Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K1.06	
	Importance Rating	3.4	

Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: Drywell instrument air/ drywell pneumatics: Plant-Specific

Question: RO #1

Which one of the following describes the response of a Relief/Safety Valve to a rising RPV pressure if a pneumatic supply is NOT available from any source?

A Relief/Safety Valve will...

A. Function in its pressure relief mode and in its safety mode

B. NOT function in its pressure relief mode but will function in its safety mode

C. Function in its pressure relief mode and NOT function in its safety mode

D. NOT function in its pressure relief mode and NOT function in its safety mode

Answer: B

Explanation (Optional):

A: Incorrect: A pneumatic supply (accumulator / continuous) is required to lift in the Pressure Relief mode.

B: Correct: A pneumatic supply is not required in the Safety mode. In the safety mode, whenever reactor pressure is greater than spring pressure the valve will open.

C: Incorrect: In the safety mode, whenever reactor pressure is greater than spring pressure the valve will open. A pneumatic supply is NOT required.

D: In the safety mode, whenever reactor pressure is greater than spring pressure the valve will open. A pneumatic supply is NOT required.

Technical Reference(s): Tech Spec Bases, page B 3.4.4-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-218000C01, Automatic Depressurization System (As available)
RBO-11, System Loss & Component Level Malfunction

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

Question History:

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

10 CFR Part 55 Content: 55.41 3
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 K1.05	
	Importance Rating	3.9	

Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: Remote shutdown system: Plant-Specific

Question: RO #2

While at rated conditions, a fire in the Relay Room results in entry into N2-SOP-78, Control Room Evacuation. The following occur:

- All SOP-78 Control Room actions are completed before evacuating the Control Room
- Before any action is taken outside the Control Room, the Automatic Depressurization System (ADS) spuriously initiates and all ADS valves open.
- Shortly thereafter all appendix R disconnect switches on panels 2CES*PNL415, 416 and 417 are placed in the actuate position.

Which one of the following describes the response of the ADS valves?

A. All ADS valves will close when the disconnect switches are placed in actuate.

B. All ADS valves remain open and cannot be closed unless solenoid fuses are removed.

C. The four ADS valves controlled from Remote Shutdown Panel, 2CES*PNL405 close. The remaining three will not close until solenoid fuses are removed.

D. The four ADS valves controlled from Remote Shutdown Panel, 2CES*PNL405 will close when their switches are taken out of the NORMAL position. The remaining three will not close until solenoid fuses are removed.

Answer: A

Explanation (Optional):

A: Correct: Per N2-OP-78, Remote Shutdown System Attachment 2, disconnect switches SW1-2CESA02 (Div 1) and SW1-2CESB02 (Div 2) isolate solenoids for the 4 ADS valves controlled from the remotes shutdown panel. Switches SW1-2CESA04 (Div 1) and SW1-2CESB04 (Div 2) isolate remainder of ADS circuitry.

B: Incorrect: Per N2-OP-78, Remote Shutdown System Attachment 2, disconnect switches SW1-2CESA02 (Div 1) and SW1-2CESB02 (Div 2) isolate solenoids for the 4 ADS valves controlled from the remotes shutdown panel. Switches SW1-2CESA04 (Div 1) and SW1-2CESB04 (Div 2) isolate remainder of ADS circuitry.

C: Incorrect: When the disconnect switches are repositioned, the ADS solenoids lose power and the valves close. See A & B above.

D: Incorrect: When the disconnect switches are repositioned, the ADS solenoids lose power and the valves close. See A & B above.

Technical Reference(s): N2-OP-78, Remote Shutdown System Attachments 2 and 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-296000C01, Remote Shutdown System RBO-1, System Function & Purpose (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K2.02	
	Importance Rating	2.7	

Knowledge of electrical power supplies to the following: Analog trip system logic cabinets

Question: RO #3

Which one of the following power supply failures would result in a loss of the RPS "A" Analog Trip Units on panel 2CEC*PNL609?

Loss of...

A. 2VBB-UPS1A

B. RPS UPS 2VBB-UPS3A

C. RPS MG set 2RPM-MG1A

D. Emergency UPS 2VBA*UPS2A

Answer: B

Explanation (Optional):

A: Incorrect: RPS UPS 2VBB-UPS3A is the power supply

B: Correct: Correct power supply

C: Incorrect: RPS UPS 2VBB-UPS3A is the power supply

D: Incorrect: RPS UPS 2VBB-UPS3A is the power supply

Technical Reference(s): N2-OP-97 section B.1.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-212000C01, Reactor Protection System RBO-04, System & Component Power Supplies (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

.

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 K2.01	
	Importance Rating	2.6	

Knowledge of electrical power supplies to the following: SRM channels/detectors

Question: RO #4

The plant is in Mode 4 when 600V distribution panel 2NJS-PNL600 is de-energized.

Assuming no operator actions which one of the following is the affect of this loss on the Source Range Monitors (SRM)?

A. SRM Channels A and C de-energize immediately.

B. SRM Channels A and C de-energize after several hours.

C. SRM Channels B and D de-energize immediately.

D. SRM Channels B and D de-energize after several hours.

Answer: D

Explanation (Optional):

A: Incorrect: 2NJS-PNL600 supplies the chargers associated 2BWS-BAT3B/3D which supplies SRM B and D. SRM A and C are supplied by BWS-PNL300A, which is fed by 2NJS-PNL-500. Plausible if the candidate does not understand the AC and 24/48 VDC system configuration.

B: Incorrect: 2NJS-PNL600 supplies the chargers associated 2BWS-BAT3B/3D which supplies SRM B and D. SRM A and C are supplied by BWS-PNL300A, which is fed by 2NJS-PNL-500. Plausible if the candidate does not understand the AC and 24/48 VDC system configuration.

C: Incorrect: SRM B and D are affected but do not immediately de-energize because on the loss of power to the 24/48 VDC battery chargers, 2BWS-BAT3B/3D supply power and are sized to maintain voltage for 4 hours. Plausible in that if the battery chargers are not available, the SRMs would immediately de-energize.

D: Correct: 2NJS-PNL600 powers chargers 2BWSCHGR3B1 and 2BWS-CHGR3D1. Although both chargers are lost, the SRMs remain energized via 2BWS-BAT3B/3D. These batteries are sized to maintain voltage for up to 4 hours before voltage begins to decay. If the chargers are not restored, SRMs will eventually de-energize.

Technical Reference(s): N2-OP-73B, 24/48 VOLT D.C. DISTRIBUTION system, page 3. N2-ELU-01, Attachments 73B & 92 (sheet 3 of 5) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-215002C01, Source Range & Intermediate Range Monitoring Systems RBO-04, System & Component Power Supplies (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K3.03	
	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Systems with D.C. components (i.e. valves, motors, solenoids, etc.)

Question: RO #5

The plant is at rated conditions when a complete loss of Normal 125 VDC Power bus 2BYS-SWG001A occurs.

Which one of the following is a direct plant response to this event?

A. Main turbine immediately trips

B. BOTH Recirc pumps immediately trip

C. Loss of Main Generator protective relays control power

D. Loss of control power to 2NPS-SWG001/2/3 load breakers

Answer: C

Explanation (Optional):

A: Incorrect: Main turbine will trip if RPM is < 1300. Since plant is at rated power, turbine will not trip.

B: Incorrect: Only the "A" Recirc pump will trip. Both pumps trip when DC bus 2BYS-SWG002A(B) is lost.

C: Correct: The loss of 2BYS-SWG001A causes a loss of Main Generator protective relays control power and results in Annunciator 852604, Generator protective relays control power failure.

D: Incorrect: Load side breaker control power is lost when DC bus 2BYS-SWG001B fails.

Technical Reference(s): SOP-4, Attachment 5, pg 11 of 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-263000C01, Plant DC Electrical Distribution RBO-11, System Loss & Component Malfunction (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 K3.05	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Rod worth minimizer Plant-Specific

Question: RO #6

A plant startup in progress with the following:

- Reactor power is 30%
- The Rod Worth Minimizer (RWM) is in the Transition Zone
- The UPS supply breaker to the RWM trips.

Other than scramming the reactor which one of the following describes the current control rod movement capability?

A. Rods can be withdrawn OR inserted using the reactor manual control system.

B. Rods CANNOT be withdrawn OR inserted using the reactor manual control system.

C. Rods can only be inserted if the Continuous Insert push button is used. Rods CANNOT be withdrawn using any method.

D. Rods can only be inserted if the Continuous Insert push button is used. Rods can only be withdrawn if the Continuous Withdraw mode is used.

Answer: B

Explanation (Optional):

A: Incorrect: Loss of power to RWM removes permissive to move rods. Plausible in that most RWM blocks (other than hardware and critical self test faults) are bypassed in the Transition Zone.

B: Correct: Loss of power to RWM removes permissive to move rods.

C: Incorrect: Loss of power to RWM removes permissive to move rods. Plausible in that Continuous insert works independently of the timer circuit. However, it will not override a RWM block.

D: Incorrect: Loss of power to RWM removes permissive to move rods. Plausible in that Continuous insert works independently of the timer circuit. However, it will not override a RWM block. Also most blocks from RWM are bypassed in the Transition zone.

Technical Reference(s): Facility Lesson Plan N2101201002C01, pages 108 &132. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-201002C01, Reactor Manual Control System RBO-8, Interrelated Systems (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K4.02	
	Importance Rating	3.7	

Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: High pressure isolation Plant-Specific

Question: RO #7

The following conditions exist:

- The reactor is in Mode 3.
- RHR Loop "B" has just been placed in Shutdown Cooling
- Reactor pressure has risen to 150 psig

Which one of the following would indicate that a failure had occurred in the automatic response of the RHR system?

A. RHR*P1B pump is running

B. RHS*MOV2B, PMP 1B SDC SUCT VLV is open

C. RHS*MOV40B, SDC B RETURN THROTTLE is closed

D. RHS*MOV8B, HEAT EXCHANGER 1B INLET BYPASS VALVE is in mid position (throttled)

Answer: A

Explanation (Optional):

A: Correct: The pump should have tripped on loss of suction flowpath when MOV112 and 113 closed.

B: Incorrect: SDC suctions MOV 112 and 113 close on Hi reactor pressure not the pump suction valve.

C: Incorrect: This valve gets a closed signal on high reactor pressure. The valve responded correctly to the event.

D: Incorrect: This valve gets an auto open signal on an initiation and does not close on an isolation signal.

Technical Reference(s): Facility lesson plan N2101205000C01. Page 81 describes the loss of suction trips on the RHR pumps. Page 83 describes the Group 5 isolation signals and response. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP N2101205000C01, RHR, RBO-5, System Operation, Control and Instrumentation (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K4.05	
	Importance Rating	3.4	3.6

Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Dispersal of boron upon injection into the vessel

Question: RO #8

The plant is at rated conditions when the following indications are received.

- Annunciator 601706, HPCS System INOP alarms
- Amber System Status Light "HPCS Line Break" illuminates

Which one of the following describes the impact if a Standby Liquid injection is required?

If the system is actuated Standby Liquid will inject into the...

A. core shroud region of the reactor vessel.

B. downcomer region of the reactor vessel.

C. drywell and will NOT reach the reactor vessel.

D. core shroud region but NOT the HPCS spray header.

Answer: B

Explanation (Optional):

A: Incorrect: Alarm annunciates on high pressure when CS injection line pressure rises. This occurs when the CS injection line rises to downcomer pressure when the line breaks outside the shroud but inside the vessel.

B: Correct: Alarm annunciates on high pressure when CS injection line pressure rises. This occurs when the CS injection line rises to downcomer pressure when the line breaks outside the shroud but inside the vessel. SBL injection would still be via the downcomer.

C: Incorrect: Alarm would not alarm for a break in the drywell.

D: Incorrect: Alarm annunciates on high pressure when CS injection line pressure rises. This occurs when the CS injection line rises to downcomer pressure when the line breaks outside the shroud.

Technical Reference(s): Pages 83 and 108 of HPCS Lesson Plan N2101209002C01 for discussion of HPCS line break detection and where SLS ties into HPCS. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-209002C01, HPCS RBO-5, System Operation, Control, & Instrumentation (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 K5.02	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: Breaker control

Question: RO #9

The plant was at rated condition when the following occur:

- A complete loss of off-site power occurred
- Emergency Diesel Generators (EDGs) start and re-power their respective buses.
- Ten minutes later, all Division I 125 VDC is lost.

Which one of the following is the current status of 2EGS*EG1 and its output breaker 101-1?

	EGS*EG1	BRKR 101-1
A.	Tripped	Open
B.	Tripped	Closed
C.	Running	Open
D.	Running	Closed

Answer: B

Explanation (Optional):

A: Incorrect: Control power has been lost to the output breaker.

B: Correct: Loss of DC control power will de-energize the EDG trip relays tripping the diesel, and breaker 101-1 will not open due to no power to its trip coil.

C: Incorrect: The EDG will trip because a Loss of DC control power will de-energize the EDG trip relays.

D: Incorrect: The EDG will trip because a Loss of DC control power will de-energize the EDG trip relays.

Technical Reference(s): N2-SOP-04, Attachment 2, page 2 of 10. (Attach if not previously provided)
N2-ARP-01, Att. 25, 852111,
EGSBC03

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-263000C01 (As available)
Plant DC Electrical Distribution
RBO-8, Interrelated Systems

Question Source:	Bank #	32598	Minor editorial changes
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K5.03	
	Importance Rating	3.0	

Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: Changing detector position

Question: RO #10

A reactor plant startup is being conducted in accordance with N2-OP-101A.

- The reactor is critical and SRM/IRM overlap data has just been completed.
- All SRMs are reading between 5 X10E4 and 1X10E5 cps
- All IRMs are on mid scale on range 1
- The operator has selected both the SRMs and the IRMs for withdraw.

Which one of the following will be the first automatic protective action as the detectors are withdrawn?

A. SRM INOP trip

B. IRM Downscale rodblock

C. SRM Downscale rodblock

D. IRM Detector NOT fully inserted rod block

Answer: D

Explanation (Optional):

A: Incorrect: SRM INOP trip will not be generated.

B: Incorrect: Although IRMs will go downscale, this rod block is bypassed with the IRMs on Range 1.

C: Incorrect: Although this will occur, it will NOT be the First, when the withdraw function is selected the first action will be the movement of the SRM and IRM detectors from the core. Immediately the IRMs will be detected to be not fully inserted The SRMs would have to be moved significantly to lower their counts from 5 X10E4 and 1X10E5 cps to <100 cps.

D: Correct: When the withdraw function is selected the first action will be the movement of the SRM and IRM detectors from the core. Immediately the IRMs will be detected to be not fully inserted This rod block is active whenever the mode switch is not in run.

Technical Reference(s): ARP603442, Control Rod Out Block
Page 79 of Facility Lesson Plan N2101215002C01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-215002C01, Source & Intermediate Range Monitoring RBO-5 System Operation, Control, and Instrumentation (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 2
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 K6.01	
	Importance Rating	3.2	

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Plant air systems

Question: RO #11

A plant startup is in progress, with the following:

- Feedwater Pump A is in service
- All three Feedwater Pump Suction MOVs, CNM-MOV84A, B and C, are open
- Low Flow Control Valve 2FWS-LV55A is controlling RPV level in AUTO
- 2FWS-LV55A is currently 40% open
- Instrument air to ALL Feedwater valves is lost

Which one of the following describes the effect on 2FWS-LV55A position and reactor water level?

	2FWS-LV55A Position	Reactor Water Level
A.	Fails closed	Lowering, below normal level
B.	Fails as is	Constant, normal level
C.	Fails as is	Lowering, below normal level
D.	Fails open	Rising, above normal level

Answer: C

Explanation (Optional):

A: Incorrect: Low Flow Control Valve 2FWS-LV55A fails as is on loss of air.

B: Incorrect: Feed pump min flow valves fail open on air, diverting water from the reactor resulting in a lowering level.

C: Correct: Low Flow Control Valve 2FWS-LV55A fails as is on loss of air. Feed pump min flow valves fail open on loss of air, diverting water from the reactor resulting in a lowering level.

D: Incorrect: Low Flow Control Valve 2FWS-LV55A fails as is on loss of air, not open

Technical Reference(s): N2-SOP-19, Loss of Instrument Air, section 2.0
Facility Feedwater Control System lesson plan, N2101259002C01, page 98
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-259002C01, Feedwater Control System
RBO-11, System Loss and Component Level Malfunction
(As available)

Question Source: Bank # 54097 Minor editorial changes made
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 3
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 K6.01	
	Importance Rating	2.7	

Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Valves

Question: RO #12

The plant is at rated conditions when a complete loss of pneumatics to 2CCP-TV108, RB CLOSED LOOP COOLING TEMPERATURE CONTROL VALVE, occurs.

Which one of the following describes the effect of this failure on the CCP heat exchanger?

A. MORE flow is directed to SHELL side of heat exchanger.

B. MORE flow is directed to TUBE side of heat exchanger.

C. LESS flow is directed to SHELL side of heat exchanger

D. LESS flow is directed to TUBE side of heat exchanger.

Answer: A

Explanation (Optional):

A: Correct: On a loss of air, 2CCP-TV108 positions itself to provide maximum cooling. The valve in the heat exchanger discharge header will fail open and the valve in the heat exchanger bypass line fails closed, which directs MORE flow through the SHELL side of the heat exchanger.

B: Incorrect: Service Water flow is through the TUBE side of the heat exchanger and is unaffected by changes in TV108 position. TV108 controls CCP flow through the SHELL side.

C: Incorrect: TV108 fails to maximum cooling position, which directs more flow through the SHELL side of the heat exchanger.

D: Incorrect: Service Water flow is through the TUBE side of the heat exchanger and is unaffected by changes in TV108 position. TV108 controls CCP flow through the SHELL side.

Technical Reference(s): Facility Lesson Plan for CCP (N2101208000C01), pages 23 & 31. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: none

Learning Objective: LP #2101-20800C01, RBCLC (As available)
RBO-2, Function & Location of Major Components
RBO-7 Design & Operational Considerations

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A1.03	
	Importance Rating	3.8	

Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Reactor water level

Question: RO #13

Following a small break on a Recirc Loop the following conditions exist:

- Low Pressure Core Spray is the only injection system available
- RPV water level has been restored
- Low Pressure Core Spray Injection Valve, 2CSL*MOV104, is throttled and Core Spray Flow is 2000 gpm

Then, the recirc loop under goes a double guillotine shear. In response to the lowering RPV level the operator fully opens 2CSL*MOV104.

Assuming the Core Spray functions as designed the Core Spray injection will:

A. Raise level to the level of the Main Steam Lines

B. Stabilize level at approximately -14 inches (actual level)

C. Stabilize level at approximately -62 inches (actual level)

D. Be insufficient to recover level to above the Bottom Of the Active Fuel

Answer: C

Explanation (Optional):

A: Incorrect: During a design bases accident, one ECCS is expected to re-flood to the top of the Jet pump suctions and then spill over to the downcomer and out the break. Break flow will prevent any significant re-flooding of the downcomer.

B: Incorrect: During a design bases accident, one ECCS is expected to re-flood to the top of the Jet pump suctions and then spill over to the downcomer and out the break. Break flow will prevent any significant re-flooding of the downcomer. -14 inches is the Top Of the Active Fuel.

C: Correct: -62 inches is the level of the jet pump suctions. During a design bases accident, one ECCS is expected to re-flood to the top of the Jet pump suctions.

D: Incorrect: During a design bases accident, one ECCS is expected to re-flood to the top of the Jet pump suctions and then spill over to the downcomer and out the break. Break flow will prevent any significant re-flooding of the downcomer but will provide sufficient core cooling. Level will NOT lower to the bottom of the fuel.

Technical Reference(s): Tech Spec Bases 3.4.3, Jet Pumps
Lesson Plan #2101-209001C01, Low Pressure Core Spray, page 96
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-209001C01, LPCS
RBO-12, EOP Implementation
(As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 2

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000 A1.01	
	Importance Rating	3.7	

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including: RCIC flow

Question: RO #14

Following a reactor scram and loss of off-site power, RCIC is manually started to control RPV level. Current conditions are:

- RPV level is 180 inches and slowly rising
- RCIC Flow Controller 2ICS*FC101 is in Auto
- RCIC Flow Controller 2ICS*FC101 setpoint has been reduced to 400 gpm.

In order to control RPV level, ICS*MOV124, Test Bypass to Condensate Storage Tank, is opened and ICS*FV108, Test Bypass to Condensate Storage Tank, is throttled in mid position.

Which one of the following are the responses of RPV injection flowrate and RCIC Flow indication on RCIC Flow Controller 2ICS*FC101, if ICS*FV108 is opened further?

	RPV injection flowrate	RCIC Flow Controller indication
A.	Lower	Remain the same
B.	Lower	Lower
C.	Raise	Remain the same
D.	Raise	Raise

Answer: A

Explanation (Optional):

A: Correct: Further opening of the Test Bypass will divert additional flow to the CST. The controller sees the increase in flow and closes down on the turbine, keeping flow at 400 gpm.

B: Incorrect: Controller indication will remain the same. Plausible in that if the candidate needs to consider where the test return line is relative to the flow transmitter.

C: Incorrect: Injection flow will lower. Plausible in that the candidate needs to consider how the valve opening affects discharge pressure in the injection line and on flow to the RPV.

D: Incorrect: Injection flow will lower.

Technical Reference(s): RCIC procedure N2-OP-35, page 23. RCIC "BIG NOTES" for location of flow transmitter. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-217001C01, RCIC RBO-3, Functional Arrangement (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 A2.05	
	Importance Rating	3.0	

Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Fan trips

Question: RO #15

The plant is operating at 100% power with a normal Reactor Building ventilation configuration. Reactor Building differential pressure is -0.6 inches WG when the following sequence of events occur:

Time:

1300: Drywell pressure rises to 14 psig
1300: GTS Trains A and B automatically start
1310: GTS Train A is shutdown by placing Train A Initiation control switch in Auto After Stop
1315: GTS Train B Fan trips due to a blown control power fuse

Which one of the following describes the impact of these events on Reactor Building differential pressure (DP) and the actions required to restore RB differential pressure?

	Reactor Building DP...	Actions Required
A.	Becomes less negative	GTS Train A must be manually restarted
B.	Becomes less negative	Confirm automatic restart of GTS Train A
C.	Becomes more negative	Defeat high Drywell pressure interlocks and restart HVR
D.	Becomes more negative	Confirm automatic restart of GTS Train A

Answer: B

Explanation (Optional):

A: Incorrect: Manual restart of Train A is NOT required to restore DP. The train will auto restart.

B: Correct: GTS Train A restarts because the high Drywell pressure (>1.68 psig) initiation signal is still present and with no running GTS train RB differential pressure will degrade toward 0. When pressure reaches -0.25 inches WG, the manually shutdown train will automatically restart.

C: Incorrect: With no running GTS train RB differential pressure will degrade toward 0 and not be maintained above -0.25 inches. The actions to defeat high drywell pressure interlocks and restart HVR are only authorized by EOP-SC. If dp never reached 0, then EOP-SC would not be entered.

D: Incorrect GTS Train A will not autostart on low train flow.

Technical Reference(s): N2-OP-61B, Page 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-261000C01, SGTS RBO-5, System Operation, Control, & Instrumentation (As available)

Question Source: Bank # 2005 NRC Exam. RO #45 .

Modified Bank #

New

Question History: 2005 NRC Exam. RO #45

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A2.02	
	Importance Rating	3.5	

Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Pump trips

Question: RO #16

Following a loss of coolant accident and loss of off-site power the following conditions exist:

- RHR Loops A and B are injecting
- NO other ECCS injection sources are available
- Reactor pressure is 100 psig and steady
- Reactor level is 50 inches and steady

While in these conditions RHR Pump A trips and cannot be restarted.

Which one of the following describes an Alternate Injection System that N2-EOP-6 Attachment 6 specifies to re-establish level control?

A. Fire water cross tied to RHR Loop B

B. Service Water cross tied to RHR Loop B

C. Fire water cross tied to RHR Loop A

D. Service Water cross tied to RHR Loop A

Answer: C

Explanation (Optional):

A: Incorrect. Attachment 6 specifies that the RHR loop is not being used prior to cross tying to fire water. Loop B is currently being used for injection.

B: Incorrect: Loop B is currently being used for injection. Attachment 5 requires the loop be out of service prior to cross tie.

C: Correct: Attachment 6 provides direction to align firewater provided the loop is currently NOT being utilized. Reactor pressure is also low enough to allow firewater injection.

D: Incorrect: Attachment 5 does not provide instructions for cross tying service water to loop "A" (can only be cross connected to loop "B")

Technical Reference(s): N2-EOP-06, attachments 5 and 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-205000C01, RHR (As available)
RBO-12, EOP Implementation

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

Question History:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Facility:	Nine Mile Point Unit 2
Vendor:	GE
Exam Date:	2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 A3.04	
	Importance Rating	3.1	

Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Operation of the governor control system on frequency and voltage control

Question: RO #17

With the plant at rated conditions the following occur:

- Drywell pressure rises to 2 psig
- The Standby Diesel generators start and run unloaded.

Two minutes later the following 2EGS*EG1 indications are observed in the control room:

- Engine RPM: 608 RPM
- Generator VARs: 0 VARS
- Generator Frequency: 60.8 Hz
- Generator AC output voltage: 4170 VAC

Given these indications, which one of the following is correct?

A. The Auto Voltage Regulator has failed and the voltage regulator has shifted to manual.

B. The Primary Control Circuit (Control Circuit #1) has failed and the engine is operating in Parallel Mode.

C. The electronic speed control governor has failed and the hydraulic governor is controlling engine speed.

D. A complete failure of the electronic-hydraulic control governor has occurred and engine speed is being controlled by the overspeed governor.

Answer: C

Explanation (Optional):

A: Incorrect: Voltage is acceptable. Plausible in that candidate must understand that VARS would be zero if unloaded.

B: Incorrect: If the Primary circuit failed the control circuit would shift to the Redundant circuit. The difference is that the Redundant Circuit will ONLY control in the Isochronous Mode.

C: Correct: A failure of the electronic speed control governor will shift control to the hydraulic governor which will control the engine at ~608 RPM. The frequency indicated is the frequency associated with a 12 pole engine operating at 608 RPM.

D: Incorrect: The overspeed governor activates at 660 RPM and trips the engine.

Technical Reference(s): Facility Lesson Plan 2101-264001C01, Standby Diesel Generators page 36 and 74. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-264001C01, Standby Diesel Generators (As available)
RBO-1, System Function & Purpose
RBO-2, Function & Location of Major Components

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209002 A3.04	
	Importance Rating	3.7	

Ability to monitor automatic operations of the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) including: System flow: BWR-5,6

Question: RO #18

Following a plant transient, High Pressure Core Spray (HPCS) auto initiates and injects.

HPCS recovers level and injection flow stops when CSH*MOV107, Pump 1 Injection Valve, closes on RPV high level.

Current plant conditions are as follows:

- RPV level is +180 inches and lowering
- Drywell pressure 1.2 psig and rising
- No operator actions have been taken up to this point.

Then Drywell pressure rises above 2 psig.

Which one of the following is correct regarding the response of CSH*MOV107 and the required actions, if ANY, to restore injection?

	CSH*MOV107 will...	Required Actions
A.	Automatically open	None
B.	Remain closed	Depress the Hi WTR Level Seal-In Reset
C.	Remain closed	Depress the Manual Initiation Seal-In Reset
D.	Remain closed	Place the control switch for CSH*MOV107 to the OPEN position

Answer: B

Explanation (Optional):

A: Incorrect: CSH*MOV107 is interlocked closed until level lowers to level 2 or the high water level reset is depressed.

B: Correct: CSH*MOV107 will re-open if high level reset is depressed.

C: Incorrect: The initiation will reset but CSH*MOV107 will not re-open until level lowers to level 2.

D: Incorrect: CSH*MOV107 cannot be opened manually until the high level is reset.

Technical Reference(s): ARP 601718, HPCS Reactor Water Level High (Attach if not previously provided)
LP #2101-209002C01, HPCS, page 49

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-209002C01, HPCS RBO-5, System Operation, Control, & Instrumentation (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A4.04	
	Importance Rating	3.2	

Ability to manually operate and/or monitor in the control room: LPRM back panel switches, meters and indicating lights

Question: RO #19

The reactor is at 100% power when the Key Lock switch on LPRM Chassis 1 is placed in the INOP position.

How will this affect the Input Status lights for APRM 1 on Two-Out-Of-Four Logic Module, MDL 1?

A. All status lights will remain extinguished.

B. The OPRM status light will illuminate. All others will be extinguished.

C. The HIGH/INOP status light AND the OPRM status light will illuminate.

D. The HIGH/INOP status light will illuminate. All others will be extinguished.

Answer: C.

Explanation (Optional):

A: Incorrect: Placing the LPRM keylock in INOP will INOP both the associated APRM and OPRM.

B: Incorrect: Placing the LPRM keylock in INOP will INOP both the associated APRM and OPRM.

C: Correct: Per N2-OP-92 precaution 15, Placing an APRM slave (LPRM chassis) keylock switch in the INOP position will be detected by the APRM master and place the APRM channel in INOP (just as if the APRM keylock switch has been placed in INOP). This will trip the OPRM input to the voter as well.

D: Incorrect: Placing the LPRM keylock in INOP will INOP both the associated APRM and OPRM. Plausible in that the reactor is at full power where the OPRM is not normally enabled.

Technical Reference(s): N2-OP-92 precaution 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-215003C01, Power Range Neutron Monitoring & RBM (As available)
RBO-2, Function & Location of Major Components

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 A4.04	
	Importance Rating	3.1	

Ability to manually operate and/or monitor in the control room: IRM back panel switches, meters, and indicating lights.

Question: RO #20

A plant Startup is in progress with the following:

- All APRMs are reading 6%
- The Reactor Mode Switch is in STARTUP with final checks in progress for placing the mode switch to RUN.

Which one of the following identifies the effect of placing the IRM A Mode Switch (S1) in the Standby position on panel 2CEC*PNL606?

A. Nothing happens at this power level.

B. Annunciator IRM UPSCALE / INOPERABLE alarm only.

C. Control Rod block and Annunciator IRM UPSCALE / INOPERABLE alarm only.

D. Half scram, Control Rod block and Annunciator IRM UPSCALE / INOPERABLE alarm.

Answer: D

Explanation (Optional):

A: Incorrect: ½ scram and rod block. Plausible because of the high power level.

B: Incorrect: ½ scram and rod block. Plausible because of the high power level.

C: Incorrect: ½ scram and rod block. Plausible because of the high power level.

D: Correct: Taking the mode switch out of operate generates a half scram and rod block

Technical Reference(s): IRM / SRM Facility Lesson plan N2101215002C01, pages 72, 79, 80, & 81. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-215002C01, Source & Intermediate Range Monitoring Systems
RBO-5, System Operation, Control, & Instrumentation (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 2.2.12	
	Importance Rating	3.7	

Equipment Control: Knowledge of surveillance procedures: 223002 PCIS/Nuclear Steam Supply Shutoff.

Question: RO #21

Primary Containment Isolation Valve testing is being conducted on containment purge AOVs in accordance with N2-OSP-CPS-Q001, Primary Containment Purge System Valve Operability Test.

Which one of the following would be the correct method for determining whether an AOV's closure time was acceptable for a fully open AOV?

	Start the timing when the...	Stop the timing ...
A.	control switch is taken to the close position	when the green closed indication illuminates
B.	green closed indication illuminates	when the red open indication extinguishes
C.	control switch is taken to the close position	when the red open indication extinguishes
D.	green closed indication illuminates	two seconds after red open indication extinguishes

Answer: C

Explanation (Optional):

A: Incorrect: The green light will illuminate as soon as the valve goes intermediate.

B: Incorrect: The valve has to leave the full open seat before the intermediate limit switch picks up. This would result in an incomplete timing of the valve.

C: Correct: Sect. 4.2.1 Measuring Valve Stroke times for all valves except Solenoid Operated Valves.
- Measure opening stroke time from the time the control switch is placed to OPEN until the green indicating light de-energizes.
- Measure closing stroke time from the time the control switch is placed to CLOSE until the red indicating light de-energizes.

This simulates the receipt of the PCIS signal and the action of the valve in response.

D: Incorrect: The green light will illuminate as soon as the valve goes intermediate.

Technical Reference(s): N2-OSP-CPS-Q001, page 2 section 4.2.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-223003C01, Containment Vent & Purge Systems RBO-10, Operational Actions & Sequence (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 2.1.28	
	Importance Rating	4.1	

Conduct of Operations: Knowledge of the purpose and function of major system components and controls. Shutdown Cooling

Question: RO #22

Which one of the following RHR valves are electrically interlocked with RHS*MOV2A, PMP 1A SDC SUCT VLV, and the reason for that interlock?

If RHS*MOV2A is OPEN then...

A. RHS*MOV24A, LPCI A Injection Valve, will not open to prevent bypassing the recirculation loop.

B. RHS*MOV4A, RHR Pump 1A Minimum Flow Valve, will not open to prevent draining the RPV to the suppression pool.

C. RHS*FV38A, RHR A Return to the Suppression Pool, will not open to prevent draining the RPV to the suppression pool.

D. RHS*MOV24A, LPCI A Injection Valve AND RHS*MOV40A, SDC A Injection valve, cannot both be opened at the same time to prevent running out the RHR pump.

Answer: C

Explanation (Optional):

A: Incorrect: No such interlock. Plausible in that RHS*MOV24A would bypass the Recirc loop which is the normal path for SDC.

B: Incorrect: No such interlock. Plausible in that the min flow valve opening would drain the RPV if in SDC. (Valve is disabled when in SDC).

C: Correct: If in SDC none of the suppression pool cooling /spray valves can be opened.

D: Incorrect: No such interlock. Plausible in that both valves being open would create two parallel discharge paths.

Technical Reference(s): ESK-6RHS04, RHR SDC MOV 2RHS*MOV2A (Attach if not previously provided)
RHR Facility Lesson Plan, N2101205000C01 Page 81.

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-205001C01, RHR System (As available)
RBO-5, System Operation, Control, & Instrumentation

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 2.4.46	
	Importance Rating	4.2	

Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions. Source Range Monitor

Question: RO #23

Following a Reactor scram the RO places the mode switch to Shutdown and commences inserting all IRMs and SRMs. Which one of the following SRM alarms would be expected as the detectors are being inserted?

A.
- SRM Downscale
- SRM Detector Position Abnormal

B.
- SRM Short Period
- SRM Upscale/Inoperable

C.
- SRM Short Period
- SRM Detector Position Abnormal

D.
- SRM Downscale
- SRM Upscale/Inoperable

Answer: B

Explanation (Optional):

A: Incorrect: The SRMs should not go downscale immediately after a scram. Given the -80 second period, neutron flux is still significantly high many minutes after a scram.

B: Correct: An upscale trip is expected as the detectors are driven into the core and move into high flux regions. A short period is also expected. The detectors interpret the rise in counts as they are inserted as a short period.

C: Incorrect: The SRM Detector Position Abnormal alarm is not expected as count rate will remain above 100 cps (setpoint for alarm).

D: Incorrect: The SRMs should not go downscale immediately after a scram. Given the -80 second period, neutron flux is still significantly high many minutes after a scram.

Technical Reference(s): N2-OP-92 page 14 and LP 2101-215002C01 page 19. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-215002C01, Source and Intermediate Range Monitoring Systems
RBO-5, System Operation, Control, and Instrumentation (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 A2.07	
	Importance Rating	3.0	

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Energizing a dead bus

Question: RO #24

The plant is at rated power when the following occur:

- 2ENS*SWG101 de-energizes when Reserve Transformer 2RTX-XSR1A locks out
- Diesel Generator 1 (EG 1) automatically starts
- EG 1 output breaker, 101-1, does NOT automatically close

Which one of the following is the potential impact of immediately attempting to manually close EG 1 output breaker, 101-1 and what actions are required by N2-SOP-3, Loss of AC Power, prior to manually closing the breaker?

	Impact of Immediately Closing 101-1	Required Actions
A.	Overloading EG 1	Place selected Bus loads in Pull to Lock ONLY
B.	Energizing a faulted bus	Verify there are no faults on the Bus ONLY
C.	Overloading EG 1	Place selected Bus loads in Pull to Lock AND Place Bus 101 Synchronizing Switch to On
D.	Energizing a faulted bus	Verify there are no faults on the Bus AND Place Bus 101 Synchronizing Switch to On

Answer: D

Explanation (Optional):

A: Incorrect: without an ECCS signal none of the large loads have closed on the bus therefore there is NOT a danger of overloading EG 1, therefore there is no requirement in SOP-3 to place large loads in PTL. Procedurally, the synch switch is required to be on prior to closing the diesel output breaker.

B: Incorrect: Procedurally, the synch switch is required to be on prior to closing the diesel output breaker.

C: Incorrect: without an ECCS signal none of the large loads have closed on the bus therefore there is NOT a danger of overloading EG 1, therefore there is no requirement in SOP-3 to place large loads in PTL.

D: Correct: A potential fault on 2ENS*SWG101 would prevent EG 1 from automatically closing on the bus. SOP-3 requires the operator to verify the bus is not faulted by verifying the bus fault annunciators (852147 and 852148) are clear. With EG 1 already running the procedure directs placing the Synch Switch in on and then manually closing the breaker.

Technical Reference(s): N2-SOP-03 Flowchart and specifically Detail A. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-SOP03C01, Loss of AC RBO-2, Operational Actions and Sequence (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K3.09	
	Importance Rating	3.2	

Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: The magnitude of heat energy that must be absorbed by the containment during accident/transient conditions

Question: RO #25

The reactor is at 100% power and 100% core flow when the following occurs:

- Main Turbine trip.
- RPS trips but NO rod insertion occurs.
- After the initial RPV pressure response, RPV pressure is stabilized between 970 psig to 1000 psig.

Which one of the following describes the amount of energy that is being generated 30 to 45 seconds after the transient begins?

(Assume that all systems respond as expected but there continues to be zero rod motion and no operator actions have taken place other than stabilizing RPV pressure).

	Approximate Reactor Power	Basis for Power Lowering
A.	Within Bypass Valve capacity	Recirc pumps are at slow speed
B.	Greater than Bypass Valve capacity	Recirc pumps are at slow speed
C.	Within Bypass Valve capacity	Recirc pumps have tripped and Feedwater flow has runback
D.	Greater than Bypass Valve capacity	Recirc pumps have tripped and Feedwater flow has runback

Answer: D

Explanation (Optional):

A: Incorrect: 30 to 45 seconds after the event occurs , with the recirc pumps tripped, reactor power lowers but remains >25% which is greater than bypass valve capacity. The LFMG sets are tripped and the feed flow runback occurs 25 seconds after the high pressure trip.

B: Incorrect: The LFMG sets are tripped and the feed flow runback occurs 25 seconds after the high pressure trip.

C: Incorrect: 30 to 45 seconds after the event occurs , with the recirc pumps tripped, reactor power lowers but remains >25% which is greater than bypass valve capacity.

D: Correct: The LFMG sets are tripped and the feed flow runback occurs 25 seconds after the high pressure trip. 30 to 45 seconds after the event occurs , with the recirc pumps tripped, reactor power lowers but remains >25% which is greater than bypass valve capacity.

Technical Reference(s): Power to Flow Map (Attach if not previously provided)

RRCS Facility Lesson Plan (N2101294008C01), pages 66 & 67

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-129008C01, RRCS RBO-7, Design and Operational Considerations (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K4.03	
	Importance Rating	2.8	

Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Securing of IAS upon loss of cooling water

Question: RO #26

The plant is operating at rated conditions when a complete loss of CCP Mini-Loop cooling water to the Instrument Air system occurs.

Which one of the following is correct regarding the impact on the Instrument Air Compressors?

	The operating Compressor trips on...	Backup Compressors...
A.	low cooling water flow	are blocked from starting
B.	high outlet air temperature	are blocked from starting
C.	low cooling water flow	start and eventually trip
D.	high outlet air temperature	start and eventually trip

Answer: D

Explanation (Optional):

A: Incorrect: There is no low cooling water flow interlock. Plausible because there is a low cooling water flow alarm. The backup air compressors will start and eventually also trip on high outlet air temperature.

B: Incorrect: The backup air compressors will start and eventually also trip on high outlet air temperature.

C: Incorrect: There is no low cooling water flow interlock. Plausible because there is a low cooling water flow alarm.

D: Correct: The compressors trip on high outlet air temperature. There is no interlock to prevent the lagging and backup compressors from starting. The backup air compressors will start and eventually also trip on high outlet air temperature.

Technical Reference(s): Facility Lesson Plan for Instrument Air N2101278001C01, Pages 104 and 105
ARP 851259
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-278001C01, Service, Instrument, and Breathing Air
RBO-5, System Operation, Control, and Instrumentation
(As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 8
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	271000	K1.04	
	Importance Rating	2.7		

Knowledge of the physical connections and/or cause-effect relationships between OFFGAS SYSTEM and the following: Condensate system

Question: RO #27

Which of the following components are cooled by the Condensate System?

1. Offgas Condensers A and B
2. Steam Jet Air Ejector Intercondensers
3. Steam Jet Air Ejector Precoolers
4. Mechanical Vacuum Pump Seal Cooler

A. Components 1 and 3

B. Components 2 and 4

C. Component 1 ONLY

D. Component 2 ONLY

Answer: D

Explanation (Optional):

A: Incorrect: Off gas condensers are cooled by TBCCW.

B: Incorrect: Mechanical Vacuum Pump Seal Cooler is cooled by Service Water.

C: Incorrect: Off gas condensers are cooled by TBCCW.

D: Correct: Steam Jet Air Ejector Intercondensers are cooled by the Condensate System.

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

N2-OP-3, Condensate and Feedwater, Page 5

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-255000C01, Condenser Air Removal (As available)
RBO-2, Function and Location of Major Components

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000 K1.02	
	Importance Rating	3.8	

Knowledge of the physical connections and/or cause effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: PCIS/NSSSS

Question: RO #28

Which one of the following describes the potential effect of a loss of RDS Backfill Injection prior to a depressurization event?

	RPV water level indication will	PCIS water level isolations will occur
A.	Indicate lower than actual	above the actual setpoint
B.	Indicate lower than actual	below the actual setpoint
C.	Indicate higher than actual	above the actual setpoint
D.	Indicate higher than actual	below the actual setpoint

Answer: D

Explanation (Optional):

A: Incorrect: Displacing water causes indicated level to be higher than actual, not lower. When indicated level lowers to 159.3, actual level is lower, therefore isolations that should have occurred at the actual level will occur below the actual water level.

B: Incorrect: Displacing water causes indicated level to be higher than actual, not lower.

C: Incorrect: When indicated level lowers to 159.3, actual level is lower, therefore isolations that should have occurred at the actual level will occur below the actual water level.

D: Correct: RDS Backfill keeps the reference leg full of water and prevents the accumulation of dissolved gasses in solution that may come out of solution during sudden depressurization causing a loss of density in the reference leg that results in lowering the dp across the transmitter which correlates to indicated level being higher than actual. When indicated level lowers to 159.3, actual level is lower, therefore isolations that should have occurred at the actual level will occur below the actual water level.

Technical Reference(s): N2-OP-101C, Plant Shutdown, precaution # 12, page 24, and page 68 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-216001C01, Reactor Vessel Instrumentation
RBO-2, Function and Location of Major Components (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201002 K3.03	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: Ability to process rod block signals

Question: RO #29

The plant is at rated conditions when annunciator 603303, ROD DRIVE CONTROL SYSTEM INOPERABLE, alarms.

Which one of the following describes the significance of this annunciator as it relates to the Reactor Manual Control System (RMCS) and the probable cause of this alarm?

	Effect on Control Rod motion	Probable cause
A.	Rod motion is NOT impacted	Loss of RMCS permissive signal
B.	Rod motion is NOT impacted	Loss of RDS Drive Water DP
C.	Rod withdrawal is blocked	Loss of RMCS permissive signal
D.	Rod withdrawal is blocked	Loss of RDS Drive Water DP

Answer: C

Explanation (Optional):

A: Incorrect: Per the associated ARP a rod block is generated to prevent rod motion when rod blocks may be disabled.

B: Incorrect: Per the associated ARP a rod block is generated to prevent rod motion when rod blocks may be disabled. A loss of RDS Drive Water DP will not cause the RMCS to become inoperative and is not an input to this annunciator.

C: Correct: Per the associated ARP a rod block is generated to prevent rod motion when rod blocks may be disabled. The annunciator is caused by a:
1. Low Voltage
2. Computer Failure
3. Master Test Pushbutton depressed
4. Loss of permissive signal from (fcn. Of Rx Mode Switch):
   a. RPIS (INOP)
   b. NMS
   c. RSCS
   d. RWM
   e. Scram Discharge Volume
   f. Scram Discharge Volume Bypass
   g. Refuel Platform
   h. Service Platform
5. Clock Failure

D: Incorrect: Per the associated ARP a rod block is generated to prevent rod motion when rod blocks may be disabled. A loss of RDS Drive Water DP will not cause the RMCS to become inoperative and is not an input to this annunciator.

Technical Reference(s): ARP for ROD DRIVE CONTROL SYSTEM INOPERABLE, 603303 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-201002C01, Reactor Manual Control System RBO-5, System Operation, Control, and Instrumentation (As available)

Question Source: Bank #

Modified Bank #

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	239001	K4.06	
	Importance Rating	3.1		

Knowledge of MAIN AND REHEAT STEAM SYSTEM design feature(s) and/or interlocks which provide for the following: Allows for removal or prevents escape of radioactive steam from systems that have leaky MSIV's

Question: RO #30

Twenty minutes after a fuel failure, reactor scram, and MSIV isolation, leakage past the MSIVs is indicated. Conditions are as follows:

- N2-EOP-MSL has been entered
- Main Condenser Vacuum is 0 inches
- Main Steam Line Radiation and Turbine Building Release limits of N2-EOP-MSL have been exceeded

Given these conditions, which one of the following system lineups is available to limit the escape of radioactive steam?

Place the...

A. Main Steam Line Drains in service

B. Mechanical Vacuum Pump in service

C. Bypass Opening Jack momentarily to open

D. Steam Jet Air Ejectors in service utilizing Aux Boiler Steam

Answer: D

Explanation (Optional):

A: Incorrect: This lineup is not authorized by procedure at NMP2. Plausible in that opening a steam line drain would create a flow path to the main condenser. Additionally there is no condenser vacuum.

B: Incorrect: The mechanical vacuum pump cannot be placed in service with the high main steam line radiation.

C: Incorrect: This lineup is not authorized by procedure at NMP2. Plausible in that opening a turbine bypass would create a flow path from the main steam lines to the main condenser. Additionally, there is no condenser vacuum.

D: Correct: N2-EOP-MSL directs that the SJAEs be placed in service per N2-EOP-6. EOP-6, attachment 16 lines up the SJAEs utilizing Aux Steam.

Technical Reference(s): N2-EOP-MSL (Attach if not previously provided)

N2-EOP-6 Att. 16, page 196

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPMSLC01 E)-2, Operational Actions and Sequence (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 K5.04	
	Importance Rating	2.7	

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER CLEANUP SYSTEM: Heat exchanger operation

Question: RO #31

During a reactor plant startup Reactor Water Cleanup (WCS) is being used to aid in RPV level control by rejecting reactor water to the main condenser.

Which one of the following is correct regarding the WCS flow through the Regenerative Heat Exchanger (RHX) and the Non Regenerative Heat Exchanger (NRHX) when 2WCS-FV135, REJECT FLOW CONTROL MANUAL CONTROL, is opened?

A. WCS flow to the NRHX remains the same

B. WCS flow to the NRHX decreases

C. WCS return flow from the filter demineralizers to the RHX increases

D. WCS return flow from the filter demineralizers to the RHX decreases

Answer: D

Explanation (Optional):

A: Incorrect: Flow through the NRHX increases. The reject flow is downstream of the filter demins.

B: Incorrect: Flow through the NRHX increases. The reject flow is downstream of the filter demins.

C: Incorrect: Return flow through the RHX decreases. The reject flow is downstream of the filter demins and before the RHX.

D: Correct: WCS flow branches downstream of the filter demins. One branch goes to the reject line and the other branch is the return flow to the RHX. Therefore, as reject flow goes up, RHX flow goes down.

Technical Reference(s): LP N2101204000C01, Reactor Water Cleanup, Figure 5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP N2101204000C01 Reactor Water Cleanup, RBO-3, Functional Arrangement (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006 K6.05	
	Importance Rating	2.7	

Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) : Steam flow input: P-Spec(Not-BWR6)

Question: RO #32

The reactor is at 40% power with the following:

- One of the four Main Steam Line Flow Transmitter inputs to the Feedwater Level Control System failed downscale
- Feedwater Level Control has been placed in Single Element Control.

Following this failure the RWM will be operating.....

A. Below the Low Power Set Point, alarms and rod blocks due to control rod mispositionings are enforced.

B. Above the Low Power Alarm Point, alarms and rod blocks due to control rod mispositionings are not enforced.

C. In the Transition Zone, alarms are active but rod blocks due to control rod mispositionings are not enforced.

D. In the Transition Zone, alarms are not active and rod blocks due to control rod mispositionings are not enforced.

Answer: C

Explanation (Optional):

A: Incorrect. Rod blocks are not enforced until steam flow is less than LPSP. LPSP is reached when steam flow is < 25%. With power initially at 40%, a single steam flow indicator failing will result in a total steam flow signal lowering to 30%.

B: Incorrect. With power initially at 40%, a single steam flow indicator failing will result in a total steam flow signal lowering to 30%. LPAP is between 25 and 40%.

C: Correct. Transition zone is set when steam flow is between 25% and 40%. With power initially at 40%, a single steam flow indicator failing will result in a total steam flow signal lowering to 30%. At 30% steam flow alarms are active but rod blocks are not enforced.

D: Incorrect: Although in transition zone, alarms are active.

Technical Reference(s): N2-OP-95A, RWM, Page 3 for setpoints of LPSP and LPAP and what happens at these levels. Facility lesson plan N2101201002C01 Reactor Manual Control, pages 44 and 133 for blocks enforced in these areas. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP-2101-201002C01, RMCS RBO-8, Interrelated Systems (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	233000 A1.07	
	Importance Rating	2.7	

Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: System temperature

Question: RO #33

Which one of the following is the effect of a Loss of Coolant Accident (LOCA) coincident with a total Loss of Offsite Power (LOOP) on Fuel Pool temperatures and what actions are required to re-establish temperature control?

	Fuel Pool temperature	To re-establish Fuel Pool Cooling
A.	Lowers	Manually close the Filter/Demin Bypass Valve, 2SFC-HIC113 only.
B.	Rises	Wait 60 seconds then restart the Fuel Pool Cooling Pump only.
C.	Lowers	Manually close the Filter/Demin Bypass Valve, 2SFC-HIC113 and place the Filter Demins back in service.
D.	Rises	Wait 60 seconds then restart the Fuel Pool Cooling Pump and line up Service Water Backup Cooling to the SFC heat exchanger.

Answer: D

Explanation (Optional):

A: Incorrect. Fuel Pool temperature rises because the SFP pumps trip on a LOOP/LOCA and must be manually restarted. Filter/Demin Bypass Valve, 2SFC-HIC113 initially fails closed on a momentary loss of power. The valve will then return to its normal position because this valve is always controlled in manual.

B: Incorrect: To re-establish cooling Service Water Backup Cooling must be line-up.

C: Incorrect: Fuel Pool temperature rises because the SFP pumps trip on a LOOP/LOCA and must be manually restarted. Filter/Demin Bypass Valve, 2SFC-HIC113 initially fails closed on a momentary loss of power. The valve will then return to its normal position because this valve is always controlled in manual. Placing the filter/demins in service will not effect fuel pool cooling temperatures.

D: Correct: On a Loss Of Coolant Accident (LOCA), the running Pump will continue to run but a Pump cannot be started or restarted during the first 60 seconds after receipt of the LOCA signal. If a Loss of Offsite Power (LOOP) and a LOCA occur simultaneously, the running Pumps will trip. A Pump may be manually restarted after 60 seconds using N2-SOP-03. The Reactor Building Closed Loop Cooling Water supply to the SFC Heat Exchanger is also lost on a LOOP/LOCA and Service Water Backup Cooling to the SFC Heat Exchanger must be manually lined up by the operator.

Technical Reference(s): N2-OP-38, Spent Fuel Pool Cooling, page 5
N2-SOP-38, Loss of SFC, page 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-SOP03C01, Loss of AC RBO-3 Operational Actions and Sequence
LP #2101-233000C01, SFC and Cleanup
RBO-2, Functional Arrangement (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202002 A2.08	
	Importance Rating	3.3	

Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: FCV lockup: BWR-5,6

Question: RO #34

The plant is at rated conditions when annunciator 602105, RECIRC FCV A MOTION INHIBIT, alarms due to associated Hydraulic Power Unit failure.

Which one of the following is a concern while troubleshooting and repairing this condition and a procedurally authorized action to address the issue?

	Concern	Action Taken
A.	Recirc loop flow mismatch	Close the Loop A Hydraulic Fluid Outside Isolation valves.
B.	Power level exceeds 3467 MWt	Close the Loop A Hydraulic Fluid Outside Isolation valves
C.	Recirc loop flow mismatch	Manually trip the "A" Recirc pump after a scram if the reactor scrams
D.	Inadequate Recirc Pump NPSH	Manually trip the "A" Recirc pump after a scram if the reactor scrams

Answer: A

Explanation (Optional):

A: Correct: Per precaution # 25 in OP-29, flow forces on the valve cause the FCV to drift close. If the Hydraulic Power Supply is to be shutdown at power at the discretion of the

SM, close the appropriate loop outside hydraulic isolation valve by taking the LOOP A(B) HYDR FLUID OUTSIDE ISOL switch to the CLOSE position.

B: Incorrect: Per precaution # 25 in N2-OP-29, flow forces on the valve cause the FCV to drift close NOT open therefore reactor power would lower.

C: Incorrect: Flow mismatch is not concern after the scram, additionally both pumps will shift to low speed after the scram.

D: Incorrect: A reactor scram will cause the pump to shift to slow speed. Adequate NPSH will still be maintained due to the height of the water inside the vessel. (page 26 of Facility Recirc Lesson Plan) Plausible in that the required NPSH is a function of flow through the pump. Following a scram the reduced feed flow lowers the available NPSH. Incorrect:

Technical Reference(s): N2-OP-29, Precaution #25 and Note on top of page 14 regarding closing the isolation valves. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-202001C01, Reactor Recirculation System RBO-9, Precautions, Limitations, and Operations Fundamentals (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215001 A3.03	
	Importance Rating	2.5	

Ability to monitor automatic operations of the TRAVERSING IN-CORE PROBE including: Valve operation: Not-BWR1

Question: RO #35

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

- A TIP trace is in progress
- The TIP is being operated in the MANUAL MODE
- The TIP is out of its shield chamber and running into the core but has NOT yet reached the CORE TOP LIMIT

With these initial conditions a feed water transient results in RPV level lowering to 100 inches before recovering. Ten minutes later RPV level is 180 inches and steady.

Which one of the following sets of indications would signify that the TIP system had responded as designed to the transient?

(Assume no operator actions.)

A. Amber Squib Monitor light: OFF
Amber Shear Valve Monitor light: OFF
Red Ball Valve Open light: ON
White In-Core light: ON

B. Amber Shear Valve Monitor light: OFF
Red Ball Valve Open light: ON
White In-Core light: OFF
White In-Shield light: ON

C. Amber Shear Valve Monitor light: OFF
Green Ball Valve Closed light: ON

White In-Core light: OFF
White In-Shield light: ON

D. Amber Squib Monitor light: ON
Amber Shear Valve Monitor light: ON
White In-Core light: ON
White In-Shield light: ON

Answer: C

Explanation (Optional):

A: Incorrect: These are indications that the TIP never received an isolation signal. RPV level dropping to 100 inches would actuate the automatic withdraw.

B: Incorrect: Although the TIP withdrew and is in the shield, the ball valve did not close. The automatic withdraw sequence will also close the ball valve.

C: Correct: These are indications that the TIP has withdrawn and is now in the shield. This is an expected response to an isolation signal.

D: Incorrect: These are indications that the shear fired. The shear does not automatically fire.

Technical Reference(s): Facility lesson plan N2101215001C01, TIP, pages 41, and 42 for what the lights mean. Page 46 for description of what happens on an isolation signal. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-215001C01, TIP RBO-5 System Operations, Control, and Instrumentation (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201003 A4.02	
	Importance Rating	3.5	

Ability to manually operate and/or monitor in the control room: CRD mechanism position: Plant-Specific

Question: RO #36

Following a scram from full power all control rods indicate fully inserted with the exception of Control Rod 20-37. Additional Information is as follows:

- Rod 20-37 indicates Blank on the Four Rod Display
- Blue Scram Light for rod 20-37 IS illuminated
- Full-In Light for Rod 20-37 is NOT illuminated
- OD-7 Control Rod Position Printout displays rod 20-37 as XX.
- No other rod position information is available.

Given the above information which one of the following is correct regarding the position of rod 20-37?

A. The rod's position cannot be determined.

B. The rod fully inserted but is beyond full in.

C. The rod is stuck in between notch positions.

D. The rod is at 00 but the reed switch for position 00 is stuck open.

Answer: A

Explanation (Optional):

A: Correct: A double X (XX) indicates that the RPIS is receiving abnormal data. With these indications, the rod could be anywhere from fully withdrawn to beyond full in.

B: Incorrect: The green background is maintained even with the control rods inserted beyond 00 as they would be following a scram.

C: Incorrect: The rods position cannot be determined given these indications. All notch positions show up as even numbers. Odd numbers are indicated by dashes (--).

D: Incorrect: The green full in light would be lit if this was the case.

Technical Reference(s): Scram procedure, N2-SOP-101C, discussion section 5.2 (Attach if not previously provided)
LP 2101-201002C01 page 33

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPC5C01, Failure to Scram, EO-2, Operational Actions and Sequence (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215002 2.1.20	
	Importance Rating	4.6	

Conduct of Operations: Ability to interpret and execute procedure steps. (RBM)

Question: RO #37

In accordance with N2-OP-92, Neutron Monitoring, which one of the following describes:

(1) How the Rod Block Monitor (RBM) operates as a function of APRM input AND

(2) How the "nulling" or renormalization of a RBM channel is performed?

(Assume a peripheral control rod is NOT currently selected)

A. (1) RBM "A" is provided an input signal from APRM 1 and with an alternate signal from APRM 3 ONLY.

(2) ONLY by first selecting any peripheral control rod and then reselecting any other control rod including the originally selected control rod.

B. (1) RBM "B" is provided an input signal from APRM 2 and with an alternate signal from APRM 3 ONLY.

(2) By first selecting another control rod and then reselecting the originally selected or any other control rod.

C. (1) RBM "A" is provided an input signal from APRM 1 and with an alternate signal from APRMs 3 or 4.

(2) ONLY by first selecting any peripheral control rod and then reselecting any other control rod including the originally selected control rod.

D. (1) RBM "B" is provided an input signal from APRM 2 and with an alternate signal from APRM 3 or 4.

(2) By first selecting another control rod and then reselecting the originally selected or any other control rod.

Answer: D

Explanation (Optional):

A: Incorrect – RBM "A" is provided a backup signal from APRMs 3 & 4.

B: Incorrect – RBM "B" is provided a backup signal from APRMs 3 & 4.

C: Incorrect – selection of any other control rod will renormalize the RBM.

D: Correct – IAW N2-OP-92, Page 7 system description - The APRM provides Simulated Thermal Power to the RBM which uses that power level to determine which RBM power range setpoint is enabled. RBM A receives a signal from APRM 1 (APRM 3 and then 4 are alternates). RBM B receives a signal from APRM 2 (APRM 4 and then 3 are alternates).
The RBM is calibrated to a fixed (constant) reference each time a non-peripheral rod is selected. At the time of rod selection the RBM takes input from the surrounding LPRMs at various core heights and averages these readings. A "nulling" operation is performed which establishes the pre-rod motion value. This value is normalized to 100%. Renormalization is allowed by deselecting and reselecting the rod.

Technical Reference(s): N2-OP-92 page 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 2101-215003C01, Power Range Monitoring and RBM (As available)
RBO-1, System Function and Purpose

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

Question History:

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

10 CFR Part 55 Content:	55.41	6
	55.43	

Comments:

Facility:	Nine Mile Point Unit 2
Vendor:	GE
Exam Date:	2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	234000 K4.03	
	Importance Rating	3.4	

Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and/or interlocks which provide for the following: Protection against inadvertently lifting radioactive components out of the water.

Question: RO #38

Which one of the following describes when the Refueling Platform Main Hoist Raise Block is received?

When the ______

(1) Fuel Hoist Interlock occurs
(2) Slack Cable Interlock occurs
(3) Safety Travel Interlock is received and TRAVEL OVERRIDE Switch is in NORM
(4) Normal Up Limit is reached while HOIST OVERRIDE push button is not depressed

A. (1), (2) and (3)

B. (2), (3) and (4)

C. (1), (3) and (4)

D. (1), (2) and (4)

Answer: C

Explanation (Optional):

A: Incorrect: Slack Cable interlock (2) is a Hoist Lower Block

B: Incorrect: Slack Cable interlock (2) is a Hoist Lower Block

C: Correct – IAW N2-OP-39 Section 5.4

D: Incorrect - Slack Cable interlock (2) is a Hoist Lower Block

Technical Reference(s): N2-OP-39 Section 5.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-234000C01, Fuel Handling and Reactor Servicing Equipment RBO-5, System Operation, Control, and Instrumentation (As available)

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

Question History:

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EK1.01	
	Importance Rating	4.1	

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Drywell integrity: Plant-Specific

Question: RO #39

The plant was at full power when a LOCA occurred.

Which one of the following failures / actions could result in exceeding the Primary Containment Pressure Limit (PCPL) and subsequent containment failure?

A. Initiating Drywell sprays with Suppression Pool level above 217 feet.

B. One set of Suppression Chamber to Drywell vacuum breakers failing closed.

C. Initiating Drywell sprays when outside the limits of the Drywell Spray Initiating Limit.

D. A break in a drywell to Suppression Chamber downcomer above the level in the Suppression Pool.

Answer: D

Explanation (Optional):

A: Incorrect – Initiating drywell spray at this level might challenge containment integrity but the failure mechanism would be due to exceeding the negative pressure rating due to the vacuum breakers being covered.

B: Incorrect – Both check valves failing closed would not result in a challenge to the PCPL. Plausible in that both valves failing open would also result in steam bypass to the suppression chamber and possible challenge to the PCPL.

C: Incorrect – Initiating drywell sprays when outside the limits of the Drywell Spray Initiating Limit could result in containment failure but the failure mechanism would be due to exceeding the negative pressure rating.

D: Correct – The PCPL is a high pressure limit based on maintaining containment integrity. The lowest pressure is 45 psig which is also the design pressure of the containment. The UFSAR worst-case steam bypass analysis assumes a total bypass leakage path area of less than 0.1 ft^2 (primarily drywell floor seams, downcomer and SRV piping penetrations, and vacuum breakers). The bypass path area of a single downcomer (assuming a double-ended rupture) would be almost 30 times larger. Although no specific data exists, NMPC Engineering is certain that the 45 psig primary containment design pressure would be exceeded.

Technical Reference(s): LP #2101-223001C01, Primary Containment and Suppression Pool, pages 59, 122, and 135. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-223001C01, Primary Containment and Suppression Pool (As available)
RBO-3, Functional Arrangement
RBO-11, System Loss and Component Malfunction

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

Question History:

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

10 CFR Part 55 Content: 55.41 3
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EK1.03	
	Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Safety/relief valve tailpipe temperature/pressure relationships

Question: RO #40

The plant is operating at rated power when the following occur:

- Safety Relief Valve fails open
- Tailpipe temperature on the failed open SRV is indicating 320°F

Which one of the following is the pressure of the steam discharging into the Suppression Pool from the stuck open Safety Relief Valve?

A. 14 to 16 psia

B. 65 to 75 psia

C. 85 to 95 psia

D. 115 to 125 psia

Answer: B

Explanation (Optional):

A: Incorrect – This is atmospheric pressure. The SRV is discharging to the Suppression Chamber which is close to atmospheric pressure.

B: Correct – An open SRV is a constant enthalpy throttling process. Using the Mollier Diagram and using ~1020 psig (1035 psia) the enthalpy is 1190 BTU/lbm. Following this line straight down until it intersects the 320°F constant temperature line. This intersect point occurs at the 70 psia constant pressure line.

C: Incorrect – This number is based on misunderstanding the process and is the pressure corresponding to moving along the 320°F onstant temperature line, but then choosing pressure at the Saturation Line which is 90 psia

D: Incorrect – This number is based on misunderstanding the process and is the pressure corresponding to dropping down on the constant enthalpy line, but then choosing pressure at the Saturation Line which is 120 psia.

Technical Reference(s): Mollier Diagram (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Mollier Diagram

Learning Objective: LP #2101-239001C01, Main Steam and Reheater System RBO-2, Function and Location of Major Components BWR Generic Fundamentals, Thermodynamics, Chapter 3 – Steam/Mollier Diagram (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

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

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

Question: RO #41

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

- Annunciator 849105 FIRE DETECTED PNL127 SW STAIR/237 (for the Control Building El 237) alarms
- Fire is confirmed
- HVC*ACU1A, CONTROL ROOM AC FAN tripped
- HVC*ACU2A, RELAY ROOM AC FAN tripped

Which one of the following identifies the actions required to be taken for Control Building Ventilation (HVC) and the reason?

A. Defeat cross divisional interlocks to prevent a Control Room evacuation due to smoke infiltration.

B. Actuate Appendix R disconnects to prevent tripping the Division II ACUs due to faulty electrical circuits.

C. Defeat cross divisional interlocks to ensure Control Room Envelope temperature can be maintained 90°F or less.

D. Actuate Appendix R disconnects to place HVC in a lineup that ensures the Control Room Envelope pressure does not become negative.

Answer: C

Explanation (Optional):

A: Incorrect - N2-OP-53A H.15.0 Note 2 states this is required to maintain Control Room Envelope temperature below 90°F not to maintain positive pressure. This action is not taken prevent a Control Room evacuation to due smoke infiltration.

B: Incorrect - The Appendix R switches do not prevent ACU tripping. This function is performed by the cross divisional interlock override switch.

C: Correct - Per ARP 849105, N2-OP53A off normal section H.14.0 is required to be performed immediately, to defeat the HVC cross divisional interlocks. Implementation of this section directs the starting of Div II ACUs and using the cross divisional interlock key lock override switch to prevent loss of Div II components because of fire affecting Div I components. Note 2 states this is required to maintain Control Room Envelope temperature below 90°F.

D: Incorrect - Actuating Appendix R disconnects will not place the Control Room and Relay Room HVC ACUs in the correct lineup. ACUs do not realign when Appendix R switches are actuated.

Technical Reference(s): ARP 849105 corrective action b, N2-OP-53A section H.15.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-288003C01, Control Building Ventilation RBO-5, System Operation, Control, and Instrumentation (As available)

Question Source: Bank # 22851

Modified Bank # (Note changes or attach parent)

New

Question History: Question #20 on 2005 NRC exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	295019 AK2.08	
	Importance Rating	2.8	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Plant Ventilation

Question: RO #42

The plant is operating at rated conditions.

Which one of the following describes the effect of a loss of instrument air to the following Reactor Building isolation dampers and the response of the Reactor Building ventilation system.

- 2HVR*AOD1A/B
- AOD9A/B
- AOD10A/B

	Dampers fail	Reactor Building ventilation
A.	Open	No other automatic actions occur
B.	Closed	No other automatic actions occur
C.	Open	Reactor Building supply and exhaust fans trip
D.	Closed	Reactor Building supply and exhaust fans trip

Answer: D

Explanation (Optional):

A: Incorrect – the dampers fail closed tripping all supply AND exhaust fans.

B: Incorrect – the dampers fail closed tripping all supply AND exhaust fans.

C: Incorrect – the dampers fail closed.

D: Correct - A loss of Instrument Air to the Reactor Building will cause the Reactor Building isolation dampers 2HVR*AOD1A/B, AOD9A/B AND AOD10A/B to close, tripping all supply AND exhaust fans. The resulting low above/below Refuel Floor air flow will auto start the lead emergency recirculation unit cooler, 2HVR*UC413B.

Technical Reference(s): N2-OP-52 Rev.8 page 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-288001C01, Reactor Building Ventilation
RBO-8, Interrelated Systems (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EK2.09	
	Importance Rating	4.0	

Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: Reactor water level

Question: RO #43

Emergency Procedure N2-EOP-C5, Failure To Scram, specifies under certain conditions to terminate and prevent injection and lower RPV level to below 100 inches before restoring injection.

Which one of the following is the EOP bases for performing this action?

A. Increases the concentration of boron being injected into the core.

B. Allows preheating the feedwater to lower the possibility of thermal hydraulic instabilities.

C. Reduces the amount of natural circulation in the core following the tripping of the Recirc Pumps.

D. Increases the amount of inlet subcooling to further reduce power via the temperature coefficient.

Answer: B

Explanation (Optional):

A: Incorrect - These steps were added to C5 to address thermal instabilities. Plausible in that lowering level will reduce the volume of water inside the shroud.

B: Correct: Per EOP bases, 100 inches is 24 inches below the feedwater spargers. When injection is commenced, feedwater will fall though a steam space and be preheated before entering the core region. This reduction in subcooling reduces the probability of thermal hydraulic oscillations since the step Q5 of C5 tripped the Recirc pumps.

C: Incorrect – These steps were added to C5 to address thermal instabilities. Plausible in that lowering level will reduce natural circulation some amount. The steps to lower level down to as low as -14 inches are specifically designed to achieve this result.

D: Incorrect: By lowering Reactor water level inlet subcooling is lowered (decreased) not increased.

Technical Reference(s): NMP2 EOP bases document, pages 12-18 and 19. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPC5C01, Failure to Scram, EO-2, Operational Actions and Sequence (As available)

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

Question History:

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

10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EK2.03	
	Importance Rating	3.6	

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Reactor water level indication

Question: RO #44

While executing the EOPs the following conditions exist:

- The Narrow Range RPV Level indicators are steady +152 inches.
- The hottest reactor building temperature is 170 degrees.
- All drywell temperature instruments are now pegged high at > 350 degrees
- RPV pressure is stable at 1000 psig.

Utilizing Table C from N2-EOP-RPV, which one of the following is correct regarding the Narrow Range RPV level indication?

C	Minimum Indicated Levels	
Instrument	Hottest Reactor Building or Drywell Area Temperature	
	≤ 350°F	> 350°F
Shutdown Range	195 in.	250 in.
Upset Range	190 in.	260 in.
Narrow Range	150 in.	155 in.
Wide Range	25 in.	25 in.
Fuel Zone	-150 in.	-160 in.

A. CANNOT be used. Flashing of the reference legs may be occurring.

B. CANNOT be used. Actual RPV Level may be below the variable leg tap.

C. CAN be used for trending purposes. Indicated level is lower than actual.

D. CAN be used for trending purposes. Indicated level is higher than actual.

Answer: B

Explanation (Optional):

A: Incorrect: Plausible in that drywell temperature has just gone off scale high. However flashing does not occur until saturation temperature is reached (~550 degrees at this pressure). Additionally even, if the temperature did reach 550 degrees, per Caution A, they can still be used provided that they are NO indications of flashing.

B: Correct: The minimum useable level for the Narrow Range is 150 inches when temperatures are less than 350 degrees and 155 inches when temperature is > 350 degrees in either the drywell or the reactor building. Per EOP Tech Bases document and EOP-RPV control, Caution A, the Narrow Range level indicators cannot be used when level is below the Minimum Indicating level.

C: Incorrect: Level is below the minimum indicated level and must be assumed to be below the variable leg tap and therefore cannot be used.

D: Incorrect: Level is below the minimum indicated level and must be assumed to be below the variable leg tap and therefore cannot be used.

Technical Reference(s): NMP2 EOP bases document, page 14-6. N2-EOP-RPV, Detail A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP # 2101-216000C01,Reactor Vessel Instrumentation RBO-12, EOP Implementation (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

	New	X	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 AK3.01	
	Importance Rating	3.3	

Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Raising reactor water level

Question: RO #45

During a loss of shutdown cooling direction is given in N2-SOP-31, Loss of Shutdown Cooling, to raise level if no RHR pump or Recirculation Pump can be started.

Which one of the following identifies the reason for raising RPV water level to 227 to 243 inches?

To flood the...

A. dryer assembly to promote natural circulation.

B. steam separators to promote natural circulation.

C. dryer assembly to provide long term decay heat removal.

D. steam separators to provide long term decay heat removal.

Answer: B

Explanation (Optional):

A: Incorrect - the dryer assembly is higher than the steam separator (above 243 inches)

B: Correct - the steam separators are flooded to connect flow from inside to outside the shroud to aid in natural circulation

C: Incorrect – long term heat removal is not the reason and the dryer assembly is higher than the steam separator (above 243 inches)

D: Incorrect - long term heat removal is not the reason

Technical Reference(s): N2-SOP-31, Loss of SDC (Attach if not previously provided)
N2-OP-31, RHR, Section D.37
Reactor vessel/internals figure 4
(LP #2101-101001C01)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-SOP31C01, Loss of SDC (As available)
EO-2, Operational Actions and Sequence

Question Source: Bank # 51878
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 2
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

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

Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: Ventilation isolation

Question: RO #46

Refueling is in progress when an irradiated fuel bundle is dropped in the Spent Fuel Pool causing a refuel floor high radiation alarm and a reactor building isolation.

Which one of the following describes a reason for the reactor building ventilation system response?

A. To reduce refuel floor radiation levels as quickly as possible.

B. To ensure the greatest amount of air dilution prior to discharge.

C. To prevent the spread of contamination to other parts of the Reactor Building.

D. To ensure that air discharged from the refuel floor goes through a filtration system.

Answer: D

Explanation (Optional):

A: Incorrect – the ventilation response will not assist and in reducing rad levels on the refuel floor.

B: Incorrect – Air dilution is not the design criteria for standby gas treatment.

C: Incorrect: The isolation stops the unfiltered release and directs the ventilation through SBGT filtration, it will not stop the spread of contamination. through filter trains and discharging the processed air to the main stack.

D: Correct –In the event of an accident condition, the Standby Gas Treatment System (GTS) prevents leakage of radioactive gases and particulates to the environment by maintaining a negative pressure in the Reactor Building by exhausting air (4000 cfm)

Technical Reference(s): LP #2101-261000C01, Standby Gas Treatment System, Page 65
N2-SOP-39, Refuel Floor Events, Page 2
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-261000C01, Standby Gas Treatment System
RBO-5, System Operation, Control, and Instrumentation
(As available)

Question Source: Bank # WTS #2262
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 2
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EK3.03	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: RCIC operation: Plant-Specific

Question: RO #47

Procedure N2-EOP-PC, Primary Containment Control, contains the following caution:

CAUTION: Operating ECCS or RCIC with suppression pool water level below El. 195 ft may cause system damage.

Which one of the following describes the hazard of NOT complying with this caution?

A. RCIC exhaust line pressure oscillations could potentially cause system damage.

B. NPSH and vortex limits may be exceeded if RCIC is aligned to its alternate suction.

C. Reduced cooling water flow to the RCIC turbine bearings may cause system damage.

D. RCIC steam will discharge into the Suppression Chamber air space and overpressurize containment.

Answer: B

Explanation (Optional):

A: Incorrect – this is a concern at lower RCIC RPM.

B: Correct – IAW the approved RCIC training material; EOPs caution that operating RCIC with suppression pool water level below El. 195 ft may cause system damage. NPSH and vortex limits for these systems should be observed, if possible, but may be exceeded if necessary to maintain adequate core cooling. EOP-6 Attachment 29 is used when suppression pool level is below elevation 195 feet to determine the actual limits. If necessary, EOP-6 Attachment 29 refers to use of EOP-6 Attachments 3 and 4 for throttling ECCS and RCIC flows to control within applicable limits.

C: Incorrect – SP level does not affect cooling water flow to the RCIC turbine bearings.

D: Incorrect –The exhaust flow rate of RCIC is no greater than the steam generated by decay heat. The basis for determining the PCPL takes into account containment venting capable of removing decay heat.

Technical Reference(s): N2-EOP-PC, Primary Containment Control (Attach if not previously provided)
LP #2101-EOPPCC01, Primary Containment Control, Page 36
LP #2101-217000C01, RCIC, Page 179-180

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPPCC01, Primary Containment Control (As available)
EO-2, Operational Actions and Sequence
LP #2101-217000C01, RCIC
RBO-11, System Loss and Component Level Malfunction

Question Source: Bank # WTS # 2834
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AA1.01	
	Importance Rating	3.6	

Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Grid frequency and voltage.

Question: RO #48

Following a grid disturbance the following conditions exist:

- Main Generator load is 800 megawatts electric (MWe)
- Main Generator reactive loading is 200 MVARs to the Generator
- Main Generator voltage regulation is in AUTO

System Power Control now requests that the Main Generator MVAR loading be changed to 100 MVARs lagging.

This will be accomplished by going to:

A. RAISE on the AC Voltage Regulator Control Switch

B. LOWER on the AC Voltage Regulator Control Switch

C. RAISE on the DC Voltage Regulator Control Switch

D. LOWER on the DC Voltage Regulator Control Switch

Answer: A

Explanation (Optional):

A: Correct: Voltage regulation is in AUTO which places the AC regulator in control. Initial MVAR loading is in the leading direction. To establish 100 MVARs in the lagging direction must go to raise which will increase excitation.

B: Incorrect: Going to Lower will increase VAR loading in the leading direction. Plausible in that the absolute value of the target MVAR loading is less than the initial MVAR loading.

C: Incorrect: When in AUTO, the AC regulator is in control. Plausible in that if voltage regulation was in manual, this would be the correct action.

D: Incorrect: When in AUTO, the AC regulator is in control. Plausible in that the absolute value of the target MVAR loading is less than the initial MVAR loading.

Technical Reference(s): N2-OP-68, section E.4.18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-245C01, Main Generator System (As available)
RBO-6, Major Operating Parameters

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

Question History:

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EA1.02	
	Importance Rating	3.6	

Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool spray: Plant-Specific

Question: RO #49

Following a plant transient, the following conditions exist:

- Drywell Pressure is 3.8 psig
- Drywell Temperature is 200°F
- Suppression Chamber Pressure is 2.6 psig
- Suppression Pool Temperature is 92°F
- Suppression Pool Level is 200 feet

Which one of the following actions is required?

Initiate...

A. Suppression Pool Cooling ONLY

B. Suppression Chamber Sprays AND Drywell Sprays ONLY

C. Suppression Chamber Sprays AND Suppression Pool Cooling ONLY.

D. Suppression Chamber Sprays AND Suppression Pool Cooling AND Drywell Sprays.

Answer: C

Explanation (Optional):

A: Incorrect – SP spray is also required

B: Incorrect – DW spray is not permitted per curve

C: Correct – above 90 degrees SP temperature and below 10 psig SP Pressure – both sprays and cooling are put in service per EOP-PC step PCP-2 and SPT-3. Drywell sprays are not initiated based on the DWSIL curve.

D: Incorrect - DW spray is not permitted per curve

Technical Reference(s): N2-EOP-PC (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPPCC01, Primary Containment Control
EO-2. Operational Actions and Sequence (As available)

Question Source: Bank # NRC 2008 Q #51
Modified Bank # (Note changes or attach parent)
New

Question History: NRC 2008 Q #51

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA1.01	
	Importance Rating	3.7	

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

Question: RO #50

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

- Division I Diesel Generator 2EGS*EG1 is operating in parallel with offsite power for monthly surveillance testing
- 2EGS*EG1 is supplying 3960 - 4400 KW to bus

Then Offsite breaker R-50 trips open

Which one of the following describes the effect on 2EGS*EG1 and the Electrical Distribution circuit breakers?

DIV 1 Diesel 2EGS*EG1...

A. continues to run with its output breaker 101-1 closed. Offsite feeder breaker 101-13 is tripped open.

B. continues to run with its output breaker 101-1 and Offsite feeder breaker 101-13 tripped open.

C. trips on overspeed and its output breaker 101-1 is open. Offsite feeder breaker 101-13 is closed.

D. trips on overspeed and its output breaker 101-1 and Offsite feeder breaker 101-13 are tripped open.

Answer: A

Explanation (Optional):

A: Correct - R-50 trip results in a loss of offsite Line 5. Loss of offsite power causes offsite Feeder 101-13 to trip open. Diesel Generator 2EGS*EG1 continues to run with is output breaker 101-1 closed supplying the emergency switchgear.

B: Incorrect - Diesel Generator 2EGS*EG1 continues to run with is output breaker 101-1 closed supplying the emergency switchgear.

C: Incorrect - Diesel Generator 2EGS*EG1 will not trip on overspeed even on a loss of full load.

D: Incorrect - Diesel Generator 2EGS*EG1 will not trip on overspeed even on a loss of full load.

Technical Reference(s): LP #2101-264000C01, Standby Diesel Generator and Auxiliaries (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-264000C01, Standby Diesel Generator and Auxiliaries (As available)
RBO-8, Interrelated Systems

Question Source: Bank #2005 NRC Exam Q #3 22834
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 AA2.03	
	Importance Rating	3.3	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Actual core flow

Question: RO #51

The plant is in Single Loop operation, following an unplanned Recirc Pump trip. The following conditions exist:

- Indicated drive flow in the operating Recirculation loop is 21,000 gpm
- Indicated total core flow on Control Room Panel P603 is 22 Mlbm/hr

Which one of the following identifies the current relationship between indicated total core flow on P603 flow recorder and actual total Core Flow and the reason for the difference?

Indicated total core flow is…

A. higher than actual due to idle loop flow being added to the total core flow summing network.

B. higher than actual due to idle jet pump flow being added to the total core flow summing network.

C. lower than actual due to idle loop flow being subtracted from the total core flow summing network.

D. lower than actual due to idle jet pump reverse flow being subtracted from the total core flow summing network.

Answer: C

Explanation (Optional):

A: Incorrect – at less than 22 kgpm when one recirculation pump is not running, the forward/reverse flow logic network automatically subtracts the measured flow in the idle loop from the measured flow in the active loop.

B: Incorrect – at less than 22 kgpm when one recirculation pump is not running, the forward/reverse flow logic network automatically subtracts the measured flow in the idle loop from the measured flow in the active loop.

C: Correct - NOTE 2 following Step H.6.0 of N2-OP-29, Reactor Recirculation System, states "When calculating total core flow in single loop operation and the operating loop drive flow is less than 22,000 gpm, Loop Flows should be added instead of subtracted". In addition, Step 6.2 & 6.3 of N2-RESP-07, When one recirculation pump is not running, the forward/reverse flow logic network automatically subtracts the measured flow in the idle loop from the measured flow in the active loop and displays the true core flow on the P603 chart recorder (B22-R613) and computer point NSSFA101 (total core flow), and NSSFA01S (total core flow - smooth).
If in single loop operation with less than 22,000 GPM flow, flow in the shutdown loop will be positive instead of negative as assumed by the summing network (Step 6.2). Therefore in this situation, a calculated total core flow must be substituted into computer point NSSFA101 and NSSFA01S prior to demanding a core monitoring case.

D: Incorrect - at less than 22 kgpm when one recirculation pump is not running, the forward/reverse flow logic network automatically subtracts the measured flow in the idle loop from the measured flow in the active loop (not jet pump reverse flow

Technical Reference(s): N2-RESP-07, section 6.2 and 6.3 (Attach if not previously provided)
N2-OP-29, Note 2 at step H.6.0

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-101001C01, Reactor Vessel (As available) and Internals
RBO-11, System Loss and Component Malfunction

Question Source: Bank # 22833
Modified Bank # (Note changes or attach parent)
New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	2
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018 AA2.04	
	Importance Rating	2.9	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: System flow

Question: RO #52

A small LOCA has occurred with the following conditions:

- Drywell pressure is 2.1 psig
- Drywell temperature is 170°F
- EDGs have started and are running unloaded
- EOP-RPV and EOP-PC have been entered

Which one of the following describes the status of CCP flow to the drywell unit coolers?

CCP is ...

A. NOT isolated and will only isolate if off site power is also lost.

B. NOT isolated and will only isolate if RPV level lowers below 108.8 inches.

C. isolated, but can be procedurally restored using LOCA OVERRIDE switches.

D. isolated and cannot be procedurally restored until drywell pressure lowers below 1.68 psig.

Answer: C

Explanation (Optional):

A: Incorrect – A CCP isolation to the drywell unit coolers has occurred on high drywell pressure.

B: Incorrect - A CCP isolation to the drywell unit coolers has occurred on high drywell pressure.

C: Correct – A CCP isolation to the drywell unit coolers occurs on a group 8 isolation signal (1.68 psig Drywell pressure or 108.8 RPV level). EOP-6, ATTACHMENT 24 allows restoration of drywell cooling as long as drywell temperature is below 250°F.

D: Incorrect – CCP to the drywell unit coolers can be restored using the LOCA override switches in accordance with EOP-6, Attachment 24.

Technical Reference(s): EOP-6 Att.24, Drywell Unit Cooler Operation with LOCA Signal (Attach if not previously provided)
LP 2101-208000C01 page 36

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-208000C01, Reactor Building Closed Loop Cooling RBO-5, System Operation, Control and Instrumentation (As available)

Question Source: Bank # 32864

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

Facility:	Nine Mile Point Unit 2
Vendor:	GE
Exam Date:	2010
Exam Type:	R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295005	AA2.03	
	Importance Rating	3.1		

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

Question: RO #53

The plant was operating at 22% power when grid instabilities caused a Main Generator Lockout Relay Trip.

Which one of the following describes the plant status?

A. The Turbine Stop Valves, Control Valves, and Combined Intermediate Valves are closed.

B. The Turbine Stop Valves and Control Valves are closed. The Combined Intermediate Valves are open.

C. The Turbine Stop Valves are closed. The Control Valves and Combined Intermediate Valves are open.

D. The Turbine Stop Valves and Combined Intermediate Valves are closed. The Control Valves are open.

Answer: A

Explanation (Optional):

A: Correct – All valves close on a turbine trip

B: Incorrect – The Combined Intermediate Valves also close on a turbine trip.

C: Incorrect - The Control valves and Combined Intermediate Valves also close on a turbine trip.

D: Incorrect – The control valves also close on a turbine trip

Technical Reference(s): ARP 851109, Turbine Trip (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-245000C01, Main Turbine and Auxiliaries (As available)
RBO-5, System Operation, Control, and Instrumentation

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

Question History:

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295004	AK2.03	
	Importance Rating	3.3		

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: DC Bus loads

Question: RO #54

The plant is operating at rated conditions, when a loss of 125VDC Bus 2BYS-SWG001C occurs.

Which one of the following identifies the Uninterruptible Power Supplies (UPS's) that will lose DC power?

A. 2A and 2B

B. 1D and 3B

C. 1A, 1C and 1G

D. 1B, 1G and 3A

Answer: D

Explanation (Optional):

A: Incorrect- DC power from *SWG002A and B.

B: Incorrect- DC power from SWG001B

C: Incorrect- DC power from SWG001A except UPS 1G

D: Correct - UPS 1B, 1G AND 3A receive backup DC power from 2BYS-SWG001C

Technical Reference(s): N2-OP-71D, Pages 14, 15, 24 and 36 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-262002C01, Uninterruptible Power Supply Systems
RBO-4, System and Component Power Supplies (As available)

Question Source: Bank # 32702
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 6
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295038	2.2.25	
	Importance Rating	3.2		

Equipment Control: Knowledge of bases in Technical Specifications for limiting conditions for operations and safety limits. (High Off-site release)
Question: RO #55

Which one of the following is the bases for Technical Specifications Iodine 131 limit?

Limits the maximum amount of ...

A. TEDE dose received by an individual at the exclusion area boundary

B. CEDE dose received by an individual at the exclusion area boundary

C. TEDE dose received by an individual at the protected area boundary

D. CEDE dose received by an individual at the protected area boundary

Answer: A

Explanation (Optional):

A: Correct – Per TS 3.4.8 bases - Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within 10% of the limits of 10 CFR 50.67
Per 10 CFR 50.67 - An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[2] total effective dose equivalent (TEDE). 10% of this limit would therefore be 2.5 Rem.

B: Incorrect – The limit is based on TEDE (not CEDE) per 10 CFR 50.67 which is the TS bases description

C: Incorrect – The protected area boundary is different than the exclusion area boundary and is not what is specified in 10 CFR 50.67

D: Incorrect – The limit is based on TEDE not CEDE per TS bases 3.4.8. The protected area boundary is different than the exclusion area boundary and is not what is specified in 10 CFR 50.67

Technical Reference(s): 10 CFR 50.67 b(2)(i) and TS bases 3.4.8 background (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPRRC01, Radioactivity Release Control
EO-3, Application of the Emergency Plan (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295006	2.4.50	
	Importance Rating	4.2		

Emergency Procedures / Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (Scram)

Question: RO #56

The plant was operating at rated conditions with APRM 1 inoperable and bypassed, when the "A" Reactor Recirc pump trips. Following the trip, the following conditions exist:

- Core flow 40%
- Reactor power 60%
- Annunciator 603218, OPRM TRIP ENABLED, alarmed
- OPRM Trip Enabled status lights are illuminated for OPRM 2 and 3
- The remaining OPRM status lights are extinguished.
- CRS has determined that OPRM #4 is INOP.

When the APRM Numac Displays on P608 are checked the message "OPRM TRIP ENABLE" is displayed on OPRM 2 and 3 only. NO other OPRM messages are displayed.

Which one of the following is required by station procedures?

A. Insert the first 4 CRAM rods

B. Place the reactor mode switch to Shutdown

C. Raise core flow with the "B" Recirculation Pump to exit the Exit Region

D. Shift APRM recorders to fast speed AND monitor for core oscillations

Answer: B

Explanation (Optional):

A: Incorrect: Only 2 OPRMs are available. OPRM 1 is not available due to APRM 1 being bypassed. OPRM 4 failed to enable when power and flow lowered. SOP-29 directs that if Core flow AND power are within the OPRM Dependent Stability Region and less than 3 OPRMs are available then the reactor should be scrammed. Per the single loop power flow map, core conditions are within the OPRM Dependent Stability Region. Plausible in that this would be the action if at least 3 OPRMs were available.

B: Correct: Per ARP 603218, OPRMs should have enabled. Only 2 of the three available did so. Per SOP-29, if operating in the OPRM Dependent Stability Region and less than 3 are available, the reactor is to be scrammed.

C: Incorrect: The reactor should be scrammed. Plausible in that if the 2 loop power to flow map is used, the reactor is operating in the exit region.

D: Incorrect: The reactor should be scrammed. Plausible in that if in the Heightened awareness region and < 3 OPRMs were available this would be a required action.

Technical Reference(s): ARP 603218 (Attach if not previously provided)

N2-SOP-29, Sudden Reduction In Core Flow
Single Loop Power to Flow Map

Proposed References to be provided to applicants during examination: Single and Two Loop Power to Flow Maps

Learning Objective: LP 2101-SOP29C01, N2-SOP-29, SUDDEN REDUCTION IN CORE FLOW, LO 3, Operational Actions and Sequence. (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295016	2.4.9	
	Importance Rating	3.8		

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (Control room evacuation)

Question: RO #57

The following plant conditions exist:

- A fire required entry into N2-SOP-78, Control Room Evacuation
- All Offsite power is lost
- The Reactor is shutdown
- Reactor pressure is 200 psig
- 2EGS*EG1 and 2EGS*EG3 started and powered the emergency buses
- Plant is being controlled from the Remote Shutdown Panel

Which one of the following strategies is to be used, per N2-OP-78, Remote Shutdown System and N2-SOP-78, Control Room Evacuation?

A. Open four (4) ADS Valves then place Alternate Shutdown Cooling in service.

B. Open four (4) ADS Valves then place Normal Shutdown Cooling in service.

C. Continue operating RCIC then place Alternate Shutdown Cooling in service.

D. Continue operating RCIC then place Normal Shutdown Cooling in service.

Answer: C

Explanation (Optional):

A: Incorrect: With RCIC still available, Pseudo LPCI injection is NOT required, so opening 4 ADS valves to reduce pressure, to allow injection is not performed, per N2-SOP-78.

B: Incorrect: With RCIC still available, Pseudo LPCI injection is NOT required, so opening 4 ADS valves to reduce pressure, to allow injection is not performed, per N2-SOP-78.

C: Correct: Per N2-OP-78, Remote Shutdown System, Alternate Shutdown Cooling is used if offsite power is lost, because Normal Shutdown Cooling cannot be placed in service. RCIC is still available to reduce pressure and provide makeup to the RPV.

D: Incorrect: Normal Shutdown Cooling cannot be placed in service with the loss of offsite power. Power is not available to the RCS loop isolations valves (2RCS*MOV10 and MOV18).

Technical Reference(s): N2-SOP-78, Control Room Evacuation flowchart; N2-OP-78, Remote Shutdown System, page 45 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-296000C01, Remote Shutdown System, RBO-7, Design and Operational Considerations (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295031	EK1.01	
	Importance Rating	4.6		

Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Adequate core cooling

Question: RO #58

The following conditions exist:

- A small break LOCA is in progress.
- Appropriate EOPs have been entered.
- No injection sources are available.
- RPV level is approaching the TAF.

Which one of the following describes why RPV Blowdown is delayed until water level has lowered to the Minimum Zero-Injection RPV Water Level?

A. RPV Blowdown before this level reduces the time the core remains adequately cooled during these conditions.

B. Steam Cooling will NOT maintain fuel cladding temperature below 1500°F if blowdown is performed with water level above TAF.

C. RPV Blowdown before this level produces insufficient steam mass removal rate for adequate core cooling.

D. Steam cooling will NOT maintain fuel cladding temperature below 1800°F if blowdown is performed with water level above TAF.

Answer: A

Explanation (Optional):

A: Correct - Per EOP-C3, (Page 10-14 of EOP Bases Document), it states "opening the SRVs before RPV water level reaches the MZIRWL would reduce the time over which the core remains adequately cooled with no injection".

B: Incorrect – The temperature limit in steam cooling is 1800 degrees. This value is not reached until level has lowered to -58". If the blow down is commenced before TAF, temperature will be well below 1800 degrees. However the time available before adequate core cooling is lost will be reduced since the blowdown was performed before the temperature limit was reached.

C: Incorrect – The concern is time permitted for adequate cooling with no injection and not steam mass removal rate generated heat.

D: Incorrect – -58" is the level at which steam cooling would be unable to preclude clad temperature from reaching 1800 degrees and is the Minimum Zero-Injection RPV Level. It is not the reason you delay a blowdown. TAF is -14 inches.

Technical Reference(s): EOP-C3 bases Rev 6 page 10-14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPC3C01, Steam Cooling EO-1, EOP Flowchart, Function, and Purpose (As available)

Question Source: Bank # WTS 4955

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022 AK1.01	
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS: Reactor pressure vs. rod insertion capability

Question: RO #59

Given the following:

- The reactor is critical with a heatup in progress.
- Reactor pressure is 550 psig.
- "A" CRD pump is out of service for maintenance.
- "B" CRD pump tripped and cannot be restarted.
- 5 different CRD accumulator trouble alarms were received due to low accumulator pressure. The associated control rods are fully withdrawn.
  - 18-27 @ 920 psig
  - 26-51 @ 1020 psig
  - 38-43 @ 1000 psig
  - 46-27 @ 960 psig
  - 54-23 @ 900 psig

Which one of the following describes the current manual control rod insertion capability using the Reactor Manual Control System (RMCS) and the Technical Specification status of the control rod scram function?

	Manual insertion capability	Scram function
A.	NOT available	Not Degraded
B.	Available	Degraded
C.	NOT Available	Degraded
D.	Available	Not Degraded

Answer: C

Explanation (Optional):

A: Incorrect - The first half is correct since RMCS is not available; however scram function is not fully assured. See explanation for answer C.

B: Incorrect - The first half is incorrect since RMCS is not available without drive pressure. The second half is correct since scram function may be degraded. See explanation for answer C.

C: Correct - Per TS Bases 3.1.4 Background – "If the reactor pressure is low, such as during startup, the accumulator will fully insert the control rod within the required time without assistance from reactor pressure."At least 2 withdrawn control rods have low accumulator pressures rendering them inoperable (<940 psig) Rods may or may not fully insert or they may not insert fast enough under the given conditions, so the scram function may be degraded.

With both CRD pumps off, there is no drive pressure, so manual insertion using RMCS is not possible. Furthermore, CRD charging water pressure is also lost. In addition, reactor pressure is less than 900 psig, so reactor pressure alone may not complete the scram.

D: Incorrect - Both parts of this answer are wrong. See the explanation for answer C.

Technical Reference(s): TS bases 3.1.4 background
N2-SOP-30, Control Rod Drive Failures, section 5.3 and 5.4
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-201000C01, Control Rod Drive System
RBO-2, Function and Location of Major Components
RBO-11, System Loss and Component Level Malfunction
(As available)

Question Source: Bank # WTS #1582

Modified Bank # (Note changes or attach parent)

New

Question History:

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

10 CFR Part 55 Content:	55.41	2
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295009	AK2.03
	Importance Rating	3.1	

Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following:
Recirculation system

Question: RO #60

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

At Time = 0 (T=0)

- A failure to scram occurred
- Reactor power is at 50%.
- Reactor Pressure has remained at 1000 psig during the event
- Main Turbine is still on line
- Reactor Water level is 106 inches and slowly lowering

Which of the following identifies the status of the Recirc Pumps at T +15 seconds?

A. NOT running

B. Running at Low Speed but will trip in 10 seconds

C. Running at High Speed but will shift to Low Speed in 10 seconds

D. Running at Low Speed and will remain running until tripped manually

Answer: A

Explanation (Optional):

A: Correct – the recirc pumps trip at RPV level 2 (108.8")

B: Incorrect – the 25 second time delay is related to high reactor pressure and APRMs ≥ Downscale

C: Incorrect - the 25 second time delay is related to high reactor pressure and APRMs ≥ Downscale

D: Incorrect – Manual action is not required, the recirc pumps trips at RPV level 2 (108.8")

Technical Reference(s): N2-OP-29, Reactor Recirculation System Section D.6.d (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-202001C01, Reactor Recirculation System RBO-5, System Operation, Control, and Instrumentation (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295020	AK3.02
	Importance Rating	3.3	

Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Drywell/containment pressure response

Question: RO #61

The plant was operating at rated conditions when I&C testing caused an RPV Level 2 Group 8 signal to be generated.

Which one of the following describes the response of the Drywell Unit Coolers and the effect, if any, on Drywell pressure?

	Drywell Unit Cooler Fans	Drywell pressure
A.	Trip	Rises
B.	Trip	Remains the same
C.	Operating	Rises
D.	Operating	Remains the same

Answer: A

Explanation (Optional):

A: Correct - The Level 2 signal will cause a group 8 isolation which isolates CCP flow to the drywell and trips the drywell fans. Drywell pressure will rise.

B: Incorrect – The fans will trip, CCP flow will isolate on a group 8 isolation signal and Drywell pressure will rise.

C: Incorrect – The fans will trip on a group 8 isolation signal.

D: Incorrect – The fans will trip and CCP flow will isolate.

Technical Reference(s): N2-SOP-60, Loss of Drywell Cooling, Page 3 (Attach if not previously provided)
N2-OP-13, RBCLC, Page 5

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-223004C01, Drywell Cooling (As available)
RBO-11, System Loss and Component Level Malfunction

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

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

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

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

Question: RO #62

A LOCA has occurred with the following plant conditions:

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

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

A. Continue to inject with the pump

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

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

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

Answer: A

Explanation (Optional):

A: Correct – N2-EOP-SC specifies that a system is not to be isolated if needed for EOP actions. LPCS is maintaining adequate core cooling.

B: Incorrect – N2-EOP-SC specifies that a system is not to be isolated if needed for EOP actions. LPCS is maintaining adequate core cooling. Plausible in that this alarm constitutes the Max Safe Value.

C: Incorrect – N2-EOP-SC specifies that a system is not to be isolated if needed for EOP actions. LPCS is maintaining adequate core cooling. Plausible in that the system would be isolated if it wasn't needed for level control.

D: Incorrect – Plausible in that exceeding two max safes is a EOP-SC decision point. SC-4 does not let you isolate systems that are needed for EOP actions.

Technical Reference(s): N2-EOP-SC, Secondary Containment, SC-4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPSCC01, Secondary Containment Control EO-2, Operational Actions and Sequence (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295007 AA2.02	
	Importance Rating	4.1	

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor power

Question: RO #63

Given the following:

- Reactor is operating at 70%.
- MSIV B outboard isolation valve slowly drifts closed.

Which one of the following is correct regarding reactor pressure and power as the MSIV is closing and after it is fully closed?

	Reactor pressure...	Reactor power...
A.	rises and then returns to normal	rises and stabilizes at a higher power
B.	rises and stabilizes at a higher pressure	rises and then returns to normal
C.	rises and then returns to normal	rises and then returns to normal
D.	rises and stabilizes at a higher pressure	rises and stabilizes at a higher power

Answer: D

Explanation (Optional):

A: Incorrect: Reactor pressure would not return to normal, both pressure and power would rise and remain higher.

B: Incorrect: Reactor power would not rise then return to normal, both pressure and power would rise and remain higher.

C: Incorrect: Reactor pressure and power would not rise then return to normal, both pressure and power would rise and remain higher.

D: Correct: MSIVs isolate at 140% of flow, three open MSIVs at 70% power (flow) would be well below their high flow setpoint (~93% flow) so no further isolations would occur and with no isolations no scram would occur. Reactor pressure would rise because the increased flow through the individual steam lines causes higher steam line D/Ps. The higher pressure would cause higher reactor power.

Technical Reference(s): N2-OP-1, Sect. H.1.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 2101-239001C01, Main Steam & Reheater System (As available)
RBO-11,System Loss and Component Level Malfunction

Question Source: Bank # 33836
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295002 2.4.11	
	Importance Rating	4.0	

Emergency Procedures / Plan: Knowledge of abnormal condition procedures. (Loss of Main Condenser vacuum)

Question: RO #64

The plant is operating at 50% power with the following:

- Annunciator 851358, TURBINE CNSR A/B/C VACUUM LOW alarmed
- Main Condenser Vacuum is 24.4 inches Hg vac. and slowly degrading

Which one of the following actions is FIRST required by N2-SOP-09, Loss of Vacuum?

A. Scram the reactor IAW N2-SOP-101C, Reactor Scram.

B. Immediately trip the Main Turbine per N2-SOP-21, Turbine Trip.

C. Reduce reactor power IAW N2-SOP-101D, Rapid Power Reduction

D. Verify the Main Turbine has tripped and enter N2-SOP-101C, Reactor Scram.

Answer: C

Explanation (Optional):

A: Incorrect – A scram would be required if offgas inlet pressure could not be restored and maintained <19 psia and the reactor was still critical.

B: Incorrect - A turbine trip would be required if vacuum was <24.6" Hg and the turbine was loaded to <30% (363 MWe).

C: Correct – Per N2-SOP-9, First reduce power to attempt to stabilize vacuum.

D: Incorrect – The Main Turbine will not auto trip until vacuum lowers to 22.1 inches.

Technical Reference(s): N2-SOP-9, Loss of Condenser Vacuum (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-SOP09C01, Loss of Condenser Vacuum
EO-2, Operational Actions and Sequence (As available)

Question Source: Bank #

Modified Bank # NRC 2008 Q #83– initial conditions and correct answer changed (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029 EK2.07	
	Importance Rating	3.1	

Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: Drywell/containment water level

Question: RO #65

Following a loss of coolant accident the following conditions exist:

- Reactor pressure is 800 psig and lowering
- Suppression Pool level is 212 feet and rising

Which one of the following is the limiting component for these conditions?

A. SRV Tailpipes

B. Vacuum Breakers

C. Drywell Spray Ring

D. Suppression Chamber Spray Sparger

Answer: A

Explanation (Optional):

A: Correct – Per EOP-PC Bases Rev 5 page 5-21, at that SP level you have reached the SRV tailpipe level limit as given in EOP-PC Table N. This level is the highest suppression pool water level at which opening an SRV will not result in exceeding the code allowable stresses in the tail pipe, tail pipe supports, quencher, or quencher supports.

B: Incorrect – The bottom of the vacuum breaker openings is at 227.25 feet and is not the concern per EOP Bases at this SP level (Per EOP bases page 5-28).

C: Incorrect – The drywell spray ring is >240 feet and is not the concern per EOP Bases at this SP level.

D: Incorrect – the SP spray sparger is at elev. 231 ft and is not the concern per EOP Bases at this SP level (Per EOP bases page 5-27)

Technical Reference(s): EOP-PC bases page 5-21, EOP-PC STEP SPL-2, Table N (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPPCC01, Primary Containment Control EO-2, Operational Actions and Sequence (As available)

Question Source: Bank # 51144

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 3

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	G1	2.1.41	
	Importance Rating	2.8		

Conduct of Operations: Knowledge of the refueling process.

Question: RO #66

During a Refuel Outage, with the Reactor Mode Switch in Refuel, the following conditions exist:

- Annunciator 603442, Control Rod Out Block, alarms
- One Control Rod is selected
- All Control Rods are inserted to position 00

Which one of the following is a possible cause of the annunciator?

The Refuel Bridge is ...

A. NOT over the core with the Grapple full up and loaded

B. NOT over the core with the Grapple full down and loaded

C. over the core with the Grapple full up and the Trolley Hoist is loaded

D. over the core with the Grapple full down and the Trolley Hoist is NOT loaded

Answer: C

Explanation (Optional):

A: Incorrect: plausible; would be true if over the core

B: Incorrect: plausible; would be true if over the core

C: Correct: With the Refuel Bridge over the core with the Grapple up, but an auxiliary hoist loaded a Control Rod Block will be initiated.

D: Incorrect: plausible; would be true if any hoist was loaded

Technical Reference(s): N2-OP-39, Sect 5.4.2, pg 30 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 2101-234000, Fuel Handling & Reactor Servicing Equipment RBO-5, System Operation, Control and Instrumentation (As available)

Question Source: Bank # Minor editorial changes.

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G1 2.1.36	
	Importance Rating	3.0	

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations.

Question: RO #67

The plant is in Mode 5 with core alterations in progress. Which one of the following conditions REQUIRES that Secondary Containment be OPERABLE?

A. Any core alteration regardless of the time since plant shutdown.

B. Any core alteration if performed within 1 day of initial plant shutdown.

C. Movement of irradiated fuel regardless of the time since plant shutdown.

D. Movement of recently irradiated fuel if performed within 1 day of initial plant shutdown.

Answer: D

Explanation (Optional):

A: Incorrect: Tech Spec 3.6.4.1 requires secondary containment to be operable during movement of any recently irradiated fuel assembly. Tech Spec bases defines recently irradiated as the bundle having been part of a critical core within the past 24 hours. Additionally, although moving a fuel assembly inside the core would be a core alteration, there are other activities that are core alterations but would not be required to meet this spec.

B: Incorrect: Tech Spec 3.6.4.1 requires secondary containment to be operable during movement of any recently irradiated fuel assembly. Core alterations encompass more than just movement of irradiated fuel assemblies.

C: Incorrect: Tech Spec 3.6.4.1 requires secondary containment to be operable during movement of any recently irradiated fuel assembly. Tech Spec bases defines recently irradiated as the bundle having been part of a critical core within the past 24 hours.

D: Correct: Tech Spec 3.6.4.1 requires secondary containment to be operable during movement of any recently irradiated fuel assembly. Tech Spec bases defines recently irradiated as the bundle having been part of a critical core within the past 24 hours.

Technical Reference(s): Tech Spec 3.6.4.1 and associated bases. Tech Spec definition of core alterations. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-290001C01, Secondary Containment; RBO-15 Application of Technical Specifications (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 13

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.2	
	Importance Rating	4.6		

Equipment Control: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Question: RO #68

A plant startup and power ascension is in progress. The following conditions exist:

- Reactor Power is 26%
- Generator output is 250 MWe
- Load Limit setpoint is 250 MWe because of a generator winding issue
- Annunciator 851150, TURBINE BYPASS VALVE OPEN alarms and #1 Bypass Valve opens

Which one of the following actions is required to close the #1 Bypass Valve?

A. Insert control rods.

B. Raise Pressure Set.

C. Lower Pressure Set.

D. Lower Bypass Opening Jack setpoint.

Answer: A

Explanation (Optional):

A. Correct - With the Load Limit set at 250 MWe the turbine cannot accept any more load. Since the EHC systems wants to open the control valves, but can't because of the load limit the bypass valve is opening to control pressure. The only way to close the bypass valves is to lower reactor power and hence pressure by inserting the control rods.

B. Incorrect - Reactor power is providing more power than the generator can accept. Raising or lowering pressure setpoint will change the controlling setpoint but steam supply will still exceed steam demand from the turbine.

C. Incorrect - Reactor power is providing more power than the generator can accept. Raising or lowering pressure setpoint will change the controlling setpoint but steam supply will still exceed steam demand from the turbine.

D. Incorrect - lowering the Bypass Opening Jack setpoint would not affect BPV position because the pressure regulator is in control. Additionally the Bypass Opening Jack is used to open the bypass valves.

Technical Reference(s): ARP 851150, Turbine Bypass Valve Open (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-248000C01, Main Turbine EHC (As available)
RBO-10, Operational Actions and Sequence

Question Source: Bank # 34074
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	G2	2.2.13	
	Importance Rating	4.1		

Equipment Control: Knowledge of tagging and clearance procedures.

Question: RO #69

Which one of the following is an example of a "Group Tagging" situation and how would protected personnel sign on to this type of clearance?

	Covered under Group Tagging	Sign onto
A:	Entry into the suppression chamber to conduct inspections	Specific Radiation Work Permit
B:	Reactor disassembly on the Refueling Floor	Specific Radiation Work Permit
C:	Entry into the suppression chamber to conduct inspections	Operating Permit for Personnel Protection
D:	Reactor disassembly on the Refueling Floor	Operating Permit for Personnel Protection

Proposed Answer: A

Explanation (Optional):

A: Correct: An individual that requires protection of a Tagout for general area entries of the drywell and Suppression Chamber/Torus. The use of the Specific Radiation Work Permit for these areas provides this protection.

B: Incorrect: A group tagout is only used for entry into the drywell and Suppression Chamber/Torus.

C: Incorrect: The use of Operating Permits for personnel protection is to permit resting of devices such as hydraulic valves, dampers, SRM and IRM drives, etc.

D: Incorrect: A group tagout is only used for entry into the drywell and Suppression Chamber/Torus. The use of Operating Permits for personnel protection is to permit resting of devices such as hydraulic valves, dampers, SRM and IRM drives, etc.

Technical Reference(s): CNG-OP-1.01-1007, CLEARANCE AND SAFETY TAGGING, Page 12. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #S101-ADMPTPSC04, Clearance and Safety Tagging TO-2, Implement Clearance and Safety Tagging (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3 2.3.14	
	Importance Rating	3.4	

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Question: RO #70

Core Alterations are in progress with the following:

- An irradiated fuel bundle is being moved from the reactor cavity to Spent Fuel Pool
- The bundle becomes ungrappled and falls into the reactor vessel downcomer area. (Between the vessel wall and the shroud)
- The bundle integrity is maintained

Which one of the following workers is at greatest risk of radiation overexposure?

A. I&C Tech at SLS Tank

B. Refuel SRO on the Bridge

C. Mechanic working on SRVs.

D. RP Tech at Refuel Floor Access Point.

Answer: C

Explanation (Optional):

A: Incorrect: Worker closest to the bundle with the least amount of shielding will be at greatest risk. SLC Tank is in Secondary Containment, which is shielded by Primary Containment wall.

B: Incorrect: Worker closest to the bundle with the least amount of shielding will be at greatest risk. SRO on the bridge is shielded by water level within the cavity.

C: Correct: Worker closest to the bundle with the least amount of shielding will be at greatest risk. SRVs are in the Drywell at the approximate elevation of the downcomer.

D: Incorrect: Worker closest to the bundle with the least amount of shielding will be at greatest risk. RP Tech at the access point is shielded by water level within the cavity

Technical Reference(s): Primary Containment Big Notes (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Question 11 on 2005 NMP2 NRC Exam

Modified Bank # (Note changes or attach parent)

New

Question History: Question 11 on 2005 NMP2 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 12

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3 2.3.12	
	Importance Rating	3.2	

Radiation Control: Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question: RO #71

The plant is in a refuel outage, with the following:

- An accident on the refueling floor caused a serious injury to a worker
- The worker is on the Refueling Platform
- Radiation levels in the area of the injured operator are 2500 mRem/hr
- Emergency exposure limit for life saving operations has been authorized

In accordance with EPIP-EPP-15, Emergency Health Physics Procedure, which one of the following is the maximum stay time for the individual providing life saving assistance?

A. 2 hours

B. 4 hours

C. 6 hours

D. 10 hours

Answer: D

Explanation (Optional):

A: Incorrect. Based on incorrect limit of 5 Rem federal limit. 5R/2.5 R per hr= 2hours

B: Incorrect. Based on incorrect limit of 10 Rem limit (property). 10R/2.5R per hr = 4 hours

C: Incorrect. Based on correct limit of 15 Rem federal limit lens of eye. 15R/2.5R per hr = 6 hours

D: Correct. Based on correct limit of 25 Rem limit (life saving). 25R/2.5R per hr = 10 hours

Technical Reference(s): EPIP-EPP-15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: S-100-EPL001C07 TO-01 (As available)

Question Source:	Bank #		
	Modified Bank #	Question Number 70 on 2009 NRC Exam	Changed the radiation field and added lifetime exposure to worker to make two distracters plausible. All four choices changed.
	New		

Question History: Question Number 70 on 2009 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.17	
	Importance Rating	3.9		

Emergency Procedures / Plan: Knowledge of EOP terms and definitions.

Question: RO #72

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

- RCIC Full Flow Test surveillance is in progress
- Suppression Pool temperature rises to 91° F due to RCIC operation

Which one of the following identifies the proper EOP implementation requirements?

A. Entry is NOT required because no emergency condition exists.

B. Entry is NOT required because no entry conditions have been met.

C. Must be entered but can be exited since no emergency condition exists.

D. Must be entered and can only be exited when entry conditions are not met.

Answer: C

Explanation (Optional):

A: Incorrect: EOP-PC entry conditions exists, so EOP entry IS required. The EOP can be exited when it is determined that an emergency does not exist.

B: Incorrect: EOP-PC entry conditions exists, so EOP entry IS required. The EOP can be exited when it is determined that an emergency does not exist.

C: Correct: "An EOP may be exited if the exit conditions of the procedure are met or any time it is determined that an emergency no longer exists." The determination is not dependent upon the status of entry conditions...". EOP-PC entry conditions exists, so EOP entry IS required. The EOP can be exited when it is determined that an emergency does not exist.

D: Incorrect: "An EOP may be exited if the exit conditions of the procedure are met or any time it is determined that an emergency no longer exists."

Technical Reference(s): EOP Basis, pg 2-6, NER-2M-039 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01, Primary Containment Control
RBO-2, Operational Actions and Sequence (As available)

Question Source: Bank # 51145
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.12	
	Importance Rating	4.0		

Emergency Procedures / Plan: Knowledge of general operating crew responsibilities during emergency operations.

Question: RO #73

With the plant at 100% power a fire occurs in the Control Room and the CRS announces that they have entered SOP-78, Control Room Evacuation.

Which one of the following are the required ATC operator actions before leaving the Control Room?

Shutdown the Reactor, ...

A. close the MSIVs and trip the feedwater pumps.

B. trip one feedwater pump and verify 2FWS-LV10s in Auto

C. verify emergency diesel generators running unloaded and main turbine tripped.

D. main turbine tripped, house loads transferred and turbine bypass valves controlling pressure.

Answer: A

Explanation (Optional):

A: Correct: IAW SOP-78, the ATC operator must perform the following:
1. Mode switch to SHUTDOWN.
2. Confirm ALL rods full in.
3. Close the MSIV's.
4. Trip feedwater pumps.
5. Verify 2FWS-LV10s in MANUAL AND closed.

6. Verify Main Turbine tripped.
7. Confirm House Loads transferred OR Diesels energizing buses.

B: Incorrect: Both feedwater pumps are tripped and the 2FWS-LV10s are placed in MANUAL.

C: Incorrect: There is no direction to start the EGs, they would only be started if house loads failed to transfer in which case they would be loaded.

D: Incorrect: The MSIVs are closed pressure is not controlled by the turbine bypass valves.

Technical Reference(s): N2-SOP-78, (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 2101-SOP78, Control Room Evacuation CE-2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2 2.2.38	
	Importance Rating	3.6	

Equipment Control: Knowledge of conditions and limitations in the facility license.

Question: RO #74

The plant is operating at 100% power. What is the maximum Technical Specifications amount of Reactor Coolant System Leakage (gpm) allowed for continued plant operation?

	Unidentified	Identified
A.	2	20
B.	2	25
C.	5	20
D.	5	25

Answer: D

Explanation (Optional):

A: Incorrect: 2 gpm is the maximum change in unidentified leakage in 24 hours. However the max unidentified is 5 gpm.

B: Incorrect: 2 gpm is the maximum change in unidentified leakage in 24 hours. However the max unidentified is 5 gpm.

C: Incorrect: Total identified leakage is 25 gpm.

D: Correct: 5 gpm is the maximum unidentified leakage in 24 hours and 25 is the maximum identified within 24 hours.

Technical Reference(s): T.S. 3.4.5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-101001C01, Reactor Vessel and Internals (As available)
RBO-14, Application of Technical; Specifications

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

Question History:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	G4	2.4.29	
	Importance Rating	3.1		

Emergency Procedures / Plan: Knowledge of the emergency plan.

Question: RO #75

A Site Area Emergency has been declared at NMP 2. Which Emergency Response Facility provides the location for the overall response to an emergency, once all the Emergency Response Facilities are activated?

A. Control Room

B. Technical Support Center

C. Operations Support Center

D. Emergency Operations Facility

Answer: D

Explanation (Optional):

A: Incorrect: - The Control room manages the plant the EOF provides the overall response.

B: Incorrect: - The TSC provides technical support to the EOF

C: Incorrect: - The OSC provides support to the EOF.

D: Correct: The TSC, OSC and EOF are activated during an Alert, Site Area Emergency or General Emergency, or when directed by the SM/ED or ED/RM.
The Emergency Director/Recovery Manager (ED/RM) is responsible for managing all aspects of the NMP response to an emergency at NMPNS. The ED/RM operates from the EOF.

Technical Reference(s):	EPIP-EPP-13 EPIP-EPP-23 EPIP-EPP-18	(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: NS-EPL001-EPP11-TO-01 (As available)

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

Question History:

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

10 CFR Part 55 Content:	55.41	12
	55.43	

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

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

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Operational Status of Offsite Circuit.

Question: SRO #76

The plant is operating at rated conditions. You receive notification from System Power Control that area thunderstorms have caused unstable grid conditions. They also inform you that the Load Flow Program is NOT in service.

- Line 5 voltage on P852 is reading 109 KV
- Line 6 voltage on P852 is reading 113 KV

Which one of the following is required?

A. Declare both Line 5 and Line 6 inoperable and assess the condition for reportability.

B. Declare Line 5 inoperable and assess the condition for reportability.

C. Declare both Line 5 and Line 6 inoperable. Reportability is NOT required to be addressed unless a complete loss of Offsite Power occurs.

D. Declare Line 5 inoperable. Reportability is NOT required to be addressed unless a complete loss of Offsite Power occurs.

Answer: B

Explanation (Optional):

A: Incorrect: With Line 6 voltage at 113KV, it is not yet inoperable with voltage greater than 110 KV.

B: Correct: IAW SOP-70 Discussion Section 5.3 0- The Load Flow computer at National Grid runs a program that looks at 115 KV voltage for Line 5 and Line 6. This program determines if there is sufficient voltage (depending on grid loading) to supply NMP 2 ECCS loads during a LOCA (plant trip). If the voltage is too low, a contingency low voltage alarm is received at Power Control. If the contingency low voltage alarm is received OR actual voltage is less than 110 KV with the load Flow computer unavailable, then the offsite power source cannot perform its function under accident conditions and is, therefore, inoperable. The 110 KV value was selected because the actual low voltage value of 109.25 KV cannot be read on the Control Room instrumentation. If the 115 kV voltage lowers to the point where both Line 5 and Line 6 are inoperable, the event may be reportable under 10 CFR 50.72 (b)(3)(v) for an Event or Condition That Could Have Prevented Fulfillment of a Safety Function, based on NUREG 1022, Section 3.2.7.

C: Incorrect: With Line 6 voltage at 113KV, it is not yet inoperable with voltage greater than 110 KV.

D: Incorrect: Reportability must be addressed per the SOP.

Technical Reference(s): N2-SOP-70 Rev.1 Section 5.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-SOP70C01, Major Grid Disturbances
EO-3, Operational Actions and Sequence (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1,2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295019	AA2.01
	Importance Rating		3.6

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system pressure

Question: SRO #77

With the plant operating at rated conditions the following sequence occurs:

- Alarm CRD SCRAM AIR HEADER PRESSURE HIGH/LOW, 603306, is received.
- Scram air header pressure is reported as 59 psig and slowly lowering.
- Ten seconds later, alarm CONTROL ROD DRIFT, 603443 annunciates. Operators identify that control rod 34-23 has drifted to the fully inserted position.

Which one of the following describes the required actions and the reasons for those actions?

A. Enter N2-SOP-08, Unplanned Power Changes, and direct reactor power be reduced by 40 MWe using recirc flow control. This will improve the margin to thermal limits during potentially unanalyzed rod patterns.

B. Enter N2-SOP-08, Unplanned Power Changes, and monitor for additional rod drifts. If two rods drift direct that the reactor be scrammed. This is necessary to prevent potentially exceeding thermal limits due to crossing blade tips.

C. Enter N2-SOP-19, Loss of Instrument Air, and direct the backup pressure control valve, PCV-19A(B), be placed in service. This is necessary to re-establish normal scram air header pressure to maintain required scram times.

D. Enter N2-SOP-19, Loss of Instrument Air, and direct that the reactor be scrammed. This is necessary to prevent the development of unanalyzed rod patterns, which could result from rods individually scramming.

Answer: D

Explanation (Optional):

A: Incorrect: The reactor should be scrammed. Plausible in that the Rod Drift ARP directs entry into SOP-08. SOP-08 also directs power be reduced by 40 MWe for a single rod scram.

B: Incorrect: A scram criteria exists IAW SOP-19. Plausible in that this is the required action in SOP-08 if two rods drift.

C: Incorrect: Plausible in that ARP for CRD SCRAM AIR HEADER PRESSURE HIGH/LOW directs entry into SOP-19. SOP-19, attachment 3 directs shifting PCV valves provided differential pressure across the in-service PCV and filter is >9 psid. However, the override on the loss of air flowchart requires the scram.

D: Correct: ARP for CRD SCRAM AIR HEADER PRESSURE HIGH/LOW directs entry into SOP-19. SOP-19 directs that a scram be inserted if pressure lowers to less than 60 psig. This is necessary to prevent the development of unanalyzed rod patterns, which could result from rods individually scramming when air pressure is lost to their respective scram valves.

Technical Reference(s): ARP for CRD SCRAM AIR HEADER PRESSURE HIGH/LOW, 603306 (Attach if not previously provided)
SOP-19, Loss of Instrument Air Flowchart
LP N2101212000C01, RPS, page 200 for reason why reactor is scrammed.

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 2101-SOP19C01, Loss of Instrument Air, EO-2, Operational Actions and Sequence (As available)

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

Question History:

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295001	AA2.04
	Importance Rating		3.1

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Individual jet pump flows: Not-BWR-1&2

Question: SRO #78

Reactor Power unexpectedly LOWERED from 85% to 80% power.

The following indications are noted:

- Loop A Jet Pump Flow LOWERED to 33 Mlbm/hr.
- Loop B Jet Pump Flow INCREASED to 42 Mlbm/hr.
- Jet Pump 3 and Jet Pump 4 Flow LOWERED from 3.5 to 0.5 Mlb/hr.

Which one of the following actions is required as a result of these conditions?

A. Declare Jet Pumps 3 and 4 INOPERABLE. Be in MODE 3 within 12 hours because the ability to reflood the core following a Loss of Coolant Accident is NOT assured.

B. Declare Jet Pumps 3 and 4 INOPERABLE. Be in MODE 3 within 12 hours because Loop coastdown characteristics assumed in the Loss of Coolant Accident analysis will NOT be preserved.

C. Declare Reactor Recirculation Loop A to be “not in operation” within 2 hours because the ability to reflood the core following a Loss of Coolant Accident is NOT assured. Plant operation may continue in Single Loop.

D. OPEN Loop A FCV to raise Loop A Jet Pump Flows to re-establish MATCHED Reactor Recirculation Loop Flows within 2 hours because Loop coastdown characteristics assumed in the Loss of Coolant Accident analysis will NOT be met. Plant operation may continue in Two Loop.

Answer: A

Explanation (Optional):

A: Correct: Jet Pump 3/4 common riser separation is indicated. JP 3 and JP 4 flow deviates from remainder of Loop A by >10%. With this condition, it is required to be in Mode 3 within 12 hours because the ability to reflood the core following a LOCA is not assured

B: Incorrect: and plausible; Basis is not correct for Jet Pumps INOPERABLE.

C: Incorrect: and plausible; Recirculation Loops are mismatched, continued operation in Single Loop is NOT permitted because Jet Pumps 3/4 are INOPERABLE.

D: Incorrect: and plausible; Recirculation Loops are mismatched, continued operation in Single Loop is NOT permitted because Jet Pumps 3/4 are INOPERABLE.

Technical Reference(s): Tech Spec LCO 3.4.1 and Tech Spec LCO 3.4.3 and associated SR 3.4.3.1. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.4.1 & 3.4.3 – NO BASES

Learning Objective: LP #2101-101001C01, Reactor Vessel and Internals RBO-14, Application of Technical Specifications (As available)

Question Source: Bank # NMP 2008 NRC Exam SRO Question #1

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295021	2.2.38
	Importance Rating		4.5

Equipment Control: Knowledge of conditions and limitations in the facility license: (Loss of SD Cooling)

Question: SRO #79

RHR "A" is in shutdown cooling with the following:

- Plant is in Mode 4
- Both Reactor Recirculation pumps are secured
- RPV water level is 185 inches
- Reactor coolant temperature is 150°F and slowly lowering

With these conditions, 2RHS*MOV112, Suction Isolation Valve, now fails shut and cannot be reopened.

Which of the following describes (1) the required procedural action and (2) the LCO time constraint to verify an alternate method of decay heat removal?

A. (1) Enter N2-SOP-31R, Refueling Operations Alternate Heat Removal, and place an Alternate Decay Heat Removal Loop in service
(2) Within a maximum of one hour.

B. (1) Enter N2-SOP-31R, Refueling Operations Alternate Heat Removal, and place an Alternate Decay Heat Removal Loop in service
(2) Within a maximum of two hours.

C. (1) Enter N2-SOP-31, Loss of SDC, and maximize WCS cooling per N2-OP-37, Reactor Water Cleanup
(2) Within a maximum of one hour.

D. (1) Enter N2-SOP-31, Loss of SDC, and maximize WCS cooling per N2-OP-37, Reactor Water Cleanup
(2) Within a maximum of two hours.

Answer: C

Explanation (Optional): KA match reasoning – Technical Specification and procedures are part of conditions and limitations in the facility license.

A: Incorrect – No entry conditions exist for SOP-31R as stated in the stem (RPV flooded up).

B: Incorrect - No entry conditions exist for SOP-31R as stated in the stem.

C: Correct – The loss of 2RHS*MOV112, the common suction valve, precludes the use of SDC, therefore, IAW SOP-31, WCS must be maximized. TS 3.9.9.A.1 - Verify an alternate method of decay heat removal is available for the inoperable RHR shutdown cooling subsystem.

D: Incorrect – The TS LCO is one hour.

Technical Reference(s): N2-SOP-31, Loss of SDC
TS 3.9.9, Residual Heat Removal (RHR) – Low Water Level
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-SOP31C01, Loss of Shutdown Cooling
EO-2, Operational Actions and Sequence
(As available)

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

Question History:

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295016	2.4.34
	Importance Rating		4.1

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects: (Control Room Abandonment)

Question: SRO #80

The plant was operating at rated conditions when a fire in the control room required its evacuation. The CRS has entered N2-SOP-78, Control Room Evacuation. The initial actions in the control room for the Operator At-The-Controls were completed prior to evacuation.

It has been 25 minutes since the evacuation and the operators have yet to fully establish plant control. The fire has just been reported extinguished.

Which one of the following describes a method to establish plant control and what is the emergency action level that has been met.

A. EHC will automatically control RPV pressure with the bypass valves. An SAE must be declared.

B. Operators will cycle the SRVs to control reactor pressure. An SAE must be declared.

C. EHC will automatically control RPV pressure with the bypass valves. An Alert must be declared.

D. Operators will cycle the SRVs to control reactor pressure. An Alert must be declared.

Answer: B

Explanation (Optional):

A: Incorrect – Per N2-SOP-78, pressure control is established with SRVs. The initial ATC operator actions close the MSIVs which precludes use of bypass valves for pressure control.

B: Correct – Per N2-SOP-78, SRVs are used to control RPV pressure. EAL 7.2.4 applies due to plant control not being established with 15 minutes of control room evacuation.

C: Incorrect - Per N2-SOP-78, pressure control is established with SRVs. The initial ATC operator actions close the MSIVs which precludes use of bypass valves for pressure. An SAE is required.

D: Incorrect – An SAE is required

Technical Reference(s): N2-SOP-78, Control Room Evacuation (Attach if not previously provided)
EPIP-EPP-2, EAL 7.2.4

Proposed References to be provided to applicants during examination: EPIP-EPP-2 chart

Learning Objective: 2101-SOP-78C01, Control Room Evacuation (As available)
CE-1

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis x

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295037	2.4.6
	Importance Rating		4.7

Emergency Procedures / Plan: Knowledge of EOP Mitigation strategies: (Scram present and APRMs not downscale)

Question: SRO #81

An ATWS is in progress. Following the actions to terminate and prevent injection the following conditions exist:

- Bypass valves failed to open and cannot be opened
- Reactor pressure is being maintained 800-1000 psig and 2 SRVs are open
- Suppression Pool average water temperature is 120 Degrees F
- Suppression Pool level is 199.8 feet
- Control rod insertion has not been established
- SLS failed to inject and cannot be started
- No alternate boron injection system is injecting

When indicated level reaches -40 inches Fuel Zone (FZ), direction is given to reestablish injection and maintain indicated level -70 to -40 inches (FZ)

With indicated water level at -40 inches (FZ), RPV injection is re-established. Ten (10) seconds later indicated water level is -10 inches (FZ) and reactor power is 5%

Which one of the following is the correct action in response to this transient?

A. Terminate and prevent injection again.

B. Perform a RPV Blowdown per EOP-C2.

C. Direct a new level control band of -70 to -10 inches (FZ).

D. Reduce the injection rate until in the assigned -70 to -40 inches band.

Answer: A

Explanation (Optional):

A: Correct - Level rise will cause reactor power to increase and exceed 4%. Override conditions (L-5) are met to terminate and prevent injection until reactor power lowers below 4% or level is at TAF (-52" FZ at 800 psig).

B: Incorrect - There is initially a 20°F margin to HCTL, and rise in suppression pool temperature will not require RPV Blowdown at this time. Reactor pressure will be reduced in a controlled manner to stay within the HCTL if it is being challenged, and then if exceeded and cannot restore within HCTL a blowdown would be performed.

C: Incorrect - Level rise will cause reactor power to increase and exceed 4%. Override conditions are met to terminate and prevent injection until reactor power lowers below 4% or level is at TAF(-52" FZ at 800 psig).

D: Incorrect - Restoring to the directed level control band is not appropriate, Override conditions are met to terminate and prevent injection again.

Technical Reference(s): N2-EOP-C5,Failure to Scram (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPC5C01, Failure to Scram, EO-2, Operation Actions and Sequence (As available)

Question Source: Bank # 22911

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295023	AA2.03
	Importance Rating		3.8

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Airborne contamination levels

Question: SRO #82

Refueling is in progress. The refuel floor SRO notices that spent fuel pool level is rapidly lowering. The refuel floor is evacuated.

- All the Refuel Floor Radiation Monitors are alarming and off-scale
- SGTS has initiated and Reactor Building HVAC has isolated

45 minutes later, the following conditions exist:

- Stack monitor reads 4.1E6 μCi/sec.
- Field monitoring reports indicate an external dose exposure at the site boundary of 110 Mrem/Hr

Based upon these conditions which one of the following was the correct initial Emergency Plan classification and what IF ANY re-classification is required?

(1) Initial classification
(2) Classification at 45 min

A. (1) Alert
(2) Alert

B. (1) Alert
(2) SAE

C. (1) SAE
(2) SAE

D. (1) SAE
(2) GE

Answer: B

Explanation (Optional):

A: Incorrect – An SAE is required after 45 minutes

B: Correct - An Alert is initially required per EPIP-EPP-02-EAL-1.4.2 because the isolation occurred. An SAE is required after 45 minutes based on EAL 5.2.4

C: Incorrect – Initially an Alert is required

D: Incorrect – Initially an Alert is required. After 45 minutes, the field survey readings have not reached the GE levels.

Technical Reference(s): EPIP-EPP-02, EAL Chart (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EPIP-EPP-02 Chart

Learning Objective: LP #S703-EROEALC04, EAL Overview and Bases Objective – 6, Radioactivity Release Category (As available)

Question Source: Bank #

Modified Bank # NRC 2009 #98 Time changed from 30 to 45 mins. Bank question did not contain any field monitoring data. This question has site boundary exposures at 110 Mrem/hr.

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295015	AA2.02
	Importance Rating		4.2

Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Control rod position

Question: SRO #83

The reactor was operating at 92% power with 2CSH*P1, HPCS Pump, in operation in Full Flow Test mode, Suppression Pool to Suppression Pool, as retest following maintenance. A transient has occurred which resulted in a scram and the following conditions:

- Suppression Pool Level: 197 feet and lowering slowly
- Reactor Level: 35 inches and lowering slowly
- Reactor Pressure: 1048 psig and rising slowly
- Control Rod 30-31: Position 24
- Control Rod 18-47: Position 08

Which of the following is correct concerning these conditions?

A. 2CSH*MOV118, Suppression Pool Suction Valve, remains open. N2-EOP-PC, RPV and C5 are required to be entered, and OP-31 or OP-33 are used for Suppression Pool level control.

B. 2CSH*MOV118, Suppression Pool Suction Valve, closes. ONLY EOPs N2-EOP-PC and RPV are required to be entered.

C. 2CSH*MOV118, Suppression Pool Suction Valve, remains open. ONLY EOPs N2-EOP- PC and RPV are required to be entered, and OP-31 or OP-33 are used for Suppression Pool level control.

D. 2CSH*MOV118, Suppression Pool Suction Valve, closes. N2-EOP-PC, RPV and C5 are required to be entered.

Answer: A

Explanation (Optional):

A: Correct - MOV118 opens automatically upon either a high level in the suppression pool (>200.58' elevation) or a low level in CST 'B' (<97" of H_2O). It does not auto close. EOP-RPV is entered on the scram due to low RPV level, EOP-C5 is entered because more than one control rod is positioned beyond "00".
IAW EOP-PC, OP-31 and/or 33 may be used to address SP level.

B: Incorrect - 2CSH*MOV118 remains open and EOP-C5 is entered

C: Incorrect - EOP-C5 is required to be entered.

D: Incorrect - 2CSH*MOV118 remains open

Technical Reference(s): N2-EOP-RPV, RPV Control (Attach if not previously provided)
N2-EOP-PC,Primary Containment Control
N2-OP-33, page 4

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-209002C01, High Pressure Core Spray System (As available)
RBO-3, Functional Arrangement

Question Source: Bank # 54010
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295014	2.4.31
	Importance Rating		4.1

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures: (Inadvertent Reactivity Addition)

Question: SRO #84

Reactor power is on IRM range 8 with a reactor plant startup in progress when the following sequence of events occurs.

- Alarm IRM UPSCALE,603207,annunciates
- The reactor operator reports all IRMs are rising and up ranges IRM channels
- Alarm APRM TRIP SYSTEM UPSCALE, 603208, annunciates
- Reactor power is reported as 17% and rising
- The reactor operator places the mode switch to Shutdown
- The reactor operator reports that all control rods have inserted.

Based on these indications, which one of the following is correct regarding entry into the emergency operating procedures and the emergency plan?

A. Entry into N2-EOP-C5, Failure to Scram is required and a Site Area Emergency is required.

B. Entry into N2-EOP-RPV, RPV Control is required and an Alert Declaration is required.

C. Entry into the EOPs is NOT required but an Alert Declaration is required.

D. Entry into N2-EOP-RPV, RPV Control is required but entry into the Emergency Plan is NOT required.

Answer: B

Explanation (Optional):

A: Incorrect: Although RPS failed to trip on the APRMs, entry into as C5 is not required because placing the mode switch to shutdown achieved rod insertion. C5 is only entered from N2-EOP-RPV if manual action to scram is unsuccessful. The SAE is plausible in that it addresses a failure to scram condition but is only exceeded if both auto AND manual circuits fail.

B: Correct: An entry condition to N2-EOP-RPV was exceeded when reactor power exceeded the APRM setpoint scram settings and the reactor did not scram. An alert is required based on EAL 2.2.1. This EAL is exceeded when a RPS scram signal is exceeded and an automatic scram fails to result in a control rod pattern that will achieve reactor shutdown status under all conditions. This occurred when the RPS failed to auto trip when the APRM set point was exceeded.

C: Incorrect: An entry condition to N2-EOP-RPV was exceeded when reactor power exceeded the APRM setpoint scram settings and the reactor did not scram. Plausible in that manual action achieved reactor shutdown and no other indications of an emergency are provided.

D: Incorrect: An alert is required based on EAL 2.2.1. Plausible in that placing the mode switch in shutdown achieved a control rod pattern that will ensure the reactor remains shutdown under all conditions.

Technical Reference(s): EAL 2.2.1 (Attach if not previously provided)

N2-EOP-RPV

Proposed References to be provided to applicants during examination: EAL Chart only

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295007	AA2.01
	Importance Rating		4.1

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure

Question: SRO #85

Given the following conditions:

- A pressure transient occurred with steam dome pressure peaking at 1345 psig.
- Pressure was subsequently reduced and is currently 920 psig.

From the list below, select the actions required.

Action 1 - Insert all insertable control rods within 2 hours.
Action 2 - Place the plant in cold shutdown within 24 hours while maintaining cooldown rate below 100 degrees F/hour.
Action 3 - Place the plant in cold shutdown within 24 hours disregarding any cooldown rate limitations.
Action 4 - Obtain NRC approval prior to resuming critical operation.

A. Actions 3 and 4 ONLY

B. Actions 1 and 2 ONLY

C. Actions 1 and 4 ONLY

D. Actions 1, 3 and 4 ONLY

Answer: C

Explanation (Optional):

A: Incorrect – Action 3 is not required by Technical Specifications

B: Incorrect – Action 2 is not required by Technical Specifications

C: Correct – TS 2.2 Safety Limits requires all insertable control rods to be inserted within 2 hours. 10 CFR 50.36 requires NRC approval prior to going critical because a safety limit was violated.

D: Incorrect – Action 2 is not required by Technical Specifications

Technical Reference(s): TS 2.2 (Attach if not previously provided)
10CFR50.36, Technical Specifications

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-101001C01, Reactor Pressure Vessel and Internals (As available)
RBO-14, Application of Technical Specifications

Question Source: Bank # WTS 2098
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 A2.05
	Importance Rating	3.4

Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of SBLC tank heaters

Question: SRO #86

The reactor is shutdown following a scram from rated conditions. Reactor pressure is 940 psig and stable.

While completing post scram actions, annunciator 601711, SLCS TANK 1 TEMPERATURE HIGH/LOW, alarms. The field operator reports that local tank temperature is 69 degrees and that the breakers for both heaters are tripped.

Assuming that temperature continues to drop while troubleshooting the tripped breakers....

(1) How much time is available before reactor coolant temperature must be less than 200 degrees AND
(2) What is the reason for this action given these plant conditions?

A. (1) 32 hours
(2) The standby liquid solution may not meet its design criteria in responding to an ATWS.

B. (1) 44 hours
(2) The standby liquid solution may not meet its design criteria in responding to an ATWS.

C. (1) 32 hours
(2) The standby liquid solution may not meet its design criteria in maintaining the proper pH in the suppression pool following a LOCA.

D. (1) 44 hours
(2) The standby liquid solution may not meet its design criteria in maintaining the proper

pH in the suppression pool following a LOCA.

Answer: D

Explanation (Optional):

A: Incorrect: The allowable time is 44 hours (8 hours of condition B plus the 36 of condition C). 32 is plausible because candidate needs to know that even though all rods are in the 12 hours that would have been used to achieve mode 3 is still available to achieve mode 4 as discussed on page 1.3-3 of the tech spec. (8 hours of condition B plus only 24 hours of condition C). Also, the plant is in MODE 3 and SLC is not required to perform its ATWS function during MODES 3, 4, or 5.

B: Incorrect: The plant is in MODE 3 and SLC is not required to perform its ATWS function during MODES 3, 4, or 5.

C: Incorrect: The allowable time is 44 hours.

D: Correct: The allowable time is 44 hours. Mode 4 must be achieved due to the LOCA concern.

Technical Reference(s): Section 3.1.7 and associated bases. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Section 3.1.7 of Tech Spec (no bases)

Learning Objective: LP #2101-211000C01, Standby Liquid Control System (As available)
RBO-14. Application of Technical Specifications

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level		SRO
	Tier #		2
	Group #		1
	K/A #	218000	A2.03
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of air supply to ADS valves: Plant-Specific

Question: SRO #87

RPV Blowdown is currently being executed in accordance with N2-EOP-C2. Drywell pneumatics has just been restored to the ADS valves when the following indications and reports are received:

- All seven ADS valves are open
- Reactor pressure is 180 psig and lowering
- Annunciator 601515, ADS AIR HEADER B TROUBLE, alarms
- 2IAS*SOVY186, ADS HEADER B LO FLOW SUPPLY VLV is open
- 2IAS*SOVX186, ADS HEADER B HI FLOW SUPPLY VLV is open
- ADS Header B pressure is 60 psig and lowering
- All ADS accumulator pressures associated with ADS Header B indicate approximately 170 psig and slowly lowering

Given the above conditions:

(1) What is the status of the ADS valves associated with ADS AIR HEADER B and
(2) What action should the SRO direct to mitigate the consequences of these indications?

A: (1) ADS valves are open but will eventually close as accumulators depressurize.
(2) If less than 6 SRVs are open, maintain RPV pressure less than 40 psi above Suppression Chamber pressure, in accordance with N2-EOP-C2.

B: (1) ADS valves are cycling between open and close due to low pneumatic pressure.
(2) If less than 6 SRVs are open, maintain RPV pressure less than 40 psi above Suppression Chamber pressure, in accordance with N2-EOP-C2.

C: (1) ADS valves are open but will eventually close as accumulators depressurize.
(2) Direct that ADS Header B and ADS Receiver 2IAS*TK 5 be re-pressurized via Emergency Truck Fill, in accordance with ARP 601515.

D: (1) ADS valves are cycling between open and close due to low pneumatic pressure.
(2) Direct that ADS Header B and ADS Receiver 2IAS*TK 5 be re-pressurized via Emergency Truck Fill, in accordance with ARP 601515.

Answer: A

Explanation (Optional):

A: Correct: ADS valves are being held open by their accumulators, as pressure lowers the valves will close and cannot be opened. The pneumatic symptoms provided are indicative of a pneumatic supply line break in the drywell. ADS Air Header B supplies air to ADS valve accumulator tanks TK35, TK36, TK37 AND TK38. The three other ADS valve accumulators are still operable. Additionally the non ADS valves are un-effected. IAW N2-EOP-C2, RPV Blowdown with less than 6 SRVs open, maintain RPV pressure less than 40 psig above suppression chamber pressure.

B: Incorrect: ADS valves are being held open by their accumulators.

C: Incorrect: Aligning the Emergency Fill would not restore pressure due to the break inside the drywell.

D: Incorrect: ADS valves are being held open by their accumulators. Aligning the Emergency Fill would not restore pressure due to the break inside the drywell.

Technical Reference(s): ARP 601515, ADS AIR HEADER B TROUBLE
N2-EOP-C2
N2-OP-34, Page 35
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-218000C01, Automatic Depressurization System
RBO-5, System Operation, Control, and Instrumentation
(As available)

Question Source: Bank #

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

Question History:

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215005 A2.05
	Importance Rating	3.6

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of recirculation flow signal

Question: SRO #88

The plant is at 40% power with a power ascension in progress. Core flow is 55 Mlbm/hr. APRM #3 has failed downscale and is bypassed.

With these initial conditions, the Recirc Loop "A" flow transmitter inputting to APRM #1 fails UPSCALE.

Which one of the following describes the plant response and required operator action(s)?

A. Rod block and a ½ scram on RPS "A"
Enter LCO 3.3.1.1 A. No additional action is required for RPS due to existing ½ scram.

B. Rod Block ONLY.
Enter LCO 3.3.1.1 A. If an inoperable APRM channel is not restored within 12 hours, insert an APRM upscale trip.

C. Rod Block ONLY.
Enter LCO 3.3.1.1 C. If an inoperable APRM channel is not restored within 1 hour, insert an APRM upscale trip.

D. Rod block and a ½ scram on RPS "A"
Enter LCO 3.3.1.1 B. No additional action is required for RPS due to existing ½ scram.

Answer: B

Explanation (Optional):

A: Incorrect. A ½ scram will not occur. The Recirc flow transmitter failing upscale will cause the flow biased scram setpoint to increase.

B: Correct: Per T.S. table 3.3.1.1-1, 3 APRMs are required. With one less than the required number an APRM must be restored within 12 hours or the channel placed in a tripped condition. A rod block will occur due to >10% difference in total recirc flow signals from the APRMs. APRM #1 summer will have an output > 75%. Other APRM flow summers will indicate ~ 50% at this core flow.

C: Incorrect: LCO 3.3.1.1 C does not apply as full scram capability still exists

D: Incorrect. A ½ scram will not occur. The Recirc flow transmitter failing upscale will cause the flow biased scram setpoint to increase.

Technical Reference(s): TS LCO 3.3.1.1 and associated table (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.3.1.1 and associated table (no setpoints/values)

Learning Objective: LP #2101215003C01, Power Range Neutron Monitoring and Rod Block Monitor
RBO-11, System Loss and Component Level Malfunction (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	400000	2.1.23
	Importance Rating		4.4

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation: (Component Cooling Water)

Question: SRO #89

The plant is at rated power when the following alarms and reports are received:

- Alarm 601246 REACTOR BLDG CLOSED LOOP COOLING SYS TROUBLE
- CCP Surge Tank level is out of sight low
- Alarm 601232,RBCLCW PUMPS 3A/3B/3C AUTO TRIP/FAIL TO START
- Alarm 601234 RBCLCW PUMPS 3A/3B/3C DISCH PRESS LOW
- Alarm 601252 RBCLC PUMP 1A/1B/1C AUTO TRIP/FAIL TO START
- CCP-P3A, B and C are all tripped
- CCP-P1A, B and C are all tripped

Which one of the following actions are required?

A. Direct a Rapid Power Reduction be performed per N2-SOP-101D and heat loads be secured as required to maintain component temperature within limits. If component temperatures cannot be maintained, direct a scram and enter N2-SOP-101C, Reactor Scram.

B. Direct a scram and enter N2-SOP-101C, Reactor Scram. Direct that both Recirc pumps be tripped and enter N2-SOP-29, Sudden Reduction in Core Flow.

C. Enter N2-SOP-60, Loss of Drywell Cooling, and if drywell temperature cannot be maintained less than 150 degrees direct a scram and enter N2-SOP-101C, Reactor Scram.

D. Direct WCS pumps and one Recirc pump be tripped and enter N2-SOP-29, Sudden Reduction in Core Flow, to rapidly lower power and reduce heat load. If component temperatures cannot be maintained, direct a scram and enter N2-SOP-101C, Reactor Scram.

Answer: B

Explanation (Optional):

A: Incorrect: N2-SOP-13 directs that a scam be inserted if all CCP pumps are tripped. Plausible in that N2-SOP-14, Loss of CCS does direct a rapid power reduction be performed.

B: Correct: N2-SOP-13 directs that the reactor be scrammed and both recircs be tripped if all CCP pumps are loss.

C: Incorrect: N2-SOP-13 directs that the reactor be scrammed if all CCP pumps are tripped. Plausible in that a loss of drywell cooling would also occur in this event.

D: Incorrect: N2-SOP-13 directs that the reactor be scrammed if all CCP pumps are tripped. Plausible in that the actions suggested would reduce heat load and reactor power.

Technical Reference(s): N2-SOP-13, Loss or Degraded CCP System (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-208000C01, Reactor Building Closed Loop Cooling System RBO-11, System Loss and Component Level Malfunction (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	203000	2.2.40
	Importance Rating		4.7

Equipment Control: Ability to apply Technical Specifications for a system: (Residual Heat Removal /Low Pressure Coolant Injection)

Question: SRO #90

The plant was operating at 100% when the following sequence of events occurred:

- On 5/1 at 08:00 – Division I Diesel Generator, EGS*EG1, is declared Inoperable.
- On 5/1 at 13:00 – Determination is made that the motor for RHS*MOV24B, LPCI B Injection Valve, will not develop enough torque during accident accidents to open the valve.

Which one of the following actions is required by Technical Specifications?

A. Restore OPERABILITY of EGS*EG1 by 08:00 on 5/4 and restore LPCI B by 11:00 on 5/8

B. The plant must be in at least Mode 3 by 01:00 on 5/2 and at least Mode 4 by 01:00 on 5/3

C. Entry is required to LCO 3.0.3 at 13:00 on 5/1 and shutdown must commence by 14:00

D. Entry is required to LCO 3.0.3 by 17:00 on 5/1 and shutdown must commence by 18:00

Answer: D

Explanation (Optional):

A: Incorrect: Only includes the actions for the inoperable diesel generator and the LPCI Injection Valve without considering the interaction between the two.

B: Incorrect: Incorrect application of LCO 3.0.3.

C: Incorrect: LCO 3.0.3 is not required to be entered until 17:00

D: Correct: Tech Spec 3.8.1 Condition B, Required Action B.2 becomes required when RHS*MOV24B becomes inoperable at 13:00. Four hours later (at 17:00) RHS*P1A and CSL*P1 must be declared inoperable. With 2 LPCI and CSL inoperable Tech Spec 3.5.1 Action H.1 requires entry into LCO 3.0.3. LCO 3.0.3 requires a shutdown and must commence with 1 hour (by 18:00).

Technical Reference(s): Tech Spec 3.8.1, 3.5.1, and 3.0.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Tech Spec 3.5.1 and 3.8.1

Learning Objective: LP #2101-264001C01, Standby Diesel Generator and Auxiliaries (As available)
RBO-14, Application of Technical Specifications

Question Source: Bank # 34095
Modified Bank # (Note changes or attach parent)
New

Question History: SRO #15 on 2005 Audit Exam

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	290002	A2.04
	Importance Rating		4.1

Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Excessive heatup/cooldown rate

Question: SRO #91

The reactor was at rated conditions when a trip of both Recirc Pumps resulted in a manual scram. Current plant conditions are:

- Reactor pressure: 800 psig and lowering
- Reactor water level: 184 inches and steady
- Cooling down to the Main Condenser
- RPS is tripped.
- Reactor Water Cleanup is isolated.
- Neither Recirc Pump can be started

Which one of the following is:

(1) A concern in this situation AND
(2) An action that the SRO should direct that is in accordance with station procedures that would be effective in mitigating the concern?

A. (1) Vessel stratification causing excessive cooldown of the bottom head.
(2) Maximize CRD Cooling Water to increase natural circulation in accordance with N2-OP-30, Control Rod Drive.

B. (1) Excessive cooldown of the Recirc loops resulting in exceeding the delta temperature limits between the loops and reactor coolant temperature.
(2) Restore Reactor Water Cleanup and secure Reactor Water Cleanup Bottom Head suction to maximize flow from the Recirc Loops in accordance with N2-OP-37, Reactor Water Cleanup System

C. (1) Vessel stratification causing excessive cooldown of the bottom head.
(2) Reset the reactor scram as soon as possible in accordance with N2-SOP-101C, Reactor Scram.

D. (1) Excessive cooldown of the Recirc loops resulting in exceeding the delta temperature limits between the loops and reactor coolant temperature.
(2) Reduce the cooldown rate and if necessary close the MSIVs in accordance with N2-OP-101C, Plant Shutdown.

Answer: C

Explanation (Optional):

A: Incorrect: Increasing the CRD cooling water flow would increase the cooldown of the bottom head.

B: Incorrect: the concern is excessive cooldown of the bottom head.

C: Correct: Per N2-SOP-29, Attachment 1, Recovering from Sudden Reduction in Recirc Flow, page 3 of 6, the concern is excessive cooldown of the bottom head. One of the corrective actions provided is to reset the scram which will reduce the cooldown of the bottom head.

D: Incorrect: the concern is excessive cooldown of the bottom head.

Technical Reference(s): N2-SOP-29, Attachment 1, Recovering from Sudden Reduction in Recirc Flow, pages 3 and 4 of 6.
N2-SOP-101C, Reactor Scram, flowchart
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-SOP29C01, Sudden Reduction in Core Flow
Obj.-3, Operational Actions and Sequence
(As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New	X	
Question History:		
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	201003	2.2.22
	Importance Rating		4.7

Equipment Control: Knowledge of limiting conditions for operations and safety limits: (Control Rod and Drive Mechanism)

Question: SRO #92

The reactor is operating at rated conditions with control rods 10-15 and 14-15 declared "slow". There are no other "slow" rods in the core.

While doing a control rod exercise, control rod 18-15 is inserted to position 46. All efforts to restore the rod to position 48 are unsuccessful. Additionally, the control rod cannot be inserted by any means.

Which one of the following describes the required Tech Spec actions?

A. Disarm the stuck control rod within 2 hours;
No other action is required to be completed within the next 24 hours.

B. Disarm the stuck control rod within 2 hours;
Exercise all remaining operable rods at least one notch.
No other action is required to be completed within the next 24 hours.

C. The reactor must be in Mode 3 within 12 hours

D. Restore at least one of the three control rods to operable status within 8 hrs. If not restored then be in Mode 3 within the following 12 hours.

Answer: C

Explanation (Optional):

A: Incorrect: Separation criteria is not met which requires the plant to be shutdown. Plausible in that if not recognized this would be partially correct.

B: Incorrect: Separation criteria is not met which requires the plant to be shutdown. Plausible in that if not recognized this would be correct.

C: Correct: Required action 3.1.3.A.1 for a stuck rod requires that separation criteria be met. Separation criteria "b" is not met in that the stuck rod is adjacent to a slow rod which is in turn adjacent to another slow rod (Tech Spec bases page B 3-1.3-4). Therefore required action E.1 specifies the plant be in mode 3 in 12 hours.

D: Incorrect: Separation criteria is not met which requires the plant to be shutdown

Technical Reference(s): Tech Spec 3.1.3.A and associated bases (Attach if not previously provided)
Core Map

Proposed References to be provided to applicants during examination: Tech Spec 3.1.3.A and associated bases (need for separation criteria)
Core Map

Learning Objective: LP #2101-201001C01, Control Rod Drive System (As available)
RBO-14, Application of Technical Specifications

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

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: R

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	259001	2.1.19
	Importance Rating		3.8

Conduct of Operations: Ability to use plant computers to evaluate system or component status: (Reactor Feedwater)

Question: SRO #93

The reactor is at rated conditions when the following process computer generated alarms are received over a one minute period:

- CECBC54,PMS-3D CORE MARGIN ALARM
- NSSQB10, CORE THERMAL POWER
- NSSQB10S, 1 MIN SMTHD CT POWER

The RO reports the following additional indications:

- FW Temperature Computer points NSSTA101, 102, 103, 104 all indicate 421 degrees
- Computer point FWSFU02, Feedwater Flow Line A LEFM Adjusted, indicates 7.9 E6 Mlbm/hr and very slowly rising
- Computer point FWSFU03, Feedwater Flow Line B LEFM Adjusted indicates 7.5 E6 Mlbm/hr and steady
- Feedwater flow indications in the control room are both steady at 7.5 E6 Mlbm/hr
- Feed Pump Amps are steady
- APRM power is steady at 100%

Which one of the following is consistent with these indications and describes the required actions?

A. A loss of feed water heating has occurred. Enter N2-SOP-08, Unplanned Power Changes, and direct a power reduction to 85%.

B. Computer point FWSFU02 is failing. Enter N2-REP-11, Independent Methods of Determining Core Thermal Power, and verify core thermal power.

C. The process computer is inoperable. Verify at least three non licensed operators are on duty in accordance with Tech Spec 5.2.2.a.

D. Licensed power level is being exceeded. Enter N2-SOP-101D, Rapid Power Reduction and direct the first 4 CRAM rods be inserted.

Answer: B

Explanation (Optional):

A: Incorrect: Feedwater temperatures are nominal and power is not really increasing.

B: Correct: With the exception of this computer point all other indications indicate that power is steady at 100%. Per N2-OP-91A, Process Computer, this sensor has direct input to the core thermal power calculation. Also per N2-OP-91A, the required compensatory action is to enter N2-REP-11.

C: Incorrect: Just the one point is failing. Plausible in that Tech Specs does specify that 3 NLOs are required if the computer is not available for longer than 8 hours.

D: Incorrect: Actual power level is not increasing. Plausible in that the three computer alarms would indicate that it is.

Technical Reference(s): N2-OP-91A, Attachment 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #2101-283001C01, Process Computer
RBO-11, System Loss and Component Level Malfunction (As available)

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

Question History:

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

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.1.5	
	Importance Rating		3.9

Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Question: SRO #94

The plant is in Mode One. Which of the following situations would justify the shift manager initiating a Work Hour Limits waiver IAW CNG-SE-1.01-1002 "FATIGUE MANAGEMENT AND WORK HOUR CONTROLS"?

Note: Assume that each of these situations would result in the operator(s) exceeding the 10CFR26 Work Hour Limits.

Situation 1: Holding shift operators over to assist in restoring the plant to rated conditions following a Recirc pump trip during the previous shift.

Situation 2: Holding an operator over past the operator's end of shift to complete a Tech Spec required surveillance so as to avoid a turnover in the middle of the surveillance.

Situation 3: Holding shift operators over to assist in restoring the plant equipment required to exit a 12 hour hot shutdown LCO.

A. None of the situations require a waiver

B. Situation 1

C. Situation 2

D. Situation 3

Answer: D

Explanation (Optional):

A: Incorrect: A waiver is justified for situation 3. This situation is considered a Condition Adverse to Safety as discussed in CNG-SE-1.01-1002.

B: Situation 1 does not justify a waiver. Plausible in that the recirc pump would have lowered power >20%. Waivers are justified if the activity would prevent an unplanned change in power of >20%. Per the note on page 7, it is not appropriate if the change has already occurred and the plant is stable

C: Incorrect: Situation 2 does not justify a waiver. Plausible in that CNG-SE-1.01-1002 does address tech spec required equipment but only as it relates to the recovery of failed equipment vice a surveillance.

D: Correct: Per CNG-SE-1.01-1002, this situation is a Condition Adverse to Safety and use of the waiver process is appropriate.

Technical Reference(s): CNG-SE-1.01-1002, Page 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP #S101-ADMTPSC16, Fatigue Management (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2.2.18	
	Importance Rating		3.9

Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Question: SRO #95

The plant is in mode 5. Procedure NIP-OUT-01, SHUTDOWN SAFETY, specifies that for the current plant conditions, the Shutdown Safety Criteria for:

- Reactor Decay Heat Removal is N=2.
- Reactor Inventory Letdown Control is N=1

The ONLY authorized systems available at this time to provide these Key Safety Functions are:

Reactor Decay Heat Removal	Reactor Inventory Letdown Control
RHS A SDC	Division 1 RHR Reject to Radwaste
RHS A SFC Assist	
Div 1 Alternate SDC	

How would the overall plant Shutdown Safety Color Code be classified when completing the Shift Outage Safety Assessment and the reason for that classification? Consider only these two Key Safety Functions (KSFs) when making this determination.

A. Green. Because both KSFs satisfy the Shutdown Safety Criteria

B. Orange. Because Reactor Inventory Letdown Control is not N+1

C. Yellow. Because Reactor Inventory Letdown Control is not N+1

D. Yellow. Because Reactor Decay Heat Removal is not N+2.

Answer: C

Explanation (Optional):

A: Incorrect: Green requires all KSFs to be N+1.

B: Incorrect: Orange is defined as one or more KSF are < N, with contingency plans in place. Both are > N.

C: Correct: Yellow is defined as all KSF are at least N. Both are at least N.

D: Incorrect: Yellow is correct but the reason is because Reactor Inventory Letdown Control is not N+1

Technical Reference(s): NIP-OUT-01, SHUTDOWN SAFETY, page 14, 38, 39, & 47. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NIP-OUT-01 pages 1 thru 25 , no attachments

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 (1) This is a procedure required to operate the facility. The procedures are part of the facility license and contain administrative conditions for operation.

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.13
	Importance Rating		3.8

Radiation Control: Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question: SRO #96

Given the following:

- Entry into the TIP Room is required.
- TIP Room is LOCKED and posted as VERY HIGH RADIATION AREA.
- General Supervisor Radiation Protection approval has been obtained.

Which one of the following identifies who controls the access key and the additional approval(s) required for entry per GAP-RPP-08, Control of High, Locked High and Very High Radiation Areas?

	Key Control	Additional Approval(s) Required
A.	Radiation Protection	Shift Manager AND Plant Manager
B.	Radiation Protection	Shift Manager ONLY
C.	Shift Manager	Shift Manager AND Plant Manager
D.	Shift Manager	Shift Manager ONLY

Answer: A

Explanation (Optional):

A: Correct. Per GAP-RPP-08 step 3.3.2; For Locked High Radiation Areas barriers, keys shall be controlled by RP. Per step 3.4.2; Entry into Very High Radiation Areas shall be approved by GSRP, the SM and the Plant General Manager.

B: Incorrect. Per step 3.4.2; Entry into Very High Radiation Areas shall be approved by GSRP, the SM and the Plant General Manager., not just the Shift Manager.

C: Incorrect. Key is controlled by RP for areas other than the Main Steam Tunnel, which is controlled by the Shift Manager.

D: Incorrect. Key is controlled by RP for areas other than the Main Steam Tunnel, which is controlled by the Shift Manager. Per step 3.4.2; Entry into Very High Radiation Areas shall be approved by GSRP, the SM and the Plant General Manager., not just the Shift Manager.

Technical Reference(s): GAP-RPP-08, Sect 3.3.2 and 3.4.2. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank #

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G2.4.20	
	Importance Rating		4.3

Knowledge of operational implications of EOP warnings, cautions, and notes.

Question: SRO #97

Following a transient, plant conditions are as follows:

- RPV Water Level is 170 inches, stable.
- ONLY High Pressure Core Spray (CSH) is injecting at 2000 gpm.
- Reactor Pressure is 900 psig.
- Suppression Pool Water Level is 195 feet.
- Suppression Pool Temperature is 155°F.
- Drywell Pressure is 3.5 psig.
- Suppression Chamber Pressure is 3.0 psig.

Which one of the following actions is required?

A. REDUCE CSH Injection Flow to avoid cavitation.

B. REDUCE CSH Injection Flow to avoid vortexing.

C. REDUCE RPV Pressure to protect Containment Integrity.

D. REDUCE RPV Pressure to allow Low Pressure Injection Systems to inject.

Answer: C

Explanation (Optional):

A: Incorrect - Suppression Pool Level is LOWER than normal, but does not exceed NPSH Limits for CSH Pump. (SP Temp < 235°F, complies with 5 psig NPSH curve)

B: Incorrect - Suppression Pool Level is LOWER than normal, but does not exceed Vortex Limits for CSH Pump. (SP Level is >192 feet)

C: Correct - With the Heat Capacity Limit violated 900 psig / 155°F the Suppression function may not protect from exceeding the Primary Containment Pressure Limit.

D: Incorrect - RPV Water Level is LOWER than normal, but does not require LP Injection.

Technical Reference(s): N2-EOP-PC, Heat Capacity Limit, N2-EOP-6 Attachment 29, CSH NPSH and Vortex Curves (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Heat Capacity Limit, CSH NPSH and Vortex Curves,

Learning Objective: LP #2101-EOPPCC01, Primary Containment Control EO-2, Operational Actions and Sequence (As available)

Question Source: Bank # WTS 1554

Modified Bank # (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #		G2.4.22
	Importance Rating		4.4

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Question: SRO #98

A LOCA is in progress with the following conditions:

- RPV water level is being maintained at 30 inches with Feedwater
- Suppression Pool Level is 200 feet
- Drywell Pressure is 10 psig and slowly rising
- Both recirc pumps and all drywell coolers are tripped
- Drywell Sprays are available but are not yet in service

With these conditions drywell temperature is reported to be 345 degrees and slowly rising.

Which one of the following describes the actions that are to be taken to control Primary Containment Temperature per the EOPs?

A. Immediately Spray the Drywell to lower drywell temperature. Then, immediately Blowdown, regardless of drywell temperature response, to avoid containment failure.

B. Immediately Spray the Drywell to lower drywell temperature. Blowdown only if sprays do not reduce drywell temperature.

C. Immediately Blowdown to lower drywell temperature. Then initiate drywell sprays.

D. Immediately Blowdown to lower drywell temperature. Do NOT initiate drywell sprays.

Answer: B

Explanation (Optional):

A: Incorrect - An immediate blowdown is not required since DW Sprays have not yet been placed in service and are required to be in service.

B: Correct - N2-EOP-PC directs spraying the drywell with the current conditions. Blowdown is not warranted, since all applicable actions have not yet been taken to reduce temperature. Per EOP bases, if drywell temperature is already above 340 degrees, sprays are still preferable to a blow down.

C: Incorrect - DW Sprays have not yet been placed in service and are required to be in service.

D: Incorrect – DW Sprays have not yet been placed in service and are required to be in service.

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

NMP2 EOP Bases, page 5-16.

Proposed References to be provided to applicants during examination: DSIL Curve

Learning Objective: LP 2101-EOPPCC01, N2-EOP-PC, EO-2, Operational Actions and Sequence (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		G2.2.36
	Importance Rating		4.2

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question: SRO #99

The plant is operating at rated conditions with Division III SWGR on Line 6. Offsite line maintenance has resulted in Line 5 being declared inoperable at 12:00 on May 1, 2010.

Given the following:

08:00 on May 3: Div 1 DG is declared INOP.
12:00 on May 3: Line 5 is declared OPERABLE

16:00 on May 4: Div 3 DG is declared INOP
14:00 on May 5: Div 1 DG is declared OPERABLE.

06:00 on May 6: Line 5 is declared INOP for ongoing line maintenance.
08:00 on May 6: Div 3 DG is declared OPERABLE (delay in obtaining parts).

Which one of the following describes correct implementation of Tech Specs for this sequence of events?

A. Entry into REQUIRED ACTION requiring a plant shutdown existed at NO TIME during this sequence of events. Restore line 5 to operable status no later than 06:00 on May 9.

B. Entry into REQUIRED ACTION requiring a plant shutdown existed at NO TIME during this sequence of events. Restore line 5 to operable status no later than 12:00 on May 7.

C. At 18:00 on May 4, TS 3.8.1, REQUIRED ACTION F.1 was required to be entered but this action was exited as permitted by LCO 3.0.2 before the shutdown was required to be initiated. Restore line 5 to operable status no later than 06:00 on May 9.

D. At 16:00 on May 4, TS 3.8.1 REQUIRED ACTION G.1 was required to be entered and the plant was required to be placed into MODE 3 but this action was not completed. Restore line 5 to operable status no later than 12:00 on May 7.

Answer: B

Explanation (Optional):

A: Incorrect - The statement for restoring Line 5 to operable ".....no later than 06:00 on May 9" is incorrect. This choice is based on Completion Time of 72 hours which started on 06:00 on May 6, but does not account for the limit imposed by the modified time zero (6 days from discovery of failure to meet the LCO from initial entry into Condition A at 12:00 on May 1). Not applying the modified time zero results in a completion time of 06:00 on May 9 results, which is incorrect.

B: Correct - Correctly applying the modified time zero on initial entry into CONDITION A when Line 5 is declared inoperable at 12:00 on May 1 results in Line 5 restoration with completion time 6 days later on May 7 at 12:00. The completion time of 72 hours which started on 06:00 on May 6 (last Line 5 inop condition results in completion time is 06:00 on May 9) is limited to less time because of the "modified time zero – 6 days from discovery of failure to meet the LCO" which IS DISCUSSED IN THE TS BASES. Per TS BASES B3.8.1, RA A.3 DISCUSSION: the third completion time for required action A.3 established a limit on the maximum time allowed for any combination of required AC sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. The 6 day completion time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. The "AND" connector between the 72 hours and 6 day completion times apply simultaneously, and the more restrictive must be met. The completion time of required action A.3 allows for an exception to the normal "time zero" for beginning the allowed outage time clock. This exception results in establishing the "time zero" at the time the LCO was INITIALLY NOT MET, instead of at the time Condition A was re-entered. Therefore, Line 5 must be operable no later than 6 days from declaring the LCO statement not met (which was 12:00 on May 1) which means that Line 5 must be operable by 12:00 on May 7 to avoid entering Condition F which is a Condition/Required Action for a plant shutdown.

C: Incorrect - IF Condition F or Condition G was required to be entered, then "entry into a REQUIRED ACTION" that required a plant shutdown existed. Condition F would be required if Associated Completion Time of Condition A,B,C,D or E is not met. The correct restoration time for restoring Line 5 is also incorrect. See discussion on choice A.

D: Incorrect - IF Condition F or Condition G was required to be entered, then "entry into a REQUIRED ACTION" that required a plant shutdown existed. Worst case was 2 sources inoperable when Div 1 and Div 3 EDGs were inoperable together between 5/4 1600 and 5/5 1400. Since at no time were "Three or more required AC sources inoperable", then Condition G was not required to be entered.

Technical Reference(s): TS LCO 3.8.1.1 & Bases (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.8.1 (No Bases)

Learning Objective: LP #2101-264001C01, Standby Diesel Generator and Auxiliaries
RBO-14, Application of Technical Specifications (As available)

Question Source: Bank # NRC 2005/WTS 1535
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments:

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		G2.1.32
	Importance Rating		4.0

Ability to explain and apply all system limits and precautions.

Question: SRO #100

The plant is at rated conditions.

Surveillance testing of RCIC is required and it is necessary to place "A" RHR into Suppression Pool Cooling.

Giving these conditions, which one of the following is required prior to placing "A" RHR into Suppression Pool Cooling and the bases for it?

A. Declare the Suppression Pool Cooling mode of "A" RHR inoperable. This is because the suppression pool cooling valves will be inoperable in the open position if a loss of offsite power and the associated diesel failed to start were to occur while aligned for suppression pool cooling.

B. Declare the LPCI mode of "A" RHR inoperable. This is because "A" RHR may not inject into the vessel within the analyzed time frame if a LOCA occurs while aligned for suppression pool cooling.

C. Declare the LPCI mode of "A" RHR inoperable. This is because a loss of offsite power while aligned for suppression pool cooling will drain the RHR loop and create a potential for subsequent water hammer.

D. Declare all modes of "A" RHR inoperable. This is because a loss of offsite power while aligned for suppression pool cooling will drain the RHR loop and create a potential for subsequent water hammer.

Answer: B

Explanation (Optional):

A: Incorrect: LPCI mode is declared inoperable. Plausible in that the situation described would be true.

B: Correct: LPCI is declared inoperable per procedure N2-OP-31 due to a concern that with the pump running injection may not occur within the analyzed time frame if a LOCA were to occur.

C: Incorrect: Plausible in that this is a precaution in OP-31 but is not the reason for declaring the system inoperable.

D: Incorrect: Plausible in that a drain down of the loop would affect all modes of operation but this is not the reason for declaring the system inoperable.

Technical Reference(s): N2-OP-31 Page, 41 (Attach if not previously provided)
DER-1998-000557

Proposed References to be provided to applicants during examination: None

Learning Objective: LP N2101205000C01, RHR, RBO-9, Precautions, Limitations, and Operations Fundamentals (As available)

Question Source: Bank # WTS 1530
Modified Bank # (Note changes or attach parent)
New

Question History:

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

10 CFR Part 55 Content: 55.41
55.43 2

Comments: