction.
C. Thermal overloads NOT reset: 480V Reactor MOV Boards 3B and 3C, compartments 11D and R11A.
D. PCIS Group II isolation. PSC PUMP SUCTION INBD and OUTBD ISOL VALVEs, 3-FCV-75-57 and 58, fail closed on loss of control air.
E. Both pumps are running, but pump discharge pressure is below 60 psig. A minimum discharge pressure of 55 psig is required to reach El 645'.

Automatic Action: Low level switch starts both pumps.

Operator Action:

A. **VERIFY** both PSC Head Tank Pumps are running. ☐
B. **VERIFY** power available to pumps. ☐
C. **CHECK** PSC PUMP SUCTION INBD and OUTBD ISOL VALVEs, 3-FCV-75-57 and 58, open. ☐
D. **IF** the alarm does **NOT** reset, **THEN** **DISPATCH** personnel to check pumps locally. ☐
E. **IF** the PSC Head Tank Pumps System will **NOT** maintain the RHR and Core Spray systems charged above TRM Limits, **THEN** **REFER TO** 3-OI-75, Section 8.5 or 8.10 to align the condensate transfer system to each loop. ☐
F. **REFER TO** Tech Spec Section 3.5.1. ☐

References:

3-45E620-3	3-47E610-75-1
3-47E814-1	
Technical Specifications 3.5.1	TRM 3.5.4

QUESTION 29

Given the following conditions on Unit 3:

- The Reactor is shutdown in Mode 5
- RHR Loop II is in Shutdown Cooling
- 3-HS-74-149, RHR SYSTEM II MIN FLOW INHIBIT SWITCH is in INHIBIT
- 3-HS-74-30A, RHR SYSTEM II MIN FLOW VALVE HANDSWITCH on panel 3-9-3 is taken to the OPEN position and released.

Which ONE of the following completes the statement below?

3-FCV-74-30, RHR SYSTEM II MIN FLOW BYPASS VALVE will travel full open __(1)__ and __(2)__.

A. (1) immediately
(2) travel full close

B. (1) immediately
(2) remain open

C. (1) after a 10 second time delay
(2) travel full close

D. (1) after a 10 second time delay
(2) remain open

Answer is: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	205000 K5.02	
	Importance Rating	2.8	

205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) K5.02 Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : Valve operation

Explanation: **Answer A – CORRECT:** First Part: Opening RHR SYSTEM II MIN FLOW VALVE, 3-HS-74-30A, from 3-PNL-9-3 with the RHR SYSTEM II MIN FLOW INHIBIT switch, 3-HS-74-149, in INHIBIT will cause the minimum flow valves to travel full open immediately. Second Part: The valve will return to full closed unless the RHR SYSTEM II MIN FLOW VALVE, 3-HS-74-30A, is placed in closed position to break the OPEN seal in contacts.

B– Incorrect – First Part: Correct- See A. Second Part: Incorrect. Plausible because with the RHR SYSTEM II MIN FLOW INHIBIT switch NOT in INHIBIT, the min flow valve would stay open.

C – Incorrect – First Part: Incorrect. Plausible because the min flow valve normally opens with the RHR SYSTEM II MIN FLOW INHIBIT switch NOT in INHIBIT, and the RHR system II flow rate <5800 GPM after a 10 second time delay. Second Part: Correct-See A.

D – Incorrect – First Part: Incorrect- See C. Second Part: Incorrect- See B.

Technical Reference(s): 3-OI-74 Rev 0112; 3-45E779-9 Rev 014

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):OPL171.044 ILT Obj. #3

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

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

3.6 **Interlocks (continued)**

4. Suction Path Interlocks:

   a. An RHR pump will not start or will trip, if running, unless its corresponding torus suction valve is open or the SDC suction valve and the SDC suction supply valves, 3-FCV-74-47 and 48, are open.

   b. The torus suction valves cannot be opened unless the corresponding pumps SDC suction valve is fully closed.

   c. The SDC suction valves cannot be opened unless the corresponding pumps TORUS suction valve is fully closed.

5. RHR Minimum Flow Valve Interlocks

   a. The RHR minimum flow valves auto close if both pumps in the corresponding loop are off and either pump's SDC suction valve is open.

   b. The minimum flow valves open (after a 10 second TD) and close on a low flow of 5800 gpm. The tolerance of the flow switch may allow the setpoint of the min flow valve to be from 4500 gpm to 7000 gpm. Operation outside of this expanded range should be investigated.

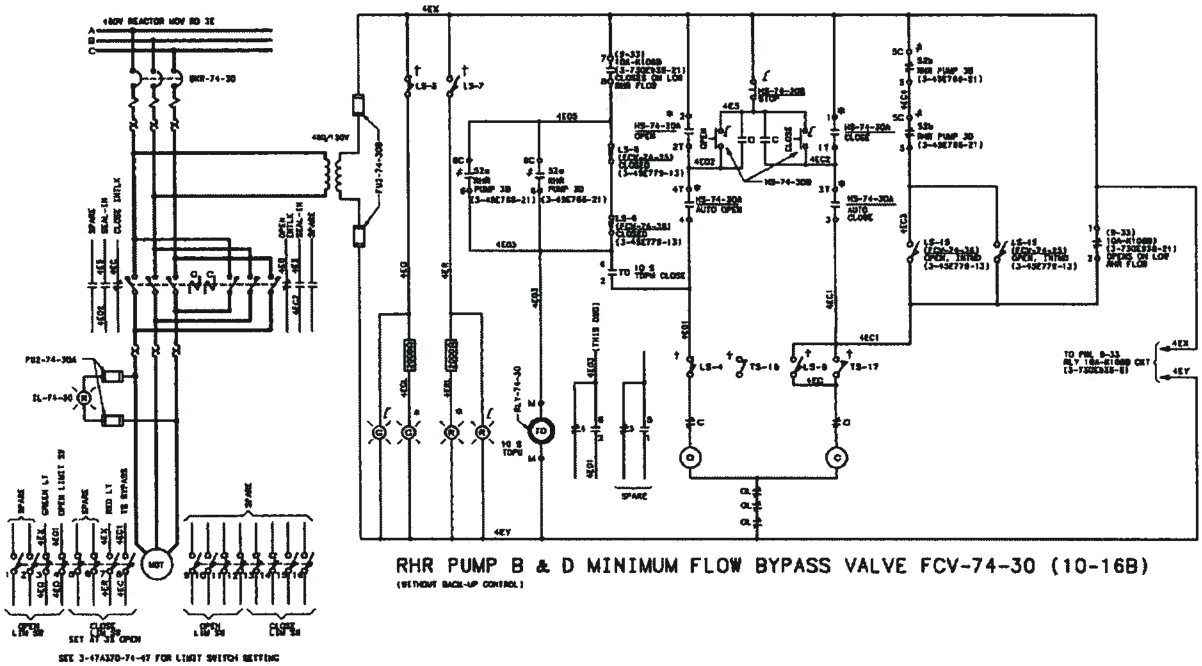
   c. If 3-HS-74-148(149) RHR SYSTEM I (II) MIN FLOW INHIBIT switch is in the INHIBIT position, the pumps on that loop do not have automatic minimum flow protection.

   d. Placing the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 3-HS-74-148(149), in INHIBIT, will simulate a high flow and the minimum flow valve will remain closed regardless of flow.

   e. Opening RHR SYSTEM I(II) MIN FLOW VALVE, 3-HS-74-7A(30A), from 3-PNL-9-3 with the RHR SYSTEM I(II) MIN FLOW INHIBIT switch, 3-HS-74-148(149), in INHIBIT will cause the minimum flow valves to travel full open and full close unless the RHR SYSTEM I(II) MIN FLOW VALVE, 3-HS-74-7A(30A), is placed in closed position to break the OPEN seal in contacts.

   f. [PROC] Misalignment of the RHR SYSTEM I(II) MIN FLOW INHIBIT Switch, 3-HS-74-148(149), with the respective RHR loop in standby readiness, can cause inadvertent damage to that loop RHR pump(s) should RHR pump(s) auto start. [BFA-880700003P]

3-45E779-9 Rev 014



QUESTION 30

Given the following Unit 2 plant conditions:

- EOI-1, RPV Control, and EOI-2, Primary Containment Control, have been entered
- Reactor water level initially lowered to (-)69 inches.
- After water level recovery, the High Pressure Coolant Injection (HPCI) Pump Injection Valve (FCV-73-44) was manually closed and HPCI was placed in pressure control to remove decay heat.

Subsequently, Condensate Storage Tank (CST) level dropped below 6800 gallons.

Which ONE of the following describes the status of the HPCI system (assume no other operator actions have occurred)?

A. HPCI would be operating in pressure control with suction from the CST.

B. HPCI would be pumping to the CST with suction from the Suppression Pool.

C. HPCI would be operating at shutoff head with suction from the Suppression Pool.

D. The HPCI turbine would trip on overspeed due to loss of suction during the transfer.

Correct Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	206000K3.02	
	Importance Rating	3.8	

K&A K3.02 Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE COOLANT INJECTION SYSTEM will have on following: Reactor pressure control: BWR-2,3,4

Explanation: **Answer– C- CORRECT.** At ~7000 gallons in the CST, HPCI auto swaps from CST suction to Suppression Pool (Torus) suction. When this occurs the CST Test Return Isolation valve receives a close signal from the Torus suction valves opening; to prevent pumping the Torus to the CST. Therefore, with the HPCI injection valve previously closed, HPCI would be operating at shutoff head without minimum flow protection. Note - the minimum flow valve would be closed due to lack of a valid initiation signal.
In order to answer this question correctly the candidate must determine the following:

1. Recognize that the HPCI initiation signal is reset to allow HPCI to be placed in Pressure Control.
2. Recognize that the HPCI Pressure Control lineup if from the CST and back to the CST.
3. Recognize that the current CST level would initiate a suction swap to the Suppression Pool.
4. Recognize that HPCI would not receive a trip signal as the suction valves re-aligned.
5. Recognize that the CST Test Isolation Valve will auto close on low CST level.

A – Incorrect – This assumes the low CST level has NOT initiated a suction swap to the Torus. This is plausible since most procedures list the setpoint for the auto swap as an elevation above sea level versus a "gallons" setpoint.

B – Incorrect – This lineup would occur if the HPCI Test Isolation Valve did NOT receive a close signal following the suction swap logic initiation. This is plausible since the ONLY auto closure interlock of the HPCI Test Isolation Valve is under this specific condition.

D– Incorrect – HPCI will not trip on low suction pressure under this specific condition. The Torus suction valves begin to open before the CST suction valve closes in a "make-before-break" fashion. This is plausible since closure of the suction path to HPCI typically results in a low suction trip.

Technical Reference(s): OPL171.042, 2-OI-73

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.042 V.B.5

Question Source:	Bank: X Modified Bank: New
Question History:	Previous NRC: BFN 0610 #3
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X

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

3.7 Interlocks

A. When any of the following signals are received, HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27, and HPCI SUPPR POOL INBD SUCT VALVE, 2-FCV-73-26 automatically open, unless a HPCI isolation signal is present.

1. Suppression Pool Level High at +5.25 in.

2. HPCI Pump Suction Condensate Header Level Low at approximately 7000 gallons (El. 552'6" on 2-LS-73-56A and -56B).

B. When HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27 and HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26 are fully open, HPCI CST SUCTION VALVE, 2-FCV-73-40, automatically closes.

3.7 Interlocks (continued)

C. When either HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27, or HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26, is FULL OPEN, the HPCI/RCIC CST TEST VLV, 2-FCV-73-36, and HPCI PUMP CST TEST VLV, 2-FCV-73-35, will close.

BFN 0610 NRC #3

0610 RO Final Examination

3. Given the following plant conditions:

    - Unit 2 reactor water level initially lowered to (-)69 inches.
    - Conditions required entry into EOI-1, "RPV Control" and EOI-2, "Primary Containment Control."
    - After water level recovery, the High Pressure Coolant Injection (HPCI) Pump Injection Valve (FCV-73-44) was manually closed and HPCI was placed in pressure control to remove decay heat.
    - Subsequently, Condensate Storage Tank (CST) level dropped below 6800 gallons.

    Which ONE of the following describes the status of the HPCI system (assume no other operator actions have occurred)?

    A. HPCI would be operating in pressure control with suction from the CST.

    B. HPCI would be pumping to the CST with suction from the Suppression Pool.

    C. HPCI would be operating at shutoff head with suction from the Suppression Pool.

    D. The HPCI turbine would trip on overspeed due to loss of suction during the transfer.

Answer: A

QUESTION 31

Given the following conditions on Unit 1:

- Reactor Power is 100%
- 1-SR-3.5.1.1 (HPCI), Maintenance of Filled HPCI Discharge Piping, has just been performed and failed to meet the Acceptance Criteria

Which ONE of the following completes both statements below?

In accordance with the Unit 1 Tech Spec requirements, __(1)__ must be verified OPERABLE by administrative means immediately.

Failure to maintain the HPCI discharge piping full of water can result in __(2)__.

A. (1) RCIC
(2) water hammer during HPCI initiation

B. (1) RCIC
(2) pressure locking of the HPCI injection valve

C. (1) ADS
(2) water hammer during HPCI initiation

D. (1) ADS
(2) pressure locking of the HPCI injection valve

Answer is: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	206000 K5.07	
	Importance Rating	2.8	

206000 High Pressure Coolant Injection System K5.07 Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM : System venting: BWR-2,3,4

Explanation: **Answer A – CORRECT:** First Part: In accordance with T.S 3.5.1 (ECCS Operating) Condition C, if HPCI System is inoperable then, RCIC must be verified operable by administrative means immediately. Second Part: Gas accumulation in the discharge piping of HPCI can result in water hammer or a system pressure transient.

B – Incorrect – First Part: Correct- See A. Second Part: Incorrect: Although pressure binding of the HCPI discharge valve is a concern with flex wedge type gate valves. This concern was alleviated with a hole drilled into the downstream disc, not by ensuring the HPCI discharge is maintained full of water. *Additional info on pressure locking plausibility*: Pressure locking occurs on the discharge valve, when the valve is subject to a high d/p. This causes the seat to leak and the bonnet to become pressurized. When the line pressure is removed, the pressure in the bonnet locks the valve preventing re-opening. This can happen with HPCI, if leakage occurs in the downstream check valve to the FW line subjecting the line to FW pressure, then being subsequently lost due to a loss of pressure in the FW line (loss of FW transient/line break).

C – Incorrect -- First Part: Incorrect. Although ADS is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails, there is no T.S. requirement to verify ADS operability immediately if HPCI is declared inoperable. In addition, if HPCI and one or more ADS valves are both inoperable then LCO. 3.03 is to be entered immediately. Second Part: Correct-See A.

D – Incorrect – First Part: Incorrect -See C. Second Part: Incorrect –See B.

Technical Reference(s):1- SR 3.5.1.1 Rev 0008; Unit 1 T.S. 3.5.1; 1-OI-73 Rev 0025;

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

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

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

T.S. 3.5.1 ECCS-Operating

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

NOTE

LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	Immediately 14 days
H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. OR HPCI System and one or more ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

T.S. 3.5.1 ECCS-Operating

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days

T.S. 3.5.1 ECCS-Operating (Bases)

BASES

BACKGROUND (continued)

water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. With HPCI taking suction from the condensate storage tank and injecting to the reactor vessel, there is sufficient inventory in the tank such that the high suppression pool level suction transfer will occur before a low condensate header level would be created. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1174 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open (for CS and RHR they are already open) to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using the pressure suppression chamber head tank or condensate head tank. The HPCI System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water.

1-OI-73 Rev 0025

3.0 PRECAUTIONS AND LIMITATIONS (continued)

II. When HPCI SYSTEM FLOW/CONTROL, 1-FIC-73-33 ALM light flashes, the DATA PROTECT battery needs to be replaced (HPCI operable) for all other conditions of the "FAIL" or "ALM" lights HPCI is required to be considered "NOT OPERABLE". When Service Request/Work Orders are initiated for flow controller alarms or features, they should contain a request for the Instrument Mechanics to use the CHECK ALARM feature on the tuning panel to obtain fault and alarm codes. This information is useful during apparent or root cause analysis performed by the technical staff to document the cause the cause of the failure.

JJ. Any time the HPCI System is in Standby and the controller is placed in "MANUAL" HPCI is to be considered "NOT OPERABLE", however HPCI is still available.

KK. The HPCI flow controller 1-FIC-73-33 is a "FLOW X10" controller. 5300 gpm on the controller digital display will read 530. The steps in this procedure which list a flow value are displayed as follows "flow as read on the digital display followed by the actual flow in gpm", i.e., a flow of 1250 gpm is shown as "125 (1250 gpm)" a flow of 5300 gpm is shown as "530 (5300 gpm)".

LL. RCIC SUPP CHBR TURB EXH VAC RELIEF VLV, 1-FCV-071-0059, is common to both Unit 1 and 2. This valve is normally de-energized in the open position and is required to be re-energized and closed to minimize leakage from primary containment following a LOCA when HPCI and RCIC are shut down and no longer required.

MM. When the HPCI piping is drained for any reason from the suction line to the discharge valve 1-FCV-73-44, running HPCI in the CST to CST mode for a minimum 15 minutes will ensure the system is dynamically vented free the system of voids. [Generic Letter 2008-01]

NN. To eliminate the potential for pressure locking for HPCI PUMP INJECTION VALVE, 1-FCV-73-44, a 1/4" hole was drilled in the downstream side of the disc to allow the bonnet pressure to equalize with the downstream pressure. [DCN 69964]

QUESTION 32

Which ONE of the following completes the statement below?

During ATWS conditions, Core Spray injection is a concern because it _________ .

A. may result in a large power excursion

B. creates a rapid pressure reduction resulting in an uncontrollable cooldown rate

C. results in inadequate re-mixing of boron located in the lower plenum as water level is raised

D. creates a steam blanket at the top of the fuel bundles which inhibits heat transfer via steam flow past the fuel

Answer is: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	209001 K1.14	
	Importance Rating	3.7	

209001 Low Pressure Core Spray System K1.14 Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Reactor vessel

Explanation: **Answer A – CORRECT:** The systems listed in Steps C5-13 and C5-16 for use in controlling RPV water level comprise all those which inject outside of the core shroud. These are used, preferentially, because the flow path outside the core shroud mixes the relatively cold injected water with the warmer water in the lower plenum prior to it reaching the core. Core Spray injects outside the shroud spraying relatively cold water directly on the fuel which could lead to fuel damage or a cold water induced power excursion

B – Incorrect –This is could cause a rapid pressure reduction but is not the overriding factor for not using Core Spray under these conditions. This is plausible since high volume Core Spray injection at close to the maximum injection pressure would cause a rapid pressure reduction.

C – Incorrect –This is plausible because the objective of raising RPV level once Hot Shutdown Boron Weight has been injected is to re-mix the Boron located in the lower plenum. This is accomplished by raising the RPV level thereby increasing natural circulation through the vessel which results in quickly mixing the Boron and distributing it throughout the core region. Core Spray injection inside the shroud does not mix with the borated water in the lower plenum, but as water level is raised above the feedwater spargers, natural circulation will re-mix the Boron located in the lower plenum.

D – Incorrect – This phenomenon, referred to as Counter Current Flow Instability, is plausible but is only of significant concern with the core completely uncovered and is the basis for removing Spray Cooling from the EPG definition of Adequate Core Cooling.

Technical Reference(s):EOIPM-0-V-K Rev ; Boiling Water Reactor GE BWR/4 Technology Advanced Manual-NRC training material.

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.205 Obj. 6; OPL171.045 Obj. B.6

Question Source:	Bank: X Modified Bank: New
Question History:	Previous NRC: BFN 0610 #4
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis

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

1.0 CONTINGENCY #5, LEVEL/POWER CONTROL BASES (continued)

DISCUSSION: C5-25

Injection into the RPV is re-established to maintain adequate core cooling. Irrespective of whether the reactor is shutdown, injection is controlled to make up the mass of steam being rejected through open MSRVs and, if possible, to keep the core submerged. Injection is increased slowly to preclude the possibility of large reactor power excursions due to the rapid injection of relatively cold, unborated water under conditions where the reactor may not be shutdown. The Minimum Steam Cooling RPV Water Level is specified as the lower limit for control of RPV water level to provide the widest possible control band.

The systems listed in this step are the same preferred systems listed earlier in Steps C5-13 and C5-16—those that are relatively easy to align, provide high quality water, and either inject outside the core shroud or inject borated water. When RPV injection is restored or increased under failure-to-scram conditions, injection systems should be aligned and operated in the manner that minimizes the potential for core instabilities and power excursions while still accomplishing the objectives of the EOI ATWS strategies.

- Injection locations should be selected to optimize mixing and preheating of injection flow and minimize the potential for boron removal from the core. Injecting into the downcomer steam space provides effective mixing and preheating, thereby reducing core inlet subcooling and minimizing the potential for core instabilities and power excursions. Injecting into the downcomer region below water level and through the jet pumps provides good mixing of injection and recirculation flows but results in greater core inlet subcooling than injection into the downcomer steam space. Injection paths discharging directly into or over the core region should be used only as a last resort due to the higher potential for boron displacement and cold water induced power excursions.

- Systems providing good flow control capability should be used to avoid large, rapid increases in core flow.

- The injection rate should be increased slowly and only when reactor power is not increasing. (Refer to the definition of "slowly" in EOIPM 0-I-C.)

1.0 CONTINGENCY #5, LEVEL/POWER CONTROL BASES (continued)

DISCUSSION: C5-20

With boron injected into the lower plenum, little natural circulation and boron mixing occur if RPV water level is lowered to and maintained near the Minimum Steam Cooling RPV Water Level. Three-dimensional scale model tests indicate that the injected boron concentrates in the lower plenum and does not contribute to reactor shutdown until in-core distribution (mixing) is achieved. When an amount of boron sufficient to shut down the reactor has been injected into the RPV, mixing is accomplished by raising RPV water level in Step C5-32, thereby increasing natural circulation flow through the vessel.

The Hot Shutdown Boron Weight (HSBW, **A.72**) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions. The HSBW is utilized to assure the reactor will be shutdown irrespective of control rod position when RPV water level is raised to uniformly mix the injected boron. Refer to EOIPM Section 0-II-ZB for discussion of the HSBW.

When an amount of boron equivalent to the HSBW has been injected, RPV water level is restored to and maintained within the normal operating range. As RPV water level is increased, natural circulation flow is increased and the boron which has accumulated in the lower plenum is quickly mixed and distributed throughout the core region. This phenomenon is known as "boron remixing," thereby distinguishing it from any mixing which may have occurred in the early phase of the transient when some core flow was present.

Three dimensional scale model tests confirm the feasibility and effectiveness of this mechanism of mixing boron. Data from test results also show that, as RPV water level is raised, the time required to achieve sufficient in-core boron concentration to turn reactor power is short enough that no sustained power increase should occur. This fact is illustrated by the Boron Remixing Time Constant illustrated in Figure C5-5.

Boiling Water Reactor GE BWR/4 Technology Advanced Manual Chapter 4.0

operating at 102% of rated power for an infinite time which represents an improbable situation.

- Peak temperature criteria - limitation of the peak calculated temperature of the cladding (2200°F) is applied to the hottest-region of the [illegible] rods. This provides a substantial degree of conservatism to ensure that the core will suffer a very limited amount of core damage due to a LOCA.

- ECCS single failure criteria. These calculations also have to assume the most damaging single failure of the ECCS component or subsystem.

- Addresses reflood and refill rates of less than 1 inch per second i.e., if the reflood/refill rate drops to less than 1 inch per second, then the calculations must assume that the cooling of the core is by steam alone. This is very conservative because the water splatter carryover that will be entrained in the steam will remove heat from the cladding but is not used in the calculations.

Many years have passed since Appendix K was implemented. Calculations have been revised as a direct result of obtaining better data through research and development programs performe[illegible] NRC and private industry. Recent calculations have established that the maximum fuel clad temperatures reached during a LOCA will be approximately 900°F less than the older calculated value of 2200°F. This added margin of safety has resulted in the reduction of many restrictions in the areas of fuel operating temperatures, surveillance testing frequencies, and permissible "down times" for safety-related equipment for testing and maintenance. These changes should result in increased reactor availability and more efficient fuel burnup.

4.1.6 Meeting Changing ECCS Criteria

The initial criteria was met by an ECCS consisting of two 100% core spray (C.S.) systems and one low pressure coolant injection (LPCI) System. Original data and calculations proved that the C.S. System could by itself, terminate post accident heatup by spray action alone. Core spray or LPCI could successfully meet the peak cladding temperatures (PCT) for all large line breaks.

When the interim criteria was established, initial calculations indicated that PCT could not be maintained less than 2300°F. Several factors contributing to the inadequacy of the ECCS included:

- Establishment of the single failure criteria - the single worst failure was determined to be a failure within the LPCI system. The LPCI system included a LPCI loop selection logic that prevented opening of the injection valve supplying the "broken" recirculation loop and opened the injection valve which supplied the "good" loop thus supplying water from both divisions of LPCI. Failure of the injection valve supplying the "good" recirculation loop to open would render inoperable the entire LPCI System.

- The C.S. System was judged incapable of meeting the new PCT requirements by itself. This was due to the existing 7X7 fuel design and C.S. system test results that indicated counter current flow limiting effects and questionable spray behavior in a steam environment.

The reactor vendor and owners groups established several possible methods for meeting the interim criteria. Possible alternatives included:

- Redesign the LPCI and C.S. Systems in order to take credit for both spray and flooding.

- Redesign the fuel to limit power production by the fuel pellets.

- Take credit for water accumulation and the eventual flooding capability of the C.S. System.

- Limit PCT by limiting MAPLHGR.

A reanalysis was performed using the last two alternatives listed above and the results indicated that either the C.S. or LPCI systems could prevent PCT from exceeding 2300°F for all large pipe breaks. However, with LPCI unavailable, the C.S. System would be required to provide both spray and flooding. The flooding capability was accomplished by drilling holes in the lower core plates.

In core instrument tube vibrations on BWR/4 plants required plugging of the bypass flow holes in the lower core plate. Those holes provided part of the design core bypass flow (10%). The bypass holes also allowed the core spray water to accumulate in the bypass region to reflood the bottom head volume and then the fuel. Plugging the holes resulted in a reduction in the core sprays ability to reflood the core and maintain PCT below the specified limit. This new problem meant that on high power density cores the core spray system could not meet the final 2200°F criteria without severe MAPLHGR restrictions. This prompted General Electric to restore the bypass flow by drilling holes in the lower tie plate of the fuel assemblies.

The final acceptance criteria further restricted the maximum PCT to 2200°F. This limit made discharge pipe breaks more severe for certain vessel geometries. Also, because the LPCI System was assumed to be unavailable for those breaks, it was determined that the C.S. System may not prevent exceeding the PCT criteria in high power density cores even with combined spray and flooding capability.

Rather than placing further limits on MAPLHGR, the final acceptance criteria was met by a combination of design and physical plant changes. Those changes involved substantial changes to the LPCI System including removal of the LPCI Loop Selection Logic, permanent closure or removal of the LPCI pumps discharge lines Division I and II crossconnect valve, and total separation and independency of the two divisions.

Completing the modifications to the LPCI System provided assurance that the C.S. System in combination with all or one of the two divisions of LPCI would meet all of the new ECCS criteria for all size pipe breaks. Later development of the 8 X 8 fuel assemblies reduced MAPLHGR, thus also contributing to the margin of safety during the LOCA.

4.1.7 Vendors Response To The Final Acceptance Criteria

None of the reactor manufacturers or owners groups agreed with the Staffs proposal of 2200°F maximum cladding temperature. Westinghouse proposed a calculated temperature of 2700°F. Combustion Engineering and the utility group agreed on a calculated temperature of 2500°F because much of the data on oxidation and its effects stops at less than 2500°F. B&W suggested a more conservative figure of 2400°F because excessive metal- water reaction rates would be precluded below 2400°F. GE disagreed with the staffs position of 2200°F and stated that 2700°F was acceptable as far as embrittlement of the cladding was concerned and suggested that the interim criteria of 2300°F be retained to ensure that the core never gets into regions of metal-water reactions. Since the owners groups did not agree with each other and all had different calculated temperatures, the staff chose to retain its proposed value of 2200°F.

4. RO 209001K5.04 001/MEM/T2G1/BASIS//209001K5.04//RO/SRO/MODIFIED 11/17/07

During EOI execution, when injection from low pressure systems is required to restore and maintain RPV level, the Core Spray System is NOT on the list of preferred systems for low pressure injection IF all control rods are NOT inserted.

Which ONE of the following describes the basis for this restriction?

A. Cold water from Core Spray creates a rapid pressure reduction and cooldown rates CANNOT be controlled.

B.✓ Core Spray injects directly onto fuel bundles inside the shroud which could damage fuel and cause a power excursion.

C. Core Spray injection creates a steam blanket at the top of the fuel bundles which inhibits heat transfer via steam flow past the fuel.

D. Core Spray does NOT provide sufficient flow to maintain adequate core cooling if an ATWS power level greater than or equal to 80% occurs.

K/A Statement:

209001 LPCS
K5.04 - Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : Heat removal (transfer) mechanisms

K/A Justification: This question satisfies the K/A statement by requiring the candidate to recall the unique heat removal mechanisms of Core Spray and recall a condition where that mechanism can result in unfavorable consequences.

References: OPL171.205 rev 8, pg 60, 11.d and EOIPM Section 0-V-K

Level of Knowledge Justification: This question is rated as MEM due to the requirement to recall or recognize discrete bits of information.

0610 NRC Exam
MODIFIED FROM OPL171.205 #9

QUESTION 33

Given the following conditions for Unit 2:

- An ATWS has occurred
- ALL IRM's and SRM's are FULL IN
- ATWS actions are complete
- Reactor power on the highest reading IRM is on range 7
- Suppression pool temperature is 95°F

Which ONE of the following completes the statements below?

Reactor power on the highest reading IRM is __(1)__ than 5 percent rated power.

In accordance with 2-EOI-1, RPV Control, SLC injection __(2)__ required.

A. (1) LESS
   (2) is

B. (1) LESS
   (2) is NOT

C. (1) GREATER
   (2) is

D. (1) GREATER
   (2) is NOT

Correct Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	211000 A4.04	
	Importance Rating	4.5	

211000 Standby Liquid Control System A4.04 Ability to manually operate and/or monitor in the control room: Reactor power

Explanation: **Answer–B-CORRECT- First Part:** IRM's Reading 75 on Range 8 would correlate to a reactor power of less than 5% (100 on range 8 is 4.875% rated power), range 7 is less than range 8 .Second Part: There are no indications given in the stem that would require initiating SLC.

A– Incorrect –First Part: Correct. Second Part: Incorrect. This is plausible because IAW 2-EOI-1, SLC injection is required if Reactor power is greater than 5%, and the given reactor power is <5%. SLC injection is also required before Suppression Pool Temperature is 110°F under ATWS conditions and the given Suppression Pool temperature is less than 110° F. However, although neither of those conditions are given in the stem, both ODM 4-20 (any oscillations) and EOI-1(peak to peak >25%) require SLC injection when periodic oscillations are observed regardless of Reactor power <5% and Torus Temp <110.

C– Incorrect –First Part: Incorrect- This is plausible because a reactor power of 75 on range 9 of the IRM's is greater than 5% rated power. Second Part: Incorrect. This is plausible because IAW 2-EOI-1, SLC injection is required if Reactor power is greater than 5%, and the given reactor power is <5%. SLC injection is also required before Suppression Pool Temperature is 110°F under ATWS conditions and the given Suppression Pool temperature is less than 110° F. However, although neither of those conditions are given in the stem, both ODM 4-20 (any oscillations) and EOI-1(peak to peak >25%) require SLC injection when periodic oscillations are observed regardless of Reactor power <5% and Torus Temp <110.

D – Incorrect –First Part: Incorrect- This is plausible because a reactor power on range 9 of the IRM's is greater than 5% rated power. Second Part: Correct.

Technical Reference(s):BFN-ODM-4.20, 2-EOI-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.148 V.B.30

Question Source:	Bank:	
	Modified Bank:	
	New:	X

Question History: Previous NRC: None

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

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

4.7.3 RPV Control (EOI-1) (continued)

C. Power Leg of flowchart

If the determination is made that the reactor is subcritical by the use of nuclear instrumentation, then the subsequent actions of AOI-100-1 should be directed. If control rods remain withdrawn from the core and EOI appendices have been directed that will insert the control rods prior to the determination of subcriticality, then the appendices should continue to be used until all control rods are fully inserted into the core.

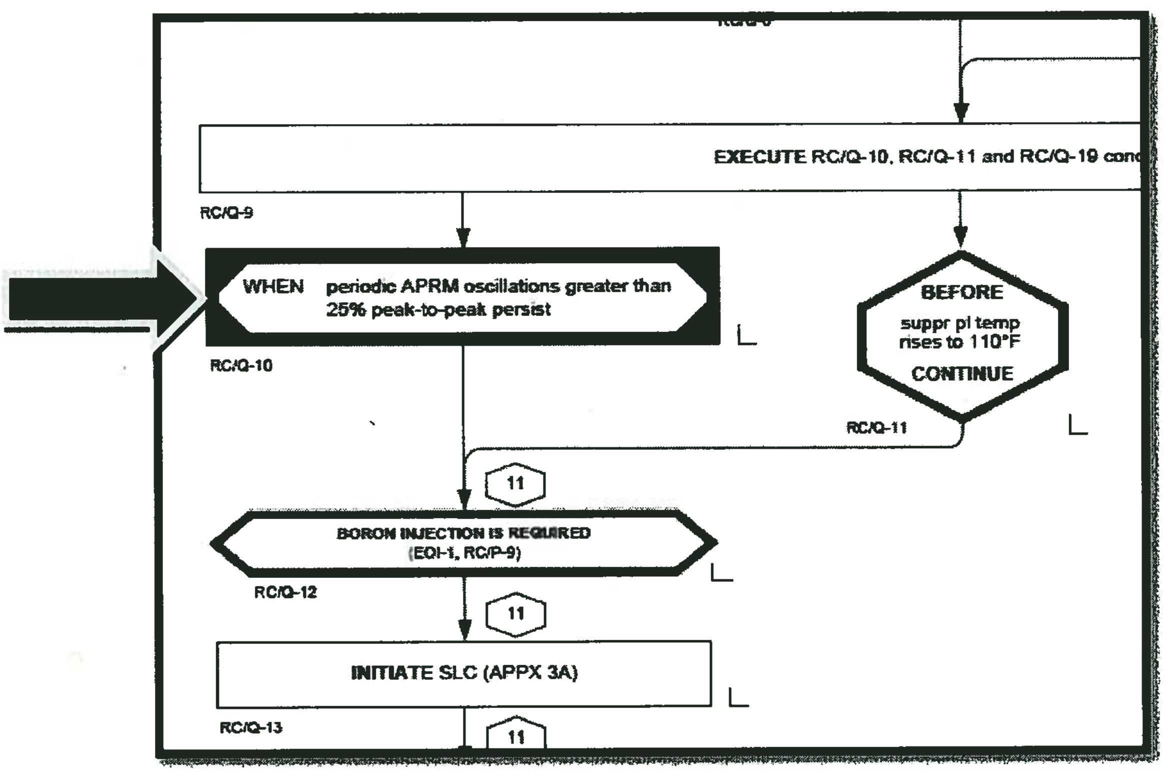
D. ATWS Actions

It is the expectation that IF all control rods CANNOT be verified fully inserted, the OATC should actuate both channels of ARI, run both Recirc pumps to minimum speed and report "*ATWS actions are complete and Reactor Power is* _____", as per AOI-100-1 Hard Card.

During an ATWS, the US should not exit RC/L and enter C-5, LEVEL/POWER CONTROL, until OATC ATWS actions per the AOI-100-1 Hard Card have been completed and Reactor Power has been reported to the US.

When EOI-1, Step RC/Q-9 is reached, IF core oscillations are observed, THEN INITIATE SLC.

When EOI-1, Step RC/Q-10 is reached, IF reactor power is greater than APRM downscale, THEN INITIATE SLC.



QUESTION 34

Given the following conditions for Unit 1:

- The plant is in MODE 1
- 1-SR-3.3.1.1.8(5), MSIV Closure - RPS Trip Channel Functional Test, quarterly test is in progress
- ALL eight RPS Main Steam Isolation Valve Closure Relays are ENERGIZED

When directed by 1-SR-3.3.1.1.8(5), 1-HS-1-27B, MSIV LINE B OUTBOARD TEST pushbutton is depressed and held in order to measure the MSIV stroke time and then released.

Which ONE of the following completes the statements below?

The MSIV LINE B OUTBOARD Closure Relay is de-energized at __(1)__ percent CLOSED.

An RPS half-scram reset __(2)__ be required as a consequence of this step.

A. (1) 10
(2) will

B. (1) 10
(2) will NOT

C. (1) 90
(2) will

D. (1) 90
(2) will NOT

Correct Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	212000 A2.03	
	Importance Rating	3.3	

212000 Reactor Protection System A2.03 Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Surveillance testing

Explanation: **Answer– B- CORRECT-** First Part: CORRECT- The MSIV LINE B OUTBOARD Closure Relay is de-energized at 10% CLOSED. Second Part: CORRECT- Because no other RPS Main Steam Isolation Valve Closure Relays besides the relays for MSIV LINE B OUTBOARD are de-energized there is no RPS half-scram.

A– Incorrect –First part: Correct- see B. Second Part: Incorrect- In order for a RPS half scram to be initiated the RPS closure relays for either A or D Main Steam lines must be de-energized. This is plausible because relays in RPS A and B are de-energized in addition this test could be performed with a fuse removed resulting in a half scram. In addition other MSIV surveillances require Fuses removed which cause a half scram

C – Incorrect – First Part: Incorrect- This is plausible because the RPS relays for MSIV Line B outboard closure are de-energized off of limit switches at <90% OPEN Second Part: Incorrect- see A.

D– Incorrect – First Part: Incorrect- See C. Second Part: Correct- See B.

Technical Reference(s): 1-SR-3.3.1.1.14(5 I); 1-OI-99

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.028 Reactor Protection System Obj.20

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis: X	
10 CFR Part 55 Content:	55.41	5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

3.3 Initiation/Isolation/Trip

A. Both RPS divisions are affected by this SR.

1. Half Scrams for the A RPS System uses the LS-4 Limit Switches on each valve as inputs to the scram circuit. Both relays in each system must be de-energized to cause a half-scrams. (ICS Point SOE035 for Reactor Trip Actuator A1 or A2.)

ICS PT	Bus	Relay	LS-4 for the following valves (<90% Open)			Fuse
SOE009	A1	RLY-99-5AK03A	1-FCV-1-14	or	1-FCV-1-15	FU1-1-15D
		RLY-99-5AK03E	1-FCV-1-26	or	1-FCV-1-27	FU1-1-27A
SOE011	A2	RLY-99-5AK03C	1-FCV-1-37	or	1-FCV-1-38	FU1-1-38A
		RLY-99-5AK03G	1-FCV-1-51	or	1-FCV-1-52	FU1-1-52A

2. Half Scrams for the B RPS System uses the LS-3 Limit Switches on each valve as inputs to the scram circuit. Both relays in each system must be de-energized to cause a half-scrams. (ICS Point SOE036 for Reactor Trip Actuator B1 or B2.)

ICS PT	Bus	Relays	LS-3 for the following valves (<90% Open)			Fuse
SOE010	B1	RLY-99-5AK03B	1-FCV-1-14	or	1-FCV-1-15	FU1-1-15E
		RLY-99-5AK03F	1-FCV-1-37	or	1-FCV-1-38	FU1-1-38B
SOE012	B2	RLY-99-5AK03D	1-FCV-1-26	or	1-FCV-1-27	FU1-1-27B
		RLY-99-5AK03H	1-FCV-1-51	or	1-FCV-1-52	FU1-1-52B

B. This procedure covers testing valves when fuses are removed prior to the performance of this procedure. It covers actions to be taken if appropriate relays are de-energized and fuses are not removed prior to testing associated valves. If relays fail to energize following a valve test, then the procedure should be stopped and the Unit Supervisor notified immediately to ensure corrective actions are performed to determine the failure.

Consideration should be given to half-scrams when removing the fuses for corrective actions to meet Tech Specs. If the fuse is removed as a corrective action, then the relay that failed to energize may be N/A'ed and recorded in Remarks Section of the Surveillance Task Sheet (STS).

C. Under no circumstances will the channel be tested with any of the 8 SCRAM SOLENOID GROUP RESET lights or 4 SYSTEM BACKUP SCRAM VALVE lights extinguished on Panel 1-9-5.

NOTES

1) All hand-switches are located on Panel 1-9-3 in the Unit 1 Control Room unless otherwise noted.
2) Section 6.5 tests the "B" Main Steam Line and inputs to the A1 and B2 RPS Circuits.
3) Prior to performing this section of the procedure, Section 4.0 and 6.1 must be verified to represent current plant configuration.
4) If the "B" Main Steam Line valves will not be tested then Section 6.5 may be N/A'd.

6.5 RPS Channel Functional - Main Steam Line B

[1] **VERIFY** the following annunciators are reset:

- REACTOR CHANNEL A AUTO SCRAM (1-XA-55-5B, window 1). ________
- REACTOR CHANNEL B AUTO SCRAM (1-XA-55-5B, window 2). ________
- MAIN STEAM LINE ISOL VLV POSN HALF SCRAM (1-XA-55-4A, window 30). ________

CAUTION
Valves should not be allowed to go more than 15 percent closed. Should hand-switch 1-HS-1-27A green indicating light fail to illuminate in approximately 5 seconds past the previous time recorded on the Operator Aid, push-button 1-HS-1-27B shall immediately be released and US notified.

NOTES

1) During the performance of the following step, valve stroke time is measured from the moment the push-button is depressed until the green light is illuminated and not when the trip relay de-energized.
2) Steps 6.5[9] and 6.5[10] must be performed concurrently.

Start of Critical Step(s)

[9] **MEASURE** MSIV stroke time by performing the following:

DEPRESS and **HOLD** Push-button MSIV LINE B OUTBOARD TEST, 1-HS-1-27B, until Hand-switch MSIV LINE B OUTBOARD, 1-HS-1-27A, red and green indicating lights are ILLUMINATED, **THEN**

RELEASE push-button 1-HS-1-27B. _______

End of Critical Step(s)

[10] **CHECK** the following relays de-energized:

- RPS CH A1 MAIN STEAM LINE 1B ISOL VLV CLOSURE, 1-RLY-099-05AK03E on Panel 1-9-15 Bay 1 (F3) DE-ENERGIZED prior to or when green indicating light ILLUMINATED. (N/A if 1-FU1-001-0027A is removed.) _____(AC)
- RPS CH B2 MAIN STEAM LINE 1B ISOL VLV CLOSURE, 1-RLY-099-05AK03D on Panel 1-9-17 Bay 3 (F7) DE-ENERGIZED prior to or when green indicating light ILLUMINATED. (N/A if 1-FU1-01-0027B is removed.) _____(AC)

NOTE
The relays in Step 6.5[12] shall be verified energized due to the impact on RPS logic and potential adverse effects on unit operation should a relay failure go undetected. If a relay is not energized, the Unit Supervisor shall be notified immediately and the remainder of the SR may be completed when the appropriate fuse is removed to meet Tech Spec actions for placing the channel in a trip condition.

[12] **CHECK** the following relays are energized:

- RPS CH A1 MAIN STEAM LINE 1B ISOL VLV CLOSURE, 1-RLY-099-05AK03E on Panel 1-9-15 Bay 1 (F3) is ENERGIZED. (N/A if 1-FU1-001-0027A is removed.)

  _______ 1st

  _______ CV

- RPS CH B2 MAIN STEAM LINE 1B ISOL VLV CLOSURE, 1-RLY-099-05AK03D on Panel 1-9-17 Bay 3 (F7) is ENERGIZED. (N/A if 1-FU1-001-0027B is removed.)

  _______ 1st

  _______ CV

NOTES
1) During the performance of this attachment, valve stroke time is measured from the moment the push-button is depressed until the green light is illuminated.
2) This attachment is used when 1-FU-1-52B is pulled for failure of LS-3 on either 1-FCV-1-51 or 1-FCV-1-52 (D MSIV Steam Line).
3) Verification of RPS CH A1 MAIN STEAM LINE 1A ISOL VLV CLOSURE, 1-RLY-099-05AK03A ensures that no half scram will occur in the A1 RPS Bus Logic when 1-FCV-1-26 or 1-FCV-1-27 is tested.

1.0 TESTING MAIN STEAM LINE B WITH 1-FU-1-52B REMOVED

[1] **CHECK** the following:

- RPS CH A1 MAIN STEAM LINE 1A ISOL VLV CLOSURE, 1-RLY-099-05AK03A relay energized on Panel 1-9-15 Bay 1 (F4). ________
- ICS data point SOE009 indicates NOTTRIP. ________
- ICS data point SOE010 indicates NOTTRIP. ________
- ICS data point SOE011 indicates NOTTRIP. ________
- ICS data point SOE012 indicates NOTTRIP. ________

[2] **NOTIFY** the Unit Operator of impending B2 RPS Bus Half-Scram. ________

[4] **MEASURE** the MSIV B OUTBOARD, 1-FCV-1-27, as follows: (N/A if **NOT** being tested.)

[4.1] **DEPRESS** and **HOLD** Push-button MSIV LINE B OUTBOARD TEST, 1-HS-1-27B, until Hand-switch MSIV LINE B OUTBOARD, 1-HS-1-27A, red and green indicating lights are ILLUMINATED, **THEN**

RELEASE push-button. ________

[4.2] **CHECK** the following relays de-energized:

- RPS CH A1 MAIN STEAM LINE 1B ISOL VLV CLOSURE, 1-RLY-099-05AK03E on Panel 1-9-15 Bay 1 (F3) DE-ENERGIZED prior to or when green indicating light ILLUMINATED. ______(AC)

- RPS CH B2 MAIN STEAM LINE 1B ISOL VLV CLOSURE, 1-RLY-099-05AK03D on Panel 1-9-17 Bay 3 (F7) DE-ENERGIZED prior to or when green indicating light ILLUMINATED. ______(AC)

[4.3] **CHECK** annunciator MAIN STEAM LINE ISOL VLV POSN HALF SCRAM (1-XA-55-4A, window 30) in ALARM. ________

[4.4] **CHECK** the following conditions exist on Panel 1-9-5:

- Annunciator REACTOR CHANNEL B AUTO SCRAM (1-XA-55-5B, window 2) is in ALARM. ________

- Annunciator MAIN STEAM LINE ISOL VLV POSN HALF SCRAM (1-XA-55-4A, window 30) can be reset, annunciator is RESET. ________

- Indicating light SCRAM SOLENOID GROUP B LOGIC RESET 1, 2, 3 and 4 (4 total) are EXTINGUISHED. ________

- Indicating light SYSTEM A BACKUP SCRAM VALVE (1-IL-99-5A/AB) (right light), is EXTINGUISHED. ________

- Indicating light SYSTEM B BACKUP SCRAM VALVE (1-IL-99-5A/CD) (right light), is EXTINGUISHED. ________

[4.5] On Panel 1-9-15

CHECK the Indicating light CONTROL ROD TEST SCRAM SOLENOID GROUP B1, B2, B3 and B4 (4 total) are EXTINGUISHED. ________

[4.6] **RESET** the RPS Half-Scram. ________

QUESTION 35

During a plant startup, IRM 'B' is indicating 34 on Range 7.

Which ONE of the following describes the plant response if IRM 'B' range select switch is placed to Range 6?

A. A Reactor Scram occurs

B. A Half Reactor Scram occurs

C. Control Rod Withdraw Block ONLY occurs

D. NEITHER a Control Rod Block, NOR a Scram occurs

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215003 K3.03	
	Importance Rating	3.7	

Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM will have on following: Rod control and information system: Plant-Specific

Explanation: **C CORRECT:** Ranging IRM 'B' down to Range 6 would result in indication of 107.4 which is above the IRM Rod Block set point of 104.6. Since the IRM Rod Block is non-coincidental, one channel will cause the trip.

A-Incorrect – Ranging IRM 'B' down to Range 6 would result in indication of 107.4 which is below the IRM Scram set point of 116.4. Additionally, only one RPS channel would be affected.

B- Incorrect – Ranging IRM 'B' down to Range 6 would result in indication of 107.4 which is below the IRM Scram set point of 116.4. Plausible because only one RPS channel is affected.

D- Incorrect – Plausible if the candidate does not remember that the Rod Block is non-coincidental.

Technical Reference(s): OPL171.020, 3-OI-92A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 0801 #35

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

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

3.0 PRECAUTIONS AND LIMITATIONS

A. In order to minimize their exposure, the IRM detectors should be fully withdrawn as soon as the reactor mode switch is in RUN.

B. All IRM rod block trips are automatically bypassed when the reactor mode switch is in the RUN position.

C. All IRM scram trips are automatically bypassed when the reactor mode switch is in RUN.

D. Only one IRM in each trip system can be bypassed at a time.

E. In order to prevent inadvertent rod withdrawal block or Reactor scram while operating either IRM Bypass selector switch;

1. Always ensure that the previously bypassed channel returns to normal status by observing the applicable High High and High or Inop status lights are extinguished prior to selecting any other channel to be bypassed.

2. After bypassing a channel, the applicable Bypassed status light should be illuminated prior to testing, operating, or working on that channel.

F. To prevent IRM detector drive damage, the CRD service platform should be locked in the stored position with the key removed to allow free movement of the IRMs.

G. The IRMs produce the following trip outputs to the Reactor Manual Control System rod withdrawal block circuitry:

1. High (> 104.6 on 125 scale).

2. Inop (module unplugged, mode switch <u>not</u> in OPERATE, HV power supply low voltage, loss of 24VDC power supply to IRM drawer).

3. Downscale (< 7.5 on 125 scale), bypassed if range switch set to position 1.

4. Detector wrong position (detector <u>not</u> full in).

3.0 PRECAUTIONS AND LIMITATIONS (continued)

H. The IRMs produce the following trip outputs to the Reactor Protection System auto-scram circuitry:

1. High-High (> 116.4 on 125 scale).

2. Inop (module unplugged, mode switch not in OPERATE, HV power supply low voltage, loss of 24VDC power supply to IRM drawer).

3. In addition, by removing the blue shorting links (2 total links), the IRMs are placed in the non-coincident trip logic where any one channel, if tripped, will produce a full reactor scram. The 2/4 Voters are also in this logic such that a trip output from any one Voter yields a full Reactor Scram.

I. The time required to drive a detector from full out to full in is approximately 3 minutes.

J. The INOP TRIP BY-PASS switches located on the IRM drawers on Panel 9-12 by-pass the IRM switch position out-of-operate trip. These switches are to be used only during testing of the IRM channels.

K. [NRC/C] Upon return to service of 24-VDC Neutron Monitoring Battery A or B, Instrument Maintenance is required to perform functional tests on SRMs and IRMs that are powered from the affected battery board. [NRC IE Inspector Follow-up Item 86-40-03]

35. 215003 K4.04

Unit 3 is starting up in Mode 2. IRM 'B' is indicating 34 on Range 7.

Which ONE of the following would result if IRM 'B' range select switch is placed to Range 6?

A. A Reactor Scram occurs.

B. A Half Reactor Scram occurs.

C. A Control Rod Withdraw Block occurs.

D. **NEITHER** a Control Rod Block **NOR** a Scram occurs.

Answer: C

QUESTION 36

A reactor startup is in progress on Unit 1 with the following conditions:

- All SRMs are fully inserted and reading between 90 and 105 cps.
- All IRMs are on Range 2 and reading 30 - 40.

Which ONE of the following completes the statements below?

The SRM RETRACT NOT PERMITTED (9-5A, Window 27) alarm ___(1)__ illuminated.

The CONTROL ROD WITHDRAWAL BLOCK (9-5A, Window 7) annunciator ___(2)__ alarm if the Unit Operator attempts to withdraw SRM "D".

A. (1) is
(2) will

B. (1) is
(2) will NOT

C. (1) is NOT
(2) will

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

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215004 A3.02	
	Importance Rating	3.4	

Ability to monitor automatic operations of the SOURCERANGE MONITOR (SRM) SYSTEM including:Annunciator and alarm signals

Explanation: **A CORRECT:** The SRM RETRACT NOT PERMITTED (9-5A,W27) alarm will be illuminated. If the Unit Operator attempted to withdraw SRM "D" the CONTROL ROD WITHDRAWAL BLOCK (9-5A, W7) annunciator will alarm because the MODE switch is NOT in RUN and the IRMs are below range 3.

B - Incorrect–First Part correct: Plausible if the applicant thinks that the alarm is NOT illuminated due to SRM counts above 100 on some SRMs. Second part: Incorrect: plausible if the applicant thinks that withdrawing the detector will result in the SRM RETRACT NOTPERMITTED annunciator but not the CONTROL ROD WITHDRAWAL BLOCK alarm. (Detector movement is always allowed.)

C - Incorrect – First Part Incorrect: Plausible if the applicant thinks that the alarm is NOT illuminated due to SRM counts above 100 on some SRMs. Second part: correct:

D-Incorrect –First Part Incorrect See C above. Second part: Incorrect: See B above

Technical Reference(s): OPL171.019, 1-ARP-9-5A, W27

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: None

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

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

SRM RETRACT NOT PERMITTED 27 (Page 1 of 1)	Sensor/Trip Point: Relay K22 — 145 cps ± 14 (TRM 100 CPS), if detector not FULL IN and IRM below Range 3 and Rx Mode Switch **NOT** in Run.

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

Probable Cause:

A. Startup (Mode 2) and SRM reading less than 100 cps.
B. SRM detectors are withdrawn to a point where the flux level is less than 100 cps.
C. Sensor failure.

Automatic Action: Control rod withdrawal block.

Operator Action:

A. **CHECK** white retract permissive light **NOT** illuminated on Panel 1-9-5 and illuminated on Panel 1-9-12. ☐
B. **CHECK** SRM reading on meter and recorder Panel 1-9-5. ☐
C. **IF** detector **NOT** full in, **THEN**
INSERT SRM detectors until indicators exceed 100 cps. ☐
D. **REFER TO** TRM Table 3.3.4-1. ☐

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

CONTROL ROD WITHDRAWAL BLOCK 7

(Page 1 of 2)

Sensor/Trip Point:

Relays	Nuclear Instrumentation
3A-K1	Refuel Equipment in Use
3A-K2	High Level In Scram Discharge Volume
	Scram Discharge Volume High Water Level Bypass
	Rx. Mode Switch in Shutdown
	PRNM (ANY APRM, OPRM or RBM)

Sensor Location: Panel 1-9-28
Elevation 593'
Aux. Inst. Room.

Probable Cause:
A. One or more sensors at or above set point.
B. Malfunction of sensor.
C. Control rod drop accident.

Automatic Action: Rod withdrawal block.

Operator Action:

A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s) and **REFER TO** operator action for alarm(s). ☐
B. **IF** alarm due to inadvertent criticality during incore fuel movements, **THEN**
REFER TO 1-AOI-79-2. ☐
C. **IF** alarm is from a control rod drop, **THEN**
REFER TO 1-AOI-85-1. ☐
D. **IF** NO corresponding alarm exists, **THEN** ☐
1. AT ICS console, **DETERMINE** if there is a refuel rod block by selecting Single Point Menu, Single Value Display, and typing DIG090, RETURN. ☐
2. **IF** rod block was from Refuel Floor, **THEN**
NOTIFY Refuel Floor Operator to have dummy plug (Refuel floor between cavity and pool, southside) checked and check jumpers in U-1 Aux. Inst. Room Panel 9-28 Bay 3, if installed per 1-OI-85 Section 8.34. ☐
3. **WHEN** IRM switches are below Range 3 with REACTOR MODE SWITCH not in RUN, **THEN**
CHECK SRM detectors NOT FULL IN. ☐
4. **WHEN** REACTOR MODE SWITCH is in START-UP position, **THEN**
CHECK IRM detectors NOT FULL IN. ☐

Continued on Next Page

QUESTION 37

Unit 1 is operating at 100% power with the following conditions:

- Control Rod 26-27 is selected
- APRM 1 is reading 75%
- All other APRMs are reading 100%

Which ONE of the following completes the statement below?

For these conditions, Rod Block Monitor (RBM) channel A will _____.

A. bypass automatically

B. immediately initiate a RBM Downscale rod block

C. automatically transfer to APRM 3 as its reference APRM

D. enforce a non-conservative RBM Upscale rod block setpoint

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	215005 K3.07	
	Importance Rating	3.2	

Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: Rod block monitor: Plant-Specific

Explanation: **D CORRECT:** The APRM's provide a reference power level for use in the Rod Block Monitor system. A Null sequence is performed every time following selection of a control rod other than a peripheral control rod. When the RBM initially performs a null sequence, it sets the initial flux level surrounding the control rod at 100%. The reference for RBM Channel A is APRM 1, with alternate APRM 3. The reference for RBM Channel B is APRM 2, with alternate APRM 4.
When the reference APRM (in this case APRM 1) reads low the RBM assumes that power is lower than it actually is and will use the trip references for that power level band to assign rod blocks. The reference trip for 75% power (117.0%) is greater than the one for 80% and above (112.0%).

A-Incorrect –Plausible because the Rod Block Monitor automatically bypasses if the Reference STP is below the Low Power Set point (LPSP), OR when Control Rod Selection inputs indicate a peripheral control rod is selected.

B- Incorrect – Plausible because the RBM looks at the reference APRM signal to determine if it is downscale. In this case it would not be.

C- Incorrect – Plausible because APRM 3 will not function as the alternate for RBM A unless the normal reference APRM 1 is Bypassed with the joystick.

Technical Reference(s): OPL171.148

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: Cooper 2011 NRC #5

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

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

2. General Description

a. RBM consists of two redundant channels for monitoring of reactor power in the immediate vicinity of a control rod selected for movement.

(1) Labeled as 'A' RBM and 'B' RBM.

b. RBM is active above 25% Simulated Thermal Power as determined by the reference APRM for the associated RBM channel. — Obj. V.B.22.d

(1) 'A' RBM receives Simulated Thermal Power (STP) input from APRM #1 with alternate APRM being APRM #3 and second alternate channel being APRM #4. — Obj. V.C.6.b

(a) Alternate APRM is automatically selected when associated primary APRM is bypassed

(2) 'B' RBM receives Simulated Thermal Power (STP) input from APRM #2 with alternate APRM being APRM #4 and second alternate channel being APRM #3. — Obj. V.B.22.b

(a) Alternate APRM is automatically selected when associated primary APRM is bypassed

(3) Both RBM channels are bypassed when APRM reference power is less than 25% STP.

(4) Both RBM channels are also bypassed when a peripheral (edge) rod is selected.

(5) RBM must have an internal control rod selected to be active.

3. Basic Operation

a. When a control rod other than a peripheral control rod is selected with STP above 25%, the LPRMs adjacent to the control rod are selected and displayed by the RBM.

b. The RBM initially performs a null sequence, whereby it sets the initial flux level surrounding the control rod at 100%

(1) Null sequence is performed every time following selection of a control rod other than a peripheral control rod.

c. The RBM then determines which setpoint to use based on STP input from the assigned APRM.

Obj. V.B.26.a
Obj. V.B.22.b

(1) Setpoints are determined based on power levels

(a) If STP is between 25% and 60%, the LOW setpoint is used (121.8%)

(b) If STP is between 60% and 80%, the INTERMEDIATE setpoint is used (117.0%).

(c) If STP is above 80%, the HIGH setpoint is used (112.0%)

The setpoints are a percentage of the initial nulled power. The power is initially nulled to 100% based on the LPRM power surrounding the rod. If power then rises to the determined percentage above the initial nulled value, a rod block is imposed.

calculation is completed.

(4) Sets the RBM Automatic Bypass if the Reference STP is below the Low Power Set point (LPSP), OR when Control Rod Selection inputs indicate a peripheral control rod is selected.

Obj. V.B.22
Obj. V.C.6

Cooper 2011 NRC #5

QUESTION: 5 6137

The plant is operating at 100% power with control rod 26-27 selected. Average Power Range Monitor (APRM) channel E is reading 80% while all other APRMs are reading 100%.

What effect will this have on Rod Block Monitor (RBM) channel A?

RBM channel A will…

a. initiate a Flow Reference Off-Normal rod block.

b. immediately initiate an RBM Downscale rod block.

c. enforce a non-conservative RBM Upscale rod block.

d. automatically transfer to APRM C as its reference APRM.

ANSWER: 5 6137

c. enforce a non-conservative RBM Upscale rod block.

QUESTION 38

Given the following conditions on Unit 2:

- Reactor Power is 50%
- RCIC System testing in progress.
- RCIC is aligned for CST-to-CST Recirc with 2-FIC-71-36A, RCIC SYSTEM FLOW CONTROLLER in MANUAL and indicating 500 gpm.

Subsequently,

- A small leak inside primary containment causes drywell pressure to rise to 2.5 psig

Which ONE of the following completes the statement below?

In response to these conditions, indicated RCIC System flow will __________.

A. remain constant at 500 gpm in CST-to-CST Recirc

B. remain constant at 500 gpm, injecting into the RPV

C. lower to 60 gpm through the RCIC Minimum Flow line

D. lower to 0 gpm with the RCIC Pump running at shut-off head

Answer is: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000 A4.08	
	Importance Rating	3.7	

217000 A4.08 Reactor Core Isolation Cooling, ability to manually operate and/or monitor in the Control Room: System Flow

Explanation: **Answer D** – Correct. HPCI has received an Auto Initiation signal and the HPCI/RCIC Test Valve, 2-FCV-73-36 automatically closes on HPCI Auto initiation. For the conditions given, RCIC does not receive an initiation, isolation, or trip signal, so RCIC Pump Injection Valve 2-FCV-71-39 does not automatically open, and the RCIC Pump Min Flow Valve, 2-FCV-71-34 will NOT automatically open.

A- Incorrect. Plausible if the candidate does not recognize that HPCI has received an Auto Initiation signal and the HPCI/RCIC Test Valve, 2-FCV-73-36 receives a close signal on HPCI Auto initiation. For the conditions given, RCIC does not receive an initiation, isolation, or trip signal, and would continue to operate CST-to-CST if the HPCI/RCIC Test Valve, 2-FCV-73-36 had not received a close signal.

B – Incorrect. Plausible if the candidate believes that RCIC has received an Auto Initiation signal. If RCIC had received an initiation signal, RCIC CST Test Valve 2-FCV-71-38 would automatically close and RCIC Pump Injection Valve 2-FCV-71-39 would automatically open, allowing RCIC to inject into the RPV at the manually set 500 gpm.

C – Incorrect. Plausible if the candidate recognizes that HPCI has received an Auto Initiation signal and the HPCI/RCIC Test Valve, 2-FCV-73-36 automatically closes. However, the given conditions do not cause a RCIC initiation signal, and the RCIC Pump Min Flow Valve, 2-FCV-71-34 will NOT automatically open on low flow if an initiation signal is NOT present.

Technical Reference(s): 2-OI-71;2-OI-73

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

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

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

3.0 PRECAUTIONS AND LIMITATIONS

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

1. High RPV water level (+51 in.); 579 in. above vessel zero. The RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, will close at +51 in. and will **RE-OPEN** when RCIC re-initiates at -45 in. RPV water level.
2. Turbine overspeed (Mechanical, 122.3% of rated speed).
3. Pump low suction pressure (10 inches Hg vacuum).
4. Turbine high exhaust pressure (50 psig).
5. Any isolation signal.
6. Remote manual trip (RCIC TURBINE TRIP push-button, 2-HS-71-9A, depressed).

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

1. RCIC steamline space temperature at ≤180°F Torus Area or ≤180°F RCIC Pump Room.
2. RCIC turbine high steam flow (150% flow, 3-second time delay.)
3. RCIC turbine steam line low pressure (73 psig).
4. RCIC turbine exhaust diaphragms ruptured (10 psig).
5. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION push-button, 2-HS-71-54, depressed, only if RCIC initiation signal is present).

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

D. In the presence of a RCIC initiation signal, the RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve will NOT auto open on low flow if an initiation signal is NOT present.

E. RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, will open on receipt of an initiation signal even with RCIC turbine manually tripped resulting in slowly draining CST to Suppression Chamber.

5.1 Automatic Initiation (continued)

K. When HPCI discharge flow is above 1,255 gpm, HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, closes. ☐

[2] If closed, the following valves open:

A. HPCI CST SUCTION VALVE, 2-FCV-73-40 (unless HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27 and HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26 are fully open). ☐

B. HPCI PUMP DISCHARGE VALVE, 2-FCV-73-34. ☐

CAUTION
Opening HPCI STEAM LINE INBD ISOL VALVE, 2-FCV-73-2 and HPCI STEAM LINE OUTBD ISOL VALVE, 2-FCV-73-3 prior to warming the downstream piping can cause water/steam hammer and possible piping damage. This should be avoided whenever possible.

[3] If closed, the following valves will NOT automatically open when an initiation signal is received and must be opened manually via handswitch operation:

A. HPCI STEAM LINE INBD ISOL VALVE, 2-FCV-73-2, using 2-HS-73-2. ☐

B. HPCI STEAM LINE OUTBD ISOL VALVE, 2-FCV-73-3, using 2-HS-73-3A. ☐

[4] If open, the following valves close:

A. HPCI PUMP CST TEST VLV, 2-FCV-73-35. ☐

B. HPCI/RCIC CST TEST VLV, 2-FCV-73-36. ☐

QUESTION 39

A fault causes a loss of 250V DC RMOV Board 3A.

Which ONE of the following completes both statements below?

If a Unit 3 RCIC automatic start signal is received, RCIC __(1)__ start.

The RCIC __(2)__ channel isolation logic is NOT functional.

A. (1) will
(2) 'A'

B. (1) will
(2) 'B'

C. (1) will NOT
(2) 'A'

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

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	217000 K2.02	
	Importance Rating	2.8	

Knowledge of electrical power supplies to the following: RCIC initiation signals (logic).

Explanation: **B CORRECT:** First Part: RCIC will start, 250V DC RMOV Bd B supplies power to Initiation logic (Div 1). Second Part: The B channel (Div 2) isolation logic does NOT have power when the 3A 250VDC RMOV board is lost.

A-Incorrect. First Part: Correct. Second Part: Incorrect. Plausible because this is easily confused. The "B" logic is powered from the "A" 250 DC RMOV board.

C- Incorrect. First Part: Incorrect. RCIC will start, 250V DC RMOV Bd B supplies power to Initiation logic (Div 1). Plausible because this is easily confused. The "B" logic is powered from the "A" 250 DC RMOV board. Second Part: Incorrect. Plausible for the same reason as the First Part.

D- Incorrect. First Part: Incorrect. RCIC will start, 250V DC RMOV Bd B supplies power to Initiation logic (Div 1). Plausible because this is easily confused. The "B" logic is powered from the "A" 250 DC RMOV board. Second Part: Correct.

Technical Reference(s): OPL171.040, ARP 3-9-3C (Window 1), 3-OI-71

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

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

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

RCIC RELAY LOGIC POWER FAILURE	1

(Page 1 of 1)

Sensor/Trip Point:

Logic Bus A: Relay 13A-K1, 13A-K24, or 13A-K40 deenergized.

Logic Bus B: Relay 13A-K34 deenergized.

Sensor Location:

	Logic Bus A	Logic Bus B
	Panel 25-31	Panel 9-33
	Rx Bldg, El 621', R-20 Q-Line	Aux Instr Rm, El 593'

Probable Cause:

A. Cleared Fuse(s)
B. Loss of 250V DC power supply to panels.

Automatic Action:

A. If Bus A fails, the automatic initiation circuit and turbine trip solenoid will **NOT** operate. Channel A isolation logic circuit is lost.
B. If Bus B fails, B channel isolation logic will be inoperative.

Operator Action:

A. **DETERMINE** which logic bus (A or B) has failed and **DISPATCH** personnel(s) to investigate the following:
   1. Logic Bus A
      a. Power supply 250V DC Rx MOV Bd 3B, Compt 8El. Loss of items listed in 2, 3, and 4. ☐
      b. Fuses 3-FU1-071-0018D (13A-F9) and 3-FU1-071-0018E (13A-F10) (10 amp) - Panel 3-25-31, fuse block CC. Loss of initiation, trip, and Logic Bus A isolation logics. ☐
      c. Fuses 3-FU2-071-0029D (3amp) - Panel 3-25-31, fuse block AA. Loss of isolation on Rupture Disc High Pressure. ☐
      d. Fuses 3-FU1-071-0013AA (13A-F28) and 3-FU1-071-13AB (13A-F29) (10 amp) - Panel 3-25-31, fuse block CC. Loss of isolation on Turbine Exhaust Pressure High and Pump Suction Low Pressure. ☐
   2. Logic Bus B
      a. Power supply - 250V DC Rx MOV Bd 3A, Compt 9A1. ☐
      b. Fuses 3-FU2-71-13A-K30 (13A-F23) and 3-FU2-71-13A-K30 (13A-F24) (10 amp) - Panel3-9-33, fuse block GG. Loss of Logic Bus B isolation logic. ☐

B. **REFER TO** Tech Spec Sections 3.3.5.2 and 3.5.3. ☐

References:

0-45E626-1, 2	3-45E620-2	3-45E626-2,3
Technical Specifications 3.3.5.2, 3.5.3		TRM 3.3.3

QUESTION 40

Which ONE of the following completes the statement below?

The Reactor water level instrument(s) __(1)__ provide(s) a confirmatory low reactor vessel water level signal to ADS initiation logic at less than or equal to __(2)__ inches.

NOTE: LIS-3-184 is Reactor Water Level A
LIS-3-185 is Reactor Water Level B
LIS-3-58A-D is Reactor Water Level A-D

A. (1) LIS-3-58A-D
(2) (-) 45

B. (1) LIS-3-58A-D
(2) (+) 2

C. (1) LIS-3-184 and LIS-3-185
(2) (-) 45

D. (1) LIS-3-184 and LIS-3-185
(2) (+) 2

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	218000 K1.03	
	Importance Rating	3.7	

218000 Knowledge of the physical connections and/or cause effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Nuclear boiler instrument system

Explanation: **D CORRECT:** The Reactor water level instruments LIS-3-184 and LIS-3-185 provide a confirmatory low reactor vessel water level signal to ADS initiation logic at ≤ (+) 2 inches.

A-Incorrect. First Part: Incorrect. Plausible because LIS-3-58A-D provide the (-) 122 inch input to ADS logic. Second part: Incorrect. Plausible because (-) 45 inches is the initiation signal for HPCI/ RCIC and is also provided by LIS-3-58A-D.

B- Incorrect. First Part: Incorrect –See A. Second Part: Incorrect- See A.

C- Incorrect. First Part: Correct- see D. Second Part: Correct- See A.

Technical Reference(s): OPL171.003, ARP-9-3C W3 and W24, ARP-9-3F W29

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
c) LT-3-184 and 3-185 (0-60")	All 3 units have Rosemount Xmitter units
Provides low reactor vessel water level signal (+2") to ADS initiation logic. Level 3	
Provides level indication on the ATU cabinets (9-81, 9-82).	
d) LT-3-203 (A-D) (0-60")	All 3 units have Rosemount Xmitter Units
Provides low reactor water level signal (+2") for reactor scram, PCIS Groups 2, 3, 6, 8 isolation, SGT initiation, Reactor Building Ventilation System isolation, and CREV units' start. Level 3	Tech Spec. >0" Use multiple indications
Provides level indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86).	
2) Emergency Systems Range (Wide Range) instruments (+60 to -155") (Referenced to instrument zero)	Part of the Analog Transmitter Trip Syster (ATTS)
Per Regulatory Guide 1.97 Emergency Range identified by Black Labels.	
LT-3-56 (A-D)	
Provides low reactor water level signal (-122") for PCIS isolation Level 1	All 3 units have Rosemount Xmitter Units
Provide level indication on the ATU cabinets (9-83, 9-84, 9-85, 9-86)	

ADS BLOWDOWN TIMERS INITIATED 11 (Page 1 of 2)			
	<u>Sensor/Trip Point</u>:		
	1-PS-064-0057	≥ +2.45 psig	(High Drywell Pressure)
	1-LIS-3-58A-D	≤ -122 inches	RPV Low-Low-Low level (Level 1)
	1-LIS-003-0184 & -0185	≤ +2.0 inches	ADS confirmatory low level (Level 3)

For CS and RHR pump press. switches **REFER TO** 1-XA-55-3C window 10, RHR or CS PUMPS RUNNING AUTO BLOWDOWN PERMISSIVE.

Sensor Location:	1-PNLA-009-0081 Aux. Instr. Room El 593'	1-PNLA-009-0082 Aux. Instr. Room El 593'	1-PNLA-009-0030 & 0033 Aux. Instr. Room El 593'

Probable Cause:

A. Possible LOCA
B. SI/SR in progress.
C. Sensor malfunction.

Automatic Action:

Relays 1-RLY-001-2E-K20(BUS B) and 1-RLY-001-2E-K9(BUS A) close immediately upon receipt of the initiating signals.

Relays 1-RLY-001-2E-K17(Bus B) and 1-RLY-001-2E-K6(Bus A) close after a time delay of 95 seconds. When all four contacts are closed, ADS is initiated.

After 95 seconds from energization of ADS timer relays 1-RLY-001-2E-K34(Bus A) and/or 1-RLY-001-2E-K35(Bus B), relays 1-RLY-001-2E-K6(Bus A) and/or 1-RLY-001-2E-K17(Bus B) will energize.

When either Logic Bus A (1-RLY-001-2E-K9 and 1-RLY-001-2E-K6) or Logic Bus B(1-RLY-001-2E-K20 and 1-RLY-001-2E-K17) are energized, ADS will initiate.

REACTOR
LEVEL LOW
ADS BLOWDOWN
PERMISSIVE

3

(Page 1 of 1)

Sensor/Trip Point:

1-LIS-003-0184	≤ +2.0 inches (Level 3)
1-LIS-003-0185	≤ +2.0 inches (Level 3)

Sensor Location:

1-LIS-003-0184	1-LIS-003-0185
1-PNLA-009-0081	1-PNLA-009-0082
El 593'	El 593'
AUX. INST. ROOM	AUX. INST. ROOM

Probable Cause:

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

Automatic Action: None

Operator Action:

A. **VERIFY** Rx water level by multiple indications. ☐
B. **DISPATCH** personnel to Aux Instrument Rm, El. 593', to check relays energized:
  1. 1-PNLA-009-0030, Relay 2E-K29. ☐
  2. 1-PNLA-009-0033, Relay 2E-K24. ☐

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

References: 1-45E620-2-1 1-47E610-3-1 GE 730E929-1 and -2

Tech Specs 3.5.1 and 3.3.5.1

RX WTR LVL LOW LOW HPCI/RCIC INIT 1-LA-3-58B 29	

(Page 1 of 1)

Sensor/Trip Point:

1-LIS-003-0058A	-45"
1-LIS-003-0058B	-45"
1-LIS-003-0058C	-45"
1-LIS-003-0058D	-45"

Sensor Location: 1-PNLA-009-0081
Auxiliary Instrument Room

Probable Cause:
A. Reactor water level low (Level 2)
B. SI/SR in progress

Automatic Action: Auto initiation of HPCI and RCIC.

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

References: 1-45E620-1-2 1-730E928-2, -3, and -4 1-47E610-3-1

QUESTION 41

Given the following conditions for Unit 1:

- Accident conditions have resulted in an EOI-directed Emergency Depressurization.
- Reactor pressure is currently 106 psig.
- ALL systems functioned as designed.

Which ONE of the following completes the statements below?

The amber HPCI AUTO-ISOL LOGIC A/B lights, on Panel 9-3, are __(1)__.

The amber (HPCI) PCIS LOGIC A/B INTITIATION lights, on the Containment Isolation Status System (CISS) Panel, are __(2)__.

A. (1) lit
   (2) lit

B. (1) lit
   (2) NOT lit

C. (1) NOT lit
   (2) lit

D. (1) NOT lit
   (2) NOT lit

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002 A3.01	
	Importance Rating	3.4	

223002 A3.01 Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: System indicating lights and alarms

Explanation: **C CORRECT:** A valid isolation HPCI signal exists. The Nuclear Steam Supply Valves to the HPCI system will, in fact, automatically isolate (CLOSE). The 'amber' HPCI AUTO-ISOL LOGIC A/B lights, at the HPCI control station on Panel 9-3, will NOT illuminate. This is a system difference from RCIC. RCIC has the same exact lights ('amber' RCIC AUTO-ISOL LOGIC A/B) in the same relative location, which WILL illuminate on the Low Reactor Pressure Isolation.

The Containment Isolation Status System (CISS) Panel amber/green lights that correspond to (HPCI) PCIS LOGIC A/B INTITATION / SUCCESS will BOTH illuminate to provide the operator a visual cue that HPCI has BOTH a demand for isolation and that it has isolated successfully.

A-Incorrect. First Part: Incorrect-Plausible because the HPCI Isolation is 105 psig on Units 2 &3. The candidate will come to the conclusion that the isolation has NOT yet occurred. Second Part: Correct- See C.

B- Incorrect. First Part: Incorrect-See A. Second Part Incorrect. Plausible for the reason stated in the First Part of A.

D- Incorrect. First Part: Correct- See C. Second Part: Incorrect- See B.

Technical Reference(s): 1(2,3)-AOI-64-2B, OPL171.042

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

constant flow.

2. HPCI Isolation

TP-10
Obj. V.B.2.c
Obj. V.C.2.c
Obj. V.D.6
Obj. V.E.7

a. Conditions causing HPCI isolation

(1) Low reactor pressure 105 psig for Units 2&3, (one out of two twice). 110 psig for Unit 1. DCN 51237

Only signal that does not seal in
Unit Difference

(2) High HPCI area temperature ≥165°F (Torus Area) or ≥185°F (HPCI Pump Room) (one out of two twice)

SER 3-05

1.0 PURPOSE

This instruction provides symptoms, automatic actions and operator actions for a Group 4 High Pressure Coolant Injection Isolation.

NOTES
1) On a normal Unit Shutdown this isolation will occur and is not considered abnormal.
2) Unless otherwise specified, all actions or indications are at Panel 2-9-3.

2.0 SYMPTOMS

A. Any one or more of the following annunciators in alarm:

1. HPCI LEAK DETECTION TEMP HIGH 2-TA-73-55 (2-XA-55-3F, Window 10).

2. HPCI TURBINE TRIPPED 2-ZA-73-18 (2-XA-55-3F, Window 11).

3. HPCI TURBINE EXH RUPTURE DISC PRESSURE HIGH 2-PA-73-20 (2-XA-55-3F, Window 17).

4. HPCI STEAM LINE FLOW EXCESSIVE 2-PDA-73-1 (2-XA-55-3F, Window 18).

B. HPCI Turbine tripped and speed lowering on HPCI TURBINE SPEED, 2-SI-73-51.

C. For isolations caused by other than low steam line pressure (100 psig), amber HPCI AUTO ISOL LOGIC A & LOGIC B lights, 2-IL-73-58A and 2-IL-73-58B, are illuminated.

D. HPCI steam line pressure below 105 psig as indicated on HPCI STM LN PRESSURE, 2-PI-73-4A.

QUESTION 42

A plant startup is in progress on Unit 1 and the following conditions exist:

- The Reactor Mode Switch is in STARTUP.
- Two Turbine Bypass Valves are open.
- Reactor pressure is 940 psig and steady.

Subsequently,

- MAIN STEAM LINE CH A FLOW HIGH (Panel 1-9-5B, Window 18) is received due to differential pressure transmitter 1-PDIS-001-0050A failing high.

Which ONE of the following describes the plant response, if any?

A. No effect since the Mode Switch is NOT in RUN.

B. One half PCIS group 1 isolation will occur ONLY.

C. A Full PCIS group 1 isolation will occur ONLY.

D. A Full PCIS group 1 isolation AND a reactor scram will occur.

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	223002 G2.4.31	
	Importance Rating	4.2	

223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off. G2.4.31 Knowledge of annunciator alarms, indications, or response procedures.

Explanation: **Answer B – CORRECT:** A half Group I isolation will occur only.

A – incorrect – plausible. However this Group I signal is not Mode Switch dependent.

C – incorrect – plausible. However a half Group I isolation will occur only

D– incorrect – plausible. However a half Group I isolation will occur only. No reactor scram will occur.

Technical Reference(s): 1-ARP-9-5B window 18

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.017 Objective A.2.a

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: Peach Bottom 2007 #43

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

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

1-ARP-9-5B Rev 19

BFN Unit 1	Panel 9-5 1-XA-55-5B	1-ARP-9-5B Rev. 0019 Page 21 of 42

MAIN STEAM LINE CH A FLOW HIGH

18

(Page 1 of 1)

Sensor/Trip Point:

Relays

16A-K3A	1-PDIS-001-0013A, 0025A	
	1-PDIS-001-0036A, 0050A	112.5 psid
16A-K3C	1-PDIS-001-0013C, 0025C	
	1-PDIS-001-0036C, 0050C	

Sensor Location: Relays, 1-PNLA-009-0015, Aux. Instrument Room.
A switches, 1-PNLA-009-0083, Auxiliary Instrument Room.
C switches, 1-PNLA-009-0085, Auxiliary Instrument Room.

Probable Cause:
A. Indicates possible break outside containment.
B. SI (or SR) in progress.
C. Sensor malfunction.

Automatic Action: Initiates one half PCIS group 1 isolation.

Operator Action:

A. **VERIFY** alarm by checking main steam flow indicators on Panel 1-9-5. ☐

1-FI-46-1	1-FI-46-3
1-FI-46-2	1-FI-46-4

B. **IF** alarm is valid on all steam lines, **THEN PERFORM** the following:
- **MANUALLY SCRAM** the Reactor ☐
- **PLACE** the Reactor Mode Switch in SHUTDOWN ☐
- **CLOSE** MSIVs. ☐
- **REFER TO** 1-AOI-100-1. ☐

C. **IF** one or more flow indicators reading low, **THEN CHECK** all MSIVs open. ☐

D. **REFER TO** 1-AOI-1-3. ☐

E. **REFER TO** Tech Spec Table 3.3.6.1-1. ☐

References: Technical Specifications
1-47E610-1-1   1-45E620-6-2   GE 1-730E927-7

43 ID: N-ILT-5007G-5D-003 Points: 1.00

A plant startup is in progress on Unit 2. The following conditions exist:

- The Reactor Mode Switch is in STARTUP.
- Two Turbine Bypass Valves are open.
- Reactor pressure is 940 psig and steady.

Which one of the following describes the plant response, if any, if "PCIS System I Main Steam Line High Flow" differential pressure transmitter DPT-2-118A fails high?

A. No effect since the Mode Switch is NOT in RUN.

B. ONLY a Half Group I Isolation will occur.

C. ONLY a Full Group I Isolation will occur.

D. a Full Group I Isolation AND a reactor scram will occur.

Answer: B

Question 43 Details

Question Type:	Multiple Choice
Topic:	N-ILT-5007G-5D-003 A plant startup is in progress on Unit 2. The following conditions exist:
System ID:	1365
User ID:	N-ILT-5007G-5D-003
Status:	Active
Always select on test:	No
Authorized for practice:	No
Difficulty:	3.00
Time to Complete:	2
Point Value:	1.00
Cross Reference:	223002 2.1.27
User Text:	
User Number 1:	0.00
User Number 2:	0.00
Comment:	Importance: RO 3.3 / SRO 3.5 Cognitive_Level: High References: PLOT 5007G, ARC 211 H-1 Justification: A. Incorrect - a half Group I isolation will occur. This Group I signal is not Mode Switch dependent. B. Correct - a half Group I isolation will occur only. C. Incorrect - a half Group I isolation will occur only. D. Incorrect - a half Group I isolation will occur only. No reactor scram will occur.

QUESTION 43

Given the following conditions for Unit 2:

- The reactor is at 100% power
- 2-AOI-1-1, Relief Valve Stuck Open, has been entered due to SRV 1-4 opening.

Which ONE of the following completes both statements below?

MAIN STEAM RELIEF VALVE OPEN (2-9-3C, window 25) is sensed off of ___(1)___.

Upon successful closure of SRV 1-4, Generator MW electric will ___(2)___.

A. (1) downstream tail pipe temperature
   (2) remain the same

B. (1) downstream tail pipe temperature
   (2) rise

C. (1) the acoustic monitor
   (2) remain the same

D. (1) the acoustic monitor
   (2) rise

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	239002 G2.4.46	
	Importance Rating	4.2	

239002 Relief/Safety Valves. Ability to verify that the alarms are consistent with the plant conditions.

Explanation: **D CORRECT:** First Part: CORRECT- MAIN STEAM RELIEF VALVE OPEN (2-9-3C, window 25) is sensed off of the acoustic monitor. Second Part: CORRECT- Upon successful closure of SRV 1-4, Generator MW electric will rise.

A-Incorrect. First Part: Incorrect- The temperature element does not generate the alarm. This is plausible as valve leakage is detected by a temperature element and an acoustic monitor on each tailpipe. However, only the acoustic monitor will generate the alarm. Second Part: Incorrect- Generator MWe will not remain the same. This is plausible if the candidate does not understand the relationship between total steam flow and generator MWs.

B- Incorrect. First Part: Incorrect- The temperature element does not generate the alarm. This is plausible as valve leakage is detected by a temperature element and an acoustic monitor on each tailpipe. However, only the acoustic monitor will generate the alarm. Second Part: CORRECT- See D.

C- Incorrect. First Part: CORRECT. Second Part: Incorrect- Generator MWe will not remain the same. This is plausible if the candidate does not understand the relationship between total steam flow and generator MWs.

Technical Reference(s): 2-AOI-1-1, 2-ARP-9-3C window 25;

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a stuck open relief valve.

2.0 SYMPTOMS

A. Annunciator MAIN STEAM RELIEF VALVE OPEN 2-FA-1-1 (2-XA-55-3C, Window 25) is in alarm due to SRV Tailpipe Flow Monitor sensing flow.

B. GENERATOR LOAD recorder, 2-XR-57-57, Panel 2-9-8, indication is lowering.

C. MAIN STEAM/TURBINE STEAM FLOW, flow recorder 2-FR-46-5, Panel 2-9-5, indication is lowering.

D. SUPPRESSION POOL WATER TEMPERATURE recorder, 2-TR-64-161 and SUPPRESSION POOL WATER TEMPERATURE recorder, 2-TR-64-162, indication is rising.

3.0 AUTOMATIC ACTION

None

MAIN STEAM RELIEF VALVE OPEN 2-FA-1-1 25

(Page 1 of 1)

Sensor/Trip Point:

2-FMT-1-4

SRV Tailpipe Flow Monitor

Sensor Location: SRV TAILPIPE FLOW MONITOR, 2-FMT-1-4, on Panel 2-9-3.

Probable Cause: Relief valve is open or leaking.

Automatic Action: None

Operator Action:

A. **CHECK** MSRV DISCHARGE TAILPIPE TEMPERATURE, 2-TR-1-1, on Panel 2-9-47 and SRV Tailpipe Flow Monitor on Panel 2-9-3 for raised temperature and flow indications. ☐

B. **REFER TO** 2-AOI-1-1. ☐

C. **IF** alarm is due to sensor malfunction, **THEN** **REFER TO** 0-OI-55 and OPDP-4. ☐

References: 2-47E610-1-1 2-45N620-2 GE 730E929-2

QUESTION 44

Unit 2 is operating at 100% power.

Which ONE of the following completes the statement below?

IF __(1)__, THEN recirculation pump speed will automatically runback to ___(2)___ rpm.

A. (1) a reactor scram occurs
   (2) 345

B. (1) total FW flow is < 19% for 15 seconds
   (2) 480

C. (1) total FW flow is < 19% for 15 seconds
   (2) 1130

D. (1) any ONE feed pump's flow lowers to < 19% and RPV level lowers to Level +27 inches
   (2) 480

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	259002 K4.01	
	Importance Rating	3.0	

259002 Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Ensuring adequate NPSH for recirculation pumps: Plant-Specific

Explanation: **B CORRECT:** The 28% limiter: IF total FW flow is < 19% for 15 seconds, THEN recirculation pump speed will automatically runback to 480 rpm.

A-Incorrect. –Plausible because scram response affects RFPT speed and runs the speed of the NON-selected pumps to 600 rpm.

C- Incorrect. Plausible because Recirc Pumps receive a 75% (1130 rpm) speed run back signal from Reactor water level at 27" (normal range) and discharge flow of individual RFP < 19%.

D- Incorrect. Plausible because Recirc Pumps receive a 75% speed run back signal from Reactor water level at 27" (normal range) and discharge flow of individual RFP < 19%.

Technical Reference(s): OPL171.007, OPL171.012, 2-ARP-9-4B W35, 2-ARP-9-6C W7,

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Vermont Yankee 2007 NRC #18

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

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

OPL171.007

4. 28% Limiter

a. The 28% (480 rpm) Limiter will initiate an automatic runback of Recirculation Pump speed if Total Feedwater Flow is <19% (15 second time delay) **OR** the pump discharge valve is not full open.

b. The purpose of the limiter is to prevent pump overheating and cavitation of the Recirculation Pumps and Jet Pumps. — ILT Objective 13

c. This limiter enforces pump speeds above 480 rpm.

1) If Recirculation Pump speed is at or below 480 rpm, this signal is automatically bypassed.

2) The operator may lower speed below 480 rpm if desired.

OPL171.012

f. The total feedwater line flow signal is used for the three element control logic.

(1) The individual density compensated feedwater line signals are output to Control Room indicators. — Obj. V.B.1

(2) The density compensated total feedwater line flow signal is output to a Control Room recorder.

(3) The total feedwater line flow is used for initiating the following interlocks:

(a) RWM Enable Setpoint (<16 % rated FF or SF) — Obj. V.B.7, Obj. V.C.6

(b) Recirc Pump NPSH interlock 28% Speed Runback (<19% rated FF)

RECIRC LOOP B FLOW LIMITER ENFORCING 2-FA-96-47 (35) (Page 1 of 1)	Sensor/Trip Point:	
	2-RLY-46-5Q 2-RLY-46-5R	Total FW flow is ≤ 19% (15 sec TD),or Discharge valve is < 90% open.
	2-RLY-46-5U 2-RLY-46-5V	Either of the following: A. Individual RFW pump flow is < 19% and Rx water level ≤ 27". B. Reactor Scram

Sensor Location:

2-RLY-46-5Q	Panel 9-18, Aux Inst Rm, El 593.
2-RLY-46-5R	Panel 9-18, Aux Inst Rm, El 593.
2-FCV-068-0079	Drywell, El 549.
2-RLY-46-5U	Panel 9-18, Aux Inst Rm, El 593
2-RLY-46-5V	Panel 9-18, Aux Inst Rm, El 593

Probable Cause:

A. Loss of one or more feed pumps.
B. Recirculation pump discharge valve **NOT** fully open.
C. Reactor Scram

Automatic Action:

A. Recirculation pump speed will be limited to 28%, if total Feedwater flow is < 19% OR Recirc pump discharge valve is < 90% open.
B. Recirculation pump speed will be run back to 75%, if individual Reactor Feed Pump flow is < 19% AND Reactor level is ≤ 27".
C. Recirculation pump speed will run back to 75% if Reactor Scram occurs.

Operator Action:

A. **IF** Recirculation Pump 2B speed is limited (with Feedwater flow and reactor level in operating limits), **THEN**
CHECK Recirculation Pump 2B discharge valve, 2-FCV-68-79, fully open. ☐
B. **IF** run back or limiting condition has occurred due to loss of one or more reactor Feedpumps, **THEN**
REFER TO 2-AOI-3-1. ☐
C. **IF** Recirc pump speed has lowered, **THEN**
REFER TO 2-AOI-68-1A or 2-AOI-68-1B. ☐
D. **REFER TO** 2-OI-68, to reset runback. ☐

References:

2-45N620-5	GE 731E320-3	FSAR 13.6.2
2-731E320RH-13	2-45E779-21	2-729E895-8

RFWCS GROSS FAILURE 1-LA-46-5C 7	<u>Sensor/Trip Point:</u> 1-XM-046-0097/54	Any gross failure of the RFW Control System or CP1001 control processors that results in lockup or failure of the system.

(Page 1 of 1)

Sensor Location: Panel 1-9-97 (behind 1-9-5)

Probable Cause: Major failure of the RFW Control System

Automatic Action:

A. RFP/RFPT speeds could lower to minimum.
B. Faulty indications may occur with instrumentation associated with RFWCS.
C. Possible reactor scram on low level at +2.0 inches.

CAUTION

Narrow Range level instruments 1-LI-3-53, 1-LI-3-60, 1-LI-3-206 and 1-LI-3-253 are unreliable during a gross failure of the RFWCS.

Operator Action:

A. **MONITOR** reactor water level using the following instrumentation: ☐
- 1-LI-3-208A and 1-LI-3-208D on Panel 1-9-5
- 1-LI-3-58A and 1-LI-3-58B on Panel 1-9-5
- 1-LI-3-208B and 1-LI-3-208C on Panel 1-9-3

B. **ATTEMPT** to control reactor water level with RFPT speed control Raise/Lower switches in MANUAL GOVERNOR (switches in depressed position with associated amber lights illuminated). ☐

References: 1-45E620-7 1-729E895-10

Vermont Yankee 2007 NRC #18

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K4.01	
	Importance Rating	3.0	

Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Ensuring adequate NPSH for recirculation pumps: Plant-Specific

Proposed Question: Common 18

The plant is operating normally at 100% power, when the feedwater flow summer fails to zero. The 9-5 Operator takes manual control of and quickly restores level to the normal band.

As a result of the failure, what additional automatic action will occur?

A. Feedwater pumps trip on sensed low flow.

B. Recirc pumps runback to minimum speed.

C. Rod Worth Minimizer rod block.

D. Recirc pumps runback to 40% positioner stroke.

Proposed Answer: B

Explanation (Optional): KA match due to interlock related to adequate NPSH for recirc pumps.

A. Incorrect -The feedwater pump low flow trip signal is not provided by the feed flow summer. Each pump has a flow element on its suction to provide low flow protection.

B. Correct – FSAR section 7.9.4.3. – the runback setpoint is based on input that feedwater flow is greater than the minimum specified for recirc or jet pump cavitation considerations (reactor recirc system NPSH interlock value).

C. Incorrect - The RWM only uses the feed flow and steam flow signals to determine when it should actuate blocks for rods that are positioned outside the prescribed sequence.

D. Incorrect –recirc runs back to minimum.

QUESTION 45

Given the following conditions on Unit 2:

- Reactor Power is 100%
- 480V D/G Aux Bd “A” has been de-energized.

Subsequently,

- A reactor scram occurs
- Reactor water level is (-) 1 inches and recovering

Which ONE of the following completes the statement below?

Differential Pressure for the operating SGT Trains are read on ________ .

A. ONLY Panel 1-9-25

B. BOTH Panel 1-9-25 and Panel 2-9-25

C. ONLY Panel 1-9-20

D. BOTH Panel 1-9-20 and Panel 2-9-20

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	261000A1.01	
	Importance Rating	2.9	

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM (SGTS) controls including: System flow

Explanation: **B CORRECT:** With 480V D/G Aux Bd A de-energized, the SGT A is without power, therefore only SGT Trains B and C are running. SGT "B" D/P is monitored on panel 1-9-25, and SGT "C" D/P is monitored on panel 2-9-25.

A-Incorrect. This is plausible as SGT Trains "B" and "C" Decay Heat Removal Dampers are powered from 480V D/G Aux Bd A. If the candidate incorrectly assumes "A" is the only train of SGT that is running, the D/P can be monitored from panel 1-9-25 ONLY.

C- Incorrect. This is plausible as SGT Trains "B" and "C" Decay Heat Removal Dampers are powered from 480V D/G Aux Bd A. If the candidate incorrectly assumes "A" is the only train of SGT that is running, the flow rate can be monitored from panel 1-9-20.

D- Incorrect. This is plausible with 480V D/G Aux Bd A de-energized, the SGT "A" is without power, therefore only SGT Trains B and C are running. SGT "B" Flow is monitored on panel 1-9-20, and SGT "C" Flow is monitored on panel 2-9-20.

Technical Reference(s): OPL171.018, 0-OI-65

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

6.0 SYSTEM OPERATIONS

NOTE
SGT System is normally in a Standby Readiness condition.

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

CAUTION
Iodine desorption is expected when an SGT train's charcoal filter temperature rises to 270°F following a LOCA.

A. SGT Train A(B)(C) Temperature:

- less than 150°F with no release of radioactive material
- less than 200°F with a release of radioactive material. ☐

PANEL 1-9-25	
SGT Train A	
0-TI-65-46	0-TI-65-44
SGT Train B	
0-TI-65-47	0-TI-65-45

PANEL 2-9-25	
SGT Train C	
0-TI-65-64	0-TI-65-64

B. SGT Train A(B)(C) Differential Pressure- less than 8" H20: ☐

PANEL 1-9-25		PANEL 2-9-25	
SGT Train A	0-PDI-65-5	SGT Train C	0-PDI-65-53
SGT Train B	0-PDI-65-27		

6.0 SYSTEM OPERATIONS (continued)

C. SGT Total Flow- sum of 0-FI-65-50B/1(2)(3) and 0-FI-65-71B/1(2)(3). ☐

PANEL 1-9-20	PANEL 2-9-20	PANEL 3-9-20
0-FI-65-50B/1	0-FI-65-50B/2	0-FI-65-50B/3
0-FI-65-71B/1	0-FI-65-71B/2	0-FI-65-71B/3

[2] **IF** an SGT train has been shut down after a LOCA **AND** the train's charcoal filter temperature reaches 150°F, **THEN**

INITIATE decay heat removal. **REFER TO** Section 8.0. ☐

INSTRUCTOR NOTES

B. Component Description

1. Piping — Obj. V.B.3, Obj. V.C.1, Obj. V.D.3, Obj. V.E.2

   a. Suction pipe from normal Reactor Zone Ventilation System exhaust

   b. Suction pipe from Refuel Zone

   c. Suction pipe from Drywell

   d. Suction from Suppression Chamber

   e. Suction pipe from HPCI gland exhauster (2")

   f. f30" discharge piping to plant stack (2)

2. Valves — Obj. V.B.4, Obj. V.C.2, Obj. V.E.2

   a. Motor-operated butterfly type

   b. The following system valves fail open (all others fail closed):

      (1) DMP 65-67 SGT Filter Bank C Outlet Damper

      (2) DMP 65-17 SGT Fan A inlet Damper

      (3) DMP 65-39 SGT Fan B inlet damper

   c. SGT valves are powered from same power as their associated SGT fan. — Obj. V.E.5

      (1) SGT 'A' - 480V D/G Aux Bd A

      (2) SGT 'B' - 480V D/G Aux Bd B

      (3) SGT 'C' - 480V SGT Bd

      (4) Except for the decay heat removal dampers which are powered from the adjacent SGT power supply as followed

         (a) SGT A - 480V D/G Aux Bd B — Obj. V.E.5

INSTRUCTOR NOTES

         (b) SGT B - 480V D/G Aux Bd A

         (c) SGT C - 480V D/G aux Bd A

QUESTION 46

Given the following conditions on panel 9-23-3:

- 0-43-203-A, 4kV COM BD A MAN/AUTO SELECT, switch is in the MAN position
- 4kV COM BD A to the NORM FDR BKR 1118 is CLOSED
- 4kV COM BD A to the ALT FDR BKR 1422 is OPEN
- While the Unit Operator is **HOLDING** 0-HS-203-A/2A, 4kV COM BD A ALT FDR BKR 1422 in the CLOSE position, he/she **PLACES** 0-HS-203-A/3A, 4kV COM BD A NORM FDR BKR 1118 to TRIP position.

Which ONE of the following completes the statement below?

The 4kV Common Board A will __(1)__ Transfer AND will be supplied from __(2)__.

A. (1) SLOW
(2) Start Bus 1A

B. (1) SLOW
(2) Start Bus 1B

C. (1) FAST
(2) Start Bus 1A

D. (1) FAST
(2) Start Bus 1B

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262001 K4.03	
	Importance Rating	3.1	

Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Interlocks between automatic bus transfer and breakers

Explanation: **C CORRECT:** Automatic delayed transfer from the normal to the alternate source is initiated by undervoltage on the normal source, subject to voltage check on the alternate source, and automatic return is initiated by normal voltage on normal source. **Manual transfers in either direction are fast type.** The ALTERNATE source to 4KV Common Board is Start Bus 1A.

A- Incorrect. First Part. Incorrect. Plausible because 4KV Shutdown Boards transfers are SLOW. Second Part: Correct.

B-Incorrect. – First Part. Incorrect. Plausible because 4KV Shutdown Boards transfers are SLOW. Second Part: Incorrect. Plausible because Start Bus 1B is the Alternate source for 4KV Common Board B.

D- Incorrect. First Part: Correct. Second Part: Incorrect. Plausible because Start Bus 1B is the Alternate source for 4KV Common Board B.

Technical Reference(s): OPL171.036, 0-OI-57A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

Illustration 1
(Page 4 of 7)

Auxiliary Power Supplies and Bus Transfer Schemes

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE	REMARKS
9	4kV Common Bd. A	Unit SS TR 1A (BKR 1118)	Start Bus 1A (BKR 1422)	Automatic delayed transfer from the normal to the alternate source is initiated by undervoltage on the normal source, subject to voltage check on the alternate source, and automatic return is initiated by normal voltage on normal source. Manual transfers in either direction are fast type.
10	4kV Common Bd. B	Unit SS TR 2A (BKR 1218)	Start Bus 1B (BKR 1522)	

ITEM	BOARD AND/OR MAIN BUS	NORMAL	ALTERNATE 1	ALTERNATE 2	ALTERNATE 3
11	4-kV Shutdown Bd. A	Shutdown Bus 1 (BKR 1614)	Diesel Generator A (BKR 1818)	Shutdown Bus 2 (BKR 1716)	Shutdown Bd. 3EA (BKR 1824)
12	4-kV Shutdown Bd. B	Shutdown Bus 1 (BKR 1616)	Diesel Generator B (BKR 1822)	Shutdown Bus 2 (BKR 1714)	Shutdown Bd. 3EB (BKR 1828)
13	4kV Shutdown Bd. C	Shutdown Bus 2 (BKR 1718)	Diesel Generator C (BKR 1812)	Shutdown Bus 1 (BKR 1624)	Shutdown Bd. 3EC (BKR 1814)
14	4kV Shutdown Bd. D	Shutdown Bus 2 (BKR 1724)	Diesel Generator D (BKR 1816)	Shutdown Bus 1 (BKR 1618)	Shutdown Bd. 3ED (BKR 1826)

8.22 Control Room Transfer of 4kV COM BD A Power Supplies (continued)

[10] To transfer 4kV COM BD A to the ALT FDR BKR 1422, **PERFORM** the following: (Otherwise **N/A**)

[10.1] **DETERMINE** whether "E" service air compressor should be placed in service <u>**OR**</u> station an operator at "F" service air compressor to restart following the transfer. ☐

[10.2] On Panel 1-9-8, **PLACE** 1-XS-202-1, 4kV BD/BUS/XFMR VOLTAGE SELECT switch to START BUS 1A and **VERIFY** voltage is between 3950 and 4400 Volts. ☐

[10.3] **PLACE** and **HOLD** 0-HS-203-A/2A, 4kV COM BD A ALT FDR BKR 1422 to CLOSE. ☐

[10.4] **PLACE** 0-HS-203-A/3A, 4kV COM BD A NORM FDR BKR 1118 to TRIP. ☐

[10.5] **CHECK CLOSED** 4kV COM BD A ALT FDR BKR 1422. ☐

[10.6] **CHECK OPEN** 4kV COM BD A NORM FDR BKR 1118. ☐

QUESTION 47

The Control Bay AUO reports that the Unit 1 Unit Preferred Inverter has the following indication lamps lit on the inverter:

- 1-IL-252-0001Q (Red Lamp) Low DC Voltage
- 1-IL-252-0001P (Red Lamp) Low DC Disconnect

Which ONE of the following completes both statements below?

The Normal DC Source, Battery Board __(1)__, is no longer in service to the inverter.

In accordance with 1-ARP 9-8B Window 35, UNIT PFD SUPPLY ABNORMAL, inverter loads will auto transfer to __(2)__.

A. (1) Four
(2) Battery Board Five

B. (1) Four
(2) the Alternate AC Source

C. (1) Five
(2) Battery Board Four

D. (1) Five
(2) the Alternate AC Source

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002 A2.04	
	Importance Rating	3.2	

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: Abnormal battery operation: BWR-1

Explanation: **D CORRECT:** The Normal DC Source to the Inverter is Battery Board 5. If a problem with Battery Board 5 occurred that caused it to become disconnected from the inverter, the 1-IL-252-0001Q (Red Lamp) Low DC Voltage and 1-IL-252-0001P (Red Lamp) Low DC Disconnect would illuminate indicating that there was no DC input to the inverter. The Static Transfer Switch would automatically shift the inverter loads to the Alternate AC Source.

A- Incorrect. First Part: Incorrect. Plausible because battery Board 4 is the Alternate DC Source. Second Part: Incorrect. Plausible because Battery Board 5 is the Normal DC source to the inverter.

B- Incorrect. First Part. Incorrect. Plausible because battery Board 4 is the Alternate DC Source. Second Part: Correct.

C-Incorrect. – First Part. Correct. Second Part: Incorrect. Plausible because battery Board 4 is the Alternate DC Source to the inverter. However shifting DC sources is a manual operation.

Technical Reference(s): 0-OI-57C, OPL171.102

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

UNIT PFD SUPPLY ABNORMAL 35

(Page 1 of 1)

Sensor/Trip Point:

Relay SE - loss of normal DC power source.
Relay TS - DC Xfer switch transfers to Emergency DC Power Source.
Regulating Transformer Common Alarm.
1-INV-252-001, INVT-1 System Common Alarm.

Sensor Location: EL 593' 250V DC Battery Board 2

Probable Cause:

A. Loss of normal DC power source
B. DC power transfer.
C. Relay failure
D. INVT-1 System Common Alarms
   1. Fan Failure Rectifier
   2. Over temperature Rectifier
   3. AC Power Failure to Rectifier
   4. Low DC Voltage
   5. High DC Voltage
   6. Low DC Disconnect
   7. Fan Failure Inverter
   8. Alternate Source Failure
   9. :Low AC Output Voltage
   10. High Output Voltage
   11. Inverter Fuse Blown
   12. Static Switch Fuse Blown
   13. Over Temperature Inverter

E. PFD Regulating XFMR Common Alarms
   1. Transformer Over temperature
   2. Fan Failure
   3. CB1 Breaker Trip
   4. CB2 Breaker Trip

Automatic Action:

A. Auto transfer to DC Power Source on Rectifier failure.
B. Auto transfer to Alternate AC supply (Regulated Transformer) on Inverter failure.

Operator Action:

A. IF 120V AC Unit Preferred is lost, THEN REFER TO 1-AOI-57-4. ☐
B. REFER TO appropriate portion of 0-OI-57C. ☐

References:

0-45E641-2	1-45E620-11	1-3300D15A4585-1
10-100467	0-20-100756	20-110437

Illustration 4
(Page 1 of 2)

UNIT PREFERRED INVERTER LAMP INDICATORS

INVERTER LAMP INDICATORS	CONDITION / ACTION	ADDITIONAL INFORMATION
1-IL-252-0001E (Red Lamp) Common Alarm	Common light that comes in along with a specific alarm	
1-IL-252-0001L (Red Lamp) Alternate Source Failure	Indicates Alternate AC source not available. Static Switch will not transfer to this power.	
1-IL-252-0001N (Red Lamp) Low AC Output Voltage	Static Switch auto transfers to Alternate. Transfer back to inverter (Normal) is inhibited.	
1-IL-252-0001H (Red Lamp) High AC Output Voltage	No Auto transfer to Alternate. Manual transfer to the Alternate supply is required.	Maintenance on the inverter is probably required.
1-IL-252-0001L (Red Lamp) Inverter Fuse Blown	Auto Static Switch transfer to Alternate. Manual / Auto transfer back to inverter (normal) blocked.	
1-IL-252-0001R (Red Lamp) Over Temperature Inverter	Ambient Temperature High and/or Inverter Fan Failure. No auto actions.	Ops should evaluate continued operation. There are two fans per rectifier / inverter bays. One should provide adequate cooling.
1-IL-252-0001U (Red Lamp) Static Switch Fuse Blown	Static Switch auto transfers to Alternate. Transfer back to inverter (normal) blocked.	
1-IL-252-0001Q (Red Lamp) Low DC Voltage (warning light)	Low DC input voltage will cause the Battery input breaker B1 to trip and load will auto transfer to alternate.	
1-IL-252-0001J (Red Lamp) High DC Voltage	No auto actions. Plant Battery/Charger 5/4 voltage is high. If voltage cannot be adjusted, then place on Alternate.	The unit is designed to be in service with DC power applied. If DC is removed, place loads on maintenance and shutdown the unit.
1-IL-252-0001P (Red Lamp) Low DC Disconnect	Static Switch auto transfers to Alternate and trips DC breaker B1.	The unit is designed to be in service with DC power applied. If DC is removed, place loads on maintenance and shutdown the unit.
1-IL-252-0001G (Red Lamp) Fan Failure Rectifier	If one fan is still running, the unit should avoid an overtemperature alarm.	If both Fans are tripped then transfer to alternate or maintenance is required.

Instructor Notes

(3) Unit Preferred Inverter 1-INV-252-001 power supplies:

- Normal AC: 480V RMOV 1A Comp 19E
- Alternate AC: via 1-XFA-251-001 from 480 RMOV1A Comp 19E
- Normal DC: Batt Bd 5, Pnl 3, Bkr 313:
- Alternate DC: Batt BD 4, Pnl 3, Bkr 324

Obj V.B.1.a
Instructor: Stress that to swap DC power supplies , the static switch must be lined up to the 'Alternate Source'

(4) Inverter operation is as follows:

a. The Normal AC supply from 480V RMOV BD1A Comp 19E is the preferred source of power to the inverter.

b. On a loss of normal AC to rectifier, the diode from the DC supply will now forward bias and supply DC to the inverter with no interruption of power. IF normal AC returns then the diode will become reversed bias and the normal AC will supply the inverter output voltage.

c. On a loss of inverter output (inverter failure) the AC load will automatically shift to the alternate supply via the regulating transformer supply with no loss of loads. When the inverter output returns, the supply to the loads must be **manually** returned to the inverter output via the static switch.

Ref 0-OI-57C Sect 8.7

QUESTION 48

The Unit 1 Unit Preferred System Inverter and Rectifier, 1-INV-252-001, inverter output fails.

Which ONE of the following completes both statements below?

If ONLY the Unit Preferred Inverter output fails, Battery Board 1 Cabinet 11 will automatically be powered through the Unit Preferred Inverter Static Switch via the ___(1)___.

When the Unit Preferred Inverter output is restored, the Unit Preferred Static Switch will ___(2)___ to the NORMAL Unit Preferred Inverter power supply.

A. (1) Unit 2 Unit Preferred MMG set
(2) automatically return

B. (1) Unit 2 Unit Preferred MMG set
(2) NOT automatically return

C. (1) Unit Preferred Regulating XFMR1
(2) automatically return

D. (1) Unit Preferred Regulating XFMR1
(2) NOT automatically return

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	262002K6.03	
	Importance Rating	2.7	

262002 UPS (AC/DC)Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (UPS) (A.C. / D.C.): Static inverter

Explanation: **D CORRECT** – First Part: On a loss of inverter output (inverter failure) the AC load will automatically shift to the alternate supply via the regulating transformer supply with no loss of loads. Second Part: When the inverter output returns, the supply to the loads must be manually returned to the inverter output via the static switch.

A- Incorrect. First Part: Incorrect. Plausible as the Alternate Emergency source is the Unit 2 MMG . Second Part: Incorrect – Must be manually returned to inverter when output restored. Plausible in that this is a feature of the inverter. However, Auto re-transfer from Alternate to Inverter has been defeated by BFN for all conditions.

B- Incorrect. First Part: Incorrect see A. Second Part: Correct – See D.

C- Incorrect. First Part: Correct – See D Second Part: Incorrect- See A.

Technical Reference(s): OPL171.102; 0-OI-57C

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: BFN 0801 #49

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

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

Instructor Notes

(3) Unit Preferred Inverter 1-INV-252-001 power supplies:

- Normal AC: 480V RMOV 1A Comp 19E
- Alternate AC: via 1-XFA-251-001 from 480 RMOV1A Comp 19E
- Normal DC: Batt Bd 5, Pnl 3, Bkr 313:
- Alternate DC: Batt BD 4, Pnl 3, Bkr 324

Obj V.B.1.a
Instructor: Stress that to swap DC power supplies , the static switch must be lined up to the 'Alternate Source'

(4) Inverter operation is as follows:

a. The Normal AC supply from 480V RMOV BD1A Comp 19E is the preferred source of power to the inverter.

b. On a loss of normal AC to rectifier, the diode from the DC supply will now forward bias and supply DC to the inverter with no interruption of power. IF normal AC returns then the diode will become reversed bias and the normal AC will supply the inverter output voltage.

c. On a loss of inverter output (inverter failure) the AC load will automatically shift to the alternate supply via the regulating transformer supply with no loss of loads. When the inverter output returns, the supply to the loads must be **manually** returned to the inverter output via the static switch.

Ref 0-OI-57C Sect 8.7

3.0 PRECAUTIONS AND LIMITATIONS (continued)

R. Unit 1 Unit Preferred normal supply to Unit Preferred Battery Bus 1 Cabinet 11 is using inverter 1-INV-252-0001. The alternate supply is through Unit Preferred Regulating Transformer XFMR1. To prevent possible inverter or transformer damage, all operation with Unit 2 MMG (Alternate Emergency) source, must be performed by a dead bus transfer.

S. UNIT PFD SYSTEM INVER AND RECTIFIER, 1-INV-252-0001 can only operate in parallel with the Alternate AC source, XFMR1, for a short period of time. Damage to the inverter/rectifier or transformer could occur if operated in parallel for an extended period.

T. Transfer of DC power from Normal Feeder Battery Board 5 to Alternate Feeder Battery Board 4 is limited by the availability of the Battery Board 4 changer. If the charger is not available, the load is not to be transferred.

U. Relays will be considered ENERGIZED if the movable contact fingers (metal plate) are pushed back away from the relay case glass front cover with movable fingers and stationary fingers making contact, and DE-ENERGIZED if the contact fingers (metal plate) are in the forward position towards the relay case glass front cover against the rest bar (movable fingers and stationary fingers are NOT touching)

49. 262002 K6.03

The Unit 1 Unit Preferred Inverter output fails.

Which ONE of the following completes the statements?

If **ONLY** the Unit Preferred Inverter output failed, the Reactor Feedwater Level Control PDS indication __(1)__ .

When the Unit Preferred Inverter output is restored, the Unit Preferred Static Switch will __(2)__ to the NORMAL Unit Preferred Inverter power supply.

A. (1) was **NOT** lost.
   (2) automatically return

B. (1) was lost.
   (2) automatically return

C. (1) was **NOT** lost.
   (2) **NOT** automatically return

D. (1) was lost.
   (2) **NOT** automatically return

QUESTION 49

The $\pm$ 24V DC Neutron Monitoring Battery charger is being operated in the equalizing mode.

Which ONE of the following completes the statements below?

In equalize, the charger output voltage to the battery will be __(1)__ than when in the float mode.

The $\pm$ 24V Neutron Monitoring Battery is capable of supplying connected loads for a MAXIMUM of __(2)__ hours.

A. (1) lower
   (2) 3

B. (1) lower
   (2) 4

C. (1) higher
   (2) 3

D. (1) higher
   (2) 4

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	263000 A1.01	
	Importance Rating	2.5	

Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate

Explanation: **C CORRECT:** In equalize, the charger output voltage to the battery will be higher than when in the float mode. The $\pm$ 24V Neutron Monitoring Battery is capable of supplying connected loads for 3 hours with each cell having a residual charge NOT less than 1.75 volts.

A- Incorrect. First Part. Incorrect- Plausible because the float voltage range (264.0-270.0 VDC) is slightly lower than the equalize charge voltage range (276.0-379.6 VDC). Second Part: Correct- see A.

B- Incorrect. First Part. Incorrect See C. Second Part: Incorrect- See B.

D-Incorrect. – First Part. Correct- see A. Second Part: Incorrect- Plausible because the SBO coping duration is 4 hours.

Technical Reference(s): 0-OI-57D, 3-SR-3.8.4.3 (SB-3EB)

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Duane Arnold 2009 NRC #14
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis :

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

5.8 Placing Unit 1 ± 24V DC Neutron Monitoring Battery A(B) in Service to Battery Board 1 (continued)

CAUTION

The ± 24V Neutron Monitoring Battery is capable of supplying connected loads for 3 hours with each cell having a residual charge NOT less than 1.75 volts at the end of the 3 hour period. Battery voltage should be monitored frequently.

Date ________

7.4 Battery Recharge

[1] **VERIFY** no load on the load bank.

________ 1st

________ CV

[2] **UNPLUG** the load bank from the Battery Discharge receptacle in 250V DC Distribution Board 3EB.

________ 1st

________ CV

[3] **DISCONNECT** load cable from positive(+) pin at the battery discharge plug. ________

[4] **DISCONNECT** load cable from negative(-) pin at the battery discharge plug. ________

[5] **DISCONNECT** the BCT-2000 Battery Capacity Test per Attachment 4.

________ 1st

________ CV

[6] **CLOSE** 3-BKR-248-03EB/DC BATTERY CHARGER 3-CHGA-248-3EB DC OUTPUT BKR.

________ Ops.

[7] **CLOSE** 3-BKR-248-03EB/AC BATTERY CHARGER 3-CHGA-248-3EB AC SUPPLY BKR.

________ Ops.

NOTE
Float Voltage for the Charger should be set near the upper end of the range.

[8] **ADJUST** charger float voltage, if required, using 3-RES-248-03EBA FLOAT VOLTAGE CONTROL ADJUST to obtain 264.0 to 270.0 VDC. ________

[9] **REQUEST** Predictive Monitoring/Electrical Maintenance to perform periodic thermal scans of 3-CHGA-248-3EB SHUTDOWN BDS 250VDC BATTERY CHARGER SB-3EB, during Equalizing Charge. ________

Date ________

7.4 Battery Recharge (continued)

[10] **PLACE** the Battery Charger in equalize using 3-TMR-248-03EB BATTERY CHARGER 3-CHGA-248-3EB TIMER, and **ADJUST** 3-RES-248-03EBB EQUALIZE VOLTAGE CONTROL ADJUST so the voltage is between 276.0 and 279.6 VDC. ________

Duane Arnold 2009 NRC #14

14. The Div 1 125 VDC battery charger is being operated in the equalize mode.

    Which one of the following describes:

    (1) the voltage relationship between the charger and the batteries

    AND

    (2) the design rating of the batteries if a loss of AC power occurred?

    a. (1) In equalize, the charger output to the battery will be a higher voltage than when in the float mode
       (2) The 125 VDC batteries are sized to supply emergency power for a 4-hour time period.

    b. (1) In equalize, the charger output to the battery will be a lower voltage than when in the float mode
       (2) The 125 VDC batteries are sized to supply emergency power for a 4-hour time period.

    c. (1) In equalize, the charger output to the battery will be a higher voltage than when in the float mode
       (2) The 125 VDC batteries are sized to supply emergency power for an 8-hour time period.

    d. (1) In equalize, the charger output to the battery will be a lower voltage than when in the float mode
       (2) The 125 VDC batteries are sized to supply emergency power for an 8-hour time period.

QUESTION 50

Which ONE of the following battery boards is the NORMAL power supply to the Unit **2** Main Turbine Emergency Bearing Oil Pump?

A. Battery Board #2

B. Battery Board #4

C. Battery Board #5

D. Battery Board #6

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	263000 K2.01	
	Importance Rating	3.1	

Knowledge of electrical power supplies to the following: Major D.C. loads

Explanation: **B CORRECT:** Battery Board #4 is the NORMAL power supply to the Unit #2 Main Turbine Emergency Bearing Oil Pump.

A-Incorrect. -Incorrect. Plausible because Battery Board #2 supplies **Unit 2** safety related equipment.

C- Incorrect. -Incorrect. Plausible because Battery Board #5 supplies the NORMAL power to the **Unit 1** Main Turbine Emergency Bearing Oil Pump.

D- Incorrect. -Incorrect. Plausible because Battery Board #6 supplies the NORMAL power to the **Unit 3** Main Turbine Emergency Bearing Oil Pump.

Technical Reference(s): 0-OI-57D Attach 3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

3.0 ATTACHMENT DATA (continued)

Performed On: ________________

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
Turbine Building - Battery Board Rm 4 - El 586'				
310	0-BKR-280-0004/310 125V INSTRUMENTATION ON PNL 3	ON	0	____ ____
313	0-BKR-280-0004/313 NORMAL SUPPLY FOR U2 PFD AC SYS MMG SET	ON	2	____ ____
314	0-BKR-280-0004/314 U-2 TURB EMER BRG OIL PUMP MOTOR NORM FDR		2	Aligned by 2-OI-47B

3.0 ATTACHMENT DATA (continued)

Performed On: ____________

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
	Turbine Building - 250V DC Battery Board 5 - El 617'			
216	0-BKR-280-0005/216 3A RFPT EMERG OIL PUMP 3A3		1	Aligned by 3-OI-3
303	0-BKR-280-0005/303 ALT SUPPLY U3 PFD AC SYSTEM MMG SET	ON	3	____ ____
313	0-BKR-280-0005/313 UNIT 1 PFD INVERTER NORM FEEDER	ON	1	____ ____
314	0-BKR-280-0005/314 UNIT 1 TURB EBOP NORMAL FEEDER		1	Aligned by 1-OI-47B

3.0 ATTACHMENT DATA (continued)

Performed On: ______________

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
	Turbine Building - 250V DC Battery Board 6 - El 617'			
215	0-BKR-280-0006/215 RFPT 1C EMERG OIL PMP 1C3		1	Aligned by 1-OI-3
216	0-BKR-280-0006/2163C RFPT EMERG OIL PUMP 3C3		3	Aligned by 3-OI-3
303	0-BKR-280-0006/303 ALTERNATE SUPPLY FOR U2 PFD AC SYSTEM MMG SET	ON	2	____ ____
313	0-BKR-280-0006/313 NOR SUPPLY U3 PFD AC SYSTEM MMG SET	ON	3	____ ____
314	0-BKR-280-0006/314 UNIT 3 TURB EBOP NORMAL FEEDER		3	Aligned by 3-OI-47B

QUESTION 51

Which ONE of the following completes both statements below?

The 3A diesel generator A/C driven lube oil soakback pump __(1)__ provides oil to the turbocharger bearing area.

If the A/C driven lube oil soakback pump is lost, the 3A D/G __(2)__ be able to start and load.

A. (1) does
   (2) will

B. (1) does
   (2) will NOT

C. (1) does NOT
   (2) will

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

Correct Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	264000 K6.03	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction ofthe following will have on the EMERGENCY GENERATORS(DIESEL/JET) :Lube oil pumps

Explanation: **Answer– A-CORRECT-** First Part:The 3A diesel generator lube oil soakback pump provides oil to turbocharger area when the Diesel Generator is not running. Second Part: If the lube oil soakback pump is lost, the 3A D/G will still be able to start and load.

B– Incorrect – First Part: Correct. Second Part: Incorrect. Plausible because part of the Prestartup/ Standby Readiness requirements in OI-82, Section 4.2 is to check that the lube oil circulating pump is in service.

C– Incorrect –First Part: Incorrect. Plausible because not every oiled component is supplied by the soakback pump. For example, the piston cooling oil subsystem supplies oil for piston cooling and lubrication of the piston pin bearing surface. Second Part: Correct.

D – Incorrect –First Part: Incorrect. Plausible because not every oiled component is supplied by the soakback pump. For example, the piston cooling oil subsystem supplies oil for piston cooling and lubrication of the piston pin bearing surface. Second Part: Incorrect. Plausible because part of the Prestartup/ Standby Readiness requirements in OI-82, Section 4.2 is to check that the lube oil pump is in service.

Technical Reference(s):OPL171.038, 3-OI-82

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: None

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

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

a. **Lubricating oil system**

b. Engine lubricating oil is supplied from the crank case which holds approximately 465 gallons. At the full mark on the dip stick this gives 236.16.gals of usable oil. The engine consumes approximately 0.98 gallons of oil per hour at full load operation.

c. The engine lubricating oil system is a combination of four subsystems: main lubricating oil, piston cooling oil, scavenging oil, and soakback lube oil ACSP and DCSP. The main oil pump, piston oil pump, and scavenging oil pump are engine-driven.

1) The main lubricating oil subsystem supplies oil at 80-90 psig to the various moving parts of the engine. The main lube oil pump takes oil from the strainer housing and discharges it into the main oil manifold, from which the main bearings, gear train, turbocharger, cam shaft and other moving parts are supplied.

2) The piston cooling oil subsystem supplies oil for piston cooling and lubrication of the piston pin bearing surface. The pump receives its oil from a common suction with the main lube oil pump and delivers oil to the two piston oil cooling manifolds, which supply the pistons.

Level must be within 2" of full mark (197 useful gallons) on diesel crankcase dipstick

Obj. NLOR/NLO-2 TP-1
OI-82, P&L for brg. Oil press. Sw. operation
Add oil to level in crankcase if <-2" on dipstick with idle speed. (<-3" while DG is running; P&L illust. pg 2 note 1. See 0-OI-82, illustration 2 for system variables.

3-OI-82

BFN Unit 3	Standby Diesel Generator System	3-OI-82 Rev. 0130 Page 11 of 219

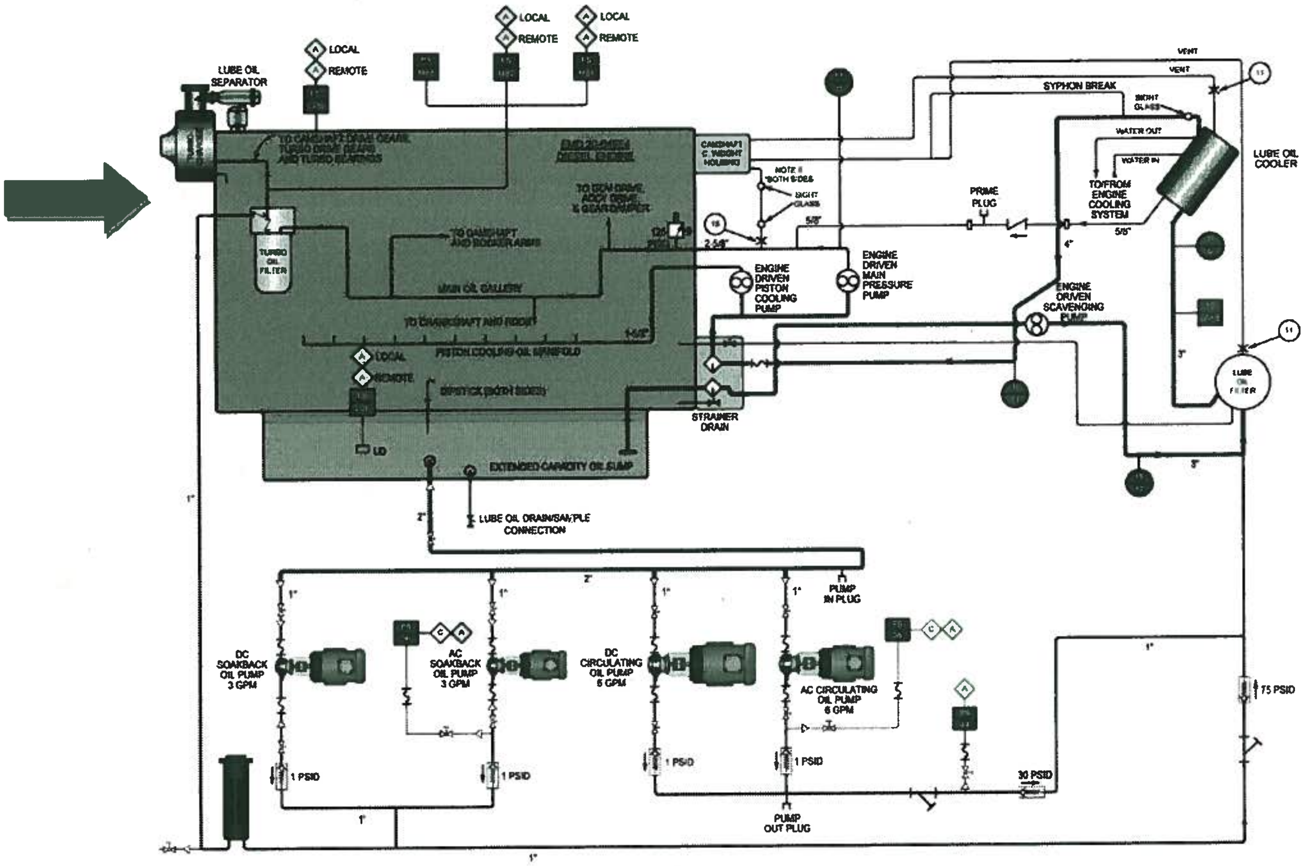
3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. If the DG has been operating >4 hours at <50% load AND the DG could not be operated >50% load prior to shutting down, then a run at 100% for 30 minutes should be performed as soon as possible by loading the DG incrementally as already described in the procedure.

F. Fast starts during the time period of 15 minutes to 3 hours after shutdown should be avoided except in an emergency condition. However, manual slow starts from the Engine Control Cabinet are allowed during this time period. This minimizes the possibility of damage to the turbocharger thrust bearing.

G. Lube oil modification for 3A, 3C and 3D Diesel Generator eliminates the vendor recommended prohibition on fast starts of the diesel generators during the 15 minute to 3 hour time period following shutdown. Due to the addition of a separate AC driven soakback pump supplying oil to the turbocharger, all the oil supplied from the AC motor driven circulating oil pump will be directed to the oil rack, maintaining it full at all times, and ensuring the rapid buildup of oil pressure upon engine start. [DCN 69454-5, 7&8]

POST 12 YEAR PM



4.2 DG 3A Prestartup/Standby Readiness Requirements (continued)

NOTE
The preferred position for the duplex Motor-driven fuel pump Discharge Filter, is "L" (left).

C. MOTOR-DRIVEN FUEL PUMP DISCHARGE FILTER, 3-FLT-018-0787A, selector lever in the "L" (LEFT) position. ☐

OR

IF MOTOR-DRIVEN FUEL PUMP DISCHARGE FILTER, 3-FLT-018-0787A, selector lever is found in the "R" (RIGHT) position, **THEN**

VERIFY an SR/WO has been initiated to clean the "L" (LEFT) filter. ☐

D. DG 3A RIGHT BANK AIR PRESSURE, 3-PI-086-0601A, between 165 and 200 psig. ☐

E. DG 3A LEFT BANK AIR PRESSURE, 3-PI-086-0602A, between 165 and 200 psig. ☐

F. EXPANSION TANK WATER LEVEL, 3-LG-82-5A, between 4.75 inches (STOP LOW) and 8.75 inches (STOP FULL). ☐

G. LO CLR LUBE OIL OUTLET TEMP, 3-TI-82-18A, greater than 85°F. (**RECORD** actual value) [PER 424092] ☐

DISTRACTOR

H. Lube oil circulating pump is in operation. ☐

- DG-3A TURBOCHARGER COMP BEARING LUBE OIL PRESS INDR, 3-PI-082-1001A, indicates between 6 and 75 psig oil pressure. ☐

BFN 1108 #51

51. 264000 K6.03 NEW/H

Which ONE of the following completes the statements below?

The lube oil circulating pump __(1)__ provide oil to the turbocharger bearing area.

If the 3A diesel generator lube oil circulating pump is lost, the 3A D/G __(2)__ be able to start and load.

A. (1) does NOT
(2) will

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

C. (1) does
(2) will

D. (1) does
(2) will NOT

CORRECT ANSWER C

QUESTION 52

The following plant conditions exist:

- "G" Air Compressor is in service
- 0-TCV-32-2945, Cooling System Heat Exchanger Bypass Valve, has failed such that the air compressor has lost its cooling

Which ONE of the following completes the statement below?

In accordance with 0-OI-32, Control Air System, the Compressor "G" will trip if ___(1)___ reaches ___(2)___.

A. (1) lube oil temperature
   (2) 115° F

B. (1) lube oil temperature
   (2) 125° F

C. (1) air discharge temperature
   (2) 115° F

D. (1) air discharge temperature
   (2) 125° F

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	300000 A3.02	
	Importance Rating	2.9	

Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: Air temperature

Explanation: **D CORRECT:** Compressor 'G' will trip if air discharge temperature reaches 125° F.

A-Incorrect. –First Part: Incorrect. Plausible because lube oil temperature is a trip parameter. Second Part: Incorrect. Plausible because the candidate may confuse this with the Compressor Discharge Air Header overpressure Relief Valve 0-RFV-032-2926, which is set to lift at approximately 115 psig.

B- Incorrect. - First Part: Incorrect. Plausible because lube oil temperature is a trip parameter. Second Part: Correct.

C- Incorrect. –First Part: Correct. Second Part: Incorrect. Plausible because the candidate may confuse this with the Compressor Discharge Air Header overpressure Relief Valve 0-RFV-032-2926, which is set to lift at approximately 115 psig.

Technical Reference(s): OPL171.054, 0-OI-33, 0-OI-32

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 1108 #52

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

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

3.0 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 - 130°F
5. Lube Oil Temperature low - 85°F
6. Air Temperature high, Stage 1 - 125°F
7. Discharge Air Temperature high - 125°F
8. Seal Air Pressure low - 6 psig

3.4 Compressor Discharge Air Header overpressure

Relief Valve 0-RFV-032-2926 is set to lift at approximately 115 psig.

Illustration 10
(Page 9 of 11)

Ingersoll-Rand Control Air Compressors A,B,C,D Operations

1.0 INGERSOLL-RAND INTELLISYS DISPLAY (continued)

ALARM	SENSOR	FUNCTION	SETPOINT
High bearing oil temperature	5OTT (0-PT-032-004 6A(BCD))	Oil temperature in manifold	>170°F (160°F when not running)

M. Control Air Compressors A, B, C, D will trip and must be manually reset on the following signals:

1. Associated 480V SD or Common Board undervoltage
2. Inlet restriction: > 13.3 psig vacuum when running unloaded or >3 psig vacuum when running loaded
3. High intercooler pressure: Intercooler pressure >43 psig and 1st stage discharge temperature is >410°F running loaded or >5 psig when running unloaded
4. High 2nd stage pressure: >140 psig
5. High line air pressure: >140 psig
6. Low bearing oil pressure: <34 psig for 2 seconds
7. High 1st stage temperature: >440°F (Max operator setpoint)
8. High intercooler air temperature: >140°F
9. High 2nd stage temperature: >486°F (Max operator setpoint)
10. High bearing oil temperature: >160-170°F
11. Main motor overload
12. Fan motor overload
13. Sensor failure
14. Emergency stop: EMERGENCY STOP pushbutton engaged

BFN 1108 NRC #52

52. 300000 K4.03 MODIFIED/L

The following plant conditions exist:

- G Air Compressor is in service
- 0-TCV-32-2945, Cooling System Heat Exchanger Bypass Valve, has failed such that the air compressor has lost its cooling

Which ONE of the following completes the statement below?

In accordance with 0-OI-32, Control Air System, the Compressor 'G' trip setpoint for __(1)__ is __(2)__.

A. (1) air discharge temperature
(2) 120°F

B. (1) lube oil temperature
(2) 130°F

C. (1) lube oil temperature
(2) 120°F

D. (1) air discharge temperature
(2) 130°F

CORRECT ANSWER B

QUESTION 53

All three Units were operating at rated power when 480V Reactor MOV Board 2B was de-energized.

Which ONE of the following systems has valve operators that are directly affected by this power loss?

A. Raw Service Water (RSW)

B. Raw Cooling Water (RCW)

C. Unit 2 Stator Water Cooling

D. Reactor Building Closed Cooling Water (RBCCW)

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	400000 K2.02	
	Importance Rating	2.9	

Knowledge of electrical power supplies to the following: CCW valves

Explanation: **D CORRECT:** The Reactor Building Closed Cooling Water (RBCCW) isolation valve 2-FCV-70-47 is powered from 480V Reactor MOV Board 2B.

A- Incorrect. - Plausible because Raw Service Water (RSW) pump 2A is powered from 480TMOV Board 2C.

B-Incorrect. - Plausible because Raw Cooling Water (RCW) supply to stator water cooling is powered from 480 Reactor MOV Board 2C.

C- Incorrect. - Plausible because Stator Water Cooling pump 2B is powered from 480 Unit Board 2B.

Technical Reference(s): 0-OI-57B/ATT-3H

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: Oyster Creek 2008 NRC #52

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

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

BFN Unit 0	Attachment 3E Unit 2 480V Shutdown Boards and Reactor MOV Boards Electrical Lineup Checklist	0-OI-57B/ATT-3E Rev. 0182 Page 20 of 33

3.0 ATTACHMENT DATA (continued)

Performed On: ____________________

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
	480V Reactor MOV Board 2B **Control Bay -Div II, Electric Board Room 2B - El 593', R-14R** **(2-45E751 - 3 & 4)**			
6A	2-BKR-070-0047 CLOSED COOLING WATER ISOLATION VALVE FCV-70-47		2	ALIGNED BY 2-OI-70

3.0 ATTACHMENT DATA (continued)

Performed On: ___________________

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV

480V Reactor MOV Board 2C
Reactor Bldg - El 565', R-13T
(2-45E751 - 5 & 6)

4E	CRD COOLING WATER PRESSURE CONTROL VALVE (PCV-85-27)		2	ALIGNED BY 2-OI-85
5A	2-BKR-064-0135 STEAM VAULT EXH BOOSTER FAN		2	ALIGNED BY 2-OI-30B
5B	BUS HT EXC ALTERNATOR CLR & H2 CLG WTR SHUTOFF MOV FCV-24-25		2	ALIGNED BY 2-OI-24
5C	CLOSED COOLING WATER SPARE PUMP SUCTION VALVE FCV-70-67		0	ALIGNED BY 2-OI-70
5E	2-BKR-085-0065 CRD PUMP 1A SUCTION ISOLATION SHUTOFF VALVE FCV-85-65		2	ALIGNED BY 2-OI-85
6B	2-BKR-024-0041 STATOR COOLING HX RCW SHUTOFF VALVE		2	ALIGNED BY 2-OI-24

3.0 ATTACHMENT DATA (continued)

480V Unit Board 2B
Turbine Bldg - El 604', T-11C
(2-45E747 - 2)

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
2C	2-BKR-225-0002B/2C MAIN FEEDER BREAKER	CLOSED	0	____ ____
3C	EMER FEEDER BREAKER	CONNECTED AND OPEN	0	____ ____
4A	2-BKR-225-0002B/4A TURBINE MOV BOARD 2B NORMAL FEEDER	CLOSED	2	____ ____
4B	2-BKR-035-0036 STATOR COOLING WATER PUMP 2B		2	ALIGNED BY 2-OI-35A

3.0 ATTACHMENT DATA (continued)

Panel/Breaker Number	Component Description	Required Position	Unit	Initials 1st/IV
	480V Turbine MOV Board 2C Turbine Bldg - El 586', T-11C (2-45E753 - 5 & 6)			
5A	2-BKR-269-02C/05A SPARE	OFF	2	____ ____
5C	2-BKR-269-002C/05C SPARE	OFF	2	____ ____
5D	2-BKR-003-0152 RFPT 2C TURNING GEAR MOTOR		2	ALIGNED BY 2-OI-3
6D	EMERGENCY SUPPLY BREAKER	CONNECTED AND OPEN	2	____ ____
7A	2-BKR-024-0927C RCW STRAINER 2C		2	ALIGNED BY 2-OI-24
7B	RAW SERVICE WATER PUMP 2A		0	ALIGNED BY 0-OI-25

Oyster Creek 2008 NRC #52

OC ILT 07-1 RO NRC Exam KEY

Question 52

The plant was at rated power when Motor Control Center 1B21A was de-energized.

Which of the following systems has valve operators that are directly affected by this power loss?

A. ESW

B. TBCCW

C. RBCCW

D. Circulating Water

Question # Answer	52 C	Question Developer Initials/Date: NTP 12/8/07

Knowledge and Ability Reference Information						RO	SRO
400000 K2.02 Knowledge of electrical power supplies to the following: CCW valves					Importance Rating	2.9	2.9
Level	RO	Tier #	2	Group #	1		
References	3004 sh. 3						
Explanation:	The plant was at power when MCC 1B21A was de-energized. Of the systems listed, only RBCCW has motor operators powered from this bus. Answer C is correct.						
References to be provided during exam:	None						
Learning Objective	2621.828.0.0035 0005 State how service water, shutdown cooling, RWCU, primary containment, AC electrical distribution and chemical treatment systems interrelate with the RBCCW system.						
Question Source	Bank		Modified Bank		New	X	
Question Cognitive Level:	Memory or Fundamental Knowledge	X 1:F	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	7	55.43				
Time to Complete: 1-2 minutes							

QUESTION 54

Which ONE of the following completes the statement below?

A loss of __(1)__ will <u>directly</u> affect the cooling of the __(2)__ .

A. (1) Raw Cooling Water (RCW)
   (2) Drywell Blowers

B. (1) Raw Cooling Water (RCW)
   (2) CRD pump speed changer and thrust bearing

C. (1) Reactor Building Closed Cooling (RBCCW)
   (2) Drywell/Torus ΔP air compressor

D. (1) Reactor Building Closed Cooling (RBCCW)
   (2) Recirculation pump Variable Frequency Drives

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	201001 K6.06	
	Importance Rating	2.8	

Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System : Component cooling water systems: Plant-Specific

Explanation: **B CORRECT:** The CRD pump speed changer and thrust bearing are cooled by Raw Cooling Water (RCW).

A-Incorrect –First Part: Correct. Second Part: Incorrect. Plausible because Drywell blowers are cooled by Reactor Building Closed Cooling (RBCCW).

C-Incorrect – First Part: Incorrect. Plausible because the Drywell/Torus ΔP air compressor cooled by Raw Cooling Water (RCW). Second Part: Incorrect.

D- Incorrect – First Part: Incorrect. Plausible because the Recirculation pump Variable Frequency Drives are cooled by Raw Cooling Water (RCW).

Technical Reference(s): OPL171.048, OPL171.047, OPL171.005

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Quad Cities 2009 NRC #54

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

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

B. RCW System Loads — Obj. V.B.3

1. Turbine Building loads: — Obj. V.C.2

   a. Generator stator coolers — Obj. V.D.2
   b. Generator exciter air coolers — Obj. V.E.3
   c. Generator hydrogen coolers
   d. Bus duct cooler
   e. Generator breaker cooing water system
   f. Condensate pump coolers — Right Unit - Different Train
   g. Condensate pump bearing oil coolers
   h. Condensate booster pump coolers
   i. Condensate booster pump motor coolers

INSTRUCTOR NOTES

   j. Condensate booster pump oil coolers — Note that U-2 main turbine and EHC oil coolers are supplied from U-3 portion of RCW header.
   k. RFP turbine lube oil coolers
   l. Off-gas pre-cooler
   m. Off-gas chiller
   n. Condenser vacuum pump seal water coolers
   o. Main turbine oil coolers
   p. EHC oil coolers
   q. Seal water injection pumps (u-3 only) — EECW automatic backup for CA Compressors
   r. Control and station service air compressors
   s. Vacuum priming pump seal water

s. Vacuum priming pump seal water

2. Reactor Building loads: — Obj. V.B.3

   a. Drywell/Torus ΔP air compressor — Obj. V.C.2
   b. Recirculation pump Variable Frequency Drive heat exchangers — Obj. V.D.2
   c. CRD pump speed changer oil cooler and pump thrust bearing cooler — Obj. V.E.3
   d. RBCCW heat exchangers — EECW automatic backup supply

2. RBCCW Heat Loads

a. Essential loop loads — Obj. V.B.2

- Drywell Blowers(10) — Obj. V.D.2
- Reactor recirculation pump motor coolers (2)
- Reactor recirculation pump seal coolers (2)
- Drywell equipment drain sump heat exchanger (1)

Quad Cities 2009 NRC #54

EXAMINATION ANSWER KEY

U.S. Nuclear Regulatory Commission 2009 SRO Written Exam (Quad Cities)

54	ID: QDC.ILT.15530	Points: 1.00

Which ONE of the following components is cooled by the Turbine Building Closed Cooling Water (TBCCW) system?

A. Generator H_2 coolers

B. CRD Pump Seal Coolers

C. Control Room HVAC Train B

D. Main Turbine Lube Oil coolers

Answer: B

Answer Explanation:

Answer: The TBCCW system supplies cooling water flow to the CRD pumps.

Distractor 1 is incorrect: Generator H_2 coolers are supplied by the Service Water system.
Distractor 2 is incorrect: Control Room HVAC Train B is supplied by the RBCCW system.
Distractor 3 is incorrect: Main Turbine Lube Oil Coolers are supplied by the Service Water system.

Reference: QCOA 3800-03 Rev 8
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 2 Group: 2

Question Source: Quad ILT Bank (QDC.ILT.01091)
Question History: N/A

10 CFR Part 55 Content: 41.2 to 41.9

Comments: None

Associated objective(s):

SR-3800-K18 (Freq: LIC=I)
LIST the plant systems which are supported by TBCCW and DESCRIBE the nature of support.

201001.K1.06 Component cooling water systems: Plant-Specific (RO=2.8 / SRO=2.8)

QUESTION 55

Unit 3 is at 88% Reactor Power with 3-SR-3.1.3.3, Control Rod Exercise Test for Withdrawn Control Rods, in progress when the following indications are received:

- APRM DOWNSCALE / OPRM INOP, (3-9-5A, Window 4) is in alarm
- APRM 1 indicates 0%

Which ONE of the following completes the statement below?

This condition will result in a Control Rod __(1)__, which requires __(2)__ to continue the surveillance.

A. (1) withdrawal block ONLY
   (2) bypassing APRM 1

B. (1) withdrawal block ONLY
   (2) placing APRM 1 Mode Switch to INOP

C. (1) withdrawal AND insert block
   (2) bypassing APRM 1

D. (1) withdrawal AND insert block
   (2) placing APRM 1 Mode Switch to INOP

Correct Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	201002A2.04	
	Importance Rating	3.2	

Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Control rod block

Explanation: **Answer– A- CORRECT-** First Part – APRM Downscale will result in a Control Rod withdrawal block ONLY with Mode Switch in RUN. Second Part – In accordance with 3-ARP-9-5A, IF APRM failed downscale, THEN BYPASS channel. REFER TO 3-OI-92B.

B – Incorrect – First Part: Correct. Second Part: Incorrect – Removing APRM Mode Switch from OPER position will result in a trip signal. Plausible in that the APRM is Inoperable.

C – Incorrect – First Part: Incorrect – A Control Rod insert block signal will not be generated from an APRM downscale. Plausible in that various RMCS signals do result in a Control Rod insert block signal. Second Part: Correct.

D– Incorrect – First Part: Incorrect – A Control Rod insert block signal will not be generated from an APRM downscale. Plausible in that various RMCS signals do result in a Control Rod insert block signal. Second Part: Incorrect – Removing APRM Mode Switch from OPER position will result in a trip signal. Plausible in that the APRM is Inoperable.

Technical Reference(s): 3-ARP-9-5A, OPL171.029

Proposed references to be provided to applicants during examination: None

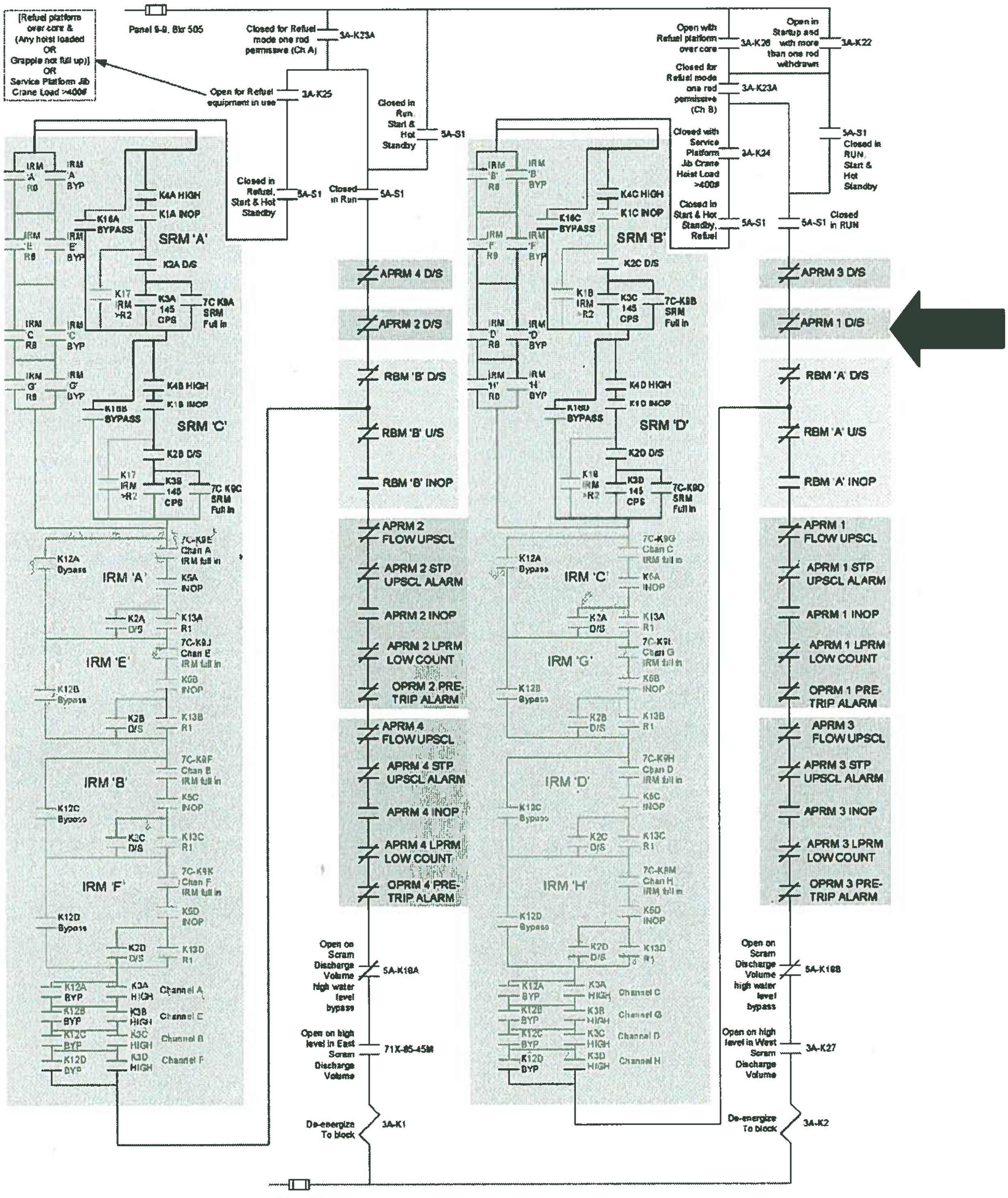
Learning Objective (As available): OPL171.029 V.B.7, OPL171.148 V.B.13

Question Source: Bank: X
Modified Bank:
New

Question History: Previous NRC: BFN 1006 #54

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

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



TP-14: ROD WITHDRAWAL BLOCK LOGIC

APRM DOWNSCALE/ OPRM INOP

4

(Page 1 of 1)

Sensor/Trip Point:

APRM Downscale	≤ 5%
OPRM Inop	Less than 23 operable cells

Sensor Location: Control Room Panel 3-9-14.

Probable Cause:

A. Unbypassed APRM channel at or below sensor set point.
B. SI or SR in progress.
C. Unit shutdown (Mode 3 or 4).
D. Failed sensor.

Automatic Action:

A. Rod withdrawal block with Rx Mode Sw. in RUN.(APRM only)

Operator Action:

A. **DETERMINE** which APRM/OPRM channel is downscale/inop. ☐
B. **IF** APRM failed downscale, **THEN** **BYPASS** channel. **REFER TO** 3-OI-92B. ☐
C. **IF** a single OPRM channel is inoperable, **THEN** **BYPASS** channel. **REFER TO** 3-OI-92B. ☐
D. **IF** the OPRM Trip Function is inoperable in MODE 1, **THEN**: **PERFORM** 3-SR-3.3.1.1.I to initiate alternate Thermal-Hydraulic Instability monitoring and required actions. ☐
E. **REFER TO** Tech Spec Table 3.3.1.1-1, TRM Table 3.3.4-1. ☐

References:

3-45E620-6	3-107E5784-20	3-107E5784-03
3-107E5784-03A	3-OI-92B	Technical Specifications
Technical Requirements Manual-TRM		

INSTRUCTOR NOTES

b. Imposed when, with three insert errors existing and an insert block present, a control rod other than one of the insert error control rods is selected.

c. In each case above, the block is applied to force the correction of the error before allowing movement of any other control rods.

d. Withdraw blocks are alarmed on the RWM operator's panel by a WITHDRAW BLOCK indicator light and status indication at all RWM display screens. — Panel 9-5 RWM ROD BLOCK annunciator

9. Insert block — Obj. V.B.8.d, Obj. V.C.3.d, OI-85 P&L

a. Imposed when a control rod is moved which exceeds the maximum number of allowable insert errors.

(1) The number of allowable insert errors may be varied through use of an off-line RWM system function.

(2) The number of allowable insert errors may be set to values of 0, 1 or 2.

(3) At Browns Ferry, 2 insert errors are allowed; 3 insert errors will cause an insert block.

b. Imposed when a withdraw error has been made, a withdraw block applied, and a control rod other than the withdraw error control rod is selected.

c. In each case above, the block is applied to force correction of the error before allowing further control rod movement.

d. Insert blocks are alarmed on the operator's panel by an INSERT BLOCK indicator light and status indication at all RWM display screens. — Obj. V.B.10; Panel 9-5 RWM ROD BLOCK annunciator

QUESTION 56

Which ONE of the following completes the statement below?

The Rod worth Minimizer (RWM) senses power below the Low Power Alarm Point (LPAP) by __________.

A. total steam flow

B. Main Turbine first stage pressure

C. average Reactor power from APRMs

D. Reactor power as calculated by the Integrated Computer System (ICS)

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	201006 K1.04	
	Importance Rating	3.1	

Knowledge of the physical connections and/or cause effect relationships between ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) and the following: Steam flow/reactor power: P-Spec(Not-BWR6)

Explanation: **A CORRECT:** The Low Power Alarm Point (LPAP) for the RWM is (27%) as sensed by Total Steam Flow.

B- Incorrect – Plausible because the Low Power Alarm Point (LPAP) for the RWM is looking for a measurement of total Reactor power. And Main Turbine first stage pressure is an indicator of Reactor power.

C- Incorrect –Plausible because the Low Power Alarm Point (LPAP) for the RWM is looking for a measurement of total Reactor power.

D- Incorrect – Plausible because the Low Power Alarm Point (LPAP) for the RWM is looking for a measurement of total Reactor power.

Technical Reference(s): OPL171.048, OPL171.047, OPL171.005

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: Columbia 2009 NRC #63

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

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

3.8 Rod Worth Minimizer (RWM)

A. The RWM system Rod Test/Touchscreen function allows any one rod to be selected and moved to any position only if all other control rods are fully inserted. To get out of the rod test, the pushbutton needs to be depressed again (otherwise any single rod in any group can be selected and withdrawn).

B. [NER/C] When the RWM is bypassed, a second licensed operator or other qualified member of the technical staff is required to verify the Control Rod Sequence is followed. [INPO SOER-84-002] (Not required with no fuel in the RPV or during single Control Rod withdrawal when the Reactor is in Modes 3, 4, or 5.)

C. 1-SR-3.3.2.1.7 is used to document independent verification of the RWM whenever the reactor is in STARTUP or RUN, below 10% power.

D. [NER/C] Activities that can directly affect core reactivity are of a critical nature and strict procedural compliance, along with conservative actions, must be followed. [INPO SOER-84-002]

E. For RWM to enforce, Total Feedwater Flow or Total Steam Flow must be <23%. To take RWM out of service automatically, Low Power Set Point (LPSP), Total Steam Flow and Total Feedwater Flow must be >23%. The Low Power Alarm Point (LPAP) for the RWM is (27%) as sensed by Total Steam Flow. When the RWM is operating in the transition zone, between the LPSP (23%) and the LPAP (27%), no rod blocks will be applied as a result of insert or withdraw errors, but the RWM will continue to provide alarm indications and error displays.

F. The monitoring functions of the RWM are automatically bypassed at power levels above the LPAP.

G. All the RWM blocks will be applied in the event of a system hardware or software failure when power is below the LPAP. At any Reactor power, when a loss of ICS 1A occurs, a select block will occur due to the loss of power and cannot be bypassed using the RWM Bypass key.

Columbia 2009 NRC #63

COLUMBIA GENERATING STATION WRITTEN EXAMINATION

QUESTION # 63 EXAM KEY MARCH 2009

Columbia is in the process of a reactor shutdown. With reactor power being reduced from 35% to 30% power, CRO1 receives a Below/LPAP Rod Worth Minimizer (RWM) alarm.

Which of the following signals causes this Rod Worth Minimizer alarm?

A. The total steam flow from all four main steam line flow instruments.

B. The average Reactor power from all APRM instruments.

C. Reactor power as calculated by PPCRS.

D. Main Turbine first stage pressure.

ANSWER: A

QUESTION TYPE: RO/SRO Closed

KA # & KA VALUE: 201006 K1.04 Knowledge of the physical connections and/or cause-effect relationships between Rod Worth Minimizer and the following: Steam flow/Reactor power (3.1 / 3.2)

REFERENCE: SD000154 Page 5 and 16

SOURCE: Bank LX00601 Modified slightly

LO: 5916 Describe the physical connection and/or cause-and-effect relationship between RMW and: a. FWLC

RATING: L2

ATTACHMENT: None

JUSTIFICATION: Per reference the steam flow inputs are summed to determine 32% power and give the alarm (A is correct).

QUESTION 57

Unit 3 was operating at 100% when the following occurred:

- 3-AOI-85-4, Loss of RPIS, has been entered for a loss of position indication on control rod 30-31.

Which ONE of the following completes both statements below?

In accordance with 3-AOI-85-4, to individually scram the rod, operators will ___(1)___ for control rod 30-31.

The Unit Operator will verify the correct rod scrammed by ___(2)___ light indication on the control rod 30-31 4-light display.

A. (1) pull the Scram Solenoid Fuses at the HCU
(2) blue

B. (1) pull the Scram Solenoid Fuses at the HCU
(2) red

C. (1) operate the Scram Test Switch at 1-PNLA-009-0016 in the Aux Instrument room
(2) blue

D. (1) operate the Scram Test Switch at 1-PNLA-009-0016 in the Aux Instrument room
(2) red

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	214000 G2.4.34	
	Importance Rating	4.2	

214000 Rod Position Information System. Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Explanation: **C CORRECT:** When efforts to move the rod to an Operable Position Indication have failed. 3-AOI-85-4 section 4.4[2] operators are directed to FULLY INSERT the CRD, individually scram the rod, and a Blue light indication is verified on for the individually scrammed rod.

A-Incorrect –First Part: Incorrect. Plausible because pulling the scram fuses at the HCU would cause the rod to scram. Second Part: Correct.

B- Incorrect – First Part: Incorrect. Plausible because pulling the scram fuses at the HCU would cause the rod to scram. Second Part: Incorrect. A Red light would indicate a ROD drift

D- Incorrect – First part: Correct. Second Part: Incorrect. A Red light would indicate a ROD drift

Technical Reference(s): 3-AOI-85-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

4.4 Alternate Methods of Determining Control Rod Position of A Single Control Rod (continued)

B. **WHEN** the above indications show the Control Rod is fully inserted and RPIS has not restored, Individually SCRAM the Control Rod at 1-PNLA-009-0016 (a key is required) by placing the individual Scram Test Switch to the "DOWN" position for 10 seconds, **THEN**

RETURN Test Switch to the normal "UP" position. ☐

C. **IF** RPIS indication has not been restored, **THEN**

DISARM electrically the CRD HCU amphenols by disconnecting and tagging them with a hold order. ☐

QUESTION 58

Unit 2 is operating at 100% power, with Feedwater Level Control in 3 element. The following indications are observed:

• RFPT speed	stable
• Total feedwater flow	stable
• Indicated level on 2-LI-3-53	25" (inches), lowering
• Indicated level on 2-LI-3-60	34" (inches), stable
• Indicated level on 2-LI-3-206	24" (inches), lowering
• Indicated level on 2-LI-3-253	33" (inches), stable

Which ONE of the following completes the statements below?

If the current plant trends continue, the reactor __(1)__ scram, and Feedwater Level Control (FWLCS) will __(2)__ element control.

A. (1) will
(2) shift to single

B. (1) will
(2) remain in three

C. (1) will NOT
(2) shift to single

D. (1) will NOT
(2) remain in three

ANSWER:**D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	216000 A3.01	
	Importance Rating	3.4	

Ability to monitor automatic operations of the NUCLEAR BOILER Instrumentation including: Relationship between meter/recorder readings and actual parameter values: Plant-Specific

Explanation: **DCORRECT:**The reactor will NOT scram, FWLC remains in 3 element at approximately same level.If two level signals are BAD or invalid, the algorithm will average the remaining twolevels and will control on that value. Algorithm determines an average level signal and compares all 4 level signals to it. If any level signal deviates from the average by more than 8" inches, then the level is declared invalid, is bypassed, and control room annunciator alarms.

A-Incorrect–First Part: Incorrect. The reactor will not scram. Level will remain approximately the same, 3 element, & automatic. Plausible if candidate assumes that 2-LI-3-53 and 2-LI-3-206 input to RPS. The RPV level 3 (+2 inch) scram comes from LIS-3-203A,B,C,andD.Second Part: Incorrect. Plausible because the candidate could incorrectly assume that 2 bad Narrow Range level inputs could cause FWLC to shift to single element control.

B- Incorrect – First Part: Incorrect- See A. Second Part: Correct-See D.

C-Incorrect –First Part: Correct- See D.Second Part: Incorrect- See A.

Technical Reference(s): 2-OI-3, Illustration 8, Page 1 of 7, OPL171.003

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.012 V.B.6

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: None

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

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

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.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	<u>4. Reactor Vessel Water Level - Low, Level 3</u> (LIS-3-203A, LIS-3-203B, LIS-3-203C, and LIS-3-203D)

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated at Level 3 to substantially reduce the heat generated in the fuel from fission. The Reactor Vessel Water Level - Low, Level 3 Function is assumed in the analysis of the recirculation line break (Ref. 6). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the Emergency Core Cooling Systems (ECCS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level - Low, Level 3 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level - Low, Level 3 Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level - Low, Level 3 Allowable Value is selected to ensure that (a) during normal operation the steam dryer skirt is not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder), and (b) for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water - Low Low Low, Level 1 will not be required.

1) Normal Control Range (Narrow Range) instruments (0 to +60") (digital; -10" to +70") (Referenced to instrument zero)

a) LT-3-53, LT-3-60, LT-3-206, and LT-3-253

(1) Provides level input signal to the Feedwater Level Control System (FWLCS) and to reactor water level indications and recorder on Panel 9-5

- high level alarm (+39")
- low level alarm (+27") are actuated as sensed by the point on the recorder.

Unit Difference:

U-1 has Foxboro IDP10-D22B Level Xmitter and U-2/3 has Foxboro 823DP level Xmitter. U-1 Xmitter has reversed sensing lines compared to U-2/3 Xmitter

QUESTION 59

Given the following conditions on Unit 2:

- RHR Pump 2A is in Suppression Chamber sprays

Which ONE of the following completes the statements below?

In accordance with 2-EOI appendix 17C, RHR System Operation Suppression Chamber Sprays, pump discharge pressure is monitored using ___(1)___. Pump discharge pressure fluctuating excessively is an indication of ___(2)__.

A. (1) RHR Sys I discharge pressure on 2-PI-74-51
   (2) chugging

B. (1) RHR Sys I discharge pressure on 2-PI-74-51
   (2) inadequate NPSH

C. (1) Suppression Pool discharge header pressure on 2-PI-74-94
   (2) chugging

D. (1) Suppression Pool discharge header pressure on 2-PI-74-94
   (2) inadequate NPSH

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	230000 A4.08	
	Importance Rating	3.0	

230000 RHR/LPCI: Torus/Suppression Pool Spray Mode A4.08 Ability to manually operate and/or monitor in the control room: Pump/system discharge pressure

Explanation: **B CORRECT:** First Part-**Correct-** Since 2A RHR pump is in service, pump discharge pressure is monitored with RHR Sys I discharge pressure using 2-PI-74-51. Second Part- **Corrrect-** In acordance with EOI Curve 2 (RHR NPSH Limits), Pump discharge pressure fluctuating excessively is an indication of inadequate NPSH, and 2-EOI-Appendix 17C directs monitoring for inadequate NSPH using Curve 2 while in suppression chamber sprays.

A-Incorrect –First Part: Correct-See B First Part. Second Part: Incorrect-Plausible as Suppression Chamber sprays are initiated prior to 12 psig in the suppression chamber in order to preclude chugging. In addition, chugging is cyclic.

C- Incorrect – First Part: Incorrect-Plausible as suppression chamber sprays are discharging to the air space of the suppression pool, however this pressure indicator reflects the suction pressure available to the ECCS pumps which take a suction off of the ECCS ring header not to the actual discharge pressure to the suppression pool. Second Part: Incorrect- See A Second Part.

D- Incorrect – First Part: Incorrect- See C First Part. Second Part: Correct- See B Second Part.

Technical Reference(s): 2-EOI-Appendix 17C

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

2-EOI-APPENDIX 17C

5. **INITIATE** Suppression Chamber Sprays as follows:

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

b. IF.....EITHER of the following exists:

- LPCI Initiation signal is NOT present,
  OR
- Directed by SRO,

THEN...**PLACE** keylock switch 2-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. ____

c. **MOMENTARILY PLACE** 2-XS-74-121(129), RHR SYS I(II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT. ____

d. IF.....2-FCV-74-53(67), RHR SYS I(II) INBD INJECT VALVE, is OPEN,
THEN...**VERIFY CLOSED** 2-FCV-74-52(66), RHR SYS I(II) OUTBD INJECT VALVE. ____

e. **VERIFY OPERATING** the desired RHR System I(II) pump(s) for Suppression Chamber Spray. ____

f. **VERIFY OPEN** 2-FCV-74-57(71), RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV. ____

g. **OPEN** 2-FCV-74-58(72), RHR SYS I(II) SUPPR CHBR SPRAY VALVE. ____

h. IF.....RHR System I(II) is operating ONLY in Suppression Chamber Spray mode,
THEN...**CONTINUE** in this procedure at Step 5.k. ____

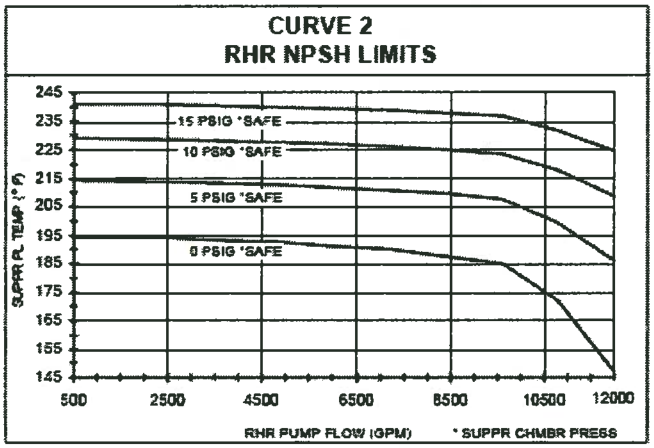
i. **VERIFY CLOSED** 2-FCV-74-7(30), RHR SYSTEM I(II) MIN FLOW VALVE. ____

j. **RAISE** System flow by placing the second RHR System I(II) pump in service as necessary. ____

k. **MONITOR** RHR Pump NPSH using Attachment 2. ____

NPSH MONITORING

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



Other indications of inadequate NPSH are:

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

LAST PAGE

QUESTION 60

When removing a spent fuel bundle from the reactor with the Main Hoist (fuel grapple), the Fuel Handling Bridge Operator raises the bundle and observes the GRAPPLE NORMAL UP indicating light illuminated.

Subsequently,

The operator uses the HOIST OVERRIDE pushbutton and raises the fuel bundle further and receives the FUEL POOL FLOOR AREA RADIATION HIGH, Panel 2-9-3A Window 1.

Which ONE of the following completes both statements below?

The FUEL POOL FLOOR AREA RADIATION HIGH, Panel 2-9-3A Window 1 annunciator alarms at __(1)__mr/hr.

During movement of recently irradiated fuel, the MINIMUM Spent Fuel Pool Water Level required above the top of irradiated fuel assemblies seated in the spent fuel storage racks is ≥__(2)__ft.

A. (1) 10
(2) 21.5

B. (1) 10
(2) 22

C. (1) 72
(2) 21.5

D. (1) 72
(2) 22

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	234000 A1.02	
	Importance Rating	3.3	

Ability to predict and/or monitor changes in parameters associated with operating the FUEL HANDLING EQUIPMENT controls including: †Refuel floor radiation levels/ airborne levels

Explanation: **A CORRECT:** Part (1) CORRECT-If the operator uses and the HOIST OVERRIDE pushbutton and raises the fuel bundle further, the FUEL POOL FLOOR AREA RADIATION HIGH annunciator would be received at 10 mr/hr on the refuel floor. Part (2) CORRECT – IAW T.S. 3.7.6: The MINIMUM Spent Fuel Pool Water Level fuel required above the top of irradiated fuel assemblies seated in the spent fuel storage racks is $\geq$ 21.5 ft.

B- Incorrect – First Part: CORRECT-See C. Plausible because the Refueling Platform moves at 10 feet per minute and the auxiliary hoist can move at 10 feet per minute. Second Part: Incorrect-See B.

C-Incorrect –First Part: Incorrect- This is plausible as this is 72mr/hr is the set point for refuel exhaust radiation high. Second Part: CORRECT-See C.

D- Incorrect – First Part: Incorrect.-See A. Second Part: Incorrect- this is plausible as 22 ft is the minimum required water level above the RPV flange during movement of irradiated fuel assemblies (T.S 3.9.6).

Technical Reference(s): 2- ARP-9-3A, T.S.3.7.6; T.S 3.9.6; 2-SIMI-90B

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

FUEL POOL FLOOR AREA RADIATION HIGH 2-RA-90-1A 1	Sensor/Trip Point:

(Page 1 of 1)

RI-90-1B	
RI-90-2B	For setpoints
RI-90-3B	**REFER TO** 2-SIMI-90B.

Sensor Location:

RE-90-1B	El 664'	R-11 P-LINE
RE-90-2B	El 664'	R-10 U-LINE
RE-90-3B	El 639'	R-10 Q-LINE

Probable Cause:

A. Change in general radiation levels.
B. Refueling accident.
C. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **CHECK** 2-RI-90-1A, 2-RI-90-2A and 2-RI-90-3A on Panel 2-9-11. ☐
B. **NOTIFY** refuel floor personnel. ☐
C. **IF** Dry Cask loading/unloading activities are in progress, **THEN** **NOTIFY** Cask Supervisor. ☐
D. **IF** airborne levels rise by 100 DAC **AND** RAD PRO confirms, **THEN** **REFER TO** EPIP-1. ☐
E. **REFER TO** 2-AOI-79-1 or 2-AOI-79-2 as applicable. ☐
F. **IF** this alarm is not valid, **THEN REFER TO** 0-OI-55. ☐
G. **IF** this alarm is valid, **THEN** **MONITOR** the other parameters that input to it frequently. These other parameters will be masked from alarming while this alarm is sealed in. ☐
H. **MONITOR** Fuel Pool Cooling system and Refer to AOI-78-1. ☐
I. **ENTER** 2-EOI-3 Flowchart. ☐

References:

0-47E600-13 2-47E610-90-1 2-45E620-3
GE 730E356 Series, TVA Calc NDQ00902005001/EDC63693

REFUELING ZONE EXHAUST RADIATION HIGH 2-RA-90-140A 34

(Page 1 of 2)

Sensor/Trip Point:

2-RE-90-140A	72 MR/HR	
2-RE-90-140B	72 MR/HR	
2-RE-90-141A	72 MR/HR	Required setting of
2-RE-90-141B	72 MR/HR	≤ 100 MR/HR.

Sensor Location: Rx Bldg, El 664' (Refuel Floor), R-10 P-LINE

Probable Cause:
A. Radiation levels have risen above alarm setpoint.
B. Refueling accident.

NOTE
Drawing 0-47E201-1 "IFSI DRY STORAGE IMPLEMENTATION NOTE (20A) requires these detectors be temporarily shielded during Dry Cask loading/unloading activities.

C. Temporary shielding not in place for monitors during Dry Cask loading/unloading activities.
D. Loss of power to NUMAC drawer.

Automatic Action:
A. Control Room and Refuel Zone ventilation isolates.
B. SGTS initiates.
C. Control Room emergency pressurization units start.

Operator Action:
A. **VERIFY** alarm condition on the following:
1. REACTOR & REFUEL ZONE EXHAUST RADIATION recorder, 2-RR-90-144 on Panel 2-9-2. ☐
2. RX & REFUEL ZONE EXH CH A RAD MON RTMR radiation monitor, 2-RM-90-140/142 on Panel 2-9-10. ☐
3. RX & REFUEL ZONE EXH CH BRAD MON RTMR radiation monitor, 2-RM-90-141/143 on Panel 2-9-10. ☐

B. **IF** Dry Cask loading/unloading activities are in progress, **THEN NOTIFY** the Cask Supervisor to place the MPC in a safe condition using MSI-0-079-DCS037 or as directed by RAD PRO. ☐
C. **NOTIFY** Unit Supervisor/SRO, Unit 1 and Unit 3. ☐
D. **IF** the TSC is **NOT** manned, **THEN EVACUATE** personnel from the refuel floor. ☐

Continued on Next Page

3.7 PLANT SYSTEMS

3.7.6 Spent Fuel Storage Pool Water Level

LCO 3.7.6 The spent fuel storage pool water level shall be $\geq$ 21.5 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

3.7 PLANT SYSTEMS

3.7.6 Spent Fuel Storage Pool Water Level

LCO 3.7.6 The spent fuel storage pool water level shall be ≥ 21.5 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level

LCO 3.9.6 RPV water level shall be ≥ 22 ft above the top of the RPV flange.

APPLICABILITY: During movement of irradiated fuel assemblies within the RPV,
During movement of new fuel assemblies or handling of control rods within the RPV, when irradiated fuel assemblies are seated within the RPV.

2-SIMI-90B

FUNCTION: Fuel Stg Pool Area Radiation Indicator/Trip Unit			
LOCATION: Control Room Panel 2-9-11			
SETTINGS	REFERENCE	ACCURACY	REFERENCE
High - 10 mR/HR Dnscl - 0.1 mR/HR	RIMS R38900207853	± 7.5% ELFS	See Attachment 4
CALIBRATION RANGE: 0.1 to 1000 mR/HR		ACTION: Direct	
INPUT 1: Current		OUTPUT 1: Indication	

QUESTION 61

Unit 1 is operating at 80% power when the following indications are received:

- HEATER C3 LEVEL HIGH, (1-9-6A, Window 17) is in alarm
- The 1C3 heater level indication is HIGH HIGH (red) on the FEEDWATER HEATER LEVEL(FWHL) ICS screen

Which ONE of the following completes the statements below?

For these heater indications, the steam from the __(1)__ to Feedwater Heater 1C3 will isolate, and the 1C3 heater __(2)__ will be open.

A. (1) 8th stage of the low pressure turbine
(2) normal level control valve ONLY

B. (1) 8th stage of the low pressure turbine
(2) normal level control valve AND the high level dump valve

C. (1) cross-around piping
(2) normal level control valve ONLY

D. (1) cross-around piping
(2) normal level control valve AND the high level dump valve

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	259001 K4.02	
	Importance Rating	2.8	

Knowledge of REACTOR FEEDWATER SYSTEM design feature(s) and/or interlocks which provide for the following: Feedwater heating

Explanation: **A CORRECT:** For these heater indications (high), the steam from the 8th stage of the low pressure turbine to Feedwater Heater 1C3 will isolate, and the normal dump valve to heater 1C4 will be open.

B-Incorrect –First Part: Correct-See A. Second Part: Incorrect. Plausible, because would be the case for a HIGH-HIGH feedwater level (red) ICS indication if heater 1C4 had an emergency dump valve.

C- Incorrect – First Part: Incorrect. Plausible because the steam supplied to 1C1 heater is from the cross around steam piping to the moisture separators. Second Part: Correct-See A.

D- Incorrect – First Part: Incorrect- See C. Second Part: Incorrect See B.

Technical Reference(s): OPL171.095, OPL171.026, 1-ARP-9-6A Window 17

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

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

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

d. #3 heater extraction steam comes from the 8th stage of the low pressure turbine. Each #3 heater has a non-return valve and an extraction steam isolation valve.

(1) #3 heater is similar in construction and operation to #1. — U-Tube

(2) Shell side pressure at full load is about 74 psia. — Saturated conditions

2. Flow Path — TP-1

a. Extraction steam to the #1 heaters comes from the cross-around steam to the moisture separators. This steam passes through an extraction non-return valve. — Obj. V.B.3, Obj. V.E.3

(1) Non-return valves prevent steam from re-entering the turbine from the heater following a turbine trip to prevent a turbine overspeed. — Obj. V.B.2.a, Obj. V.C.1.a, Obj. V.E.2.a

(2) Also prevent moisture from entering the turbine on high heater levels (level control malfunction or tube leak).

7. Heater high level isolation logic — Obj. V.B.18, Obj. V.E.18, Obj. V.C.9, TP-14

a. A high level in the #1 or #2 heaters, with a confirmatory high level from the heater level control circuit, will isolate the associated extraction steam valves and close the moisture separator level control reservoir isolation valves. Isolation can be bypassed with a keylock bypass switches.

b. A high water level in the #3, #4 or #5 heaters, with a confirmatory high level from the heater level control circuit, will isolate the associated #3 heater extraction steam valve. Isolation can be bypassed with a keylock bypass switches.

c. Two float type level switches per heater input to the heater isolation logic. — TP-10, 11, 14

HEATER C3
LEVEL HIGH

1-LA-6-43

17

(Page 1 of 1)

Sensor/Trip Point:

1-LT-006-0043A 29" H_2O
1-LT-006-0043B

Sensor Location: 1-LPNL-925-0109
El. 586', T5-E line

Probable Cause:
A. Tube leak 1C3 heater.
B. Malfunction of the following:
1. FW HTR 1C3 DRAIN TO HTR 1C4, 1-LCV-006-0043.
2. RFW HTR 1C-3 LEVEL, 1-LT-006-0043A or 1-LT-006-0043B.

Automatic Action: None

Operator Action:
A. **CHECK** the following on Panel 1-9-6 for indications of a possible tube leak:
- Rising condensate flow on CONDENSATE, 1-XR-2-29. ☐
- High/rising shell pressure on LP HEATER 1C3, 1-PI-5-66. ☐
- High/rising flow from COOLER 1C5, 1-FI-6-52. ☐

NOTE

If a Hi level occurs when a heater is not in service (extraction stm isolated) it is not required to isolate the condensate or feedwater side unless a gross tube failure is indicated. A gross tube leak would be indicated by rising flow on Condensate Flow Recorder 2-29, Panel 9-6 or elevated Heater shell pressure approaching the extraction steam header pressure.

B. **CHECK** level on ICS screen, FEEDWATER HEATER LEVEL(FWHL). ☐
- **IF** 1C3 heater indicates HIGH (yellow), **THEN**

  VERIFY proper operation of the drain and dump valves. ☐
- **DISPATCH** personnel to 1-LPNL-925-0562D to check alarm and manually control level. ☐

C. **IF** a valid HIGH HIGH level (red) is received, **THEN**

REFER TO 1-AOI-6-1B or 1-AOI-6-1C. ☐

References:

1-45E620-8-1	1-45E777-7	1-47E610-6-1
1-47E805-1	1-47E802-1	

c. The drains from the #1 heaters flow into the #2 heaters via the #1 heater drain valves. — No Hi lvl dump on the #1 htrs

(1) Level is maintained in the #2 heaters as described above via the Normal level control valve. — Obj. V.B.10.c Obj. V.E.10.c

(2) The #2 heaters also have an emergency drain valve that bypasses drain flow directly to the main condenser if the normal drain valve cannot maintain heater level within the desired band. — Hi lvl dump vlv

d. The moisture separator drains also flow into the #2 heaters. — TP-4 & 7

INSTRUCTOR NOTES

e. The #2 heater drains flow into the #3 heaters via the normal level control valve.

(1) Level control of the #3 heaters is similar to above.

(2) #3 heater does not have an emergency drain valve.

f. Drains from the #3 heaters flows into the #4 heaters.

Level control is similar to above.

QUESTION 62

Which ONE of the following completes the statement below?

A loss of __(1)__ would result in a loss of power to __(2)__.

A. (1) the ±24VDC Neutron Monitoring Battery System
   (2) Wide Range Gaseous Effluent Radiation Monitor (WRGERM), 0-RM-90-306

B. (1) the 48VDC Annunciator Battery System
   (2) Stack-Gas Radiation Monitor detectors (RM-90-147 & 148)

C. (1) 250VDC Battery Board 1
   (2) Stack-Gas Radiation Monitor detectors (RM-90-147 & 148)

D. (1) 250VDC Battery Board 2
   (2) Wide Range Gaseous Effluent Radiation Monitor (WRGERM), 0-RM-90-306

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	2	
	K/A#	272000 K2.03	
	Importance Rating	2.5	

Knowledge of electrical power supplies to the following:
K2.03 Stack gas radiation monitoring system

Explanation: **D** CORRECT – The Wide Range Gaseous Effluent Radiation Monitor (WRGERMS), 0-RM-90-306 is powered from 250VDC Battery Board 2.

A Incorrect- First part: Incorrect. Plausible because the Stack-Gas Radiation Monitors (RM-90-147 & 148) are scintillation detectors powered from low voltage DC. Second Part: Incorrect. Plausible because the Stack-Gas Radiation Monitors (RM-90-147 & 148) are scintillation detectors powered from low voltage DC electrical, but Wide Range Gaseous Effluent Radiation Monitor (WRGERM), 0-RM-90-306 is not.

B Incorrect- First Part: correct. Second Part: Incorrect. Plausible because the Stack-Gas Radiation Monitors (RM-90-147 & 148) are scintillation detectors powered from low voltage DC electrical, but not 48VDC.

C Incorrect- First part: Incorrect. Plausible because stack gas rad monitor recorder RR-90-147 is on UNIT 1 panel 9-10. Second Part: Incorrect. Plausible because Wide Range Gaseous Effluent Radiation Monitor (WRGERM), 0-RM-90-306 is powered from a 250VDC battery board.

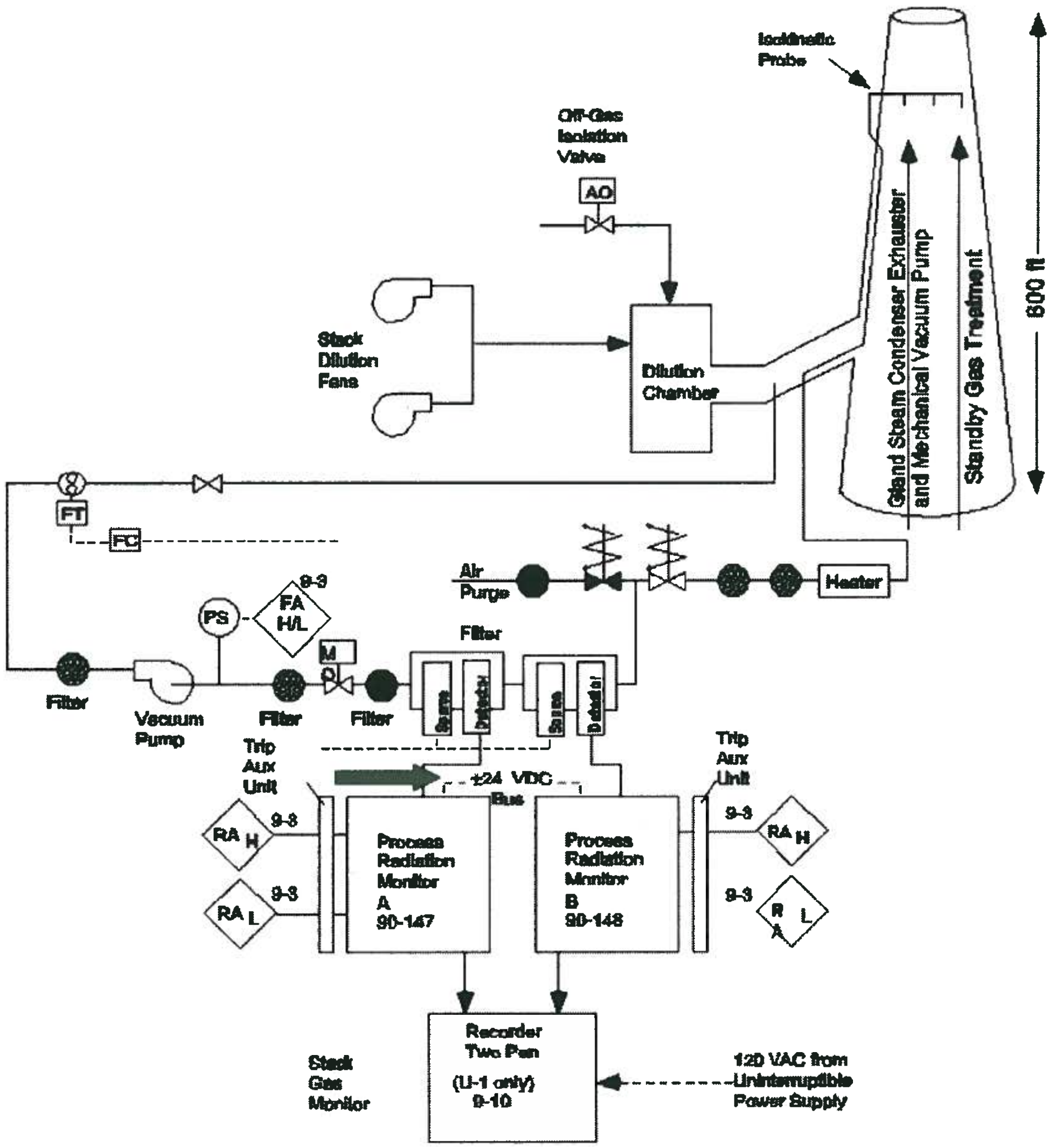
Technical Reference(s): OPL171.033

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: BFN 1306 #62
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis

10 CFR Part 55 Content: 55.41 (13) Procedures and equipment available for handling and disposal of radioactive materials and effluents.


Isokinetic Probe
Off-Gas Isolation Valve
AO
Stack Dilution Fans
Dilution Chamber
Gland Steam Condenser Exhauster and Mechanical Vacuum Pump
Standby Gas Treatment
600 ft
FT
FC
Air Purge
Heater
PS
9-3
FA H/L
M O
Filter
Filter
Vacuum Pump
Filter
Filter
Sample
Detector
Sample
Detector
Trip Aux Unit
±24 VDC Bus
Trip Aux Unit
RA H
9-3
RA L
9-3
Process Radiation Monitor A 90-147
Process Radiation Monitor B 90-148
9-3
RA H
9-3
RA L
Recorder Two Pen
(U-1 only)
9-10
Stack Gas Monitor
120 VAC from Uninterruptible Power Supply

STACK GAS
RADIATION
MONITOR
DNSC/INOP

2-RA-90-147C

20

(Page 1 of 1)

Sensor/Trip Point:

RE-90-147	RM-90-147B	Low detector output
RE-90-148	RM-90-148B	Low detector output
0-RM-90-306		Low/High detector output

Sensor Location: El 599'6" Panel 25-39.
Inside stack.

Probable Cause:

A. Source check.
B. Sensor malfunction.
C. Loss of power (Batt Bd 2, Panel 13) to detector.
D. Abnormal Temperature (High or Low) in the WRGERMS Building.
E. Stack flow abnormal

Automatic Action: None

Operator Action:

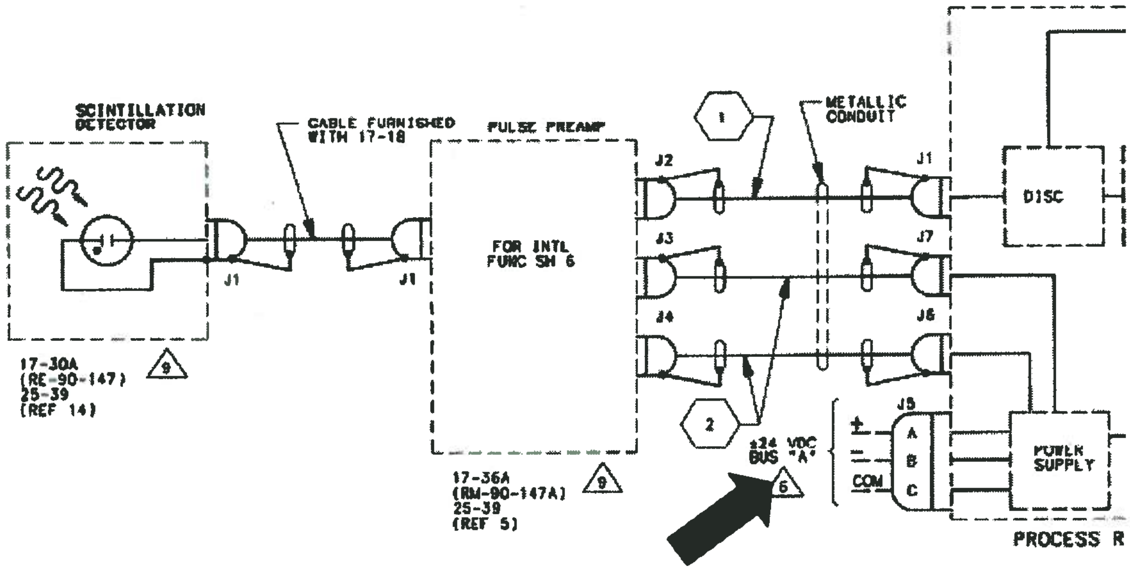
A. **CHECK** alarm condition on the following:
   1. WIDE RANGE GASEOUS EFFLUENT RADIATION MONITOR, 0-RM-90-306 on Panel 2-9-10. ☐
   2. STACK GAS RAD MONITOR C1 RATEMETER, 0-RM-90-147B on Panel 1-9-10. ☐
   3. STACK GAS RAD MONITOR C2 RATEMETER, 0-RM-90-148B. ☐

B. **IF** alarm is from 0-RM-90-306, **THEN REFER TO** 2-AOI-90-2. ☐

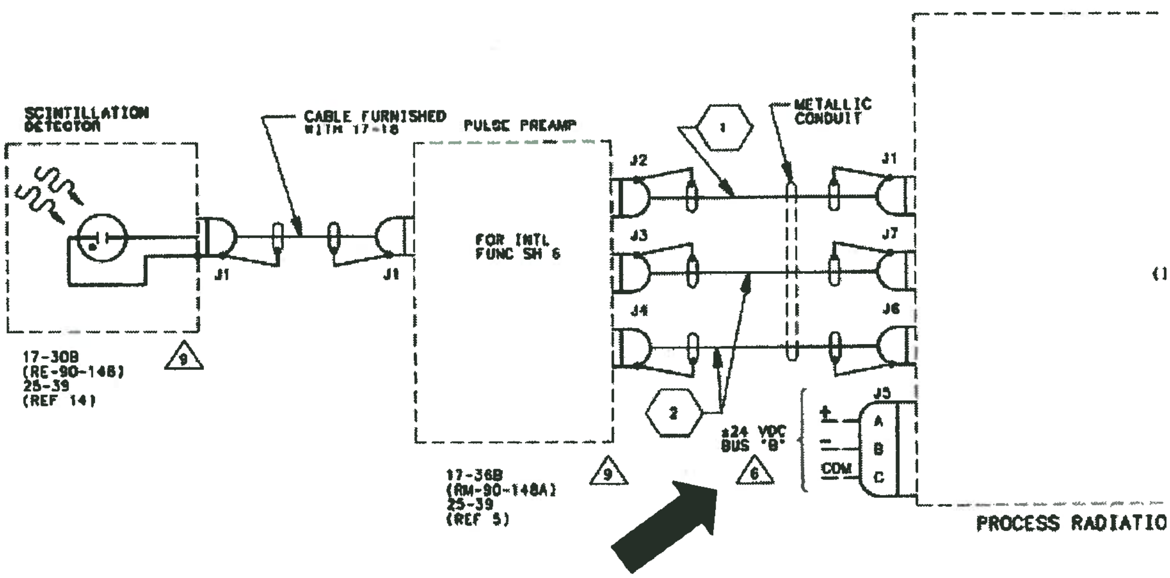
C. **CHECK** following radiation monitors on Panel 2-9-10 and if applicable, associated radiation recorders on Panel 2-9-2 for levels below alarm limits:
   1. OG PRETREATMENT RAD MON RTMR, 2-RM-90-157. ☐
   2. OFFGAS RAD MON RTMR, 2-RM-90-160. ☐
   3. OG POST-TREATMENT CHAN A RAD MON RTMR, 2-RM-90-265A. ☐
   4. OG POST-TREATMENT CHAN B RAD MON RTMR, 2-RM-90-265B. ☐

D. **NOTIFY** Unit Supervisor/SRO, Unit 1 and Unit 3. ☐
E. **NOTIFY** Chemistry to begin sampling procedures. ☐
F. **REFER TO** ODCM 1.1.2. ☐

References: 2-45E620-3 0-47E610-90-4 GE 2-729E814RF-5



SCINTILLATION DETECTOR
CABLE FURNISHED WITH 17-18
PULSE PREAMP
METALLIC CONDUIT
FOR INTL FUNC SH 6
J1
J2
J3
J4
J7
J6
J5
DISC
POWER SUPPLY
17-30A (RE-90-147) 25-39 (REF 14)
17-36A (RM-90-147A) 25-39 (REF 5)
±24 VDC BUS "A"
COM
A
B
C
PROCESS R



SCINTILLATION DETECTOR
CABLE FURNISHED WITH 17-18
PULSE PREAMP
METALLIC CONDUIT
FOR INTL FUNC SH 6
J1
J2
J3
J4
J7
J6
J5
17-30B (RE-90-148) 25-39 (REF 14)
17-36B (RM-90-148A) 25-39 (REF 5)
±24 VDC BUS "B"
COM
A
B
C
PROCESS RADIATIO

BFN 1306 #62

QUESTION 62

Which ONE of the following completes the statements below?

The Wide Range Gaseous Effluent Radiation Monitor (WRGERM), 0-RM-90-306, is powered from ___(1)___.

The Stack-Gas Radiation Monitor detectors (RM-90-147 & 148) are powered from ___(2)___.

A. (1) 250VDC Battery Board 2
(2) 48VDC Annunciator Battery System

B. (1) 250VDC Battery Board 2
(2) ±24VDC Neutron Monitoring Battery System

C. (1) 250VDC Battery Board 1
(2) 48VDC Annunciator Battery System

D. (1) 250VDC Battery Board 1
(2) ±24VDC Neutron Monitoring Battery System

Answer: **B**

QUESTION 63

Control Room Emergency Ventilation (CREV) train A is running for testing when a loss of Control Air occurs.

Which ONE of the following completes the statement below:

At 73 psig control air pressure, the _____.

A. Service Air Crosstie, 0-FCV-33-1, OPENS and train A continues to run

B. Control Bay Emergency Compressor STARTS and train A continues to run

C. Isolation dampers 0-FCO-31-150 (B,D,E,F,G) fail CLOSED and train A trips on low flow

D. CREV TRAIN A INLET DAMPER, 0-FCO-31-7211, fails CLOSED and train A trips on low flow

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	288000 K6.03	
	Importance Rating	2.7	

Knowledge of the effect that a loss or malfunction of the following will have on the PLANT VENTILATION SYSTEMS : Plant air systems

Explanation: **B CORRECT:** On a loss of plant control air decreasing air pressure at 80 psi gives local alarm and annunciation in control room, and the Emergency Control Bay compressor starts at 73 psig and stops at 100 psig automatically. The Emergency Control Bay compressor supplies compressed air upon loss of normal control air for chiller condenser controls, pressure reducing valve mounted in SUMMER/WINTER pneumatic control system, and 18 psi air used to operate sensing lines and control bay damper controls. The running CREV train will continue to run.

A- Incorrect – Plausible because the Service Air Crosstie, 0-FCV-33-1, OPENS at 85psi decreasing.

C- Incorrect – Plausible because Isolation dampers 0-FCO-31-150(B,D,E,F,G) CLOSE on a CREV initiation.

D-Incorrect –Plausible because a running CREV fan will trip on low flow at approximately 2700 cfm.

Technical Reference(s): OPL171.067, OPL171.054, 0-OI-31, 0-AOI-32-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

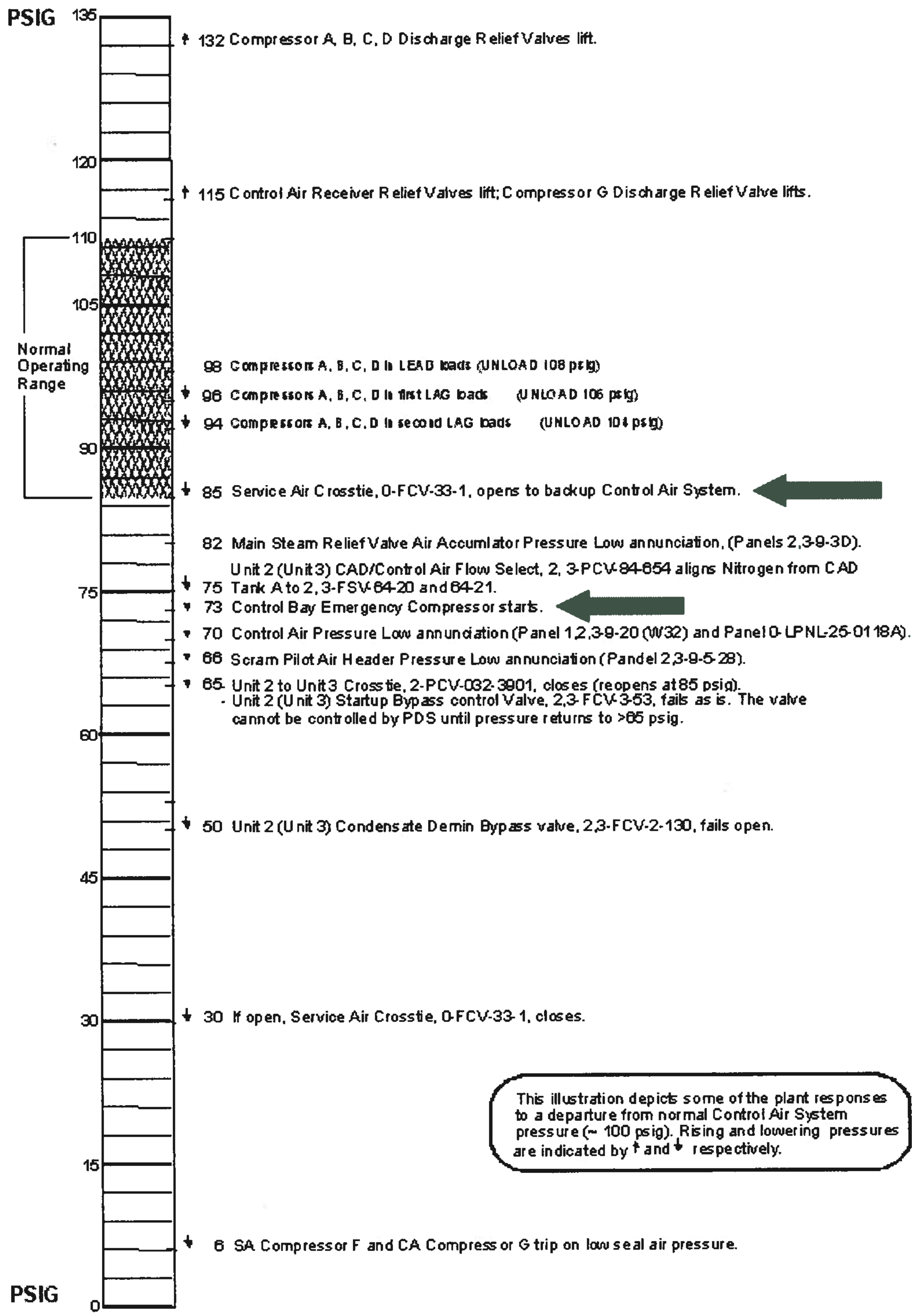
Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
will then be deenergized, with the fan continuing to run and the damper held open for approx. 30 seconds, and the damper closing and the fan turned off as discussed earlier.	
8. Emergency air compressor and receiver tank supplies compressed air upon loss of normal control air for chiller condenser controls, pressure reducing valve mounted in SUMMER/WINTER pneumatic control system, and 18 psi air used to operate sensing lines and control bay damper controls	In 2C Mech Equip Room Shown on control air print 1-47E847-6
a. Plant control air is normally supplied to the tank at 100 psi. Receiver tank air is filtered and reduced to 60 psig and further reduced to 18 psi.	
b. On a loss of plant control air decreasing air pressure at 80 psi gives local alarm and annunciation in control room, and the Emergency Control Bay compressor starts at 73 psig and stops at 100 psig automatically.	

BFN Unit 0	Loss of Control and Service Air Compressors	0-AOI-32-1 Rev. 0041 Page 5 of 35

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.



PSIG
135
120
110
105
90
75
60
45
30
15
0
PSIG
Normal Operating Range
132 Compressor A, B, C, D Discharge Relief Valves lift.
115 Control Air Receiver Relief Valves lift; Compressor G Discharge Relief Valve lifts.
98 Compressors A, B, C, D in LEAD loads (UNLOAD 108 psig)
96 Compressors A, B, C, D in first LAG loads (UNLOAD 106 psig)
94 Compressors A, B, C, D in second LAG loads (UNLOAD 104 psig)
85 Service Air Crosstie, 0-FCV-33-1, opens to backup Control Air System.
82 Main Steam Relief Valve Air Accumlator Pressure Low annunciation, (Panels 2,3-9-3D).
75 Unit 2 (Unit 3) CAD/Control Air Flow Select, 2, 3-PCV-84-654 aligns Nitrogen from CAD Tank A to 2, 3-FSV-64-20 and 64-21.
73 Control Bay Emergency Compressor starts.
70 Control Air Pressure Low annunciation (Panel 1,2,3-9-20 (W32) and Panel 0-LPNL-25-0118A).
66 Scram Pilot Air Header Pressure Low annunciation (Pandel 2,3-9-5-28).
65 Unit 2 to Unit 3 Crosstie, 2-PCV-032-3901, closes (reopens at 85 psig).
- Unit 2 (Unit 3) Startup Bypass control Valve, 2,3-FCV-3-53, fails as is. The valve cannot be controlled by PDS until pressure returns to >65 psig.
50 Unit 2 (Unit 3) Condensate Demin Bypass valve, 2,3-FCV-2-130, fails open.
30 If open, Service Air Crosstie, 0-FCV-33-1, closes.
This illustration depicts some of the plant responses to a departure from normal Control Air System pressure (~ 100 psig). Rising and lowering pressures are indicated by ↑ and ↓ respectively.
6 SA Compressor F and CA Compressor G trip on low seal air pressure.

3.5 CREV and CREV instrumentation operability issues (continued)

C. 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.

D. 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.

OPL171.067, Plant Heating, Ventilation, and Air Conditioning (HVAC) Systems, Rev. 18

Lesson Plan Content

Outline of Instruction	Instructor Notes and Methods
i. Fan overload ii. Unit low flow, less than approx. 2700 cfm -- trip is delayed for 10 seconds after fan start. iii. High heater discharge temperature, approx. 220°F iv. Low heater delta temperature(between unit inlet and heater discharge), indicating that the heater is not getting the relative humidity below 70 % -- trip is delayed for approx. 15 seconds after the heater is energized	

QUESTION 64

Given the following conditions for Units 1 and 2:

- 1B Control Bay Supply Fan is in service

Subsequently,

- The 1B Control Bay Supply Fan trips on loss of power.

Which ONE of the following completes the statements below?

The 1A Control Bay Supply Fan ___(1)___ Auto-Start.

For proper cooling the 1A Control Bay Supply Fan can be aligned to ___(2)___ Unit 1 and 2 Main Control Room AHU.

A. (1) will
   (2) ONLY the 1A

B. (1) will
   (2) EITHER the 1A or 1B

C. (1) will NOT
   (2) ONLY the 1A

D. (1) will NOT
   (2) EITHER the 1A or 1B

ANSWER:**D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	290003 K3.03	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the CONTROL ROOM HVAC will have on following: Control room temperature

Explanation: **D CORRECT:** Part (1) CORRECT: On a loss of power to the 1B supply fan the 1A supply fan will NOT auto start. In order for the 1B Control Bay Room Supply Fan to operate, logic requires the 1A to be in OFF. In this condition there is no standby feature. Part (2) CORRECT: The 1A supply fan can supply either unit 1 and 2 MCR AHU.

A- Incorrect – Part (1) Incorrect: This is plausible as the 1B supply fan auto-starts on a loss of the 1A supply fan. Part (2) Incorrect: This is plausible as Unit 3 chillers must be aligned to their respective AHU's to provide proper cooling. In addition, it's plausible that there are two separate redundant trains from supply fans down to AHU, and therefore the 1A supply fan is only aligned to 1A AHU and vice versa for 1B.

B-Incorrect –Part (1) Incorrect: See A. Part (2) Correct: See D.

C-Incorrect– Part (1) Correct: See D. Part (2) Incorrect: See A.

Technical Reference(s): 0-OI-31, FSAR Chapter 10.12

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

3.1 General (continued)

K. [NRC/C] This Operating Instruction is used for three units. Valves, electrical boards, switches, and instruments have a specific unit prefix designated unless common to more than one unit, in which case "0" prefix is used. [RPT 82-13]

L. In order for the 1B(3B) CONTROL BAY ROOM SUPPLY FANS to operate, logic requires the 1A(3A) to be in OFF. In this condition, there is no standby feature and failure of the 1B(3B) results in the loss of both Control Bay Supply Fans. If failure occurs, Reference Tech Specs for potential LCO.

M. When two Control Room AC subsystems are inoperable, an alternate method of cooling is required to be placed in service within 24 hours. Technical Specifications Bases states these alternate methods include, but are not limited to, the use of the emergency chiller, the Relay Room AHUs, and the other Unit's Control Room AC System. Reference TS Bases 3.7.4

3.3 Unit 3 Control Bay Chillers

A. Prior to closing the UNIT 3 CONTROL BAY WATER CHILLER A (B) breaker, verify associated CONTROL CIRCUIT and OIL HEATER CIRCUIT switches, located on the control bay water chiller control Panel, are in the OFF position to prevent failure of the control power transformer primary fuses. After this breaker is closed, place the CONTROL CIRCUIT and OIL HEATER CIRCUIT switches to ON.

B. In order to warm Unit 3 CONTROL BAY CHILLER LUBE OIL, the control power fuses are required to be INSTALLED and the Power Supply Breaker CLOSED. The required warm-up time is 4 hours.

C. The low chilled water flow circuit for 3A and 3B Control Bay Chillers includes a time delay. This time delay allows chilled water flow to stabilize on chiller startup without tripping the chiller on chilled water low flow.

D. U3 Control Bay Chillers and U3 Control Room and Control Bay el. 593' AHUs should be aligned in the same train for proper cooling. When realigning an AHU (or Chiller), ensure that the corresponding Chiller (or AHU) is also realigned. Sections 8.29 and 8.30 would only be used in exceptional circumstances and are not to be used in normal operations.

E. Any time Unit 3 Control Bay Chillers are swapped, the Control Bay Air Handling Units (A&B) and Chilled water supply/return for Relay Rm AHU B is required to be swapped.

FSAR CHAPTER 10.12

The control room air handling units provide ventilation to the main control room area. Two 100-percent capacity air handling units are provided, each containing: heating and cooling coils, a humidifier, controls, and motor-operated dampers. The

dampers isolate the air handling unit when on standby. The air handling cooling coils are equipped with vent and drain valves. Room return air is proportionally mixed with fresh air by manual dampers and filtered by renewable media filter cells rated at 85-percent NBS.

Fresh air is mechanically supplied for makeup to air-conditioning systems, for ventilating system requirements, and for pressurizing the Control Building. Fresh air supply systems separately serve the Units 1 and 2 air-conditioned spaces except the Electric Board Rooms, the Unit 3 air-conditioned spaces except the electric board rooms and spreading rooms. Each of the air-conditioned spaces has two 100-percent capacity supply fans.

QUESTION 65

Which one of the following completes both statements below?

To avoid pressure and temperature conditions that might cause brittle fracture failure of the Reactor Coolant Pressure Boundary, Technical Specification 3.4.9, RCS Pressure and Temperature (P/T) Limits, are applicable __(1)__.

When starting a Reactor Recirculation Pump, the difference between bottom head temperature and RPV coolant temperature must be verified __(2)__ prior to starting each recirculation pump.

A. (1) MODES 4 and 5 ONLY
   (2) within 15 minutes

B. (1) MODES 4 and 5 ONLY
   (2) within 30 minutes

C. (1) At all times
   (2) within 15 minutes

D. (1) At all times
   (2) within 30 minutes

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	2	
	Group #	1	
	K/A#	290002 K5.05	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to REACTOR VESSEL INTERNALS : Brittle fracture

Explanation: **C CORRECT:** The Pressure-Temperature curves are developed to assure that brittle fracture is prevented. They are applicable at all times. SR 3.4.9.3 is required to be performed once within 15 minutes of starting a reactor recirc pump.

A-Incorrect – First Part: Incorrect. Plausible because these are the modes the RCS will be at cold conditions when brittle fracture is of most concern. Second Part: Correct.

B- Incorrect- First Part: Incorrect. Plausible because these are the modes the RCS will be at cold conditions when brittle fracture is of most concern. Second Part: Incorrect. Plausible because SR 3.4.9.5 and 3.4.9.6 are required to be performed at 30 minute intervals.

D- Incorrect - First Part: Correct. Second Part: Incorrect. Plausible because SR 3.4.9.5 and 3.4.9.6 are required to be performed at 30 minute intervals.

Technical Reference(s): TS 3.4.9 and basis

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

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

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. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.3	-------------------------NOTE---------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	-----------------------NOTES--------------------- 1. Only required to be performed when tensioning the reactor vessel head bolting studs. 2. The reactor vessel head bolts may be partially tensioned (four sequences of the seating pass) provided the studs and flange materials are > 70°F. --- Verify reactor vessel flange and head flange temperatures are > 83°F.	30 minutes
SR 3.4.9.6	-----------------------NOTE---------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 85°F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are > 83°F.	30 minutes

QUESTION 66

Given the following conditions:

- Unit 2 has scrammed and multiple rods remain out
- Reactor Pressure is 1000 psig
- **The pictures on the next page reflect the current status of the SLC system**

Which ONE of the following completes BOTH of the statements below?

SLC __(1)__ currently injecting to the RPV. Placing 2-HS-63-6A, SLC Pump 2A/B, control switch in the START-B position will result in __(2)__ SLC pump(s) running.

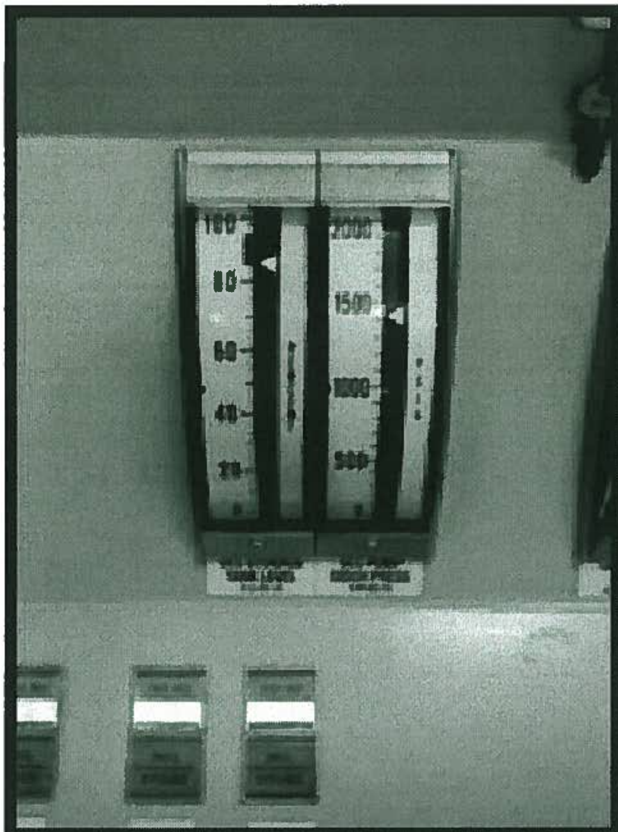
A. (1) is
(2) only the 2B

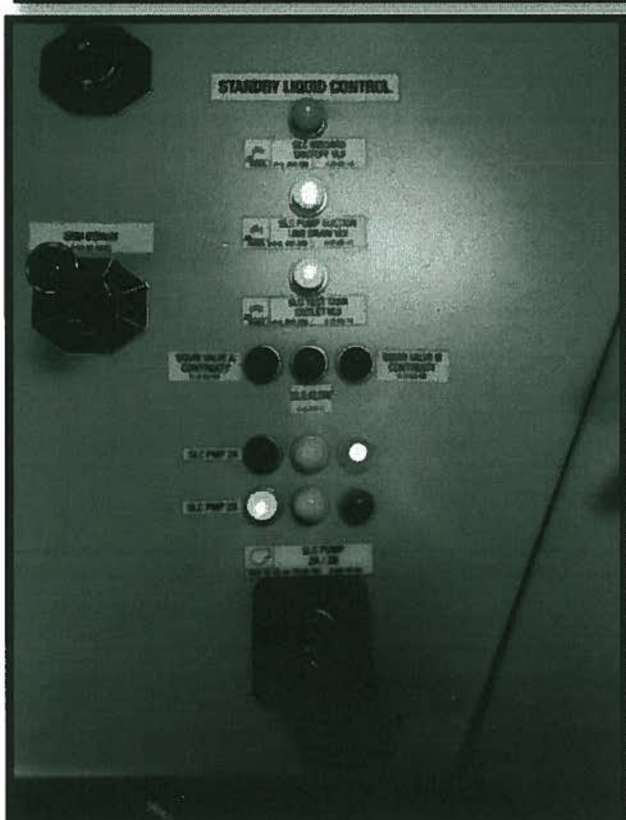
B. (1) is
(2) the 2A and 2B

C. (1) is NOT
(2) only the 2B

D. (1) is NOT
(2) the 2A and 2B

ANSWER: **C**


2000
1500
1000
500


STANDBY LIQUID CONTROL

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.1.31	
	Importance Rating	4.6	

G2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Explanation: **C CORRECT**: First Part: Correct- SLC is NOT injecting, based on too high a discharge pressure of 1300 psig and the SLC flow light remaining unlit. Second Part: Correct- If the switch is taken to Start-B, only the B pump will run based on an- interlock that prevents simultaneous operation of both SLC pumps.

A- Incorrect: First Part: Incorrect – Plausible as the continuity lights are not lit, SLC pump A red light illuminated, and SLC discharge pressure greater than RPV pressure which are all signs of SLC injection. Second Part: Correct - See D..

B- Incorrect: First Part: Incorrect – See A. Second Part: Incorrect – Plausible that taking the control switch to Start-B with the 2A pump already running would start the 2B pump and the 2A pump would remain running

D- Incorrect: First Part: Correct – See C. Second Part: Incorrect –See B.

Technical Reference(s): 2-EOI-Appendix 3A, 2-OI-63, OPL171.039

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.039 ILT Obj 4.h.

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

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

2-EOI Appendix 3A, SLC Injection

2-EOI APPENDIX-3A

SLC INJECTION

LOCATION: Unit 2 Control Room

ATTACHMENTS: None (√)

1. **UNLOCK** and **PLACE** 2-HS-63-6A, SLC PUMP 2A/2B, 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 Annunciator in alarm on Panel 9-5 (2-XA-55-5B, Window 20). ____
   - 2-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. ____
   - System flow, as indicated by 2-IL-63-11, SLC FLOW, red light illuminated on Panel 9-5, ____
   - SLC INJECTION FLOW TO REACTOR Annunciator in alarm on Panel 9-5 (2-XA-55-5B, Window 14). ____

3. IF ..... Proper system operation CANNOT be verified, THEN ... **RETURN** to Step 1 and **START** other SLC pump. ____

4. **VERIFY** RWCU isolation by observing the following:

   - RWCU Pumps 2A and 2B tripped ____
   - 2-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed ____
   - 2-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed ____
   - 2-FCV-69-12, RWCU RETURN ISOLATION VALVE closed. ____

3.0 PRECAUTIONS AND LIMITATIONS

3.1 SLC System Operation

A. The Unit SRO/RO or Shift Manager are the only persons authorized to inject SLC solution.

3.2 SLC Pump Operation

A. 2A and 2B SLC PUMP HAND SWITCHES, 2-HS-063-0006AA and 2-HS-063-0006B, are for pump starting only. The squib valves will not fire when using these control switches.

B. Starting either SLC pump from the control room fires both squib valves.

C. The SLC pumps are interlocked so that only one pump can be run at a time. Operation of both SLC pumps simultaneously may result in overpressurization of the system.

D. [II/F] SLC pump abnormal noise (similar to uncoupled or no load condition), lack of normal test tank perturbations, or smell of burnt packing may indicate the pump is air bound. These positive displacement pumps do not deliver flow if air bound. [Incident Investigation II-B-90-134]

QUESTION 67

The following conditions exist on Unit 3:

- 3-GOI-100-1A, Unit Startup, is in progress.
- Single notch withdrawal of Control Rods is required during the approach to criticality.

Following a notch withdrawal of Control Rod 30-31, the Operator observes a Reactor Period of 50 seconds.

Which ONE of the following describes the required action(s) to take based on the above conditions?

A. Shut down the Reactor until a thorough assessment has been performed.

B. Re-insert the last Control Rod pulled to achieve a stable period of greater than 60 seconds.

C. Insert Control Rods until the reactor is Subcritical. ALL Control Rods do NOT have to be inserted.

D. Stop Control Rod withdrawal AND monitor conditions, allowing power to decay to greater than 100 seconds before proceeding. Control Rod insertion is NOT required.

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.1.39	
	Importance Rating	3.6	

G2.1.39 Knowledge of conservative decision making practices.

Explanation: **B CORRECT** – 3-GOI-100-1A section 5.4, step [6.1]. Required if < 60 second period is observed.

A- Incorrect. Plausible because this is the action required in 3-GOI-100-1A section 5.4, step [6.3]required if < 5 second period is observed.

C- Incorrect. Plausible because this is the action in 3-GOI-100-1A section 5.4, step [6.2]required if < 30 second period is observed.

D- Incorrect. Plausible because withdrawing control rods to maintain a period of 100 seconds or greater is directed in 3-GOI-100-1A section 5.4, step [14].

Technical Reference(s): 3-GOI-100-1A

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 0801 #66

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

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

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

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

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

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

PERFORM the following:

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

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

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

PERFORM the following:

[6.2.1] **INSERT** control rods in accordance with 3-SR-3.1.3.5(A).

[6.2.2] **VERIFY** Reactor subcritical.

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

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

SHUT DOWN the Reactor until a thorough assessment has been performed. (**REFERENCE** 3-GOI-100-12A).

_____ Initials _____ Date _____ Time

5.4 **Withdrawal of Control Rods while in Mode 2 (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.

[12] **WHEN** in a configuration that is expected to be near critical, and Nuclear Instrument response is not as expected, **THEN**

NOTIFY Reactor Engineer and Unit Supervisor.

_______ _______ _______
Initials Date Time

[13] **IF** operation is to be suspended for greater than one hour near the point of criticality, **THEN**

PLACE the Reactor core sufficiently subcritical as directed by the Unit Supervisor and as advised by the Reactor Engineer, to avoid an inadvertent criticality. (Otherwise N/A)

_______ _______ _______
Initials Date Time

[14] **WITHDRAW** control rods to maintain a period of 100 seconds or greater as indicated on the following indicators on Panel 3-9-5:

- CHANNEL A PERIOD, 3-XI-92-7/44A.
- CHANNEL B PERIOD, 3-XI-92-7/44B.
- CHANNEL C PERIOD, 3-XI-92-7/44C.
- CHANNEL D PERIOD, 3-XI-92-7/44D.

(R) _______ _______ _______
Initials Date Time

66. G2.1.39

The following conditions exist on Unit 3:

- 3-GOI-100-1A, "Unit Startup," is in progress
- Single notch withdrawal of Control Rods is required during the approach to criticality

Following a notch withdrawal of Control Rod 30-31, the Operator observes a Reactor Period of 50 seconds.

Which ONE of the following describes the required action(s) to take based on the above conditions?

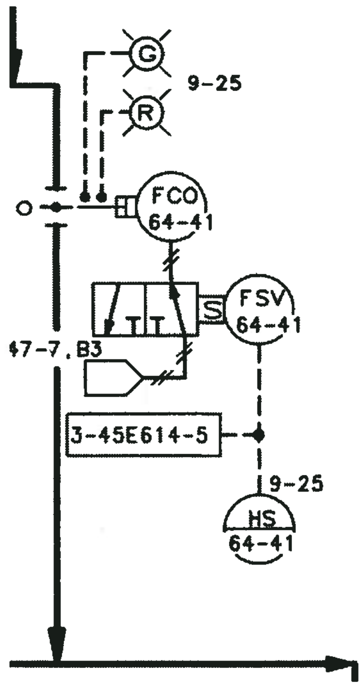
A. Shut down the Reactor until a thorough assessment has been performed.

B. Re-insert the last Control Rod pulled to achieve a stable period of greater than 60 seconds.

C. Insert Control Rods until the reactor is Subcritical. **ALL** Control Rods do **NOT** have to be inserted.

D. Stop Control Rod withdrawal **AND** monitor conditions, allowing power to decay to greater than 100 seconds before proceeding. Control Rod insertion is **NOT** required.

QUESTION 68



Given the drawing of FSV-64-41 above, which ONE of the following completes both statements below?

When the solenoid is energized, control air is __(1)__.

On a loss of air to FSV-64-41, the __(2)__ position indication on Panel 9-25 will be illuminated.

A. (1) applied to the air operator
   (2) red

B. (1) applied to the air operator
   (2) green

C. (1) vented off the air operator
   (2) red

D. (1) vented off the air operator
   (2) green

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.2.15	
	Importance Rating	3.9	

G2.2.15 Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.

Explanation: **C CORRECT:** Solenoid 3-FCV-64-41 is shown de-energized. When the damper solenoid is energized air is blocked and vented off the operator. The drawing shows that the RX ZONE EXH SGT XTIE DMPR OPR, 2-FCV-64-41, fails open.

A- Incorrect. First Part: Incorrect. Plausible if the candidate does not know which state (energized or de-energized) the solenoid is shown in the drawing. Second Part: Correct.

B- Incorrect. First Part: Incorrect. Plausible if the candidate does not know which state (energized or de-energized) the solenoid is shown in the drawing. Second Part: Incorrect. Plausible if the candidate is not familiar with the drawing indication of the failed position.

D- Incorrect.. First Part: Correct. Second Part: Incorrect. Plausible if the candidate is not familiar with the drawing indication of the failed position.

Technical Reference(s): Drawing 3-47E610-64-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: BFN 1306 #69

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

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

Attachment 1
(Page 4 of 7)

Expected System Responses

10.0 STATOR COOLING

A. STATOR CLG WTR CLR DISCH TEMP, 1-TCV-035-0054 (Y-07), fails OPEN to high stop. Minimum flow at high stop will be ≈697 gpm.

B. STATOR COOLING WATER COOLER DISCH PRESS, 1-PCV-035-0055 (Y-63), fails OPEN. The operating Stator Cooling Pump discharge valve must be throttled to maintain stator cooling water Generator inlet pressure 3 psig less than generator hydrogen pressure.

11.0 PRIMARY CONTAINMENT

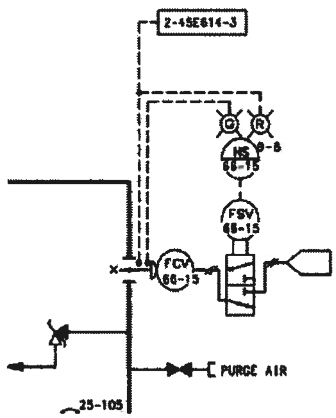
A. All valves and dampers associated with System 64, including Reactor and Refuel zone ventilation dampers fail CLOSED on loss of air EXCEPT for the following:

1. RX ZONE EXH SGT XTIE DMPR OPR, 1-FCO-064-0040, fails OPEN.
2. RX ZONE EXH SGT XTIE DMPR OPR, 1-FCO-064-0041, fails OPEN.

BFN 1306 #69

QUESTION 69

The operating crew is implementing a clearance on the component shown below.



Which ONE of the following completes the statements below?

The solenoid is shown ___(1)____.

The flow control valve fails ___ (2)___.

A. (1) energized
   (2) closed

B. (1) energized
   (2) open

C. (1) de-energized
   (2) open

D. (1) de-energized
   (2) closed

ANSWER: **D**

QUESTION 69

On Unit 1, which ONE of the following combinations of Reactor Power AND Reactor Pressure constitute a Safety Limit violation?

	Reactor Power	Reactor Pressure
A.	15%	750 psig
B.	24%	770 psig
C.	28%	775 psig
D.	32%	810 psig

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.2.22	
	Importance Rating	4.0	

Knowledge of limiting conditions for operations and safety limits.

Explanation: **C CORRECT**: With the reactor steam dome pressure < 785 psig or core flow< 10% rated core flow, THERMAL POWER shall be ≤ 25% RTP.

A- Incorrect: If Reactor Pressure greater than 785 psig, this would be a correct answer.

B- Incorrect: If Reactor Pressure greater than 785 psig, this would be a correct answer.

D- Incorrect: If Reactor Power was less than 25%, this would be a correct answer.

Technical Reference(s): Unit 1 Tech Specs, Sect. 2.0

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.087 V.B.14

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1006 #70
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X

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

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

THERMAL POWER shall be ≤ 25% RTP.

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

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

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

70. G2.2.22

Which ONE of the following combinations of Reactor Power **AND** Reactor Pressure on Unit 1 constitute a Safety Limit violation?

	Reactor Power	Reactor Pressure
A.	15%	750 psig
B.	24%	770 psig
C.	28%	775 psig
D.	32%	810 psig

QUESTION 70

Given the following conditions for Unit 2:

- Reactor power is 100%
- The running CRD pump tripped
- The standby CRD pump has been placed in service

Subsequently,

- CONTROL ROD DRIVE UNIT HIGH TEMP, (2-9-5A, Window 17) alarms

Which ONE of the following completes the statements below?

In accordance with Tech Spec 3.1.4, the affected Control Rod(s) are to be declared ___(1)___.

The ARP for CONTROL ROD DRIVE UNIT HIGH TEMP, (2-9-5A, Window 17) directs ___(2)___.

A. (1) SLOW
(2) raising CRD flow

B. (1) SLOW
(2) isolating the affected CRD HCU(s) from service

C. (1) INOPERABLE
(2) raising CRD flow

D. (1) INOPERABLE
(2) isolating the affected CRD HCU(s) from service

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.2.38	
	Importance Rating	3.6	

Knowledge of conditions and limitations in the facility license.

Explanation: **A CORRECT**: First Part: Correct - Per Tech Spec 3.1.4 Bases, temperatures are permitted to rise to 350° F prior to affecting Scram times. Over 350° F, they would be declared SLOW per note 1 of T.S. 3.1.4 -1 table; making the second part correct. ARP directs raising drive water flow.

B- Incorrect: First part: Correct- See A. Second Part: Incorrect- Plausible since this is a required action in TS 3.1.3 Control Rod Operability. .

C- Incorrect: First part: Incorrect. Plausible since high temperatures adversely affect scram times. Per T.S. 3.1.4 Bases and 2-TI-393, temperatures are permitted to rise to 350 °F prior to affecting Scram times. Then, they would be declared SLOW per note 1 of T.S. 3.1.4 -1 table in accordance with 2-TI-393 and ARP-9-5A.Second Part: Correct- See A.

D- Incorrect: First part: Incorrect- See C. Second Part: Incorrect- See B.

Technical Reference(s): U2 Tech Spec 3.1.4, 2 -ARP-9-5A, 2-TI-393

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.006 V.B.18/22

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1006 #24
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X
10 CFR Part 55 Content:	55.41 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

CONTROL ROD DRIVE UNIT TEMP HIGH 2-TA-85-7 17

(Page 1 of 2)

Sensor/Trip Point:

TE-85-7 (1 thru 185) 350°F — Alarm comes from recorders, 2-TR-085-007A1, & 2-TR-085-007B1

Sensor Location: Located on each control rod drive.

Probable Cause:

A. Insufficient cooling water flow.
B. Malfunction of sensor.
C. Leaking scram discharge valve.
D. Plugged CRD cooling water orifice.

Automatic Action: None

Operator Action:

A. **VALIDATE** high temp of CRD on recorder 2-TR-085-007A1, & 2-TR-085-007B1 (Panel 2-9-47) or on ICS. ☐

B. **IF** alarm is valid, **THEN** perform the following as directed by the Unit Supervisor: ☐
- **CHECK** cooling water pressure and flow normal on Panel 2-9-5. ☐
- **DISPATCH** personnel to check for HCU scram discharge valve leaking as indicated by elevated discharge piping temperatures for associated CRD. ☐
- **PERFORM** 2-TI-393 for control rods with high temperatures or failed thermocouples. ☐
- **REFER TO** 0-OI-55, 2-OI-85, 2-AOI-85-3. ☐
- **FLUSH** CRD to unblock restricted cooling water flow. **REFER** to 2-OI-85. ☐
- **DECLARE** the control rod, which is in alarm, "SLOW" as directed by 2-TI-393 per Tech Spec. Table 3.1.4-1 Note 1. ☐
- **RAISE** CRD Flow, as directed by Unit Supervisor, if required to keep the drives cool per "CRD Pump Operation At Elevated Flow" section of 2-OI-85. ☐

C. **IF** alarm is invalid, **THEN PERFORM** the following as directed by the Unit Supervisor: ☐
- **REFER TO** 0-OI-55. ☐
- **INITIATE** WO to determine cause of invalid alarm. ☐

Continued on Next Page

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds) REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.

BASES (continued)

LCO

The scram times specified in Table 3.1.4-1 (in the accompanying LCO) are required to ensure that the scram reactivity assumed in the DBA and transient analysis is met (Ref. 6).

To account for single failures and "slow" scramming control rods, the scram times specified in Table 3.1.4-1 are faster than those assumed in the design basis analysis. The scram times have a margin that allows up to approximately 7% of the control rods (e.g., 185 x 7% ≈ 13) to have scram times exceeding the specified limits (i.e., "slow" control rods) assuming a single stuck control rod (as allowed by LCO 3.1.3, "Control Rod OPERABILITY") and an additional control rod failing to scram per the single failure criterion. The scram times are specified as a function of reactor steam dome pressure to account for the pressure dependence of the scram times. The scram times are specified relative to measurements based on reed switch positions, which provide the control rod position indication. The reed switch closes ("pickup") when the index tube passes a specific location and then opens ("dropout") as the index tube travels upward. Verification of the specified scram times in Table 3.1.4-1 is accomplished through measurement of the "dropout" times. To ensure that local scram reactivity rates are maintained within acceptable limits, no more than two of the allowed "slow" control rods may occupy adjacent locations.

Table 3.1.4-1 is modified by two Notes, which state that control rods with scram times not within the limits of the table are considered "slow" and that control rods with scram times > 7 seconds are considered inoperable as required by SR 3.1.3.4.

Scram times can be adversely affected by high control rod drive temperatures. Temperatures over 350°F may result in a measurable delay in scram time response times for an otherwise normally performing CRD due to the potential for flashing of the hot water in the drive when the scram valves are opened. As a conservative measure, CRDs which have a

(continued)

BASES

LCO (continued)	temperature of greater than 350°F will either be classified as "slow" rods or an engineering evaluation can be performed. This LCO applies only to OPERABLE control rods since inoperable control rods will be inserted and disarmed (LCO 3.1.3). Slow scramming control rods can be conservatively declared inoperable and not accounted for as "slow" control rods.
APPLICABILITY	In MODES 1 and 2, a scram is assumed to function during transients and accidents analyzed for these plant conditions. These events are assumed to occur during startup and power operation; therefore, the scram function of the control rods is required during these MODES. In MODES 3 and 4, the control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod scram capability during these conditions. Scram requirements in MODE 5 are contained in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."
ACTIONS	A.1 When the requirements of this LCO are not met, the rate of negative reactivity insertion during a scram may not be within the assumptions of the safety analysis. Therefore, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

24. 295022 G2.2.38

Unit 2 is operating at 100% Reactor Power when the running CRD pump trips. The standby CRD pump has been placed in service. The following alarm is subsequently received:

- CONTROL ROD DRIVE UNIT HIGH TEMP, (2-9-5A, Window 17), is in alarm

Which ONE of the following identifies the required actions?

A. Declare the affected Control Rod(s) "SLOW" **AND** raise CRD flow.

B. Declare the affected Control Rod(s) "INOPERABLE" **AND** raise CRD flow.

C. Declare the affected Control Rod(s) "SLOW" **AND** isolate the affected CRD HCU(s) from service.

D. Declare the affected Control Rod(s) "INOPERABLE" **AND** isolate the affected CRD HCU(s) from service.

Answer: **A**

QUESTION 71

Given the following conditions:

- Unit 3 has entered the EOIs
- Immediate entry into a High Radiation Area by an Assistant Unit Operator (AUO) is required.
- NO RWP currently exists for this entry.

In accordance with RCI-9.1, Radiation Work Permits, which ONE of the following completes the statements below?

This High Radiation Area entry, without an RWP, must be authorized by the __(1)__.

A Radiation Protection individual escort__(2)__ required.

A. (1) Shift Manager
   (2) is

B. (1) Shift Manager
   (2) is NOT

C. (1) Radiation Protection Shift Supervisor
   (2) is

D. (1) Radiation Protection Shift Supervisor
   (2) is NOT

Correct Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.3.7	
	Importance Rating	3.5	

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Explanation: **A CORRECT**: First Part: RCI-9.1 section 3.2.17- This High Radiation Area entry, without an RWP, must be authorized by the shift Manager. Second Part: A Radiation Protection individual equipped with a dose rate monitoring device must escort the AUO.

B- Incorrect: First part: Correct- See A. Second Part: Incorrect. Plausible because according to RCI-9.1, section 3.2.8.A, personnel normally use a dose rate warning device as indicated on the RWP vice RP surveillance for High Radiation Area entry.

C- Incorrect: First part: Incorrect. Plausible because according to RCI-9.1, section 3.1.1, the RP Shift Supervisor normally approves RWPs. Second Part: Correct- See A.

D- Incorrect First part: Incorrect- See C. Second Part: Incorrect --See B.

Technical Reference(s): RCI-9.1, Radiation Work Permits

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1108 #71
Question Cognitive Level:	Memory or Fundamental Knowledge: X Comprehension or Analysis :

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

3.2.17 Emergency Situations

In emergency situations where the Shift Manager authorizes immediate entry to an area, the prior approval requirements of a RWP will be waived. If the RWP approval requirement is waived, Radiation Protection or the personnel escorted by RP must comply with radiation protection procedures for entry into high radiation areas (i.e., RP individual is equipped with radiation dose rate monitoring device and provides positive control over activities within the area to include protective recommendations for the personnel being escorted for the duration of the emergency). Radiation surveillance by virtue of RP escort is considered to be continuous coverage in this situation. The RWP must be completed when the emergency entry is completed or the emergency is over.

3.1.1 Radiation Protection Shift Supervisors (or assigned designee)

Will approve RWP/Support Requests and seven day extensions.

Will review active RWPs to ensure procedural compliance.

When assigned to Work Control will ensure Work Orders requiring entry to RWP areas will include the RWP number on the work order to the extent practical.

3.2.8 Entry to High Radiation Areas

A. Entry to high radiation areas will require the use of a dose rate meter, dose warning device or RP surveillance. Normally, personnel should use the dose warning device and this should be indicated on the RWP.

1. RP shall verify the workers trip ticket was adequately completed (see section 3.2.5 prior to the issue of a dose warning device.

71. G2.3.7 NEW/L

Unit 3 has entered the EOIs and immediate entry into a High Radiation Area by an Assistant Unit Operator (AUO) is required. NO RWP currently exists for this entry.

Which ONE of the following completes both statements below in accordance with RCI-9.1, Radiation Work Permits?

This High Radiation Area entry, without an RWP, must be authorized by the __(1)__.

A Radiation Protection individual __(2)__.

A. (1) Shift Manager
   (2) equipped with a dose rate monitoring device must escort the AUO.

B. (1) Radiation Protection Shift Supervisor
   (2) equipped with a dose rate monitoring device must escort the AUO.

C. (1) Shift Manager
   (2) escort is not required.

D. (1) Radiation Protection Shift Supervisor
   (2) escort is not required.

CORRECT ANSWER A

Tier 3: Generic.
2.3.7. Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12/45.10) RO IR: 3.5

Plausibility:"B" and "D" first part plausible because according to RCI-9.1, section 3.1.1, the RP Shift Supervisor normally approves RWPs. "C" and "D" second part plausible because according to RCI-9.1, section 3.2.8.A, personnel normally use a dose rate warning device as indicated on the RWP vice RP surveillance for High Radiation Area entry.

References
NPG-SPP-05.1, section 3.6.4
RCI-9.1, Radiation Work Permits, section 3.2.17

Radiological Worker Training RWT 010/000 Obj: "State the required actions to be taken if the work scope or radiological conditions change so that they are not within the scope of an RWP",
Obj: "Ability to extract information from an RWP".

QUESTION 72

In accordance with 2-GOI-200-2, Primary Containment Initial Entry and Closeout, entering the drywell, with the primary system at or near rated operating temperature and pressure, requires permission from which ONE of the following?

A. Site Vice President

B. Plant Manager

C. Operations Manager

D. Radiation Protection Manager

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.3.13	
	Importance Rating	3.4	

G2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Explanation: **B** CORRECT – 2-GOI-200-2 requires Plant Manager permission for drywell entries at NOT/NOP (Precaution 3.1.H) and for entries with the Mode Switch in RUN (3.2.E)

A- Incorrect Plausible if the candidate believes that only the highest on site manager must approve the entry.

C- Incorrect – Plausible if the candidate believes the Operations Manager permission is required.

D- Incorrect – Plausible if the candidate believes that the senior radiation manger is sufficient.

Technical Reference(s): 2-GOI-200-2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:	
Question History:	Previous NRC: BFN 1306 #72	
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	(12) Radiological safety principles and procedures.

2-GOI-200-2 , Primary Containment Initial Entry and Closeout

H. Permitting access to the Drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the Reactor and operating for extended periods with significant leaks in the Primary System, leak inspections are scheduled during startup periods, when the Primary System is at or near rated operating temperature and pressure. These entries require Plant Manager permission.

BFN 1306

QUESTION 72

In accordance with 2-GOI-200-2, Primary Containment Initial Entry and Closeout, entering the drywell, with the primary system at or near rated operating temperature and pressure, requires permission from which ONE of the following?

A. Plant Manager

B. Site Vice President

C. Radiation Protection Manager

D. Operations Manager

Answer: **A**

QUESTION 73

Unit 1 was at 35% Reactor Power when the Hydrogen Injection System was placed in service in Automatic / Power Determined mode in accordance with 1-OI-4, Hydrogen Water Chemistry System.

- Power is raised from 35% Reactor Power to 100% Reactor Power
- At 100% Reactor Power hydrogen flow rate indicates 20 scfm

Which ONE of the following completes the statements?

In accordance with 1-OI-4, hydrogen injection flow rate is __(1)__ the normal 100% Reactor Power flow rate.

Radiation levels in the Condenser Bay will stabilize __(2)__ expected normal full power radiation levels.

A. (1) below
   (2) below

B. (1) below
   (2) at

C. (1) above
   (2) at

D. (1) above
   (2) above

Correct Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.3.14	
	Importance Rating	3.4	

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Explanation: **D CORRECT**: First Part: In the Automatic / Power Determined Mode, the hydrogen injection system is load following. It is normally placed in service above 25% Rx power. As Rx power is increased, the hydrogen flow rate is increased to the maximum amount that the controller is set for. Normal H2 Injection Rate (100% Reactor Power) is 14 scfm. Due to current EPU spanned software installed on U1 HWC computer, Chemistry has requested Ops to input a 16 scfm H2 injection value in order to receive the required 14 scfm actual H2 injection flow at 100% Power. Second Part: Operation of HWC with injection rates above normal will cause a significant rise in radiation dose rates in steam affected areas.

A- Incorrect: Part 1 and 2 incorrect as explained above.

B- Incorrect: First part: Incorrect. H2 flow rate is NOT below normal. Plausible in that this injection flow rate is below the maximum allowed H2 injection rate of 25 scfm. Second Part: Incorrect. radiation levels are expected to increase to above normal levels.

C- Incorrect: First part: Correct. Plausible because according to RCI-9.1, section 3.1.1, the RP Shift Supervisor normally approves RWPs. Second Part: Incorrect. Radiation levels are expected to increase to above normal levels. Plausible because flow is higher than normal, but the candidate must understand the effects of H2 injection on Radiation levels.

Technical Reference(s): 1-OI-4, OPL171.220

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1006 #72
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X

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

INSTRUCTOR NOTES

(8) At the OIU, transfer mode to Automatic / Power Determined Setpoint

or responses when viewing different screens

(9) After reaching steady state oxygen flow and indication, verify offgas oxygen stabilizes at 21% ± 5%

Adjusting the oxygen controller ratio may be required. Procedure USe

5. Normal Operation

a. During operation important injection flow rate values are:

(1) Minimum hydrogen injection rate allowed to be entered on the OIU: 3 SCFM. When 3 SCFM is entered in the OIU for Automatic/Power Determined mode, hydrogen injection rate will lower automatically to a new SCFM depending on new power level

Normally used when lowering HWC for ALARA or maintenance.

(2) Normal allowed hydrogen injection rate (100% reactor power): ~12-14 SCFM.

Unit Diff:
U1 / 2 = 14 scfm
stpt @ 16 scfm
U3 = 12 scfm

(3) Maximum hydrogen injection rate allowed: U1 / 3 = 25 SCFM, U2 = 20 SCFM

Value set by Chem Lab
This is the desired setpoint. This flowrate is post noble metals injection. However U-1 will utilize this flowrate pre-noble metal injection. Noble Metal injection planned to be performed 1st cycle of operation

b. Changes in reactor power can affect the HWC System's ability to operate

(1) Reactor power maneuvers <15%,: HWC System should control injection rates adequately

(2) Reactor power maneuvers >15% with a 15 minute wait period between each 30% change: HWC System should control injection rates adequately

INSTRUCTOR NOTES

a. These compounds are circulated through the reactor coolant systems and are ultimately removed by the RWCU System

b. A smaller fraction of the N^{16} is carried over in the steam in the form of nitrogen gas (N_2) and ammonia (NH_3)

Predominate contributor to background radiation levels

4. H_2 injection alters the N^{16} carryover ratio

a. Concentrations of NO_3, NO_2, and NO decrease

b. Concentration of NH_3 increases

Ammonia

(1) A gas

(2) High water solubility

5. The net production of N^{16} is not influenced by hydrogen injection

6. The increased dose rates are due to the increased ease with which N^{16} gets out of the reactor and into the steam pipes when in the NH_3 form

We can maintain up to 2.7 ppm injection concentration.

7. The initial U2 run was the first week in Nov. 1999. Up to 90 scfm hydrogen was injected. Average MSL radiation level increased approximately 5 times normal

MSL 'B' was highest at 5.2 times normal

8. Addition of noble metals to reactor water

a. Noble metals decompose during reactor startup or shutdown

Rubidium and Iridium

b. During this time it produces a thin layer of noble metal on wetted surfaces

c. The ECP on these surfaces are reduced significantly during subsequent operation

3.0 PRECAUTIONS AND LIMITATIONS (continued)

4. Important hydrogen injection flow rate values are as follows:

   a. Minimum H_2 Injection Rate allowed to be entered on the OIU is 3 scfm. This is the injection rate normally used when lowering HWC for ALARA considerations or maintenance purposes per Section 6.0, Normal Operations. When 3 scfm is entered in the OIU for Automatic/Power Determined Mode, H_2 Injection Rate will lower automatically to a new scfm depending on the new power level, i.e., 5 scfm for 100% power; when power is lowered to 90%, the injection rate will automatically roll back to 4.5 scfm and so on.

   b. Normal H_2 Injection Rate (100% Reactor Power) is 14 scfm. Due to current EPU spanned software installed on U1 HWC computer, Chemistry has requested Ops to input a 16 scfm H2 injection value in order to receive the required 14 scfm actual H2 injection flow at 100% Power. This value, 14 scfm, is determined by Chemistry with the performance of CI-13-1, Chemistry Program. Chemistry will notify Operations should this value change). "Off Normal" operating conditions may require other injection rates which are to be coordinated with the System Engineer, Chemistry, Radiation Protection, and approved by the Unit Supervisor/SRO.

   c. Maximum H_2 Injection Rate allowed: 25 scfm.

BFN 1006 NRC #71

Examination Outline Cross-reference:
G2.3.14 (10CFR 55.41.12)
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Level	RO	SRO
Tier #	3	------
Group #	------	------
K/A #	G2.3.14	
Importance Rating	3.4	------

Proposed Question: **# 72**

Unit 1 was at 35% Reactor Power when the Hydrogen Injection System was placed in service in Automatic / Power Determined mode in accordance with 1-OI-4, "Hydrogen Water Chemistry System."

- Power is raised from 35% Reactor Power to 100% Reactor Power
- At 100% Reactor Power hydrogen flow rate indicates 20 scfm

Which ONE of the following completes the statements?

In accordance with 1-OI-4, hydrogen injection flow rate is __**(1)**__ the normal 100% Reactor Power flow rate.

Radiation levels in the Condenser Bay will stabilize __**(2)**__ expected normal full power radiation levels.

A. **(1)** above
(2) at

B. **(1)** below
(2) at

C. **(1)** above
(2) above

D. **(1)** below
(2) below

QUESTION 74

Unit 2 is at 100% power with CRD pump 2A tagged out for maintenance.

Unit 1 is in Mode 2 with a startup in progress with the following conditions:

- Reactor Pressure 850 psig
- CRD Pump 1A is in service
- Reactor Power is on Range 8 of the IRMs

Subsequently,

- CRD Pump 1A trips on an electrical fault and can NOT be restarted.

Which ONE of the following describes the required operator actions in accordance with 1-AOI-85-3, CRD System Failure?

A. Manually scram and immediately place the mode switch to shutdown.

B. Immediately attempt to place 1B CRD Pump in service, manual scram is NOT required if charging water pressure can be restored and maintained above 940 psig.

C. Immediately upon discovery of charging water header pressure less than 940 psig AND with ONE or more scram accumulators INOPERABLE, then verify all control rods with inoperable accumulators are fully inserted.

D. If charging water can NOT be restored and maintained above 940 psig within 20 minutes AND with TWO or more scram accumulators INOPERABLE with the associated control rod NOT fully inserted, then Manually scram and immediately place the mode switch to shutdown.

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.4.1	
	Importance Rating	4.6	

Knowledge of EOP entry conditions and immediate action steps.

Explanation: **A CORRECT** – With Reactor Pressure less than 900 psig and the in service Pump tripped and neither CRD Pump can be started the action is to Manually scram and immediately place the mode switch to shutdown. With CRD Pump 2A tagged out CRD Pump 1B is operating to support Unit 2 operation.

B- Incorrect – Plausible in that this would be correct if CRD Pump 1B was available to Unit 1, CRD Pump 1B is being utilized by Unit 2 100% power operation.

C- Incorrect – Plausible in that this is the correct tech spec action for given conditions.

D- Incorrect – Plausible in that this would be correct is Reactor pressure was above 900 psig.

Technical Reference(s): 1-AOI-85-3, Tech Spec 3.1.5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.005 V.B.33

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 1205 #23

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

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

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** operating CRD PUMP has failed **AND** the standby CRD Pump is available, **THEN**

PERFORM the following at Panel 1-9-5: (Otherwise **N/A**)

[1.1] **PLACE** CRD SYSTEM FLOW CONTROL, 1-FIC-85-11, in MAN at minimum setting. ☐

[1.2] **START** associated standby CRD Pump using one of the following:

- CRD PUMP 1B, using 1-HS-85-2A
- CRD Pump 1A, using 1-HS-85-1A ☐

BFN Unit 1	CRD System Failure	1-AOI-85-3 Rev. 0004 Page 7 of 12

4.1 Immediate Actions (continued)

[2] IF Reactor Pressure is less than 900 psig AND either of the following conditions exists:

- In-service CRD Pump tripped and neither CRD Pump can be started, OR
- Charging Water Pressure can NOT be restored and maintained above 940 psig, THEN

PERFORM the following: (Otherwise N/A)

[2.1] **MANUALLY SCRAM** Reactor and **IMMEDIATELY** PLACE the Reactor Mode Switch in the SHUTDOWN position . ☐

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. ☐

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each control rod scram accumulator.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig.	B.1	Restore charging water header pressure to ≥ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1	Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	<u>AND</u>		
	C.2	Declare the associated control rod inoperable.	1 hour
D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met.	D.1	------------NOTE------------ Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. ------------------------------ Place the reactor mode switch in the shutdown position.	Immediately

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0137 Page 45 of 236

5.3 Aligning CRD Pump for Standby Service

NOTE
1B CRD pump standby alignment is to U1 with1B CRD pump suction valve aligned to U1 CST. 1B CRD pump is aligned per 1-OI-85.

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0137 Page 48 of 236

6.0 SYSTEM OPERATIONS

NOTE
1B CRD pump standby alignment is to U1 with 1B CRD pump suction valve aligned to U1 CST. 1B CRD pump is aligned per 1-OI-85.

6.1 Shifting CRD Pumps From 2A To 1B

[1] **NOTIFY** Radiation Protection of impending action to shift from 2A to 1B CRD Pump. **RECORD** name and time of notification in NOMS narrative log.

(R) ___________
Initials

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

- **REVIEW** all Precautions and Limitations in Section 3.6. ☐
- **VERIFY** CRD Pump 1B is available and <u>not</u> required to support Unit 1 operations. ☐

[3] **ESTABLISH** communications between control room and the following locations:

- Control Rod Drive Pump 2A, El 541, U2 Reactor Building Northwest corner. ☐
- Control Rod Drive Pump 1B, El 541, U1 Reactor Building Northeast corner. ☐

QUESTION 23

Unit 2 is at 100% power with CRD pump 2A tagged out for maintenance.

Unit 1 is in Mode 2 with a startup in progress with the following conditions:

- Reactor Pressure 850 psig
- CRD Pump 1A is in service
- Reactor Power is on Range 8 of the IRMs

SUBSEQUENTLY, CRD Pump 1A trips on an electrical fault and can NOT be restarted

Which ONE of the following describes the required operator actions in accordance with 1-AOI-85-3, CRD System Failure?

A. Immediately attempt to place 1B CRD Pump in service, manual scram is NOT required if charging water pressure can be restored and maintained above 940 psig.

B. Immediately upon discovery of charging water header pressure less than 940 psig AND with ONE or more scram accumulators INOPERABLE, then verify all control rods with inoperable accumulators are fully inserted.

C. Manually scram and immediately place the mode switch to shutdown.

D. If charging water can NOT be restored and maintained above 940 psig within 20 minutes AND with TWO or more scram accumulators INOPERABLE with the associated control rod NOT fully inserted, then Manually scram and immediately place the mode switch to shutdown.

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #	1	
	Group #	2	
	K/A#	295022 G2.4.1	
	Importance Rating	4.6	

295022 Loss of CRD Pumps G2.4.1: Knowledge of EOP entry conditions and immediate action steps.

Explanation: CORRECT – C, with Reactor Pressure less than 900 psig and the in service Pump tripped and neither CRD Pump can be started the action is to Manually scram and immediately place the mode switch to shutdown. With CRD Pump 2A tagged out CRD Pump 1B is operating to support Unit 2 operation.

A Incorrect – plausible in that this would be correct if CRD Pump 1B was available to Unit 1. CRD Pump 1B is being utilized by Unit 2 100% power operation.

B Incorrect – plausible in that this is the correct tech spec action for given conditions.

D Incorrect – plausible in that this would be correct is Reactor pressure was above 900 psig

Technical Reference(s): 1-AOI-85-3, Tech Spec 3.1.5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.005 V.B.33

Question Source:	Bank:	
	Modified Bank:	
	New	X
Question History:	Previous NRC	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	X
	55.43	

QUESTION 75

Unit 3 is operating at 100% when the following alarms are received:

- RFPT B ABNORMAL, (3-9-6C, Window 8)
- RFPT TRIP, (3-9-6C, Window 29)
- RFP DISCH FLOW LOW, (3-9-6C, Window 32)
- REACTOR WATER LEVEL ABNORMAL, (3-9-5A, Window 8), due to Low Level

Which ONE of the following completes both of the following statements?

In accordance with 3-AOI-3-1, Loss of Reactor Feedwater or Reactor Water Level High/Low, the crew is required to __(1)__.

Unit 3 Reactor Feed Pump speeds are limited to ≤ __(2)__ in accordance with 3-OI-3, Reactor Feedwater System.

A. (1) verify automatic runback of reactor recirculation pumps
(2) 5050 rpm

B. (1) verify automatic runback of reactor recirculation pumps
(2) 5850 rpm

C. (1) reduce core flow to 50 to 60%; then, manually scram the Reactor
(2) 5050 rpm

D. (1) reduce core flow to 50 to 60%; then, manually scram the Reactor
(2) 5850 rpm

Answer: A

	Level:	RO	SRO
	Tier #	3	
	Group #		
	K/A#	G2.4.45	
	Importance Rating	4.1	

Ability to prioritize and interpret the significance of each annunciator or alarm.

Explanation: **A CORRECT:** Based on the alarms identified in the stem, RFPT B has tripped and Reactor Level has dropped less than (+) 27 inches. The 75% Limiter will initiate an automatic runback of Recirc pump speed if any individual RFP flow is < 19% AND RPV water level lowers to the low level alarm set point of + 27" (level 4). 75% corresponds to 1130 rpm. 2-AOI-3-1, "Loss of Reactor Feedwater or Reactor Water Level High/Low" subsequent actions direct the crew to verify applicable automatic actions. Part 2 correct - 3-OI-3 specifies operating limit for RFPT speed of 5050 rpm.

B – Incorrect – First Part: Correct. Second Part: Incorrect. Precautions and Limitations for 3-OI-3 say to maintain Operating RFPs for Unit 3 ≤ 5050 rpm. This distracter's plausibility is based on unit difference. Precautions and Limitations for 2-OI-3 say to maintain Operating RFPs for Unit 2 ≤ 5850 rpm

C – Incorrect – . First Part: Incorrect, The purpose of the 75% limiter is to automatically reduce reactor power to a value within the capacity of the remaining Feedwater Pumps. With that the Low Level Reactor Scram will not be challenged. Manually Scramming the Reactor would not be required or appropriate. Second Part: Correct.

D– Incorrect – First Part: Incorrect, The purpose of the 75% limiter is to automatically reduce reactor power to a value within the capacity of the remaining Feedwater Pumps. With that the Low Level Reactor Scram will not be Second Part: Incorrect. Precautions and Limitations for 3-OI-3 say to maintain Operating RFPs for Unit 3 ≤ 5050 rpm. This distracter's plausibility is based on unit difference. Precautions and Limitations for 2-OI-3 say to maintain Operating RFPs for Unit 2 ≤ 5850 rpm. challenged. Manually Scramming the Reactor would not be required or appropriate.

Technical Reference(s) 3-ARP-9-6C, 3-ARP-9-5A, OPL171.007, 3-AOI-3-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 0801 #44
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X

10 CFR Part 55 Content: 55.43 (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

OPL171.007 Reactor Recirc

4. 75% Limiter

   a. The 75% Limiter will initiate an automatic runback of Recirc pump speed if

      75% = 1130 rpm

      Any individual RFP flow is < 19% AND RPV water level lowers to the low level alarm setpoint of + 27" (level 4)

      28% = 480 rpm

      OR

      Reactor Scram Signal (Units 2 and 3 only - DCN 67325 and DCN 65487-A)

      DCN 65487-A

   b. The purpose of the limiter is to automatically reduce reactor power to a value within the capacity of the remaining feedwater pumps.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

FF. It is acceptable to SHUTDOWN Vapor Extractor for up to 24 hours since this should NOT cause any detrimental effect to the system. If the Extractor needs to be out longer than 24 hours, then an evaluation of the effects on the system should be done by System Engineering. [SEOPR 96-2/3 047-003]

GG. For operating Feed Pumps, monitor and maintain the following parameters within ranges described below.

1. RFPT Hydraulic Pressure: ≈ 200 psig (local indication).
2. Lube Oil Pressure to RFP Bearings: ≈ 15 psig (local indication).
3. Lube Oil Pressure to RFPT Bearings: ≈ 10 psig (local indication).
4. Bearing lube oil from cooler: 110°F to 120°F (obtained from Process Computer Point Id's 24-56, 24-54, and 24-52).
5. Bearing lube oil to cooler: 180°F maximum (obtained from Process Computer Point Id's TBD025, TBD032, and TBD039).
6. Maximum Oil Temp Rise across the Turbine Bearings: 50°F.
7. Vertical Vibration at RFP Bearing Supports: 2 mils double amplitude.
8. RFPT Speed: 5050 rpm maximum (3-9-6).

HH. New Flow Control Valve, 3-FCV-3-53 Start up Bypass Valve, has a hand wheel associated with it which acts as a local locking device (Dogging device). With hand wheel all the way closed, valve will respond normally from the output air signal. When hand wheel is fully in open direction, then valve is locked in open position. This is a unique acting valve and close attention to detail is required when hand wheel is manipulated. This hand wheel is not for locking valve closed.

II. Maintenance will be required to provide documentation for all leads lifted and re-landed in this procedure.

BFN 0801 #44

Examination Outline Cross-reference:	Level	RO	SRO
259002 Reactor Water Level Control	Tier #	2	----
G2.4.45 (10CFR 55.41.10)	Group #	1	----
Ability to prioritize and interpret the significance of each annunciator or alarm.	K/A #	259002G2.4.45	
	Importance Rating	4.1	----

Proposed Question: # 44

Unit 3 is operating at 100% when the following alarms are received:

- RFPT B ABNORMAL, (3-9-6C, Window 8)
- RFPT TRIP, (3-9-6C, Window 29)
- RFP DISCH FLOW LOW, (3-9-6C, Window 32)
- REACTOR WATER LEVEL ABNORMAL, (3-9-5A, Window 8), due to Low Level

Which ONE of the following completes both of the following statements?

In accordance with 2-AOI-3-1, "Loss of Reactor Feedwater or Reactor Water Level High/Low," the crew is required to __(1)__ . Unit 3 Reactor Feed Pump speeds are limited to ≤ __(2)__ in accordance with 3-OI-3, "Reactor Feedwater System."

A. (1) Verify Automatic Runback of Reactor Recirculation Pumps
(2) 5050 rpm

B. (1) Reduce Core Flow to 50 to 60%; then, manually Scram the Reactor.
(2) 5050 rpm

C. (1) Verify Automatic Runback of Reactor Recirculation Pumps
(2) 5850 rpm

D. (1) Reduce Core Flow to 50 to 60%; then, manually Scram the Reactor.
(2) 5850 rpm

Proposed Answer: A