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

SRVs

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: #1

The plant is operating at 100% power when the low pressure liquid nitrogen supply from 2GSN-TK2 to the Drywell is lost due to a spurious isolation valve closure.

Which one of the following describes the effect of this malfunction on makeup to SRV accumulators?

Makeup is...

A. still automatically available to all pressure relief and ADS accumulators.

B. lost to the pressure relief accumulators, but is still available to the ADS accumulators.

C. lost to the ADS accumulators, but is still available to the pressure relief accumulators.

D. lost to both the pressure relief accumulators and the ADS accumulators.

Proposed Answer: B

Explanation: 2GSN-TK2 supplies makeup to the (18) 2ft^3 SRV pressure relief accumulators. 2GSN*TK4 and 2GSN*TK5 separately supply makeup to the (7) 10 ft^3 ADS accumulators. Therefore, isolation of the line from 2GSN-TK2 to the Drywell results in loss of makeup to the pressure relief accumulators, but not the ADS accumulators.

A. Plausible – Makeup is lost to the pressure relief accumulators. Plausible because TK4 and TK5 are unaffected by isolation of this line, so some accumulators still automatically have makeup.
C. Plausible – Makeup is lost to the pressure relief accumulators and not the ADS accumulators. Plausible because this would be the correct answer for isolation of TK4 and TK5.
D. Plausible – Makeup to the ADS accumulators is not lost. Plausible because makeup is lost to the pressure relief accumulators, and both makeup sources are nitrogen.

Technical Reference(s): N2-OP-34, PID-105B, PID-19D, PID-19E, PID-19F

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K1.07
	Importance Rating	3.9

EDGs

Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency core cooling systems

Question: #2

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

- A loss of all offsite power occurs.
- A coolant leak develops in the Drywell.
- Drywell pressure is 4 psig and rising slowly.
- 2EGS*EG2 trips on overspeed during start and CANNOT be started.

Given the following ECCS pumps:

(1) 2CSH*P1
(2) 2CSL*P1
(3) 2RHS*P1A
(4) 2RHS*P1B
(5) 2RHS*P1C

Which one of the following lists which of these ECCS pumps are running?

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

B. (1), (4), and (5) only

C. (2), (3), (4), and (5) only

D. (1), (3), (4), and (5) only

Proposed Answer: C

Explanation: With a loss of all offsite power and 2EGS*EG2 tripped, 2ENS*SWG102 is de-energized. This bus supplies power to 2CSH*P1, HPCS. The other four given pumps are powered from 2ENS*SWG101 or 103 and are running because a LOOP/LOCA signal is present.

A. Plausible – 2RHS*P1B and 2RHS*P1C are not running and 2CSH*P1, 2CSL*P1 and 2RHS*P1A are running. Plausible because this would be correct for the loss of 2EGS*EG3.
B. Plausible – 2CSL*P1 and 2RHS*P1A are not running and 2CSH*P1, 2RHS*P1B and 2RHS*P1C are running. Plausible because this would be correct for the loss of 2EGS*EG1.
D. Plausible – 2CSL*P1 is not running and 2CSH*P1, 2RHS*P1A, 2RHS*P1B and 2RHS*P1C are running. Plausible if the Core Spray pump power supplies are mixed up.

Technical Reference(s): N2-OP-100B Section F.1.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-5

Question Source: Modified Bank – Vision SYSID 32342

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K2.01
	Importance Rating	3.2

RPS

Knowledge of electrical power supplies to the following: RPS motor-generator sets

Question: #3

Which one of the following power supply failures would result in loss of power to RPS MG Set 2RPM-MG1A?

A. 2NNS-SWG011

B. 2NNS-SWG013

C. 2NNS-SWG014

D. 2NNS-SWG015

Proposed Answer: C

Explanation: 2NJS-US5 supplies power to 2NHS-MCC008 which supplies power to the RPS MG Set 2RPM-MG1A drive motor. 2NJS-US5 is powered from 2NNS-SWG014.

A. Plausible – This is a 4160 VAC bus that powers various 2NJS-US busses, similar to 2NNS-SWG014.
B. Plausible – This is a 4160 VAC bus that powers various 2NJS-US busses, similar to 2NNS-SWG014.
D. Plausible – This supplies power to RPS MG Set 2RPM-MG1B.

Technical Reference(s): EE-1AL, EE-1X, EE-1M

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-4

Question Source: Modified Bank – 2015 NRC #3

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	209002 K2.02
	Importance Rating	2.8

HPCS

Knowledge of electrical power supplies to the following: Valve electrical power: BWR-5,6

Question: #4

The plant is shutdown following a loss of coolant accident with the following:

- HPCS injected and raised Reactor water level until 2CSH*MOV107, PMP1 INJECTION VLV, automatically closed on high level.
- 5 minutes later, annunciator 601729, HPCS PRESS PUMP 2 VALVES MOT OVERLOAD alarms.
- Computer points indicate that 2CSH*MOV107 has a thermal motor overload condition.
- The CRS has directed dispatching an Operator to inspect the associated MCC.

Which one of the following describes the correct MCC to inspect and the response of 2CSH*MOV107 if Reactor water level lowers below 108.8"?

	MCC to Inspect	2CSH*MOV107 Response
A.	2EHS*MCC102	Remains closed
B.	2EHS*MCC102	Automatically opens
C.	2EHS*MCC201	Remains closed
D.	2EHS*MCC201	Automatically opens

Proposed Answer: D

Explanation: 2CSH*MOV107 is powered from 2EHS*MCC201. The thermal overload for 2CSH*MOV107 does not prevent automatic opening of the valve on low Reactor water level. The thermal overload only bypasses the open and close seal-in function from working, which makes the valve throttleable using the Panel 601 control switch. Therefore, if Reactor water level lowers below 108.8", the valve will automatically open.

A. Plausible – 2EHS*MCC201 is the correct MCC. Plausible because 2EHS*MCC102 supplies power to CSL valves. 2CSH*MOV107 will still automatically open on low Reactor water level. Plausible because on many components a thermal overload would completely remove power from the associated motor.
B. Plausible – 2EHS*MCC201 is the correct MCC. Plausible because 2EHS*MCC102 supplies power to CSL valves.
C. Plausible – 2CSH*MOV107 will still automatically open on low Reactor water level. Plausible because on many components a thermal overload would completely remove power from the associated motor.

Technical Reference(s): ARP 601729, N2-OP-33-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209002-RBO-4

Question Source: Modified Bank – 2012 Cert #3

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K3.01
	Importance Rating	3.4

DC Electrical Distribution

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Emergency generators: Plant-Specific

Question: #5

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

- 2EGS*EG1 is running loaded to 4100 KW for surveillance testing.
- Then, Division I 125 VDC 2BYS*SWG002A de-energizes due to a sustained electrical fault.

Which one of the following describes the response of 2EGS*EG1?

A. Trips and will NOT restart under LOCA conditions.

B. Trips, but is available to restart under LOCA conditions.

C. Continues to run, but is now unloaded with the output breaker tripped open.

D. Continues to run loaded, but protection is lost for the generator and output breaker.

Proposed Answer: A

Explanation: The loss of 2BYS*SWG002A results in loss of multiple loads related to 2EGS*EG1, including generator and output breaker control power and protection circuits, and generator control and starting circuits. The result is 2EGS*EG1 trips since it is running in test mode and it is unavailable to restart, even under LOCA conditions due to loss of both primary and secondary start circuitry.

B. Plausible – 2EGS*EG1 trips since it is running in test mode and will NOT restart. Plausible since it would not have tripped if originally running due to LOCA conditions.
C. Plausible – 2EGS*EG1 trips. Plausible because it would not have tripped if originally running due to LOCA conditions. Also plausible because various circuits are de-energized related to output breaker control and protection.
D. Plausible – 2EGS*EG1 trips. Plausible because it would not have tripped if originally running due to LOCA conditions. Also plausible because various circuits are de-energized related to generator and output breaker control and protection.

Technical Reference(s): N2-SOP-4, Attachment 2; N2-OP-100A, B.11.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264000-RBO-8

Question Source: Bank – Vision SYSID 32344

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K3.19
	Importance Rating	2.8

PCIS/Nuclear Steam Supply Shutoff

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Containment atmosphere sampling

Question: #6

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

- Drywell pressure trip unit C72-N650D fails upscale.
- 603401, RPS B DRYWELL PRESSURE HIGH TRIP, annunciator alarms.

Which one of the following describes the response of the H2/O2 analyzers?

	Div. 1 H2/O2 Analyzer	Div. 2 H2/O2 Analyzer
A.	Remains un-isolated and in service	Remains un-isolated and in service
B.	Isolates	Remains un-isolated and in service
C.	Remains un-isolated and in service	Isolates
D.	Isolates	Isolates

Proposed Answer: A

Explanation: The H2/O2 analyzers isolate as part of PCIS group 8 on either high Drywell pressure or low Reactor water level. Div. 1 PCIS (logic channels A and D) isolates the Div. 1 analyzer and Div. 2 PCIS (logic channels B and C) isolates the Div. 2 analyzer. Because only one Drywell trip unit failed, only a half isolation signal is received for Div. 1 PCIS. The logic requires 2 out of 2 to actually cause an isolation, therefore both H2/O2 analyzers remain un-isolated and in service.

B. Plausible – Both H2/O2 analyzers remain un-isolated and in service. Plausible because the logic arrangement for H2/O2 analyzer isolation is unique in that each PCIS Division is assigned to a separate H2/O2 analyzer and each PCIS Division only has 2 DW pressure inputs.
C. Plausible – Both H2/O2 analyzers remain un-isolated and in service. Plausible because the logic arrangement for H2/O2 analyzer isolation is unique in that each PCIS Division is assigned to a separate H2/O2 analyzer and each PCIS Division only has 2 DW pressure inputs.
D. Plausible – Both H2/O2 analyzers remain un-isolated and in service. Plausible because for most systems, each PCIS Division controls only outboard or inboard isolation valves and each PCIS Division only has 2 DW pressure inputs.

Technical Reference(s): ARP 603401, N2-OP-82, N2-OP-83

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223002-RBO-5

Question Source: Modified Bank – 2012 NRC #6

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K4.10
	Importance Rating	2.8

SLC

Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Over pressure protection

Question: #7

A failure to scram has occurred with the following:

- RRCS failed to actuate.
- Standby Liquid Control 2SLS*P1A, PMP 1A, keylock control switch has been placed in the PUMP A RUN position.
- 2SLS*MOV1A, SLC STORAGE TK OUTLET VLV, indicates open.
- 2SLS*P1A red light is illuminated.
- 2SLS*VEX3A ready light is illuminated.
- 2SLS*P1A discharge pressure is approximately 1650 psig.

Which one of the following describes the flow of boron solution?

The boron solution is going...

A. to the equipment drain system.

B. back to the suction of 2SLS*P1A.

C. to the Reactor through a Feedwater sparger.

D. to the Reactor through the HPCS injection line.

Proposed Answer: B

Explanation: 2SLS*VEX3A ready light being illuminated indicates that the squib failed to fire. Additionally, the discharge pressure of 1650 psig is abnormally high. This indicates that the explosive valve did not open. Boron solution is therefore flowing through 2SLS*RV2A (nominally set at 1600 psig) back to the suction of the pump.

A. Plausible – Boron solution is flowing back to the suction of the SLS pump. Plausible because relief valves in the CRD system relieve to the equipment drain system and many other discharge locations would be undesirable for boron solution (Torus, CSTs, Hotwell, etc.).
C. Plausible – Boron solution is flowing back to the suction of the SLS pump. Plausible because the Feedwater sparger is one possible injection points to the Reactor vessel.
D. Plausible – Boron solution is flowing back to the suction of the SLS pump. Plausible because the HPCS sparger is the normal injection point for SLS into the Reactor.

Technical Reference(s): PID-36A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-211000-RBO-3

Question Source: Modified Bank - 2015 Cert #23

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K4.01
	Importance Rating	3.7

IRM

Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Rod withdrawal blocks

Question: #8

The plant is operating in Mode 2 with the following IRM indications:

IRM	Range	Indication
A	4	79
B	4	83
C	4	113
D	4	93
E	4	97
F	4	87
G	4	110
H	4	103

Which one of the following describes the status of rod blocks and RPS?

A. NEITHER a rod block NOR a half scram is received.

B. A rod block is received, but NOT a half scram.

C. A rod block and only a half scram are received.

D. A rod block and full scram are received.

Proposed Answer: B

Explanation: During a plant startup, IRMs are normally ranged to keep indication between 25 and 75 on the 125 scale. The upscale rod block is received at 108 on the 125 scale. The upscale RPS trip is received at 120 on the 125 scale. For the given indications, IRMs C and G are causing a rod block, but all IRMs remain below the scram setpoint.

A. Plausible – A rod block is received. Plausible because all IRMs are less than the higher 120/125 threshold and only RPS A IRMs are above the 108/125 threshold.
C. Plausible – No half scram is received. Plausible because all IRMs are above the normal threshold of 75/125 and multiple IRMs are >100.
D. Plausible – A full scram is not received. Plausible because all IRMs are above the normal threshold of 75/125 and multiple IRMs are >100.

Technical Reference(s): ARPs 603207 & 603201

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K5.01
	Importance Rating	2.5

Instrument Air

Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Air compressors

Question: #9

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

- Instrument Air compressor 2IAS-C3B is running and selected as the lead compressor.
- Instrument Air pressure is 120 psig.
- The Instrument Air system is operating under normal air loading.

Then, 2IAS-C3B trips.

For the ABOVE CONDITIONS, Instrument Air compressor...

A. 2IAS-C3A will auto-start when air pressure reaches 85 psig.

B. 2IAS-C3A will auto-start when air pressure reaches 100 psig.

C. 2IAS-C3C will auto-start when air pressure reaches 85 psig.

D. 2IAS-C3C will auto-start when air pressure reaches 100 psig.

Proposed Answer: D

Explanation: With 2IAS-C3B running as the lead compressor, the compressor selector switch must be in the "BCA" position. This makes 2IAS-C3C the lag compressor, which will automatically start at 100 psig. This also makes 2IAS-C3A the backup compressor, which would automatically start at 85 psig, if air pressure lowered that far. However, 85 psig will not be reached because 2IAS-C3C will have already started and restored air pressure since air system loading is normal.

A. Plausible – 2IAS-C3C will auto-start and restore pressure, preventing the start of 2IAS-C3A. Plausible because 2IAS-C3A would also start at 85 psig if 2IAS-C3C failed to start, or if higher than normal air system loading were present.
B. Plausible – 2IAS-C3C will auto-start and restore pressure, preventing the start of 2IAS-C3A. Plausible if 2IAS-C3A were the lag compressor with 2IAS-C3B set as the lead compressor.
C. Plausible – 2IAS-C3C will start earlier at 100 psig. Plausible because 85 psig is the setpoint for the backup compressor auto-start.

Technical Reference(s): N2-SOP-19, N2-ARP-851238, N2-OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-RBO-5

Question Source: Bank – 2014 Cert #10

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K5.01
	Importance Rating	3.8

ADS

Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM: ADS logic operation

Question: #10

A plant transient has resulted in the following:

- RHR pumps B and C are the only ECCS pumps running.
- EOP's require that an RPV blowdown be executed.

Just prior to the blowdown, the following events/actions occur:

- A complete loss of Division II 125 VDC occurs.
- Then, all four ADS Logic Manual Initiation pushbuttons are armed and depressed.

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

The ADS valves...

A. open only after the ADS timer has timed out.

B. open immediately without further operator action.

C. remain closed, but may be opened using their "B" solenoids.

D. remain closed, but may be opened using their "A" solenoids.

Proposed Answer: D

Explanation: ADS valves remain closed because ADS logic Div I does not have an ECCS pump run permissive with only RHR pumps B and C running and ADS logic Div II does not have power. The "A" solenoids still have power from Div I DC, allowing manual initiation.

A. Plausible – ADS valves will not open without further action. Plausible because this would be the response to an auto initiation if a Div 1 ECCS Pump was running.
B. Plausible – ADS valves will not open without further action. Plausible because this would be the response to the manual action taken if a Div I ECCS pump was running.
C. Plausible – "B" ADS solenoids are powered from Div II DC (2BYS*PNL201B) and therefore have no power. Plausible because this would be the correct answer if the opposite division of 125 VDC power was lost.

Technical Reference(s): N2-OP-34

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-8

Question Source: Bank – 2010 Audit #4

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K6.01
	Importance Rating	3.4

LPCS

Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM: A.C. power

Question: #11

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

- LPCS is in a normal standby lineup.
- 600 VAC 2EJS*US1 de-energizes due to a sustained electrical fault.

Then, a loss of coolant accident results in a condition requiring automatic LPCS injection.

Which one of the following describes the response of LPCS?

2CSL*P1A...

A. does NOT start.

B. starts and injects to the Reactor vessel.

C. starts and runs on minimum flow to the CSTs.

D. starts and runs on minimum flow to the Suppression Pool.

Proposed Answer: D

Explanation: Loss of 2EJS*US1 results in loss of motive power for multiple LPCS MOVs, including 2CSL*MOV104, LPCS Injection Valve, and 2CSL*MOV107, LPCS Min Flow to Suppression Pool. The normal standby lineup has MOV104 initially closed and MOV107 initially open. With loss of power to these valves, they remain in these positions even upon receipt of the valid injection signal. 2CSL*P1A still has power from the upstream 4160 VAC 2ENS*SWG101 and initiation logic still functions with power from 2BYS*PNL201A. Therefore, 2CSL*P1A starts and runs on minimum flow through the open 2CSL*MOV107, but does not inject to the Reactor vessel. Minimum flow is routed to the Suppression Pool, not the CSTs.

A. Plausible – 2CSL*P1A still starts. Plausible that the logic would be affected by this power loss and prevent pump start.
B. Plausible – The pumps runs on min flow and does not inject to the Reactor vessel. Plausible that the standby lineup would have injection aligned or a UPS would still power the injection valve.
C. Plausible – The pump runs with min flow to the Suppression Pool, not the CSTs. Plausible that the minimum flow would be to the CSTs since they are another large water source that receive various pump discharges (ex. RCIC full flow test line).

Technical Reference(s): Emergency AC Big Note, N2-OP-32, N2-OP-32-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209001-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K6.11
	Importance Rating	4.1

RHR/LPCI: Injection Mode

Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): ADS

Question: #12

The plant has experienced a loss of coolant accident with the following:

- N2-EOP-C2, RPV Blowdown, is in progress.
- Multiple ADS and SRV failures have occurred.
- Reactor pressure is 400 psig and very slowly lowering.
- Reactor water level is -60" and slowly lowering.
- All RHR pumps are running.

Which one of the following describes the status of 2RHS*MOV24A(B)(C), LPCI Injection Valve, and 2RHS*V16A(B)(C), LPCI Injection Line Check Valve?

	2RHS*MOV24A(B)(C)	2RHS*V16A(B)(C)
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: B

Explanation: Multiple ADS and SRV failures have resulted in a higher than desired Reactor pressure in a situation where injection from low pressure systems is required. Operating RHR pumps produce a discharge pressure of approximately 300 psig. With Reactor water level below 17.8" and Reactor pressure within 130 psid above RHR pump discharge pressure, 2RHS*MOV24A(B)(C) automatically open. However, until Reactor pressure lowers below RHR pump discharge pressure, 2RHS*V16A(B)(C) remain closed.

A. Plausible – Reactor pressure is still too high for 2RHS*V16A(B)(C) to open. Plausible because this would be the correct answer if Reactor pressure were lower.
C. Plausible – 2RHS*MOV24A(B)(C) has automatically opened since Reactor pressure is within 130 psig above RHR pump discharge pressure. Plausible because 2RHS*MOV24A(B)(C) was closed until Reactor pressure lowered to near the current value. 2RHS*V16A(B)(C) is closed. Plausible if the valve were immediately on the RHR pump discharge such that min flow went through this valve.
D. Plausible – 2RHS*MOV24A(B)(C) has automatically opened since Reactor pressure is within 130 psig above RHR pump discharge pressure. Plausible because 2RHS*MOV24A(B)(C) was closed until Reactor pressure lowered to near the current value. 2RHS*V16A(B)(C) is closed.

Technical Reference(s): N2-OP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-5

Question Source: Bank – Peach Bottom 2008 NRC #1

Question History: Peach Bottom 2008 NRC #1

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K6.02
	Importance Rating	2.8

UPS (AC/DC)

Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.): D.C. electrical power

Question: #13

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

- The 2VBB-UPS3A and B Manual Transfer Switches are selected to STATIC SWITCH position
- A complete loss of 2NJS-US1 occurs
- 10 seconds after 2NJS-US1 is lost, a complete loss of 2BYS-SWG001C occurs.

Which one of the following describes the effect of these electrical losses on 2VBB-UPS3A or 2VBB-UPS3B?

A. 2VBB-UPS3A loads are de-energized.

B. 2VBB-UPS3B loads are de-energized.

C. 2VBB-UPS3A loads automatically receive power from the maintenance supply via the static switch.

D. 2VBB-UPS3B loads automatically receive power from the maintenance supply via the static switch.

Proposed Answer: C

Explanation: UPS-3A receives normal AC power from 2NJS-US1, backup battery power from 2BYS-SWG001C, and maintenance power from 2NJS-US5. With the Manual Transfer Switches in the STATIC SWITCH position, the UPS's are able to automatically transfer to the maintenance supply if normal AC and DC inputs are lost. Therefore, when 2NJS-US1 and 2BYS-SWG001C de-energize, UPS-3A automatically receives power from the maintenance power supply, 2NJS-US5.

A. Plausible – UPS-3A automatically receives power from the maintenance power supply, 2NJS-US5, therefore it does not de-energize. Plausible because this would be the correct answer if the Manual Transfer Switch was in a different position or if 2NJS-US5 were lost.
B. Plausible – UPS-3B remains energized from the normal AC power source, 2NJS-US4, which is unaffected by the other electrical losses. Plausible because the power supplies to UPS-3A and 3B are similar.
D. Plausible – UPS-3B remains energized from the normal AC power source, 2NJS-US4, which is unaffected by the other electrical losses. Plausible because the power supplies to UPS-3A and 3B are similar.

Technical Reference(s): N2-OP-71D Section B, UPS Big Note, and EE-M001D

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262002-RBO-4

Question Source: Bank – 2015 NRC #12

Question History: 2015 NRC #12

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A1.02
	Importance Rating	2.8

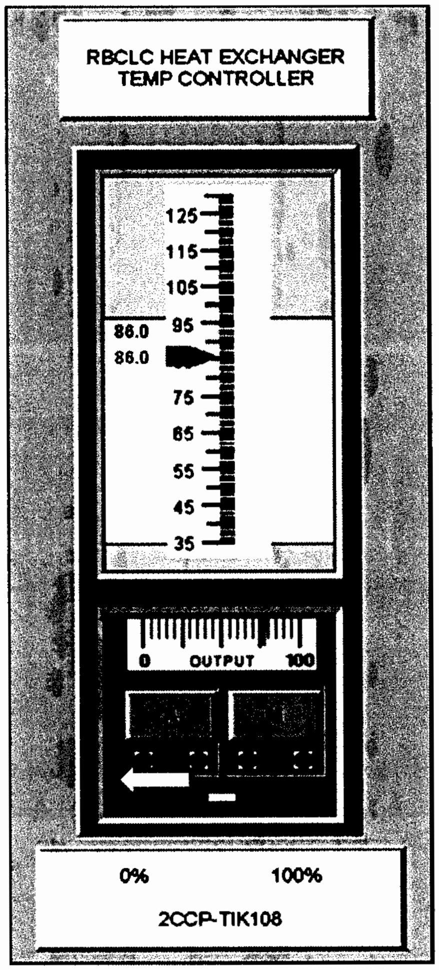
Component Cooling Water

Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: CCW temperature

Question: #14

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

- 2CCP-TIK108, RBCLC Heat Exchanger Temp Controller, has been placed in Manual due to failure of Automatic temperature control.
- The manual slider on 2CCP-TIK108 is pushed in the direction indicated by the arrow in the picture below:



Which one of the following describes the effect of this manipulation on RBCLC cooling water temperature?

RBCLC cooling water temperature...

A. lowers because RBCLC flow through RBCLC heat exchangers rises.

B. rises because RBCLC flow through RBCLC heat exchangers lowers.

C. lowers because Service Water flow through RBCLC heat exchangers rises.

D. rises because Service Water flow through RBCLC heat exchangers lowers.

Proposed Answer: A

Explanation: With 2CCP-TIK108 in Manual, pushing the slider to the left raises cooling by directing more RBCLC flow through the RBCLC heat exchangers. This causes temperature to lower.

B. Plausible – RBCLC flow through the HX's rises, not lowers. Plausible because this would be the correct answer if the arrow were reversed and because the controller is setup opposite of most controllers, where going to the left lowers flow.
C. Plausible – This operation adjusts RBCLC flow, not Service Water flow. Plausible because Service Water does flow through the other side of the heat exchangers and varying Service Water flow is another physically possible method of controlling RBCLC temperature.
D. Plausible – This operation adjusts RBCLC flow, not Service Water flow. Plausible because Service Water does flow through the other side of the heat exchangers and varying Service Water flow is another physically possible method of controlling RBCLC temperature.

Technical Reference(s): N2-OP-13, PID-13C

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-208000-RBO-5

Question Source: Modified Bank - 2010 NRC #12

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	205000 A2.08
Importance Rating	3.3

Shutdown Cooling

Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of heat exchanger cooling

Question: #15

The plant is in Mode 3 with the following:

- RHR B is operating in Shutdown Cooling (SDC).
- Both Recirc pumps are secured.
- Head spray is isolated.
- Reactor coolant temperature is 250°F and stable.
- The following valves are all throttled at an intermediate position:
  - 2RHS*MOV40B, SDC B Return Throttle
  - 2RHS*MOV8B, Heat Exchanger 1B Inlet Bypass Valve Throttle
  - 2SWP*MOV33B, Heat Exchanger 1B Service Water Outlet Valve
- RHR B total flow is 7000 gpm.
- Service Water to RHR B Heat Exchanger flow is 2000 gpm.
- Service Water temperature is 74°F.
- Five (5) Service Water pumps are in operation.

Then, the following occurs:

- 2RHS*MOV8B spuriously opens fully.
- RHR B total flow rises to 7450 gpm.
- 2RHS*MOV8B CANNOT be closed.

Which one of the following describes the resulting trend in Reactor coolant temperature and an allowable operator action to stabilize reactor coolant temperature, in accordance with N2-OP-31, Residual Heat Removal System?

	Reactor Coolant Temperature Trend	Allowable Operator Action
A.	Rises	Throttle 2RHS*MOV40B
B.	Rises	Throttle 2SWP*MOV33B
C.	Lowers	Throttle 2RHS*MOV40B
D.	Lowers	Throttle 2SWP*MOV33B

Proposed Answer: B

Explanation: 2RHS*MOV8B opening further causes flow through the RHR heat exchanger to lower. This causes less cooling. Since Reactor coolant temperature was stable with the initial flows, lowering the heat removal rate will cause temperature to rise. 2RHS*MOV40B cannot be throttle further open because the upper limit on RHR total flow is 7450 gpm, and this is already being slightly exceeded. 2SWP*MOV33B can be throttled further open to raise cooling. The upper limit on Service Water to RHR B Heat Exchanger flow is 7400 gpm, so margin still exists to this limit.

A. Plausible – 2RHS*MOV40B cannot be throttle further open because the upper limit on RHR total flow is 7450 gpm, and this is already being slightly exceeded. Plausible because this would physically have the intended effect on Reactor coolant temperature, but is procedurally not allowed.
C. Plausible – Reactor coolant temperature rises, not lowers. Plausible because the malfunction does cause RHR total flow to rise, however the additional flow is bypassing the heat exchanger, not going through it, so the net effect is less cooling.
D. Plausible – Reactor coolant temperature rises, not lowers. Plausible because the malfunction does cause RHR total flow to rise, however the additional flow is bypassing the heat exchanger, not going through it, so the net effect is less cooling.

Technical Reference(s): N2-OP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A2.01
	Importance Rating	3.3

Reactor Water Level Control

Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL 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 any number of main steam flow inputs

Question: #16

The plant is operating at 95% power when the steam flow signal for Main Steam Line B fails downscale.

Which one of the following describes automatic reactor water level response and the Immediate Operator Action, in accordance with N2-SOP-06, Feed Water Failures?

	Reactor water level	Immediate Operator Action
A.	Rises	Place 2FWS-HIC1600, FEEDWATER LEVEL MASTER CONTROL, in MAN and adjust flow.
B.	Rises	Place C33A-S2, 1 ELEMENT/3 ELEMENT FEEDWATER CONTROL, in 1 ELEMENT position.
C.	Lowers	Place 2FWS-HIC1600, FEEDWATER LEVEL MASTER CONTROL, in MAN and adjust flow.
D.	Lowers	Place C33A-S2, 1 ELEMENT/3 ELEMENT FEEDWATER CONTROL, in 1 ELEMENT position.

Proposed Answer: C

Explanation: Feedwater level control sees lower steam flow than feed flow, and therefore anticipates Reactor water level rising. Actual feed flow is lowered while actual steam flow remains the same, so Reactor water level lowers. Feed flow/steam flow mismatch associated with loss of a single steam flow input results in approximately a 12" level deviation. Resultant water level would lower to approximately 171". This would bring in annunciator 603139, Reactor Water Level High/Low, and make N2-SOP-6 entry correct. The immediate Operator action in N2-SOP-6 allows placing 2FWS-HIC1600, FEEDWATER LEVEL MASTER CONTROL, in MAN and adjusting flow.

A. Plausible – Level lowers, not rises. Plausible because level would rise if the opposite failure occurred.
B. Plausible – Level lowers, not rises. Plausible because level would rise if the opposite failure occurred.
D. Plausible – The allowable immediate Operator action is placing 2FWS-HIC1600, FEEDWATER LEVEL MASTER CONTROL, in MAN and adjusting flow. Plausible because placing FWLC in single element would remove the erroneous steam flow measurement from the control circuit and is an eventual possible solution in this situation.

Technical Reference(s): N2-259002-RBO-11, N2-SOP-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-259002-RBO-11

Question Source: Modified Bank - 2012 Cert #42

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 A3.04
	Importance Rating	3.6

RCIC

Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: System flow

Question: #17

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

- N2-OSP-ICS-Q@002, RCIC Pump and Valve Operability Test, is in progress.
- RCIC is operating in the full flow test mode with the RCIC controller in Automatic.

Then…

- A Reactor scram occurs due to trip of running Feedwater pumps.
- Reactor water level drops to 100".
- HPCS fails to start.

Which one of the following describes the RCIC flow rate and flow path 2 minutes later?

RCIC flow is approximately…

A. 220 gpm to the CST via the test return line.

B. 220 gpm to the Suppression Pool via the minimum flow line.

C. 600 gpm to the CST via the test return line.

D. 600 gpm to the Reactor via the Reactor head spray nozzle.

Proposed Answer: D

Explanation: RCIC is initially running in AUTO with 600 gpm to the CST through the test return line. With Reactor water level <108.8", RCIC receives an injection signal. This signal isolates the full flow test path to the CST and opens the injection path to the Reactor. With the RCIC flow controller in AUTO, RCIC will inject 600 gpm to the Reactor. The RCIC system is designed to come up to rated speed and injection within 30 seconds.

A. Plausible – RCIC will inject 600 gpm to the Reactor. Plausible because the initial flow path was to the CST and 220 gpm is the min flow rate.
B. Plausible – RCIC will inject 600 gpm to the Reactor. Plausible because 220 gpm is the min flow rate and the Suppression Pool is the normal min flow path.
C. Plausible – RCIC will inject 600 gpm to the Reactor. Plausible because this is the initial flow rate and path in N2-OSP-ICS-Q@002.

Technical Reference(s): N2-OSP-ICS-Q@002, N2-OP-35

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-217000-RBO-5

Question Source: Bank – 2014 Cert #17

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	215004 A3.04
Importance Rating	3.6

Source Range Monitor

Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: Control rod block status

Question: #18

A normal plant startup is in progress following a routine refueling outage with the following:

- All IRMs are on range 1.
- SRMs indicate as follows:

SRM	Indication (cps)	Position
A	7×10^4	Fully inserted
B	5×10^2	Partially withdrawn
C	3×10^5	Fully inserted
D	1×10^3	Partially withdrawn

Which one of the following describes the status of these SRM indications?

A. All SRM indications are within the preferred band of N2-OP-100A, Plant Startup.

B. One or more SRMs are indicating outside the preferred band of N2-OP-100A, Plant Startup, but NO rod block is being generated.

C. SRMs are causing a rod block to be generated, but NO scram signal is being generated.

D. SRMs are causing a scram signal to be generated.

Proposed Answer: C

Explanation: N2-OP-101A directs maintaining SRM indications between 1×10^2 and 1×10^5 cps by withdrawing SRM detectors as required. SRMs A, B, and D are within this band. SRM C is above this band. The upper end of the band, 1×10^5 cps, is also the upscale rod block setpoint. Therefore, SRM C is causing a rod block to be generated. Since a startup is being performed per N2-OP-101A, the non-coincident shorting links are installed, such that SRMs will not cause a scram signal to be generated.

A. Plausible – SRM C is above the upper end of the band. Plausible because SRMs A, B, and D are all within the band.
B. Plausible – SRM C is causing a rod block to be generated. Plausible because SRMs A, B, and D are all OK.
D. Plausible – No scram signal is being generated. Plausible because this can occur if special testing or initial fuel loading is in progress that removes the shorting links. The setpoint during these conditions is 2×10^5 cps.

Technical Reference(s): N2-OP-101A, N2-OP-92, ARP 6032023

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 A4.03
	Importance Rating	3.0

SGTS

Ability to manually operate and/or monitor in the control room: Fan

Question: #19

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

- A steam leak develops in the Reactor Building
- Both trains of Standby Gas Treatment (GTS) start based on Reactor Building exhaust radiation levels.
- An Operator secures GTS fan A in accordance with N2-OP-61B, Standby Gas Treatment System, with the initiation signal still present.

3 minutes later, Reactor Building differential pressure drops to -0.10" WG.

Which one of the following describes the position of the GTS Train A initiation switch based on the Operator's actions, in accordance with N2-OP-61B, and the current status of GTS fan A?

	GTS Train A Initiation Switch Position	GTS Fan A
A.	Pull-to-lock	Secured
B.	Pull-to-lock	Running
C.	Auto After Stop	Secured
D.	Auto After Stop	Running

Proposed Answer: D

Explanation: N2-OP-61B section H.2.0 provides direction for securing a GTS train with an initiation signal still present. The GTS train control switch is placed and left in AUTO AFTER STOP. While this secures the fan initially, the fan automatically re-starts because Reactor Building differential pressure drops less than -0.25" WG.

A. Plausible – The control switch is in Auto After Stop. Plausible because many components would need their control switch in pull-to-lock to be secured with an initiation signal present. GTS fan A automatically restarts. Plausible because Reactor Building D/P is still maintained negative and the fan was stopped with an initiation signal present.
B. Plausible – The control switch is in Auto After Stop. Plausible because many components would need their control switch in pull-to-lock to be secured with an initiation signal present.
C. Plausible – GTS fan A automatically restarts. Plausible because Reactor Building D/P is still maintained negative and the fan was stopped with an initiation signal present.

Technical Reference(s): N2-OP-61B

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-261000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	215005 A4.03
Importance Rating	3.2

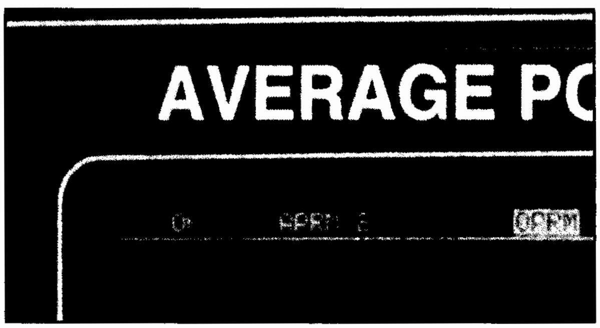
APRM / LPRM

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

Question: #20

The plant was operating at 100% power when a transient resulted in the following:

- Indications on the Average Power Range Monitor Chassis for APRMs 1 through 4 have "OPRM" displayed in inverse video, as shown in the sample picture below.



Which one of the following conditions is indicated by these displays?

OPRM trips and alarm setpoints are ___(1)___.

Reactor power is ≥23% ___(2)___ core flow is ≤75%.

A. (1) enabled
(2) OR

B. (1) enabled
(2) AND

C. (1) NOT enabled
(2) OR

D. (1) NOT enabled
(2) AND

Proposed Answer: B

Explanation: With OPRM displayed in inverse video, OPRM trips and alarm setpoints are enabled and power must be ≥23% AND core flow must be ≤75%.

A. Plausible – Both Reactor power and core flow conditions must be satisfied. Plausible that only one is required for this indication, since only one of the opposite conditions is required to clear this indication.
C. Plausible – This indicates OPRMs are enabled. Plausible because it is equally possible that this indication is for disabled OPRMs. Both Reactor power and core flow conditions must be satisfied. Plausible that only one is required for this indication, since only one of the opposite conditions is required to clear this indication.
D. Plausible – This indicates OPRMs are enabled. Plausible because it is equally possible that this indication is for disabled OPRMs.

Technical Reference(s): N2-OP-92

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-5

Question Source: Bank – 2012 NRC #19

Question History: 2012 NRC #19

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 2.2.12
	Importance Rating	3.7

AC Electrical Distribution

Knowledge of surveillance procedures.

Question: #21

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

- The regular monthly start and load test is being performed for the Division III Diesel Generator (2EGS*EG2) per N2-OSP-EGS-M@002, Diesel Generator and Diesel Air Start Valve Operability Test – Division III, Section 8.2.

Which one of the following describes the how 2EGS*EG2 is operated during this test?

2EGS*EG2 is started…

A. manually from Panel 852 and then manually paralleled with 2ENS*SWG102. 2ENS*SWG102 remains paralleled with offsite power throughout the test.

B. manually from Panel 852 and then manually paralleled with 2ENS*SWG102. 2ENS*SWG102 is disconnected from offsite power for a portion of the test.

C. by a simulated undervoltage signal and then manually paralleled to 2ENS*SWG102.

D. by an actual undervoltage signal and automatically connects to 2ENS*SWG102.

Proposed Answer: A

Explanation: N2-OSP-EGS-M@002 section 8.2 requires starting 2EGS*EG2 manually from Panel 852 and manually paralleling it with 2ENS*SWG102. 2ENS*SWG102 remains paralleled with offsite power throughout the test.

B. Plausible – 2ENS*SWG102 remains paralleled with offsite power throughout the test. Plausible that this bus would be separated from offsite power to test the EDGs capability to carry the bus by itself. Also plausible because this is done in some other test sections of the procedures.
C. Plausible – 2EGS*EG2 is manually started from Panel 852. Plausible because simulated undervoltage signals are used for testing related to 2ENS*SWG102 in N2-OSP-ENS-M001.
D. Plausible – 2EGS*EG2 is manually started from Panel 852. Plausible because other surveillance testing starts Diesel Generators by an actual undervoltage signal (N2-OSP-EGS-R003).

Technical Reference(s): N2-OSP-EGS-M@002

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 2.4.20
	Importance Rating	3.8

PCIS/Nuclear Steam Supply Shutoff

Knowledge of operational implications of EOP warnings, cautions, and notes.

Question: #22

The plant has experienced a LOCA. Conditions are as follows:

- Reactor water level is -10" and stable.
- RCIC is injecting at 600 gpm.
- NO other injection systems are available.

Then...

- Annunciator 601332, RCIC EQUIP ROOM TEMPERATURE HIGH, alarms.
- RCIC equipment room temperature is 100°F and slowly rising.
- An Operator reports a steam leak is present in the RCIC equipment room.

Which one of the following describes the required response, in accordance with the Emergency Operating Procedures?

A. Shutdown and isolate RCIC immediately.

B. Defeat the RCIC high area temperature isolation and continue use of RCIC for Reactor injection.

C. Continue use of RCIC for Reactor injection until a high room temperature isolation occurs. Then if Reactor water level cannot be maintained above -14", perform Steam Cooling.

D. Continue use of RCIC for Reactor injection until a high room temperature isolation occurs. Then if Reactor water level cannot be maintained above -14", perform an RPV Blowdown.

Proposed Answer: B

Explanation: Reactor water level is just above the top of active fuel and stable with RCIC at maximum flow. Additionally, no other injection systems are available to replace RCIC. Therefore RCIC is required by N2-EOP-RPV to maintain adequate core cooling, the main mitigating strategy of the procedure. This is in contrast to the mitigating strategy of N2-EOP-SC, which would lead to eventual isolation of RCIC. The overall EOP strategy places greater importance on adequate core cooling (as noted in N2-EOP-SC step SC-5), thus N2-EOP-RPV (and, as a result, ARP 601332) directs defeating the RCIC high area temperature isolation using N2-EOP-6.2 and continuing injection using RCIC (as noted in N2-EOP-RPV Table E1).

A. Plausible – This would be correct if RCIC was not necessary to maintain adequate core cooling. Since RCIC is needed to maintain adequate core cooling, it should be left in service and have the high temperature isolation bypassed.
C. Plausible – The high temperature isolation should be bypassed to allow continued RCIC operation for adequate core cooling. Plausible that RCIC would be allowed to run until automatic isolation to maximize injection time.
D. Plausible – The high temperature isolation should be bypassed to allow continued RCIC operation for adequate core cooling. Plausible that RCIC would be allowed to run until automatic isolation to maximize injection time. With no other injection systems available, if RCIC trips/isolates, Steam Cooling would be performed, not RPV Blowdown. Plausible because this would be correct if any other injection source were also running, such as a CRD pump.

Technical Reference(s): N2-EOP-RPV, N2-EOP-SC, ARP 601332

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-217000-RBO-12

Question Source: Bank – 2015 NRC #22

Question History: 2015 NRC #22

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 A4.01
	Importance Rating	3.7

Shutdown Cooling

Ability to manually operate and/or monitor in the control room: SDC/RHR pumps

Question: #23

The plant is operating in Mode 3 with the following:

Time (hh:mm)	Condition(s)
00:00	• RHR pump A is started for Shutdown Cooling.
00:15	• RHR pump A trips due to a worker bumping into the breaker. • Electrical Maintenance confirms that the breaker is ready to be re-closed.

Which one of the following identifies the *earliest* time that RHR pump A can be restarted without exceeding motor start limitations, in accordance with N2-OP-31, Residual Heat Removal System?

Note: Assume any other operational concerns with pump restart have been addressed satisfactorily.

A. Immediately

B. Time 00:30

C. Time 00:45

D. Time 01:00

Proposed Answer: A

Explanation: N2-OP-31 P&L 3.0 addresses motor start limitations for RHR pumps:

3.0 Observe the following RHR pump motor start limitations:

- Two starts in succession from ambient temperature after which a 60 minute wait is required prior to subsequent start attempts.
- One start from rated temperature (established after 30 minutes run time), after which a 60 minute wait is required prior to subsequent start attempts.

In the given conditions, the first bullet is applicable because the RHR pump has been running for less than 30 minutes. This bullet allows two starts. Only one start has been performed so far, at Time 00:00. Therefore, immediate RHR pump restart is allowed. If another restart is required later, a 60 minute wait would be required.

Note: The question meets the K/A because it presents a situation where RHR is operating in the Shutdown Cooling mode and tests the candidate's ability to properly operate an RHR pump (deciding when it can be restarted while adhering to pump motor start limitations) and the need to perform specific monitoring of temperatures.

B. Plausible – Immediate restart is allowed. Plausible because P&L 3.0 includes a 30 minute restriction and Time 00:30 is 30 minutes after the original pump start.
C. Plausible – Immediate restart is allowed. Plausible because P&L 3.0 includes a 30 minute restriction and Time 00:45 is 30 minutes after the pump trip.
D. Plausible – Immediate restart is allowed. Plausible because P&L 3.0 includes a 60 minute restriction and Time 01:00 is 60 minutes after the original pump start.

Technical Reference(s): N2-OP-31 P&L 3.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-9

Question Source: Modified Bank - 2012 Cert #54

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 2.1.25
	Importance Rating	3.9

RHR/LPCI: Injection Mode

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Question: #24

The plant has experienced a major accident with the following:

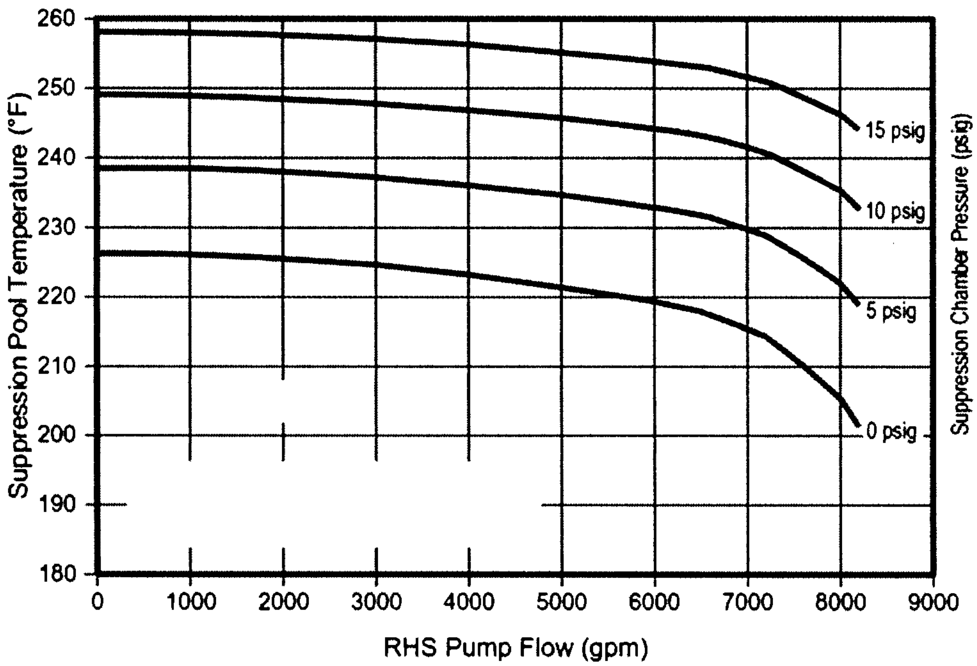
- Reactor water level is 125" and slowly rising.
- Reactor pressure is 100 psig and slowly lowering.
- Drywell pressure is 7 psig and slowly rising.
- Drywell average temperature is 200°F and slowly rising.
- Suppression Chamber pressure is 5 psig and slowly rising.
- Suppression Pool water temperature is 220°F and stable.
- Suppression Pool water level is 192.5' and stable.
- RHR loop A is injecting 7,000 gpm to the Reactor.

Note: Portions of N2-EOP-6.29 Figure 4, RHS NPSH and Vortex Limits, are provided on the following page.

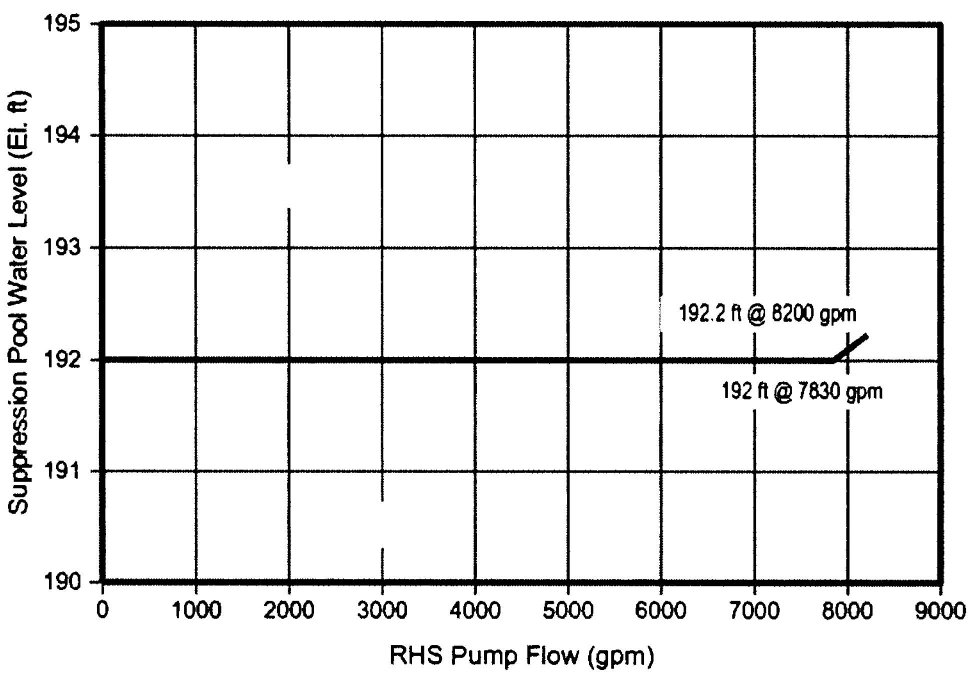
Which one of the following describes whether operation of RHR loop A is in the GOOD or BAD region of the NPSH Limit and the Vortex Limit?

	NPSH Limit	Vortex Limit
A.	GOOD	GOOD
B.	GOOD	BAD
C.	BAD	GOOD
D.	BAD	BAD

RHS NPSH Limit



RHS Vortex Limit



Proposed Answer: A

Explanation: The given conditions are between the 0 and 5 psig curves on the NPSH limit curve. The 5 psig curve is applicable based on current Suppression Chamber pressure. Operation is in the area below the curve, which is the GOOD region. The given conditions are above the Vortex limit curve, which is the GOOD region.

B. Plausible – Operation is in the GOOD region of the Vortex limit curve. Plausible because Suppression Pool level is much lower than normal, which challenges this limit, and the GOOD/BAD labels have been removed from the graph.
C. Plausible – Operation is in the GOOD region of the NPSH limit curve. Plausible because Suppression Pool water temperature is much higher than normal, which challenges this limit, operation is between two curves, and the GOOD/BAD labels have been removed from the graph.
D. Plausible – Operation is in the GOOD region of the NPSH limit curve. Plausible because Suppression Pool water temperature is much higher than normal, which challenges this limit, operation is between two curves, and the GOOD/BAD labels have been removed from the graph. Operation is in the GOOD region of the Vortex limit curve. Plausible because Suppression Pool level is much lower than normal, which challenges this limit, and the GOOD/BAD labels have been removed from the graph.

Technical Reference(s): N2-EOP-6.29

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K5.02
	Importance Rating	3.3

RPS

Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements

Question: #25

The plant is operating at 40% power with the following:

- Turbine Stop Valve (TSV) 1D drifts 50% closed.
- Then, TSV 1C drifts 50% closed.

Considering only the Turbine Stop Valve Closure scram signal, which one of the following describes the resulting operation of the Reactor Protection System?

Based on the Turbine Stop Valve Closure scram signal, a full Reactor scram…

A. occurs while TSV 1D is drifting.

B. occurs while TSV 1C is drifting.

C. does NOT occur because only these two TSVs have drifted.

D. does NOT occur because these two TSVs have NOT tripped their scram position switches.

Proposed Answer: C

Explanation: TSVs provide an input to RPS when they are less than 95% open. The logic is arranged such that this signal causes a scram only when 3 of the 4 TSVs are less than 95% open. With only 2 TSVs drifted, a full scram does not occur.

A. Plausible – A full scram does not occur. Plausible because TSV 1D drifting does provide inputs to RPS and has the potential to affect Reactor pressure and power.
B. Plausible – A full scram does not occur. Plausible because with TSV 1C and 1D drifted less than 95% open, a half scram would occur.
D. Plausible – These TSVs have drifted far enough to trip their position switches that input to RPS. Plausible because they have only drifted 50%, which still allows the majority of the flow through the valve.

Technical Reference(s): N2-212000-RBO-3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-3

Question Source: Bank – JAF 2016 NRC #10

Question History: JAF 2016 NRC #10

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K4.03
	Importance Rating	3.8

ADS

Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: ADS logic control

Question: #26

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is +5" and stable on Fuel Zone.
- Reactor water level is downscale on Wide Range and Narrow Range.
- Reactor pressure is 900 psig and slowly lowering.
- All low pressure ECCS pumps are operating properly.
- ADS has opened all ADS valves.

Then, per Shift Manager direction, the DIVISION 1 AND 2 LOGIC INITIATION SEAL-IN RESET pushbuttons are simultaneously depressed and released.

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

The ADS valves...

A. remain open, even while the pushbuttons are depressed.

B. close when the pushbuttons are depressed and then immediately re-open when the pushbuttons are released.

C. close, but will re-open in 105 seconds if conditions requiring ADS actuation remain present.

D. close and remain closed unless conditions requiring ADS actuation clear and then re-occur for at least 105 seconds.

Proposed Answer: C

Explanation: When the logic reset pushbuttons are both pushed, the 105 second timers are reset. This closes the ADS valves. When the pushbuttons are released, the timers will begin timing again, since Reactor water level is still low enough to require ADS actuation (<17.8" on Wide Range), and ADS valves will open after 105 seconds.

A. Plausible – The valves close. Plausible because the timers have already timed out, so resetting the timers could not close the valves if the logic sealed in.
B. Plausible – The valves only open after 105 seconds passes. Plausible that these buttons just remove the signal to the valves and do not reset the 105 second timers, such that the valves re-open immediately.
D. Plausible – The valves open even if the signal does not clear and the re-occur. Plausible that the reset pushbuttons seal-in until the signal clears.

Technical Reference(s): N2-OP-34, 807E155TY Sheet 5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-5

Question Source: Modified Bank - 2014 Cert #20

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290001 K1.09
	Importance Rating	2.9

Secondary Containment

Knowledge of the physical connections and/or cause-effect relationships between SECONDARY CONTAINMENT and the following: Plant air systems

Question: #27

The plant is operating at 100% power when an un-isolable Instrument Air rupture causes lowering air header pressure.

Which one of the following describes the response of Secondary Containment and the availability of Standby Gas Treatment (GTS) to start if air header pressure continues to lower?

	Secondary Containment	GTS
A.	Isolates	Can start
B.	Isolates	CANNOT start
C.	Remains un-isolated	Can start
D.	Remains un-isolated	CANNOT start

Proposed Answer: A

Explanation: On lowering Instrument Air pressure, Secondary Containment isolates because numerous dampers fail closed (2HVR*AOD1A/B, AOD9A/B, and AOD10A/B). GTS remains available and will auto-start to maintain Reactor Building differential pressure.

<u>Note</u>: The question meets the K/A by presenting a loss of air system and requiring candidates to determine the effect on Secondary Containment (cause/effect relationship)

B. Plausible – GTS remains available to start. Plausible because GTS has multiple air-operated dampers that fail closed on loss of air, however the system has accumulators to ensure it can start.
C. Plausible – Secondary Containment isolates. Plausible that Secondary Containment isolation dampers are either motor-operated or fail open on loss of air.
D. Plausible – Secondary Containment isolates. Plausible that Secondary Containment isolation dampers are either motor-operated or fail open on loss of air. GTS remains available to start. Plausible because GTS has multiple air-operated dampers that fail closed on loss of air, however the system has accumulators to ensure it can start.

Technical Reference(s): N2-OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-290001-RBO-8

Question Source: Bank – Vision SYSID 33298

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215002 K2.01
	Importance Rating	2.5

RBM

Knowledge of electrical power supplies to the following: RBM channels: BWR-3,4,5

Question: #28

The plant is operating at 100% power when power from UPS3A is lost.

Which one of the following describes the effect, if any, on the Rod Block Monitoring (RBM) system?

A. Both RBM A and B de-energize.

B. Both RBM A and B remain energized.

C. RBM A de-energizes, but RBM B remains energized.

D. RBM B de-energizes, but RBM A remains energized.

Proposed Answer: B

Explanation: Both RBM A and B receive power from UPS3A. However, this power supply is auctioneered with another power supply from UPS3B. With the loss of UPS3A, RBM A and B lose redundancy in their power supply, but remain energized without interruption with power from UPS3B.

A. Plausible – Neither RBM A or B lose power. Plausible because UPS3A is part of both their power supplies and they do lose redundancy.
C. Plausible – RBM B does not lose power. Plausible because UPS3A is part of its power supply and it does lose redundancy.
D. Plausible – RBM A does not lose power. Plausible because UPS3A is part of its power supply and it does lose redundancy.

Technical Reference(s): N2-SOP-97 Attachment 5, 105E1503TY

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-4

Question Source: Bank – 2009 NRC #28

Question History: 2009 NRC #28

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290003 K3.04
	Importance Rating	2.8

Control Room HVAC

Knowledge of the effect that a loss or malfunction of the CONTROL ROOM HVAC will have on following: Control room pressure

Question: #29

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

- Control Room Air Conditioning Unit 2HVC*ACU1A is running.
- Control Room Air Conditioning Unit 2HVC*ACU1B is in standby.

Then, the discharge damper for 2HVC*ACU1A, 2HVC*AOD6A, inadvertently closes.

Which one of the following describes the response of Control Room pressure?

Control Room pressure...

A. lowers to 0 and remains there until Operator action is taken.

B. lowers, but remains above 0. It remains at this lower pressure until Operator action is taken.

C. initially lowers, but then is restored because 2HVC*ACU1B starts on a low Control Room pressure signal.

D. initially lowers, but then is restored because 2HVC*ACU1B starts on a low flow signal from 2HVC*ACU1A.

Proposed Answer: D

Explanation: When 2HVC*AOD6A closes, Control Room pressure begins to lower because the pressurizing air from 2HVC*ACU1A is lost. 2HVC*ACU1A trips due to an interlock with 2HVC*AOD6A. The resulting low flow condition causes 2HVC*ACU1B to automatically start and restore Control Room pressure.

A. Plausible – Control Room pressure is automatically restored by start of 2HVC*ACU1B. Plausible because 2HVC*ACU1B only starts on a low flow condition if 2HVC*ACU1A trips.
B. Plausible – Control Room pressure is automatically restored by start of 2HVC*ACU1B. Plausible because 2HVC*ACU1B only starts on a low flow condition if 2HVC*ACU1A trips. Also plausible that some positive pressure would be maintained from another source, such as Computer Room or Relay Room ventilation.
C. Plausible – ACU1B starts on a low flow signal, not a low Control Room pressure signal. Plausible because GTS starts on a low RB D/P signal.

Technical Reference(s): N2-288003-RBO-5, N2-OP-53A section D.21.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-288003-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 K4.10
	Importance Rating	2.9

Main and Reheat Steam

Knowledge of MAIN AND REHEAT STEAM SYSTEM design feature(s) and/or interlocks which provide for the following: Moisture removal from steam lines prior to admitting steam

Question: #30

A plant startup from a refueling outage is in progress with the following:

- Reactor is approaching the Point of Adding Heat.

Which one of the following describes the reason why under the current conditions 2MSS*MOV111/112, Main Steam Line Drain Isolation Valves are open, and where they direct drainage?

	Reason	When they are open, 2MSS*MOV111/112 drain to the...
A.	Remove Moisture	Main Condenser.
B.	Remove Non-condensable Gases	Suppression Pool.
C.	Remove Non-condensable Gases	Main Condenser.
D.	Remove Moisture	Suppression Pool.

Proposed Answer: A

Explanation: In the Turbine Building, the four main steam lines have drains at low points connected to a common drain header that is routed to the main condenser. This drain header removes condensation from the main steam lines and routes the condensation to the main condenser. These valves direct drainage through HCV110 to the Main Condenser.

B. Plausible – The purpose of these valves is to remove condensation (moisture) from the main steam lines. Plausible because during system startup of the main steam system, there is air and non-condensable gases present in the system. These valves direct drainage through HCV110 to the Main Condenser. Plausible because the Suppression Pool would keep drainage inside Primary Containment when used as part of MSIV leakage control. Also plausible because the Reactor head vent is routed to the Suppression Pool.
C. Plausible – The purpose of these valves is to remove condensation (moisture) from the main steam lines. Plausible because during system startup of the main steam system, there is air and non-condensable gases present in the system.
D. Plausible – These valves direct drainage through HCV110 to the Main Condenser. Plausible because the Suppression Pool would keep drainage inside Primary Containment when used as part of MSIV leakage control. Also plausible because the Reactor head vent is routed to the Suppression Pool.

Technical Reference(s): N2-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-239001-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 K5.03
	Importance Rating	2.7

Recirculation

Knowledge of the operational implications of the following concepts as they apply to RECIRCULATION SYSTEM: Pump/motor cooling: Plant-Specific

Question: #31

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

Time (hh:mm)	Condition
00:00	• Seal injection flow is lost to the Recirculation pumps due to a blockage.
00:05	• Upper seal cavity temperature on both Recirculation pumps reaches 185°F with temperature continuing to rise slowly.
00:10	• Upper seal cavity temperature on both Recirculation pumps reaches 200°F with temperature continuing to rise slowly.
00:15	• Annunciator 602109, RECIRC PUMP 1A OUTER SL LEAKAGE HIGH, alarms. • Indications show that both the upper and lower seals have failed. • Drywell pressure is slowly rising.

Which one of the following identifies the *earliest* time at which a Recirculation pump trip is required, in accordance with N2-SOP-29.1, Reactor Recirculation Pump Seal Failure?

A. 00:00

B. 00:05

C. 00:10

D. 00:15

Proposed Answer: C

Explanation: Continued operation of Recirc pumps is allowed with loss of seal injection flow from RDS. However, once upper seal cavity temperature reaches 200°F, Recirculation pumps must be tripped.

A. Plausible – Continued operation of Recirc pumps is allowed loss of seal injection flow from RDS. Plausible because loss of seal injection flow cause elevated temperatures that may eventually require tripping a Recirculation pump.
B. Plausible – Recirculation pump trip is not yet required at time 00:05. Plausible because N2-SOP-29.1 lists 185°F as the Alert limit for upper seal cavity temperature, but does not require a trip until later.
D. Plausible – Recirculation pump trip is required at the earlier time of 00:10. Plausible because failure of both seals with rising Drywell pressure also requires a trip per N2-SOP-29.1.

Technical Reference(s): N2-SOP-29.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-RBO-8

Question Source: Modified Bank – Vision SYSID 33191

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	216000 K6.02
	Importance Rating	2.8

Nuclear Boiler Instrumentation

Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: D.C. electrical distribution

Question: #32

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

- 2BYS-SWG001A de-energizes due to a sustained electrical fault.

Which one of the following describes the impact on narrow range Reactor water level indications, in accordance with N2-SOP-04, Loss of DC Power?

A. All narrow range Reactor water level indications remain available unless another DC bus also de-energizes.

B. All narrow range Reactor water level indications remain available unless an AC bus also de-energizes.

C. One channel of narrow range Reactor water level indication fails, only.

D. Two channels of narrow range Reactor water level indication fail.

Proposed Answer: C

Explanation: 2BYS-SWG001A supplies power to narrow range Reactor water level transmitter LT14C. Loss of 2BYS-SWG001A causes failure of this narrow range indication.

A. Plausible – Narrow range Reactor water level channel C fails. Plausible because most Reactor water level indications are powered by multiple sources, which could be auctioneered DC sources.
B. Plausible – Narrow range Reactor water level channel C fails. Plausible because most Reactor water level indications are powered by a UPS, such that failure of one DC source would not cause their failure.
D. Plausible – Only channel C fails. Plausible because channel B is also powered from a single DC source, but it is 2BYS-SWG001B. Also plausible that channel A could be on this same source, however it is powered from a UPS.

Technical Reference(s): N2-SOP-04

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-216000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	256000 A1.09
	Importance Rating	3.1

Reactor Condensate

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: Feedwater temperature

Question: #33

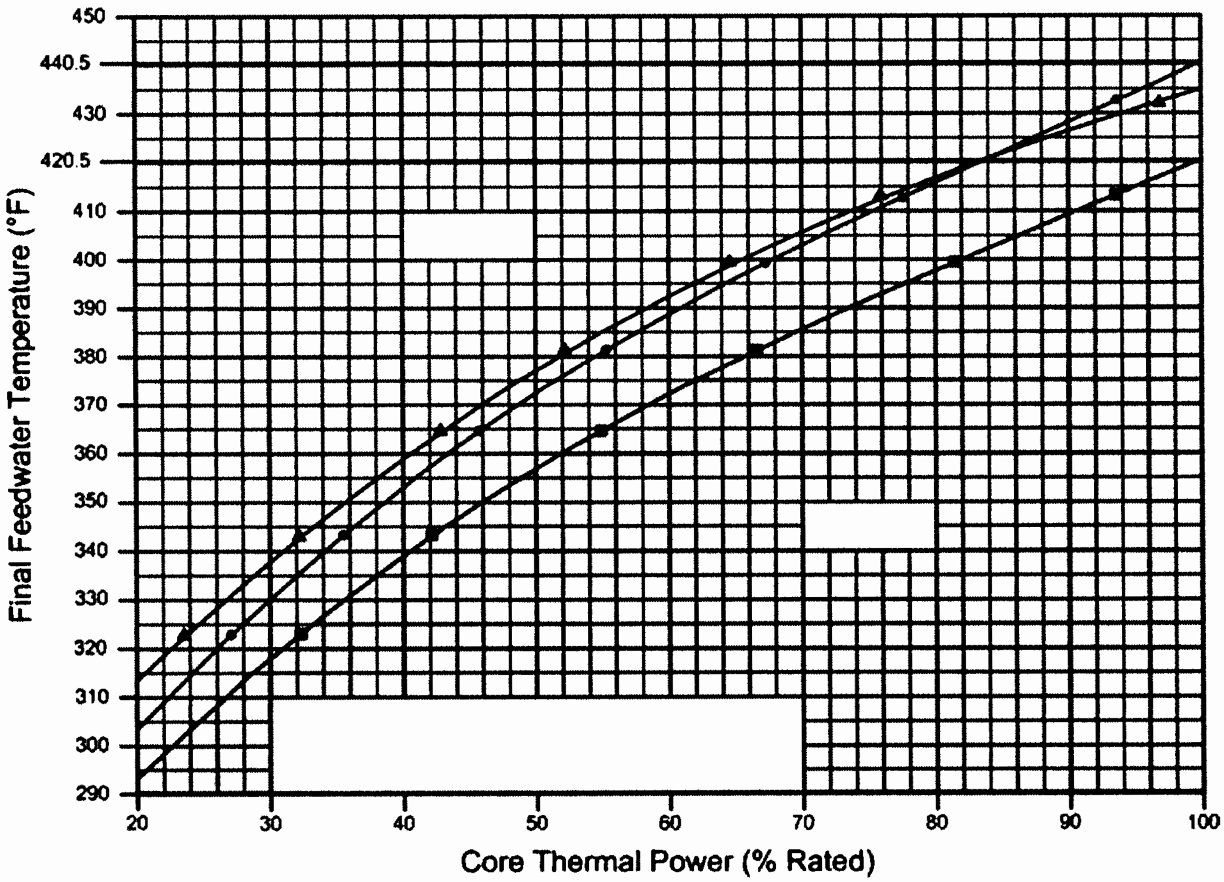
The plant is operating at 75% power with the following:

- 4^{th} point Feedwater Heater A steam is removed per N2-OP-8, Feedwater Heaters and Extraction Steam System, due to a steam leak.
- Reactor power stabilizes at 77%.
- Feedwater temperature stabilizes at 415°F.
- N2-SOP-08, Unplanned Power Changes, has been entered due to the loss of Feedwater Heating.

__Note:__ N2-SOP-08 Figure 1, Feedwater Temperature/Thermal Power Limit, is provided on the following page.

Which one of the following describes the operating point on N2-SOP-8 Figure 1 and the required action for an unacceptable Feedwater temperature, in accordance with N2-SOP-08?

	The operating point on N2-SOP-08 Figure 1 is...	If the operating point is or becomes unacceptable, then it is required to perform a Reactor...
A.	acceptable.	power reduction per N2-SOP-101D, Rapid Power Reduction.
B.	acceptable.	scram per N2-SOP-101C, Reactor Scram.
C.	NOT acceptable.	power reduction per N2-SOP-101D, Rapid Power Reduction.
D.	NOT acceptable.	scram per N2-SOP-101C, Reactor Scram.



Final Feedwater Temperature (°F)
450
440.5
430
420.5
410
400
390
380
370
360
350
340
330
320
310
300
290
20
30
40
50
60
70
80
90
100
Core Thermal Power (% Rated)

Proposed Answer: A

Explanation: The given combination of Reactor power and Feedwater temperature is in the GOOD region of the curve (acceptable). If the operating point becomes unacceptable, then a Reactor power reduction is required per N2-SOP-101D, but NOT a Reactor scram.

B. Plausible – A Reactor power reduction would be required, NOT a Reactor scram. Plausible that a scram would be required due to operation below the analyzed Feedwater temperature for the safety analysis.
C. Plausible – Feedwater temperature is acceptable. Plausible if the candidate mis-plots the operating point or does not understand which area of the curve is acceptable.
D. Plausible – Feedwater temperature is acceptable. Plausible if the candidate mis-plots the operating point or does not understand which area of the curve is acceptable. A Reactor power reduction would be required, NOT a Reactor scram. Plausible that a scram would be required due to operation below the analyzed Feedwater temperature for the safety analysis.

Technical Reference(s): N2-OP-8

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-260000-RBO-10

Question Source: Modified Bank – Vision SYSID 59227

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 A2.10
	Importance Rating	3.1

Reactor/Turbine Pressure Regulator

Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of stator water cooling: Plant-Specific

Question: #34

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

- Annunciator 851112, Generator Auxiliaries Trouble, alarms.
- Computer point GMCFC01, CLG WTR FLOW TO GEN WIND, is alarming due to a GMC flow of 450 gpm.
- NO stator water high temperature alarms are in, stator water temperatures are slightly elevated AND are STEADY.
- N2-SOP-68, Loss of Stator Water Cooling, is entered.

Which one of the following describes the status of the Generator Runback and the currently required Operator action, in accordance with N2-SOP-68?

	Generator Runback in Progress?	Operator Action
A.	No	Start the 2nd GMC pump per N2-OP-68.
B.	No	Reduce power per N2-SOP-101D, Rapid Power Reduction.
C.	Yes	Reduce power per N2-SOP-101D, Rapid Power Reduction.
D.	Yes	Scram the Reactor per N2-SOP-101C, Reactor Scram.

Proposed Answer: C

Explanation: A Generator Runback is in progress because Stator Water Cooling flow is less than 634 gpm. Because a Generator Runback is in progress, N2-SOP-68 requires lowering power per N2-SOP-101D. The more severe action of scramming the Reactor is not yet required because power has not yet been reduced to minimum Recirc flow and a complete loss of Stator Water Cooling has not occurred.

A. Plausible – A Generator Runback is in progress. Plausible that no runback would yet be received because flow still exists and no high temperature alarms have been received. Starting a 2nd GMC pump is plausible because it is part of the required actions with a runback in progress and would likely help restore flow.
B. Plausible – A Generator Runback is in progress. Plausible that no runback would yet be received because flow still exists and no high temperature alarms have been received. Reducing power is plausible because it is part of the required actions with a runback in progress and would help reduce system heat load.
D. Plausible – A Reactor scram is not yet required because power has not yet been reduced to minimum Recirc flow and a complete loss of Stator Water Cooling has not occurred. Plausible because if the runback does not clear after lowering power, then a Reactor scram will become required.

Technical Reference(s): ARP 851112, N2-OP-26, N2-SOP-68

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-253000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	2
K/A #	202002 A3.02
Importance Rating	3.4

Recirculation Flow Control

Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Lights and alarms

Question: #35

The plant is operating at 100% power when annunciator 602105, RECIRC FCV A MOTION INHIBIT, alarms.

Which one of the following identifies the resulting ability to raise and lower flow with slow and fast detent with Recirc FCV A?

	Raise with Fast Detent	Raise with Slow Detent	Lower with Fast Detent	Lower with Slow Detent
A.	Blocked	Available	Available	Available
B.	Blocked	Blocked	Available	Available
C.	Blocked	Available	Blocked	Available
D.	Blocked	Blocked	Blocked	Blocked

Proposed Answer: D

Explanation: Recirc FCV Motion Inhibit functions by automatically closing valves to trap fluid in the FCV's hydraulic cylinder. This prevents motion both to raise and lower flow regardless of using the fast or slow detent features of the flow controller.

A. Plausible – Lowering flow is also blocked. Plausible because this would be an analogous design feature to the control rod withdraw block if this were meant to limit only positive reactivity. Raising with slow detent is also blocked. Plausible because this would be a viable design feature if the purpose were to limit reactivity addition rate under certain circumstances.
B. Plausible – Lowering flow is also blocked. Plausible because this would be an analogous design feature to the control rod withdraw block if this were meant to limit only positive reactivity.
C. Plausible – Slow detent is also blocked. Plausible because this would be a viable design feature if the purpose were to limit reactivity change rate under certain circumstances.

Technical Reference(s): ARP 602105, N2-202002-RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	230000 A4.01
	Importance Rating	3.7

RHR/LPCI: Torus/Pool Spray Mode

Ability to manually operate and/or monitor in the control room: Pumps

Question: #36

Which one of the following describes the effect a loss of Reactor Building Closed Loop Cooling Water (CCP) would have on RHR pumps operating in the Suppression Pool spray mode?

A. Pumps must be shutdown because of loss of cooling to pump seals.

B. Pumps must be shutdown because of loss of cooling to pump motors.

C. Pump operation can continue and cooling can be aligned to service water.

D. Pump operation can continue and is now limited by area room temperatures.

Proposed Answer: C

Explanation: With loss of CCP, RHR pumps are affected by loss of cooling water to their seal coolers. Alternate cooling to the seal coolers can be aligned from Service Water. The pumps are allowed to continue operating during this evolution.

A. Plausible – The pumps may continue to run while alternate seal cooling is established. Plausible because this does lead to loss of seal cooling and would lead to damage if the pump were run indefinitely without alternate seal cooling aligned.
B. Plausible – The component of concern is the pump seals, not the motor, and the pumps may continue to run while alternate cooling is aligned. Plausible because operation would lead to damage if the pump were run indefinitely without alternate cooling.
D. Plausible – Room cooling is provided by Service Water, not CCP, Plausible because on loss of Service Water, RHR pump operation is limited by area temperature (P&L 4.0 – Max continuous ambient temperature = 148°F).

Technical Reference(s): N2-OP-31 P&L 14.0, ARP 601255

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-8

Question Source: Bank - 2012 Cert #34

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 2.4.21
	Importance Rating	4.0

RHR/LPCI: Containment Spray Mode

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Question: #37

The plant has experienced a loss of coolant accident with the following:

- N2-EOP-PC, Primary Containment Control, is being executed.
- Suppression Chamber spray is in service.
- Drywell spray will be required.

Which one of the following describes when the Pressure Suppression Pressure (PSP) is required to be evaluated and the action that is to be taken if you CANNOT stay inside PSP, in accordance with N2-EOP-PC?

PSP is required to be evaluated...

A. before Drywell spray is initiated. If PSP is violated, Containment venting is required.

B. before Drywell spray is initiated. If PSP is violated, an RPV Blowdown is required.

C. after Drywell spray is initiated. If PSP is violated, Containment venting is required.

D. after Drywell spray is initiated. If PSP is violated, an RPV Blowdown is required.

Proposed Answer: D

Explanation: Per N2-EOP-PC, Drywell spray is first initiated, then PSP is required to be evaluated. If PSP is violated, then an RPV Blowdown is required.

A. Plausible – PSP is required to be evaluated after Drywell spray is initiated. Plausible because another limit, the CSIL is required to be evaluated prior to initiating Drywell spray. RPV Blowdown, not Containment venting, is required if PSP is violated. Plausible because Containment venting would be required if the PCPL is violated in subsequent steps in N2-EOP-PC.
B. Plausible – PSP is required to be evaluated after Drywell spray is initiated. Plausible because another limit, the CSIL is required to be evaluated prior to initiating Drywell spray.
C. Plausible – RPV Blowdown, not Containment venting, is required if PSP is violated. Plausible because Containment venting would be required if the PCPL is violated in subsequent steps in N2-EOP-PC.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Modified Bank - 2015 NRC #47

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	2
K/A #	272000 A3.01
Importance Rating	3.8

Radiation Monitoring

Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including: Main steam isolation indications

Question: #38

A plant startup is in progress with the following:

- A Mechanical Vacuum pump is in service to establish Main Condenser vacuum.
- Reactor pressure is 150 psig and slowly rising.
- One Turbine Bypass Valve is partially open.

Then, the following Main Steam Line high radiation alarms are received:

- Annunciator 603133, MN STEAM LINE RADIATION HIGH
- Annunciator 603407, DIV 2 MN STM LINES RADIATION HI-HI/INOP

Annunciator 603107, DIV 1 MN STM LINES RADIATION HI-HI/INOP, is **NOT** yet in alarm.

Main Steam Line radiation monitor indications are as shown on the following pages.

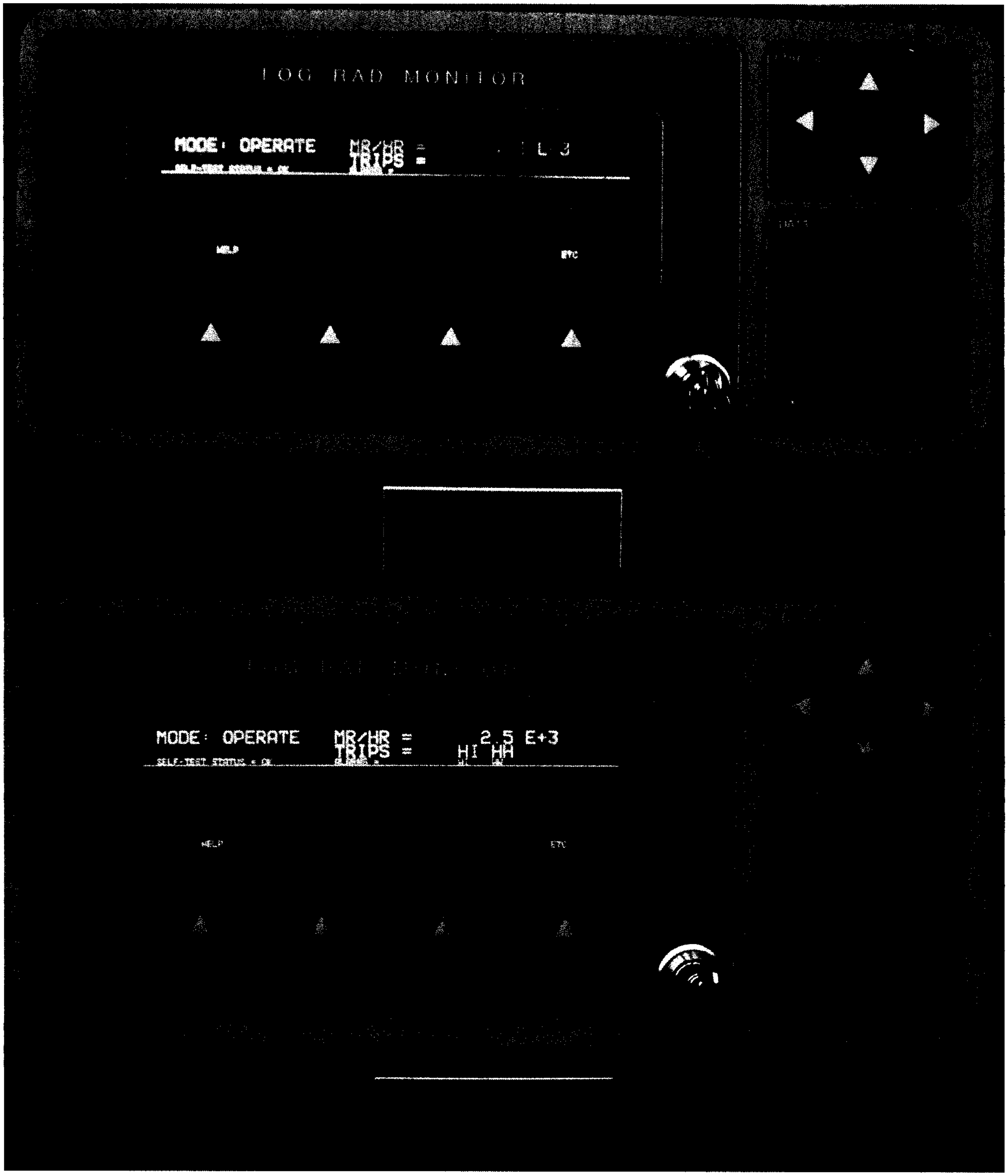
Which one of the following describes the plant response to these indications?

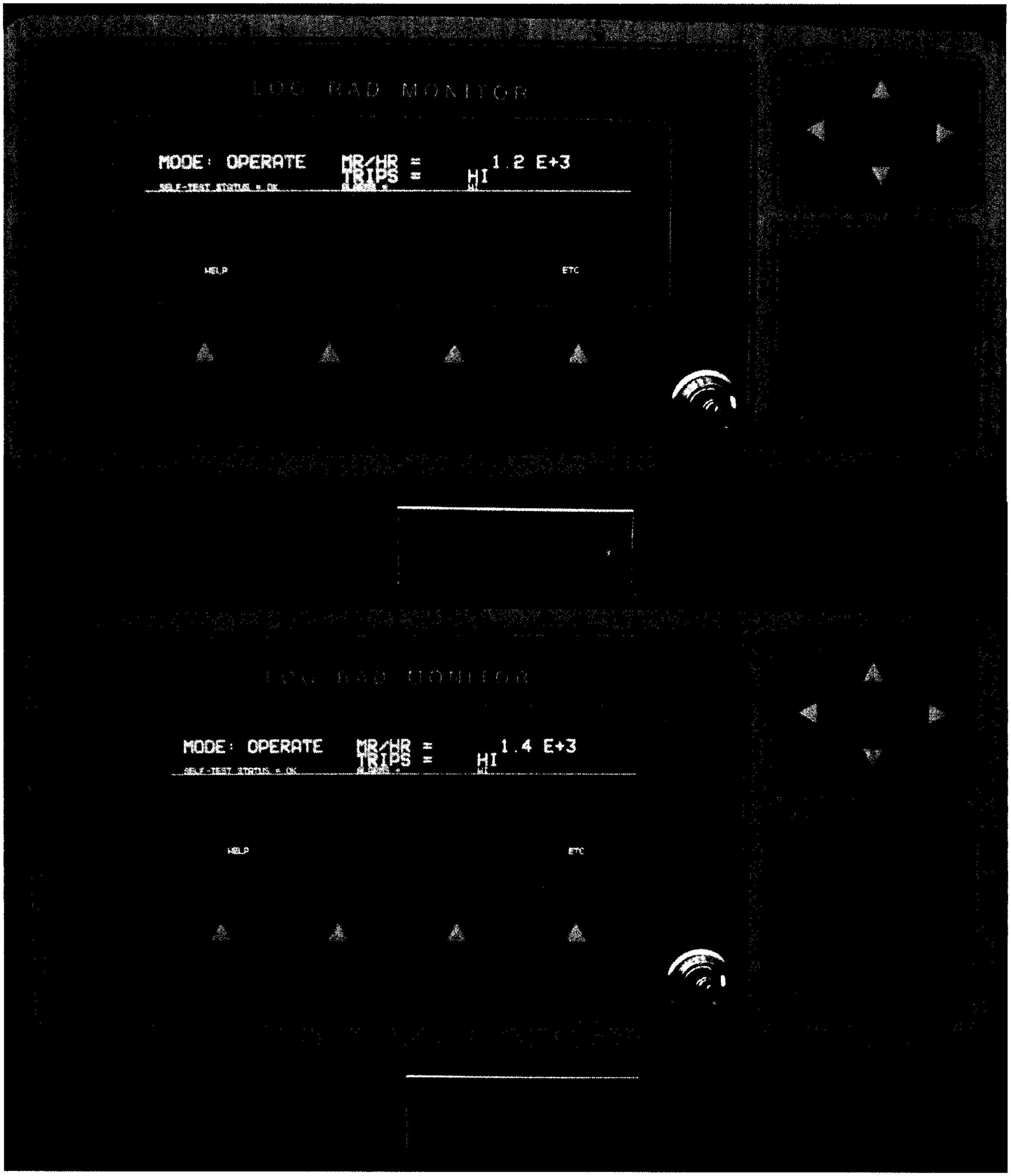
A. NO automatic plant response has occurred yet.

B. The Mechanical Vacuum pump has tripped and isolated. The MSIVs remain open.

C. The Mechanical Vacuum pump remains in service and the MSIVs have closed.

D. The Mechanical Vacuum pump has tripped and isolated and the MSIVs have closed.


LOG RAD MONITOR
MODE: OPERATE
MR/HR =
TRIPS =
HELP
ETC
MODE: OPERATE
MR/HR = 2.5 E+3
TRIPS = HI HH
SELF-TEST STATUS = OK
HELP
ETC


LOG RAD MONITOR
MODE: OPERATE
MR/HR =
1.2 E+3
TRIPS =
HI
HELP
ETC
LOG RAD MONITOR
MODE: OPERATE
MR/HR =
1.4 E+3
TRIPS =
HI
HELP
ETC

Proposed Answer: A

Explanation: Four radiation monitors (A, B, C, D) are provided on the Main Steam Lines. A and C input to Div I. B and D input to Div II. The trip logic requires at least one of the Div I inputs and at least one of the Div II inputs to be above the Hi-Hi setpoint (2871 mr/hr nominal) to cause the associated MSL rad monitor trips. These trips include:

- Trip of the Air Removal System Mechanical Vacuum pump
- Isolation of the Mechanical Vacuum pump suction valve

The given MSL rad monitor indications show 3 monitors (A, C, and D) above the Hi setpoint (1809 mr/hr nominal) but below the Hi-Hi setpoint (2871 mr/hr nominal), and 1 monitor (B) above the Hi-Hi setpoint. With only one monitor above the Hi-Hi setpoint, the trip logic is NOT satisfied, therefore the Mechanical Vacuum pump has NOT tripped. The MSIVs do NOT go closed on high MSL rad level. This automatic isolation is part of original plant designed, but has been bypassed at NMP2.

Note: The question meets the K/A because Main Steam radiation monitor indications that cause isolation are presented and the candidate must determine the status of the automatic isolation. This interpretation of K/A intent is consistent with the proposed revision of this K/A in the next K/A manual to "Main steam radiation alarms".

B. Plausible – The MVP has not yet tripped because only a Div II rad monitor is above the Hi-Hi setpoint. Plausible that this would be enough to cause isolation based on the relatively low impact of MVP trip/isolation.
C. Plausible – The MSIVs remain open. Plausible because original plant design includes MSIV isolation on Hi-Hi MSL radiation, and one rad monitor is above the Hi-Hi setpoint with the others also in alarm.
D. Plausible – The MVP has not yet tripped because only a Div II rad monitor is above the Hi-Hi setpoint. Plausible that this would be enough to cause isolation based on the relatively low impact of MVP trip/isolation. The MSIVs remain open. Plausible because original plant design includes MSIV isolation on Hi-Hi MSL radiation, and one rad monitor is above the Hi-Hi setpoint with the others also in alarm.

Technical Reference(s): ARP 603107, ARP 603407, ARP 603133

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-272000-RBO-5

Question Source: Bank – JAF 9/12 NRC #63

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AK1.01
	Importance Rating	2.7

Partial or Complete Loss of AC Power

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Effect of battery discharge rate on capacity

Question: #39

A Station Blackout is in progress with the following:

- NO operator actions have been taken in N2-SOP-01, Station Blackout.
- Both Division 1 and 2 125 VDC batteries were fully charged prior to the Station Blackout.
- Battery discharge rates for both Division 1 and 2 batteries are equal and have been for the duration of the Station Blackout so far.

Then, RCIC is placed in service.

Which one of the following identifies:

(1) the battery that will reach its minimum terminal voltage first, and

(2) the minimum time that battery is designed to maintain terminal voltage above its minimum terminal voltage following the complete loss of AC power if NO load stripping actions are taken?

A. (1) Division I
(2) 2 hours

B. (1) Division I
(2) 4 hours

C. (1) Division II
(2) 2 hours

D. (1) Division II
(2) 4 hours

Proposed Answer: A

Explanation: The only RCIC MOV supplied by Division 2 (2BYS*SWG002B) is one of the 2 turbine exhaust vacuum breakers. All the remaining loads including lights and indication are supplied from Division 1 (SWG002A). RCIC operation results in a greater battery discharge rate on Div I than Div II. The Div I and II batteries are designed to have equal capacities. Therefore, the Div I Battery will discharge faster than the Div II Battery. The batteries are designed to provide power for large load starting transients, and all DC loads in the event of AC power failure, for at least 2 hours following a loss of chargers provided that the battery had been fully charged.

B. Plausible – The minimum time is 2 hours, not 4 hours. Plausible because 4 hours is a typical time used regarding Station Blackout coping time.
C. Plausible – RCIC operation causes more DC current draw on Div I, not Div II. Plausible because RCIC does have some components powered from Div II.
D. Plausible – RCIC operation causes more DC current draw on Div I, not Div II. Plausible because RCIC does have some components powered from Div II. The minimum time is 2 hours, not 4 hours. Plausible because 4 hours is a typical time used regarding Station Blackout coping time.

Technical Reference(s): N2-OP-74A, N2-OP-35-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-217000-RBO-4

Question Source: Bank - 2012 NRC #58

Question History: 2012 NRC #58

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AK1.01
	Importance Rating	3.3

Generator Voltage and Electric Grid Disturbances

Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Definition of terms: volts, watts, amps, VARs, power factor

Question: #40

Following a grid disturbance, the following conditions exist:

- Main Generator load is 800 MWe.
- Main Generator reactive loading is 200 MVARs to the Generator.
- Main Generator voltage regulation is in AUTO.

Then, System Power Control requests that the Main Generator reactive loading be changed to 100 MVARs to the Bus.

Which one of the following describes the required manipulation to carry out System Power Control's request?

Rotate the...

A. AC Voltage Regulator control switch to RAISE.

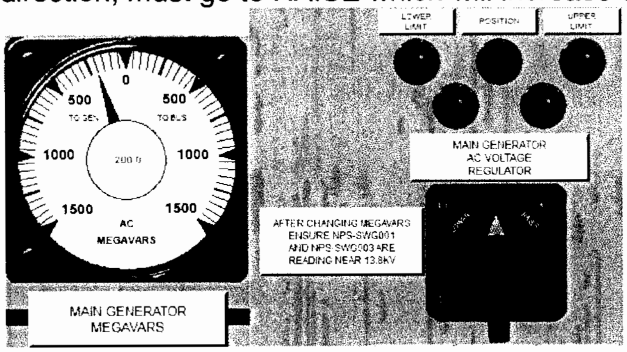
B. AC Voltage Regulator control switch to LOWER.

C. DC Voltage Regulator control switch to RAISE.

D. DC Voltage Regulator control switch to LOWER.

Proposed Answer: A

Explanation: Voltage regulation is in AUTO which places the AC Voltage Regulator in control. Initial MVAR loading is in the leading (to the Generator) direction. To establish 100 MVARs in the lagging (to the Bus) direction, must go to RAISE which will increase excitation.



Note: The question meets the K/A because an operational implication of the definitions of MWe and MVAR is how these parameters are controlled (voltage regulator modes of operation and control characteristics) and how “to the Generator” and “to the Bus” are defined.

B. Plausible – The control switch must be taken to the RAISE direction, not LOWER. Plausible if the candidate mixes up the direction of the desired MVAR change and operation of the voltage regulator.
C. Plausible – The AC Voltage regulator is in control, not the DC. Plausible because the DC Voltage Regulator would be correct if control was in manual.
D. Plausible – The AC Voltage regulator is in control, not the DC. Plausible because the DC Voltage Regulator would be correct if control was in manual. The control switch must be taken to the RAISE direction, not LOWER. Plausible if the candidate mixes up the direction of the desired MVAR change and operation of the voltage regulator.

Technical Reference(s): N2-OP-68

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-245000-RBO-5

Question Source: Bank – 2010 NRC #48

Question History: 2010 NRC #48

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AK1.02
	Importance Rating	2.9

Plant Fire On-site

Knowledge of the operational implications of the following concepts as they apply to Plant Fire On Site: Fire Fighting

Question: #41

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

- A fire has developed in the Standby Gas Treatment train A charcoal filter.
- Annunciator 870109, SBGTS TRAIN A SYSTEM TROUBLE, is in alarm with the following computer points alarming:
  - GTSTC03, GTS FLTR A ACT CHAR TEMP
  - GTSTC12, GTS FLTR A ACT CHAR TEMP

Which one of the following describes the means of fire suppression for this charcoal filter?

A. CO_2 flooding that automatically initiates.

B. Water deluge that automatically initiates.

C. CO_2 flooding that must be manually initiated.

D. Water deluge that must be manually initiated.

Proposed Answer: D

Explanation: The Standby Gas Treatment charcoal filters are provided with a water deluge fire suppression system. This suppression system is normally blocked by closure of manual valves 2FPW-V430 and 2FPW-V817, therefore it must be manually initiated.

<u>Note:</u> The question meets the K/A by presenting a situation with a fire in the plant and testing the operational implications of fire-fighting for this fire (what suppression system will be used and how it is initiated).

A. Plausible – Suppression is from water deluge, not CO_2. Plausible because the presence of water in a charcoal filter is undesirable due to an exothermic reaction, which would not be present with CO_2 usage, and CO_2 is used in other fire suppression applications. Suppression is manual, not automatic. Plausible because temperature detectors are in alarm, other systems automatically initiate, and GTS decay heat removal valves have automatic response.
B. Plausible – Suppression is manual, not automatic. Plausible because temperature detectors are in alarm, other systems automatically initiate, and GTS decay heat removal valves have automatic response.
C. Plausible – Suppression is from water deluge, not CO_2. Plausible because the presence of water in a charcoal filter is undesirable due to an exothermic reaction, which would not be present with CO_2 usage, and CO_2 is used in other fire suppression applications.

Technical Reference(s): N2-OP-61B

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-286000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK2.06
	Importance Rating	3.9

High Drywell Pressure

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Emergency generators

Question: #42

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

- The Division I EDG (2EGS*EG1) has been started, paralleled with 2ENS*SWG101 and loaded to 4400 KW.
- A steam leak develops in the Drywell.
- An automatic Reactor scram occurs due to high Drywell pressure.

Two (2) minutes later, a loss of all offsite power occurs.

Which one of the following describes the response of 2EGS*EG1?

2EGS*EG1 output breaker...

A. remains closed throughout the transient.

B. trips when a LOCA signal is received and then automatically re-closes within 10 seconds.

C. trips when a LOCA signal is received and then automatically re-closes when offsite power is lost.

D. trips when offsite power is lost and then automatically re-closes within 10 seconds.

Proposed Answer: C

Explanation: The DG output breaker receives a trip signal opening the breaker when the LOCA signal occurs. The output breaker will automatically close immediately when the bus is dead following the loss of all offsite power.

A. Plausible – The DG output breaker receives a trip signal opening the breaker when the LOCA signal occurs. The output breaker will automatically close immediately when the bus is dead. Plausible because the breaker was initially closed and there is no specific harm in keeping the DG paralleled to the bus after the LOCA signal.
B. Plausible – The output breaker does not automatically re-close until the loss of all offsite power. Plausible because 10 seconds is the time requirement for DG start, a LOCA signal automatically starts the DG, and there is no specific harm in having the DG paralleled to the bus after the LOCA signal.
D. Plausible – The DG output breaker receives a trip signal opening the breaker when the LOCA signal occurs. Plausible because the breaker was initially closed and there is no specific harm in keeping the DG paralleled to the bus after the LOCA signal. Also plausible because a loss of offsite power alone would start the DG and close the output breaker.

Technical Reference(s): N2-264001-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-5

Question Source: Bank – 2012 Cert #15

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK2.08
	Importance Rating	4.2

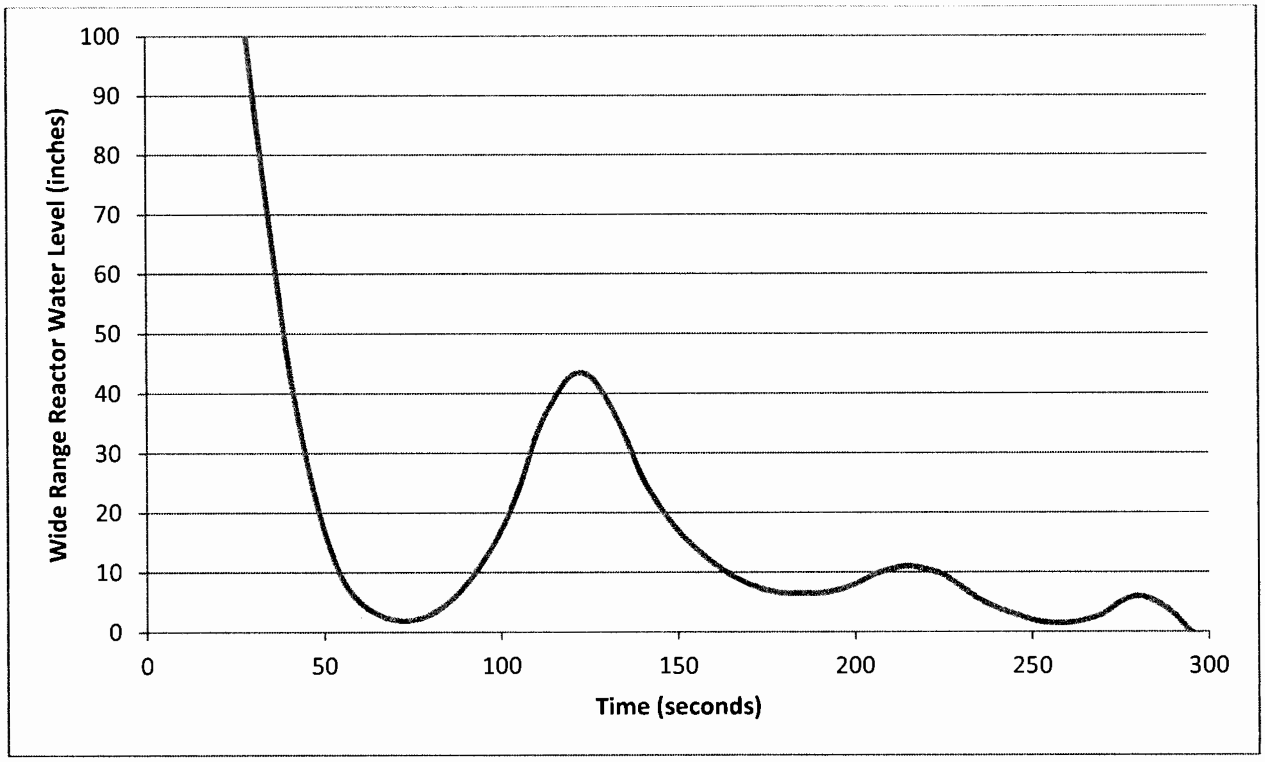
Reactor Low Water Level

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Automatic depressurization system

Question: #43

A transient results in the following:

- Reactor water level response is shown in the following graph:



- Current time is 300 seconds.
- All SRVs are closed.
- Reactor pressure is 800 psig and slowly lowering.
- All RHR and Core Spray pumps are operating per design.
- ADS has NOT been inhibited.

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

ADS has…

A. operated per design. ADS SRVs will open at approximately time = 330 seconds.

B. operated per design. ADS SRVs will remain closed unless Reactor water level drops below top of active fuel or Operator action is taken.

C. NOT operated per design. ADS SRVs should have opened at approximately time = 155 seconds.

D. NOT operated per design. ADS SRVs should have opened at approximately time = 255 seconds.

Proposed Answer: D

Explanation: ADS opens SRVs if the following conditions are met:

- low-low-low Reactor water level of 17.8" (Level 1);
- confirmatory low Reactor water level of 159.3" (Level 3);
- either RHR A or LPCS Pump running (Subchannel A) - either RHR B or C Pump running (subchannel B), and
- 105 second time delay expires.

If Reactor water level recovers above 17.8" prior to the 105 second time delay expiring, the timer resets. If Reactor water level then goes below 17.8" again, the 105 second timer starts over.

The given Reactor water level trace first goes below 17.8" at approximately 50 seconds. This would result in SRVs opening at approximately 155 seconds, except Reactor water level recovers above 17.8" at approximately time 100 seconds. This resets the timers. The timers re-initiate at approximately time 150 seconds and Reactor water level remains below 17.8", so SRVs should open at approximately time 255 seconds. Therefore, ADS has not operated properly.

A. Plausible – SRVs should be open, so ADS did not operate per design. Plausible because Reactor water level did rise again at approximately 225 seconds, but did not get high enough to reset the timers and require SRVs to open at 330 seconds.
B. Plausible – SRVs should be open, so ADS did not operate per design. Plausible if ADS logic reset needed the confirmatory level of 159.3" to clear or Operator action to be taken before the timers could re-initialize.
C. Plausible – SRVs should have first opened at approximately time 255 seconds. Plausible because if Reactor water level had not re-risen above 17.8", the first initiation of ADS would have opened SRVs at approximately 155 seconds.

Technical Reference(s): N2-OP-34, N2-218000-RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-2

Question Source: Modified Bank - NMP1 2009 Cert #58

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EK2.05
	Importance Rating	4.1

High Reactor Pressure

Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: Safety/relief valves: Plant-Specific

Question: #44

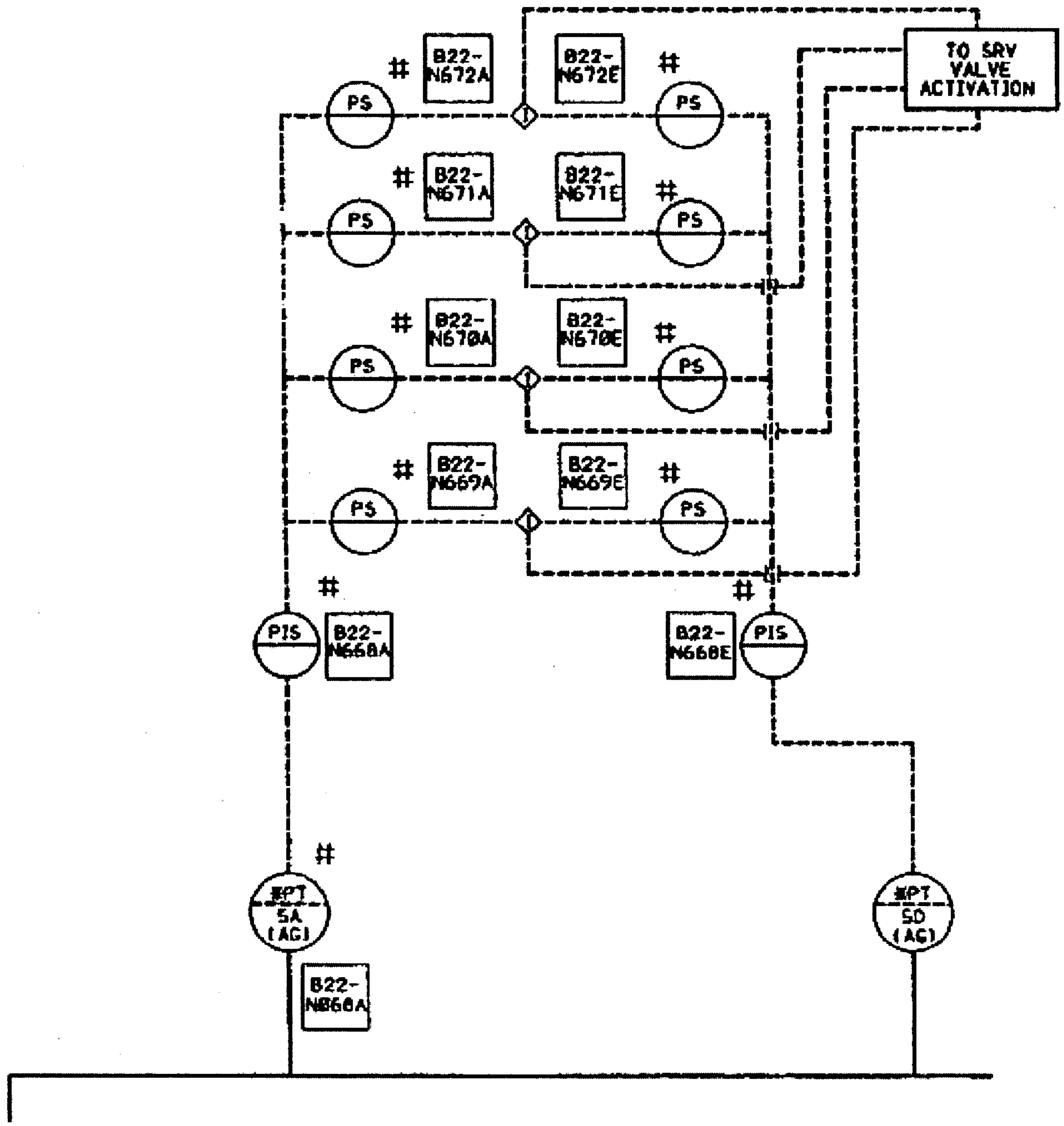
A plant startup is in progress with the following:

- The instrument root valve for Reactor pressure transmitter 2ISC*PT5A is found to be closed and CANNOT be opened.
- Reactor pressure transmitter 2ISC*PT5D is inspected and no issues are noted.
- The next page has an excerpt from PID-028A showing both pressure transmitters.

Which one of the following describes the effect of these conditions on the SRV relief and safety functions in the event of an actual high Reactor pressure condition?

	SRV Relief Function	SRV Safety Function
A.	Will operate correctly	Will operate correctly
B.	Will operate correctly	Will NOT operate correctly
C.	Will NOT operate correctly	Will operate correctly
D.	Will NOT operate correctly	Will NOT operate correctly

Excerpt from PID-028A:



Proposed Answer: C

Explanation: With PT5A isolated, it will not sense a high Reactor pressure. Both PT5A and PT5D are required to sense high pressure for SRVs to open in the relief mode. None of the SRVs can open in relief mode, even with one PT still functioning. The Safety Function of each SRV is not affected because the Safety Mode is a function of a mechanical spring rather than using air.

A. Plausible – One pressure transmitter will still sense the high Reactor pressure, however two pressure transmitters must sense the high Reactor pressure to open SRVs in relief mode.
B. Plausible – One pressure transmitter will still sense the high Reactor pressure, however two pressure transmitters must sense the high Reactor pressure to open SRVs in relief mode. The safety mode does actuate based on high Reactor pressure, but it is sensed mechanically by the SRV, not through logic involving pressure transmitters.
D. Plausible – The safety mode does actuate based on high Reactor pressure, but it is sensed mechanically by the SRV, not through logic involving pressure transmitters.

Technical Reference(s): ARP 601535 and 601536, PID-28A, GE 807E155TY Sheets 8 and 12

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-5

Question Source: Bank – 2014 Cert #24

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AK3.03
	Importance Rating	3.8

SCRAM

Knowledge of the reasons for the following responses as they apply to SCRAM: Reactor pressure response

Question: #45

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

- 2RPM-MG1A, RPS MG Set, trips.
- Two minutes later, a fault in 2VBB-UPS3B, RPS Power Supply, causes a loss of output voltage and 2VBS*PNLB100 de-energizes.

Which one of the following describes the resulting control of Reactor pressure two (2) minutes later?

Reactor pressure will be controlled...

A. by automatic cycling of the SRVs because the Reactor scrams and MSIVs close.

B. at approximately 925 psig by Turbine Bypass Valves because the Reactor scrams and MSIVs remain open.

C. at approximately 1020 psig by Turbine Bypass Valves because the Reactor scrams and MSIVs remain open.

D. at approximately 1020 psig by Turbine Control Valves because the Reactor remains operating at 100% power.

Proposed Answer: B

Explanation: Trip of 2RPM-MG1A causes a “silent half scram” on RPS A due to de-energization of the scram solenoids. Loss of power on 2VBS*PNLB100 causes a half scram on RPS B due to loss of logic power. Even though these are different failures, they combine to cause a full scram signal. Therefore, the Reactor scrams. MSIVs remain open, so Reactor pressure is controlled on Turbine Bypass Valves. Pressure is controlled at approximately 925 psig, vice the initial 1020 psig, due to the control characteristics of the EHC system.

A. Plausible – The Reactor does scram on this loss of power, however MSIVs remain open. Plausible because this would be correct for a complete loss of logic power to MSIV isolation circuitry.
C. Plausible – The Reactor does scram on this loss of power, but pressure is controlled at approximately 925 psig. Plausible because 1020 psig is the approximate normal Reactor pressure at 100% power (initial condition).
D. Plausible – This loss of power results in an automatic Reactor scram. Plausible because this would be correct if the two given failures did not combine to cause a full scram (such as if they were in the same division).

Technical Reference(s): EE-001BH, EE-MO001C

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262000-RBO-11

Question Source: Modified Bank – SSES LOC26R NRC #44

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

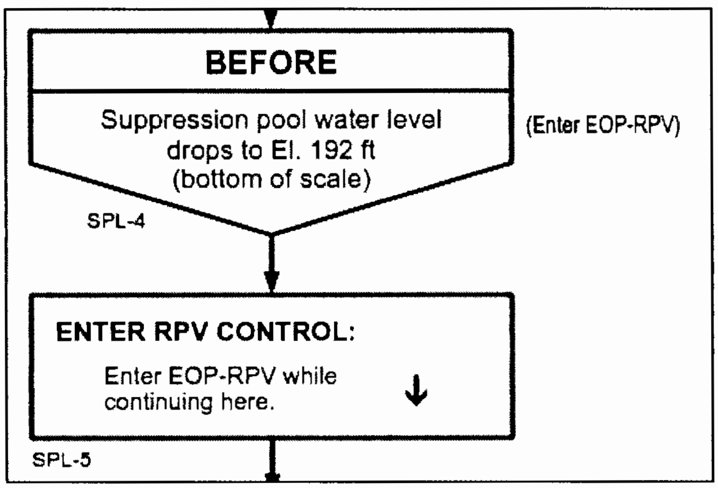
Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EK3.06
	Importance Rating	3.6

Low Suppression Pool Water Level

Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Reactor SCRAM

Question: #46

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



Which one of the following describes the reason for this requirement, in accordance with the Emergency Operating Procedures?

Assures that the Reactor is shutdown before…

A. an RPV blowdown is performed.

B. RCIC exhaust becomes uncovered.

C. required by Technical Specifications.

D. suction is lost to low pressure ECCS pumps.

Proposed Answer: A

Explanation: The EOP bases state that the reason for this requirement is to shutdown the Reactor prior to performing an RPV blowdown.

B. Plausible – The reason is to shutdown the Reactor prior to an RPV blowdown. Plausible because RCIC exhaust becoming uncovered as Suppression Pool level lowers is an operational concern.
C. Plausible – The reason is to shutdown the Reactor prior to an RPV blowdown. Plausible because TS 3.6.2.2 does require a Reactor shutdown with Suppression Pool level this low.
D. Plausible – The reason is to shutdown the Reactor prior to an RPV blowdown. Plausible because loss of NPSH to LPCS and LPCI is an operational concern with lowering Suppression Pool level.

Technical Reference(s): N2-EOP-PC, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – 2010 Cert #47

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK3.03
	Importance Rating	3.5

Control Room Abandonment

Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Disabling control room controls

Question: #47

Following a fire in the Main Control Room, N2-SOP-78, Control Room Evacuation, directs that the Appendix R Disconnect switches on 2CES*PNL415/416 (Control Building 306' west and east cable chases) be placed in the ACTUATE position.

Which one of the following describes the reason for these actions with respect to operation of RCIC?

When placed in ACTUATE, these switches...

A. install separate auto isolation circuits to ensure that fire damage will NOT prevent required isolation.

B. install separate auto initiation circuits to ensure that fire damage will NOT prevent required system initiation.

C. enable Remote Shutdown Panel controls to allow operation of the system from outside of the Main Control Room.

D. eliminate auto isolation signals and inhibit auto start signals to prevent spurious isolation or initiation of the system.

Proposed Answer: D

Explanation: Placing the SW1-2CESA02 Appendix R Disconnect switch to ACTUATE eliminates RCIC auto isolation signals and inhibits the auto start on low level. These actions are desired to prevent spurious operation of RCIC due to fire induced circuit damage.

A. Plausible – Separate auto isolation circuitry is not connected. Plausible because auto isolation circuitry is disconnected, and connecting separate circuitry outside of the control room would maintain isolation function while still getting rid of fire induced damage concerns.
B. Plausible – Separate auto initiation circuitry is not connected. Plausible because auto initiation circuitry is disconnected, and connecting separate circuitry outside of the control room would maintain initiation function while still getting rid of fire induced damage concerns.
C. Plausible – Remote Shutdown Panel controls are not enabled by this action. Plausible because this is analogous to switches on panels 2CES*PNL405.

Technical Reference(s): N2-SOP-78, N2-OP-78 Attachment 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-296000-RBO-5

Question Source: Modified Bank – 2012 NRC #45

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA1.01
	Importance Rating	3.4

Loss of Shutdown Cooling

Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water cleanup system

Question: #48

The plant is shutdown with the following:

- The Reactor Mode Switch is in SHUTDOWN.
- The Reactor Vessel Head Studs are still tensioned.
- Shutdown Cooling (SDC) is lost due to inadvertent isolation valve closure.
- Attempts to restore SDC have been unsuccessful.
- Reactor Water Cleanup (WCS) is currently in service.
- Reactor coolant temperature is 120°F and rising slowly.

Which one of the following describes the required control of WCS and the associated reason, in accordance with N2-SOP-31, Loss of SDC?

A. Isolate WCS to minimize heat addition to Reactor coolant from WCS pumps.

B. Maximize WCS flow to assist in decay heat removal.

C. Isolate WCS in anticipation of automatic isolation on high Non-Regenerative Heat Exchanger outlet temperature.

D. Maximize WCS flow to prevent automatic isolation on high Non-Regenerative Heat Exchanger outlet temperature.

Proposed Answer: B

Explanation: N2-SOP-31 directs using WCS as part of a feed and bleed with CRD/Condensate and maximizing WCS cooling per N2-OP-37. Both of these uses are to assist in decay heat removal.

A. Plausible – WCS is maintained in service. Plausible because WCS pumps do add some amount of heat to the Reactor coolant.
C. Plausible – WCS is maintained in service. Plausible because WCS would isolate if NRHX outlet temperature reaches 140°F and N2-OP-37 does direct isolating WCS filter demins since temperature if NRHX outlet temperature reaches 120°F.
D. Plausible – The reason for using WCS is to assist in decay heat removal, not specifically to prevent automatic isolation. Normal WCS operation would prevent automatic isolation unless some other failure occurred. Plausible because automatic isolation is undesirable and is a concern in the procedure used to maximize WCS flow.

Technical Reference(s): N2-SOP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP31C01 EO-2

Question Source: Bank - SSES LOC25 Cert #46

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 1
Group # 1
K/A # 295023 AA1.06
Importance Rating 3.3

Refueling Accidents

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

Question: #49

The plant is in Mode 5 with the following:

- A new fuel bundle is being loaded into the core quadrant containing SRM A.
- The following SRM indications are observed:

SRM	Initial Count Rate (cps)	Initial Period (sec)	Current Count Rate (cps)	Current Period (sec)
A	50	∞	250	40
B	60	∞	70	200
C	45	∞	50	400
D	70	∞	75	400

Which one of the following describes the operational implications of these indications?

These SRM indications are...

A. abnormal. Enter N2-SOP-39, Refuel Floor Events.

B. abnormal. Entry into N2-SOP-39, Refuel Floor Events, is NOT required.

C. normal. The fuel movement may be completed and then SRM A period should be observed to return to infinity.

D. normal. The fuel movement must be paused until the SRM A period begins to lengthen, and then the fuel movement may be completed.

Proposed Answer: A

Explanation: While some response on SRM A may be expected due to loading of a new fuel bundle in the same core quadrant, there are limitations as to how much response is expected. N2-FHP-13.3 sets a limit at 4 times the initial SRM count rate. Additionally, annunciator 603209, SRM SHORT PERIOD, is received at a period of 50 seconds. Both high SRM count rates and short period are entry conditions into N2-SOP-39, Refuel Floor Events, as indications of an inadvertent criticality.

B. Plausible – N2-SOP-39 entry is required. Plausible because this procedure has many other entry conditions and no rise in radiation levels has occurred.
C. Plausible – Some SRM A response may be expected and is allowed, however the amount of response exceeds limits. Also plausible that fuel movement could be completed because other SRMs are not in alarm.
D. Plausible – Some SRM A response may be expected and is allowed, however the amount of response exceeds limits. Also plausible that a brief pause would be warrant, such as used when withdrawing control rods and large period changes occur.

Technical Reference(s): N2-SOP-39, ARP 603209, N2-FHP-13.3 Section 6.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP39C01 EO-2

Question Source: Bank – 2014 Cert #39

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AA1.01
	Importance Rating	3.3

Partial or Complete Loss of DC Power

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

Question: #50

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

- All Emergency DC batteries are on float charge.
- 2BYS*SWG002A voltage is indicating 135 VDC.
- 2BYS*SWG002B voltage is indicating 125 VDC.

Which one of the following describes the status of these voltages, in accordance with N2-OP-74A, Emergency DC Distribution?

A. Both voltages are within the required range.

B. 2BYS*SWG002A voltage is within the required range, but 2BYS*SWG002B voltage is too low.

C. 2BYS*SWG002B voltage is within the required range, but 2BYS*SWG002A voltage is too high.

D. 2BYS*SWG002A voltage is too high and 2BYS*SWG002B voltage is too low.

Proposed Answer: B

Explanation: An Emergency DC distribution system on float charge is required to be 134.5-135.0 VDC per N2-OP-74A. 2BYS*SWG002A(B) voltage <131 VDC or ≥ 142 VDC cause an alarm. Voltage below 130 VDC may require Technical Specification entry. Based on these limitations, 2BYS*SWG002A voltage is within the required range, but 2BYS*SWG002B voltage is too low.

A. Plausible – 2BYS*SWG002B voltage is below the required voltage. Plausible because it is at the nominal voltage of the system.
C. Plausible – 2BYS*SWG002B voltage is below the required voltage. Plausible because it is at the nominal voltage of the system. 2BYS*SWG002A voltage is acceptable. Plausible because it is well above the nominal voltage of the system.
D. Plausible – 2BYS*SWG002A voltage is acceptable. Plausible because it is well above the nominal voltage of the system.

Technical Reference(s): N2-OP-74A, ARP 852208

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-10

Question Source: Bank – JAF 2016 NRC #20

Question History: JAF 2016 NRC #20

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AA2.01
	Importance Rating	2.6

Main Turbine Generator Trip

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

Question: #51

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

- High vibrations develop on the Main Turbine.
- The Reactor is scrammed and the Main Turbine is tripped.
- Main Turbine speed begins to lower and vibrations begin to rise.

Which one of the following identifies the Main Turbine rotor critical speed range, where peak vibrations can be expected, in accordance with N2-OP-21, Main Turbine System?

A. 1400-1800 rpm

B. 900-1300 rpm

C. 400-800 rpm

D. 100-400 rpm

Proposed Answer: B

Explanation: N2-OP-21 states the rotor critical speed range is 900-1300 rpm.

A. Plausible – The rotor critical speed range is 900-1300 rpm. Plausible because 1500 rpm is the speed at which the wobbulator circuit becomes active and 1600 rpm is discussed in N2-SOP-21 as a speed at which to trend vibrations to predict how vibrations will respond in the lower critical speed range. Both of these speeds are in the given range.
C. Plausible – The rotor critical speed range is 900-1300 rpm. Plausible because N2-OP-21 discusses minimizing time at the 800 rpm speed during a startup.
D. Plausible – The rotor critical speed range is 900-1300 rpm. Plausible because N2-OP-21 discusses minimizing time at the 100 rpm speed during a startup.

Technical Reference(s): N2-OP-21

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-245000-RBO-10

Question Source: Bank – Vision SYSID 91663

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

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

Partial or Complete Loss of CCW

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

Question: #52

The plant is at 100% power. Conditions are as follows:

- A tube rupture occurs in the Reactor Water Cleanup (WCS) Non-Regenerative Heat Exchanger.
- The resultant actual and indicated delta flow rate is 130 gpm.

Which one of the following describes the response of CCP surge tank level and the status of the WCS System one minute later?

	CCP Surge Tank Level	The WCS System is...
A.	Rises	Isolated
B.	Rises	Not Isolated
C.	Lowers	Isolated
D.	Lowers	Not Isolated

Proposed Answer: B

Explanation: Water will flow from the WCS system into the CCP system due to the higher pressure in WCS. This will cause CCP surge tank level to rise. Because the delta flow rate is <150.5 gpm, WCS will not isolate on high differential flow.

A. Plausible – This would be correct if the delta flow rate was >150.5 gpm.
C. Plausible – This would be correct if the delta flow rate >150.5 gpm and the WCS system was at a lower pressure than CCP.
D. Plausible – This would be correct if the WCS system was at a lower pressure than CCP.

Technical Reference(s): UFSAR Section 9.2.2.3, ARP 602313

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-208000-RBO-8

Question Source: Bank – 2015 NRC #2

Question History: 2015 NRC #2

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AA2.01
	Importance Rating	3.5

Partial or Complete Loss of Forced Core Flow Circulation

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

Question: #53

The plant is operating at 18% power with the following:

- A transient has caused Reactor core flow to lower.
- Conditions have stabilized as follows:
  - Reactor power is 15%.
  - Core flow is 20.5 Mlbs/hr.

Which one of the following describes the required control of the Reactor, in accordance with N2-SOP-29, Sudden Reduction in Core Flow?

A. Raise core flow.

B. Scram the Reactor.

C. Insert the first four CRAM rods using RMCS.

D. Reactor power and core flow are acceptable for continuous operation at their current levels.

Proposed Answer: B

Explanation: The given combination of power and flow place the plant in the Scram region of the power to flow map. N2-SOP-29 requires inserting a manual scram.

A. Plausible – A scram is required, not raising core flow. Plausible because it was lowering core flow that put the plant in the current operating region on the power to flow map.
C. Plausible – A scram is required, not inserting control rods with RMCS. Plausible because this would be required by N2-SOP-29 if operation were not in the Scram region of the power to flow map.
D. Plausible – The given combination of power and flow place the plant in the Scram region of the power to flow map. Plausible because if operation is plotted slightly to the right, then power and flow are acceptable for continuous operation.

Technical Reference(s): EM-950A, N2-SOP-29

Proposed references to be provided to applicants during examination: EM-950A (2 loop power/flow map)

Learning Objective: N2-202001-RBO-10

Question Source: Bank – 2012 NRC #40

Question History: 2012 NRC #40

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 2.1.19
	Importance Rating	3.9

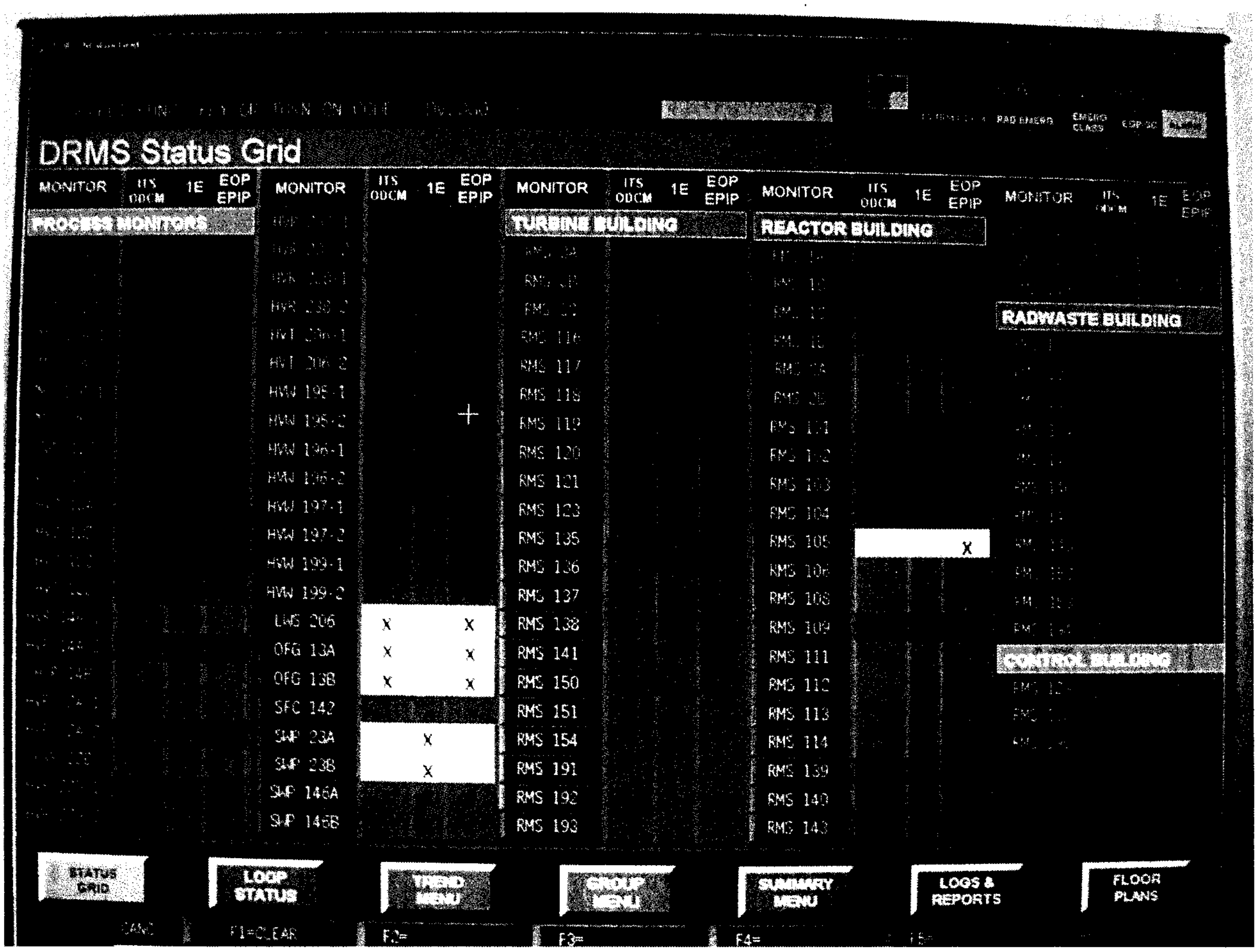
High Off-site Release Rate

Ability to use plant computers to evaluate system or component status.

Question: #54

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

- Irradiated fuel is being moved in the Spent Fuel Pool to support receipt of new fuel.
- Annunciator 851254, PROCESS AIRBORNE RADN MON ACTIVATED, alarms.
- The following indications are available on DRMS:



Which one of the following describes the status of 2HVR*AOD10B, Refueling Floor Exhaust Isolation, and 2HVR*UC413B, Reactor Building Emergency Recirculation Unit Cooler?

	2HVR*AOD10B	2HVR*UC413B
A.	Remains open	Remains in standby
B.	Remains open	Automatically starts
C.	Automatically closes	Remains in standby
D.	Automatically closes	Automatically starts

Proposed Answer: D

Explanation: The given DRMS display shows that both 2HVR-CAB14A-1 and 2HVR-CAB14B-1 are in high alarm (red). This results in 2HVR*AOD10B automatically closing and 2HVR*UC413B automatically starting.

A. Plausible – 2HVR*AOD10B automatically closes. Plausible because the annunciator could be in on the lower alert alarm (yellow), such that this damper would still be open. 2HVR*UC413B automatically starts. Plausible because the annunciator could be in on the lower alert alarm (yellow), such that this unit cooler would still be in standby. Also plausible because this is only a high rad condition, with no indication of high temperatures.
B. Plausible – 2HVR*AOD10B automatically closes. Plausible because the annunciator could be in on the lower alert alarm (yellow), such that this damper would still be open.
C. Plausible – 2HVR*UC413B automatically starts. Plausible because the annunciator could be in on the lower alert alarm (yellow), such that this unit cooler would still be in standby. Also plausible because this is only a high rad condition, with no indication of high temperatures.

Technical Reference(s): ARP 851245

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-272000-RBO-8

Question Source: Modified Bank – 2009 NRC #75

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 2.4.46
	Importance Rating	4.2

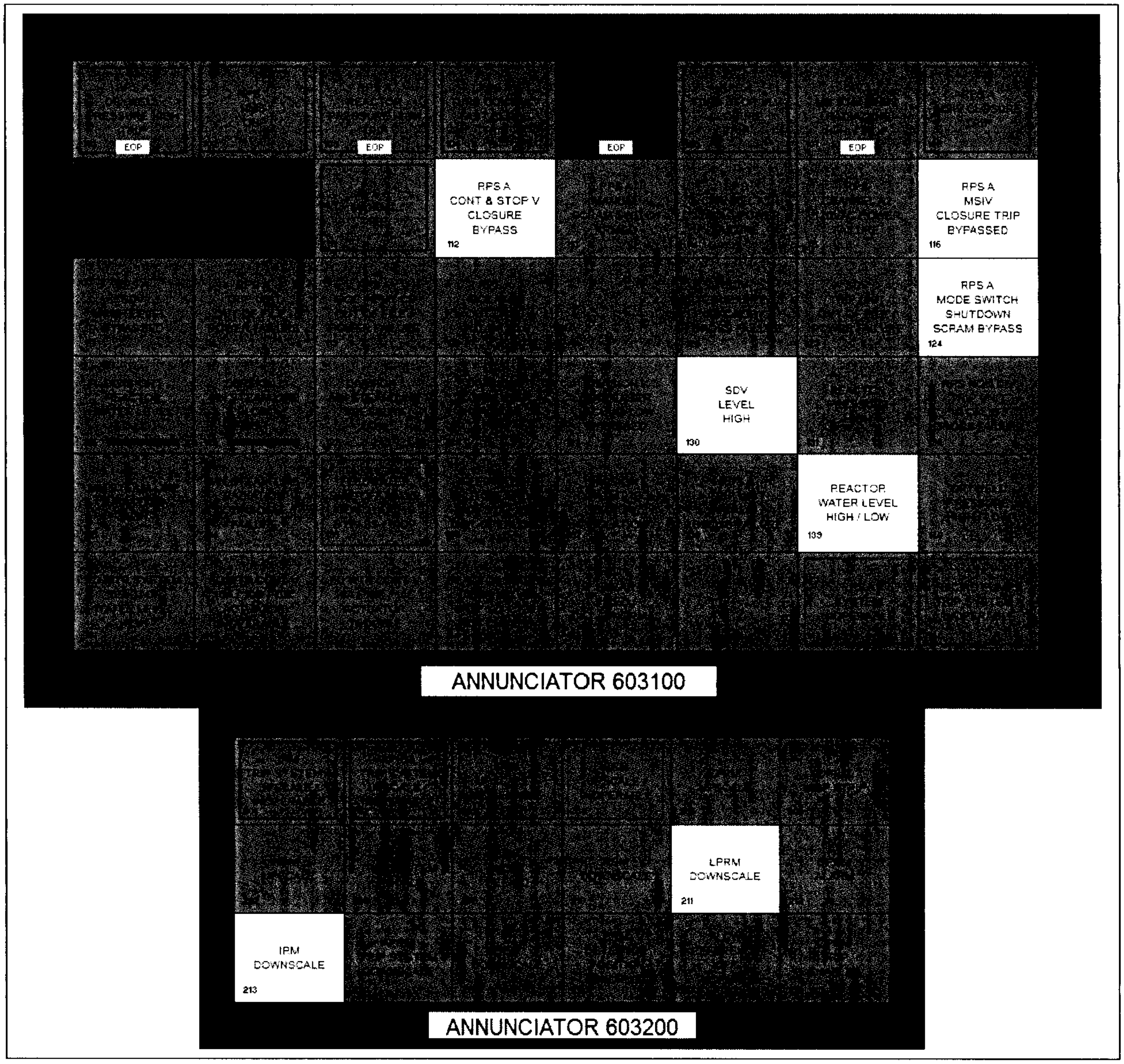
SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

Ability to verify that the alarms are consistent with the plant conditions.

Question: #55

The plant is operating at 95% power with the following:

- A manual Reactor scram is inserted.
- 10 seconds after the manual Reactor scram is inserted, the following annunciators are present on Panels 603100 and 603200 (alarms have been acknowledged):



Which one of the following describes the status of these annunciators based on expected plant conditions?

A. All annunciators are consistent with expected plant conditions.

B. Annunciator 603109 is NOT consistent with expected plant conditions.

C. Annunciator 603110 is NOT consistent with expected plant conditions.

D. Annunciator 603214 is NOT consistent with expected plant conditions.

Proposed Answer: D

Explanation: 10 seconds after a manual Reactor scram, the Scram Discharge Volume has filled to the point where the high level scram is received, therefore Annunciator 603109 being lit is consistent with expected plant response. 10 seconds after a manual Reactor scram from 95% power, Reactor water level is expected to be below the scram setpoint, therefore Annunciator 603110 being in alarm is consistent with expected plant response. Within 4 seconds after a manual Reactor scram, all control rods are expected to be fully inserted. This should result in APRMs being downscale by 10 seconds, such that Annunciator 603214 should be in alarm. Since the annunciator is extinguished, it is not consistent with expected plant response.

A. Plausible – Annunciator 603214 is extinguished when it should be in alarm and is therefore not consistent with expected plant response. Plausible because decay heat keeps thermal power above 4% for much longer than 10 seconds, but APRMs should be downscale based on a rapid drop in neutron production.
B. Plausible – 10 seconds after a manual Reactor scram, the Scram Discharge Volume is isolated and has filled to the point where the high level scram is received, therefore Annunciator 603109 being lit is consistent with expected plant response. Plausible because the candidate could think that since the setpoint for SDV level is 46.5 inches, it might take more time to fill the SDV to this level.
C. Plausible – 10 seconds after a manual Reactor scram from 95% power, Reactor water level is expected to be below the scram setpoint, therefore Annunciator 603110 being in alarm is consistent with expected plant response. Plausible because a manual scram was initiated, not an auto scram, and eventually the low level condition should clear.

Technical Reference(s): N2-SOP-101C, TS 3.1.4

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 2.1.28
	Importance Rating	4.1

Partial or Complete Loss of Instrument Air

Knowledge of the purpose and function of major system components and controls.

Question: #56

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

- A loss of all Instrument Air Compressors occurs.
- Instrument Air pressure is 98 psig and slowly lowering.

Which one of the following describes the operation of 2IAS-AOV171, INSTR AIR TO SERVICE AIR CROSSTIE VLV, as Instrument Air pressure continues to lower??

2IAS-AOV171 will ___(1)___ when Instrument Air pressure lowers to ___(2)___.

	(1)	(2)
A.	open	85 psig
B.	open	74 psig
C.	close	85 psig
D.	close	74 psig

Proposed Answer: C

Explanation: 2IAS-AOV171 is normally open and automatically closes when Instrument Air pressure lowers to 85 psig.

A. Plausible – 2IAS-AOV171 is normally open, not closed. Plausible that Instrument and Service Air would normally be separated and that this valve would open to allow Service Air to backup Instrument Air only in times of high usage.
B. Plausible – 2IAS-AOV171 is normally open, not closed. Plausible that Instrument and Service Air would normally be separated and that this valve would open to allow Service Air to backup Instrument Air only in times of high usage. 2IAS-AOV171 closes at 85 psig, not 74 psig. Plausible because 74 psig is referenced in N2-SOP-19 as the pressure at which MSIVs close.
D. Plausible – 2IAS-AOV171 closes at 85 psig, not 74 psig. Plausible because 74 psig is referenced in N2-SOP-19 as the pressure at which MSIVs close.

Technical Reference(s): N2-SOP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-RBO-5

Question Source: Bank – NMP1 2010 Audit #58

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 2.4.31
	Importance Rating	4.2

Suppression Pool High Water Temperature

Knowledge of annunciator alarms, indications, or response procedures.

Question: #57

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

- An SRV is leaking.
- Suppression Pool water temperature is 92°F and rising slowly.

Which one of the following describes the status of the Suppression Pool water temperature high annunciator (601560) and the need to enter N2-EOP-PC, Primary Containment Control?

	601560, SUPPRESSION POOL WATER TEMP HIGH	N2-EOP-PC Entry
A.	NOT in alarm	NOT required.
B.	NOT in alarm	required.
C.	In alarm	NOT required.
D.	In alarm	required.

Proposed Answer: D

Explanation: Annunciator 601560 is in alarm because temperature is greater than 85°F. N2-EOP-PC entry is required because temperature is greater than 90°F.

A. Plausible – Annunciator 601560 is in alarm because temperature is greater than 85°F and N2-EOP-PC entry is required because temperature is greater than 90°F. Plausible because 601550 is not yet in alarm (high high at 101°F).
B. Plausible – Annunciator 601560 is in alarm because temperature is greater than 85°F. Plausible because 601550 is not yet in alarm (high high at 101°F).
C. Plausible – N2-EOP-PC entry is required because temperature is greater than 90°F. Plausible because 601550 is not yet in alarm (high high at 101°F). Also plausible because this would be correct if temperature were 86-89°F.

Technical Reference(s): ARP 601550, ARP 601560, N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EA2.04
	Importance Rating	4.1

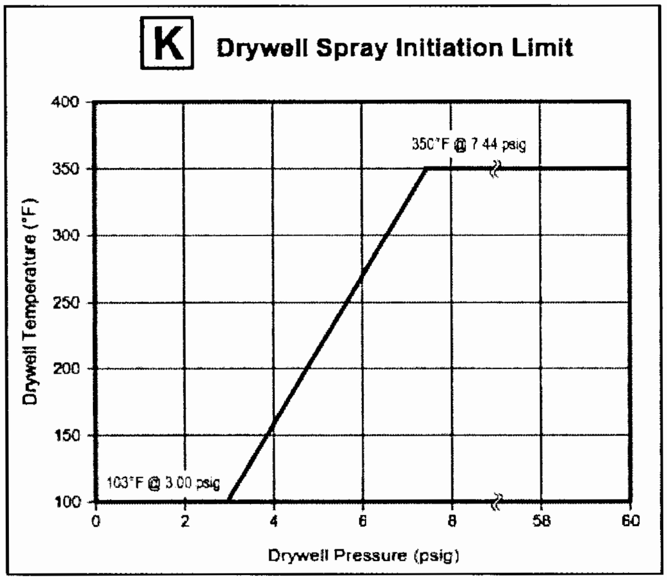
High Drywell Temperature

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell pressure

Question: #58

A loss of coolant accident has occurred with the following:

- Drywell pressure is 6.0 psig and slowly rising.
- Drywell temperature is 338°F and slowly rising.
- Suppression Chamber pressure is 5.0 psig and slowly rising.
- Suppression Pool water level is 204 feet and slowly rising.
- Suppression Chamber spray is in service.



Which one of the following describes the required operation of Drywell spray, in accordance with N2-EOP-PC, Primary Containment Control?

A. Do NOT initiate Drywell spray.

B. Initiate Drywell spray based on Drywell pressure.

C. Initiate Drywell spray based on Drywell temperature.

D. Initiate Drywell spray based on Suppression Chamber pressure.

Proposed Answer: A

Explanation: N2-EOP-PC Steps DWT-3 through DWT-8 require initiation of Drywell sprays before Drywell temperature reaches 340°F. Since Drywell temperature is 338°F and rising, Drywell spray would normally be required. However, the given combination of Drywell temperature and pressure are on the BAD side of the Drywell Spray Initiation Limit curve. Therefore, Drywell sprays are not allowed to be placed in service.

Note: The question meets the K/A by making the candidate interpret Drywell pressure and temperature on the Drywell Spray Initiation Limit curve and then determine the effect on the required actions in the EOPs for the high Drywell temperature and pressure conditions.

B. Plausible – Containment pressure is not yet high enough to require Drywell spray. Plausible because Drywell pressure is elevated and still rising with SC spray in service.
C. Plausible – Drywell spray is not allowed to be placed in service because operation is on the BAD side of the Drywell Spray Initiation Limit curve. Plausible because Drywell temperature is about to exceed the threshold of 340°F, which requires Drywell spray if on the GOOD side of the Drywell Spray Initiation Limit curve.
D. Plausible – Containment pressure is not yet high enough to require Drywell spray. Plausible because SC pressure is elevated and still rising with SC spray in service, and would require Drywell spray once it exceeds 10 psig.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Modified Bank – 2014 Cert #49

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295002 AK1.04
	Importance Rating	3.0

Loss of Main Condenser Vacuum

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

Question: #59

With the plant operating at 100% power, which one of the following is an indication of higher than normal Main Condenser air in-leakage?

	Offgas System Flow	Offgas System Pressure
A.	Rising	Rising
B.	Rising	Lowering
C.	Lowering	Rising
D.	Lowering	Lowering

Proposed Answer: A

Explanation: With the Main Condenser normally at a vacuum, rising air in-leakage causes vacuum to lower (or absolute pressure to rise) due to the presence of more non-condensable gases. This higher absolute pressure causes Offgas pressure to also rise. The rise in non-condensable gases results in more flow from the Steam Jet Air Ejectors to the Offgas system, which causes Offgas flow to rise.

B. Plausible – Offgas pressure rises, not lowers. Plausible because Main Condenser vacuum lowers.
C. Plausible – Offgas flow rises, not lowers. Plausible because Main Condenser vacuum lowers.
D. Plausible – Offgas flow rises, not lowers. Offgas pressure rises, not lowers. Plausible because Main Condenser vacuum lowers.

Technical Reference(s): N2-SOP-9, ARP 851306, ARP 122223

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-271000-RBO-8

Question Source: Bank – 2009 NRC #63

Question History: 2009 NRC #63

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295008 AK2.04
	Importance Rating	3.1

High Reactor Water Level

Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following: PCIS/NSSSS: Plant-Specific

Question: #60

The plant has experienced a Reactor scram from 100% power with the following:

- Reactor water level reached a low value of 100" following the scram.
- N2-SOP-101C, Reactor Scram, is being executed.
- Reactor water level is 210" and rising.
- It is desired to place Reactor Water Cleanup (WCS) reject in service.

Which one of the following describes the status of WCS isolation valves, 2WCS*MOV102 and 2WCS*MOV112, for placing WCS reject in service, in accordance with N2-OP-37, Reactor Water Cleanup System?

2WCS*MOV102 and 2WCS*MOV112 are...

A. open because Reactor water level did NOT go low enough to close them.

B. open because they automatically re-opened when Reactor water level was restored.

C. closed. They are re-opened by taking their control switches to OPEN. NO additional action is required prior to opening.

D. closed. They are re-opened by depressing isolation logic reset pushbuttons B22H-S32 and B22H-S33, then taking their control switches to OPEN.

Proposed Answer: D

Explanation: 2WCS*MOV102 and 2WCS*MOV112 automatically close when Reactor water level lowers below 108.8". When Reactor water level rises back above 108.8", the isolation signal remains sealed-in until manual action is taken to reset it (depressing pushbuttons B22H-S32 and B22H-S33). Once these pushbuttons are depressed and released, the valve control switches also must be taken to OPEN before the valves will re-open.

Note: The question meets the K/A by presenting a situation with high Reactor water level (210" is above the high end of the band in N2-EOP-RPV) and requiring knowledge of how PCIS/NSSSS affects an operation related to this high Reactor water level (placing WCS reject in service).

A. Plausible – 2WCS*MOV102 and 2WCS*MOV112 automatically close when Reactor water level lowers below 108.8". When Reactor water level rises back above 108.8", the isolation signal remains sealed-in until manual action is taken to reset it. Plausible because this would be correct if level did not get quite so low.
B. Plausible – 2WCS*MOV102 and 2WCS*MOV112 automatically close when Reactor water level lowers below 108.8". When Reactor water level rises back above 108.8", the isolation signal remains sealed-in until manual action is taken to reset it. Plausible because this would be the answer if the isolation signal automatically cleared when level rose above 108.8" and the valves automatically re-opened.
C. Plausible – Once these pushbuttons are depressed and released, the valve control switches also must be taken to OPEN before the valves will re-open. Plausible because nothing in the procedure has the operator take these switches to the CLOSE position prior to depressing the reset pushbuttons.

Technical Reference(s): N2-OP-83 Attachment 1, N2-OP-37 Section H.1.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295014 AK3.02
	Importance Rating	3.7

Inadvertent Reactivity Addition

Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION: Control rod blocks

Question: #61

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

- Turbine Stop Valve 1D spuriously closes.
- Reactor pressure rises to 1040 psig.
- Reactor power rises to 109%.
- Reactor water level lowers to 175".

Which one of the following describes the resulting status of control rod blocks and a Reactor scram?

	Control Rod Block	Reactor Scram
A.	Does NOT occur	Does NOT occur
B.	Does NOT occur	Occurs
C.	Occurs	Does NOT occur
D.	Occurs	Occurs

Proposed Answer: C

Explanation: Reactor power causes both control rod blocks and scrams. At rated flow conditions, APRMs cause a control rod block at a maximum of 107.5% and a Reactor scram at a maximum of 118%. TSV position, Reactor pressure, and Reactor water level can all cause Reactor scrams, but not control rod blocks. The given values of Reactor pressure and water level cause alarms, but are not severe enough to cause a Reactor scram. Additionally, only one TSV closing does not make up the TSV closure scram logic. Therefore, a control rod block occurs due to Reactor power level, but no scram occurs.

A. Plausible – A control rod block occurs due to Reactor power >107.5%. Plausible because this would be correct if Reactor power rose slightly less.
B. Plausible – A control rod block occurs due to Reactor power >107.5%. A Reactor scram does not occur because Reactor press and water level did not exceed the associated thresholds. Plausible because this would be correct if Reactor power rose slightly less but Reactor pressure or level changed slightly more.
D. Plausible – A Reactor scram does not occur because Reactor press and water level did not exceed the associated thresholds. Plausible because this would be correct if Reactor pressure or level changed slightly more.

Technical Reference(s): ARP 603442

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-201002-RBO-5

Question Source: Modified Bank – NMP1 2013 NRC #24

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295007 AA1.05
	Importance Rating	3.7

High Reactor Pressure

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: Reactor/turbine pressure regulating system

Question: #62

The plant is operating at 26% power during a startup with the following:

- Power ascension is on hold while Engineering investigates a Main Generator winding issue.
- The Load Limit has been lowered to limit Main Generator load to 250 MWe.
- The LOAD LIMIT LIMITING light is illuminated.
- The MAXIMUM COMBINED FLOW LIMIT is set to control at 115%.

Then, a Recirc Flow Control failure causes Reactor power to rise by 10%.

Which one of the following describes the response of the Electrohydraulic Control System?

A. The Control valves remain in the current position and the Bypass valves will begin to open.

B. The Control and Bypass valves remain in the current position. Reactor pressure rises until the Reactor scrams.

C. The Control valves open further until the 250 MWe Load Limit is reached. Then, the Bypass valves will begin to open.

D. The Control valves open further until the 250 MWe Load Limit is reached. The Bypass valves remain in the current position. Then, Reactor pressure rises until the Reactor scrams.

Proposed Answer: A

Explanation: With the Load Limit lowered and the LOAD LIMIT SETTING light illuminated Control valves will not open any further. With the Max Combined Flow Limit set at 115%, the Bypass valves are still capable of opening and will. The Bypass valves are sized such that they can adequately control Reactor pressure given a 10% rise in power.

Note: The question meets the K/A by giving a situation where Reactor power causes a rise in Reactor pressure which in turn causes a response in the Reactor/turbine pressure regulating system that mitigates the rise in Reactor pressure to prevent a high pressure scram, including an abnormal pressure control lineup (TCVs and TBVs open simultaneously while Main Generator is on-line).

B. Plausible – Bypass valves open and limit Reactor pressure. Plausible because this would be the response if the Max Combined Flow Limiter were lowered enough that the COMBINED FLOW LIMITED light was illuminated.
C. Plausible – With the Load Limit lowered and the LOAD LIMIT SETTING light illuminated Control valves will not open any further. Plausible because this would be the response if the LOAD LIMIT SETTING light was extinguished.
D. Plausible – With the Load Limit lowered and the LOAD LIMIT SETTING light illuminated Control valves will not open any further. Bypass valves open and limit Reactor pressure. Plausible because this would be the response if the Max Combined Flow Limiter were lowered enough to partially limit Control valve opening.

Technical Reference(s): N2-OP-23, ARP 851150, N2-248000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-248000-RBO-5

Question Source: Bank – 2012 NRC #46

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295033 EA2.03
	Importance Rating	3.7

High Secondary Containment Area Radiation Levels

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Cause of high area radiation

Question: #63

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

- Annunciator 851244, REACTOR BLDG AREA RADN MON ACTIVATED, alarms.
- Area Radiation Monitor 2RMS-RE108, CRD MAINT. ROOM, is slightly above the high alarm setpoint and slowly rising.

Which one of the following identifies the need to enter N2-EOP-SC, Secondary Containment Control, and N2-EOP-RR, Radioactivity Release Control?

	N2-EOP-SC Entry	N2-EOP-RR Entry
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: B

Explanation: 2RMS-RE108 is an area radiation monitor in the Secondary Containment. With this area radiation monitor above the alarm setpoint, N2-EOP-SC must be entered. This alarm indicates a potential radioactivity release in the Reactor Building, but does not indicate a radioactivity release offsite, therefore N2-EOP-RR entry is not required.

<u>Note:</u> The question meets the K/A by presenting indication of a high radiation condition in the Secondary Containment (annunciator 851244) and requiring the candidate to determine how the specific radiation monitor in alarm affects EOP entry conditions.

A. Plausible – This alarm indicates a potential radioactivity release in the Reactor Building, but does not indicate a radioactivity release offsite, therefore N2-EOP-RR entry is not required.
C. Plausible – This ARM does require N2-EOP-SC entry. Plausible because some of the radiation monitors that input to annunciator 851244 would not require entry into N2-EOP-SC. This alarm indicates a potential radioactivity release in the Reactor Building, but does not indicate a radioactivity release offsite, therefore N2-EOP-RR entry is not required.
D. Plausible – This ARM does require N2-EOP-SC entry. Plausible because some of the radiation monitors that input to annunciator 851244 would not require entry into N2-EOP-SC.

Technical Reference(s): ARP 851244

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-272000-RBO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 2.2.40
	Importance Rating	3.4

High Drywell Pressure

Ability to apply technical specifications for a system.

Question: #64

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

- Indications of rising leakage in the Drywell have been observed.
- Drywell pressure, average temperature, and floor drain leak rate are given in the table below.

	One (1) hour ago	**Now**
Drywell Pressure	14.99 psia	15.61 psia
Drywell Average Temperature	110°F	121°F
Drywell Floor Drain Leak Rate	1.2 gpm	6.5 gpm

Given the following Technical Specifications:

(1) 3.4.5, RCS Operational Leakage
(2) 3.6.1.4, Drywell and Suppression Chamber Pressure
(3) 3.6.1.5, Drywell Air Temperature

Which one of the following identifies the total number of these Technical Specifications that have LCO limits exceeded, if any, based on the available data?

A. 0

B. 1

C. 2

D. 3

Proposed Answer: C

Explanation: TS 3.4.5 requires Drywell unidentified leakage to be ≤ 5 gpm. Drywell floor drain data shows an unidentified leakage rate of 6.5 gpm. This exceeds the LCO limit of TS 3.4.5. TS 3.6.1.4 requires Drywell pressure to be 14.2-15.45 psia. The given Drywell pressure of 15.61 psia exceeds the LCO limit of TS 3.6.1.4. TS 3.6.1.5 requires Drywell average temperature to be ≤150°F. The given Drywell average temperature of 121°F is elevated, but within this limit.

A. Plausible – TS 3.4.5 has a limit exceeded. Plausible because unidentified leakage is 6.5 gpm, which is below the identified leakage limit of 25 gpm. TS 3.6.1.4 limit is exceeded. Plausible because Drywell pressure is still well below the scram setpoint of 1.68 psig (~16.38 psia).
B. Plausible – TS 3.4.5 has a limit exceeded. Plausible because unidentified leakage is 6.5 gpm, which is below the identified leakage limit of 25 gpm. TS 3.6.1.4 limit is exceeded. Plausible because Drywell pressure is still well below the scram setpoint of 1.68 psig (~16.38 psia).
D. Plausible – TS 3.6.1.5 limit is not exceeded. Plausible because Drywell temperature is elevated and this would be correct if temperature rose further.

Technical Reference(s): TS 3.4.5, TS 3.6.1.4, TS 3.6.1.5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295032 EA2.01
	Importance Rating	3.8

High Secondary Containment Area Temperature

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Area temperature

Question: #65

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

- A steam leak has developed in the Reactor Building.
- The US has determined access to Reactor Building elevation 206', the RCIC pump room, and the associated pipe chase is required to support EOP actions.
- The following Reactor Building temperatures have been recorded:

N2-EOP-6.28
DETERMINING REACTOR BUILDING TEMPERATURES
Table 3 Reactor Bldg Areas
(Partial Table Only)

RX BLDG RCIC PMP ROOM RX BLDG RADIOACTIVE PIPE CHASE EL 206	
2LDS*TRSH2A(1) *(P632)*	198°F
2LDS*TRSH2A(2) *(P632)*	185°F
2LDS*TRSH2B(1) *(P642)*	150°F
2LDS*TRSH2B(2) *(P642)*	162°F

RX BLDG GENL AREA EL 206	
2LDS*TRSH4A(7) *(P632)*	131°F

RX BLDG GENL AREA EL 206	
2LDS*TRSH4B(7) *(P642)*	124°F

Which one of the following identifies the number of Reactor Building Areas that have exceeded the Maximum Safe Temperature, if any, in accordance with N2-EOP-SC?

A. 0

B. 1

C. 2

D. 3

Proposed Answer: B

Explanation: The Maximum Safe Area temperature is either 135°F (if personnel access is required for support of EOP actions) or 212°F. Since access is required to the given areas, the 135°F limitation must be used. Each block of temperatures on the provided attachment represent a different General Area. All temperatures in the **RX BLDG RCIC PMP ROOM / RX BLDG RADIOACTIVE PIPE CHASE EL 206** General Area are > 135°F. This means one General Area is above the Maximum Safe Temperature.

A. Plausible – One General Area is above the 135°F Maximum Safe Temperature for personnel access. Plausible because all temperatures are below the 212°F limit if personnel access was not required.
C. Plausible – Both the RCIC pump room and pipe chase temperatures are above Max Safe, however these areas all count as one General Area.
D. Plausible – Only one General Area is above Maximum Safe Temperature. Plausible because temperatures are given for three General Areas, and all temperatures are significantly elevated.

Technical Reference(s): N2-EOP-SC, N2-EOP-6.28 Table 3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Bank – 2015 CERT #63

Question History: 2015 CERT #63

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.14
	Importance Rating	3.1

Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.

Question: #66

Given the following events:

(1) Re-energizing switchgear 2NPS-SWG001.
(2) An Injury in the plant requires Fire Brigade response.
(3) The Reactor reaches criticality during a startup.

Which one of the following identifies the event(s) that must be announced to the plant?

A. (1) and (2) only

B. (2) and (3) only

C. (1) and (3) only

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

Proposed Answer: D

Explanation: Event (1) must be announced because it is energization of a major electrical switchgear. Event (2) must be announced because it requires alerting plant personnel to an injury in the plant. Event (3) must be announced for reaching criticality.

A. Plausible – Event (3) must also be announced. Plausible because Event (3) is an expected condition during a Reactor startup.
B. Plausible – Event (1) must also be announced. Plausible because event (1) is a normal evolution confined to a single location.
C. Plausible – Event (2) must also be announced. Plausible if the candidate believes that event (2) only warrants use of radios to dispatch Fire Brigade.

Technical Reference(s): OP-AA-104-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – NMP1 2013 NRC #67

Question History: NMP1 2013 NRC #67

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.9
	Importance Rating	2.9

Ability to direct personnel activities inside the control room.

Question: #67

Given the following individuals:

(1) A qualified, non-licensed Equipment Operator who has been selected for license class but has not yet begun formal license class training.

(2) An instant SRO license candidate who is performing the pre-license class plant familiarization guide.

(3) An RO license candidate who is completing the in-plant OJT phase of license class.

Which one of the following identifies the individual(s) that may be allowed to perform reactivity manipulations in the plant under the guidance of a licensed Reactor Operator, in accordance with OP-AA-103-103, Operation of Plant Equipment?

A. (3) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: OP-AA-103-103 allows an RO to direct a non-licensed individual to perform reactivity manipulations under their guidance in certain circumstances:

- The non-licensed individual shall be an active participant in an ongoing Licensed Operator Training class.
- The individual shall have met the knowledge requirements as defined in the Licensed Operator Training program.

The knowledge requirements in the Licensed Operator Training program include completion of related Generic Fundamentals and reactivity control system training. Since individual (3) is the only one having completed those requirements to get to the in-plant phase of license class, they are the only individual allowed to perform reactivity manipulations under instruction.

B. Plausible – Individual (1) is qualified to perform non-licensed activities and will eventually perform reactivity manipulations as a license candidate, however since they are not yet in the formal training class, they are not yet allowed to perform reactivity manipulations under instruction.
C. Plausible – Individual (2) is in a preparatory phase of license class, but has not yet completed the prerequisite training for reactivity manipulations, and therefore is not yet allowed to perform reactivity manipulations.
D. Plausible – Individual (1) is qualified to perform non-licensed activities and will eventually perform reactivity manipulations as a license candidate, however since they are not yet in the formal training class, they are not yet allowed to perform reactivity manipulations under instruction. Individual (2) is in a preparatory phase of license class, but has not yet completed the prerequisite training for reactivity manipulations, and therefore is not yet allowed to perform reactivity manipulations.

Technical Reference(s): OP-AA-103-103, OP-AA-300

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – 2014 Cert #74

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.36
	Importance Rating	3.1

Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question: #68

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

(1) The heater for Hydrogen Recombiner A is tagged out of service for a breaker inspection.

(2) Diesel Generator 2EGS*EG1 CONTROL MODE switch is taken to OFF at the Engine Control Panel.

Which one of the following identifies the event(s) that will require Technical Specification Condition entry, if any?

A. Neither event

B. Event (1) only

C. Event (2) only

D. Events (1) and (2)

Proposed Answer: C

Explanation: Technical Specifications no longer require operability of the Hydrogen Recombiner heaters, therefore the first event does not require Technical Specification condition entry. Placing 2EGS*EG1 CONTROL MODE switch in OFF disables all auto-starts, which makes the Diesel Generator inoperable. Technical Specification 3.8.1 entry is required for this event.

A. Plausible – Placing 2EGS*EG1 CONTROL MODE switch in OFF disables all auto-starts, which makes the Diesel Generator inoperable. Technical Specification 3.8.1 entry is required for this event. Plausible because two other Diesel Generators remain operable and this would not require TS Condition entry in Mode 4 or 5.
B. Plausible – Placing 2EGS*EG1 CONTROL MODE switch in OFF disables all auto-starts, which makes the Diesel Generator inoperable. Technical Specification 3.8.1 entry is required for this event. Plausible because two other Diesel Generators remain operable and this would not require TS Condition entry in Mode 4 or 5. Technical Specifications no longer require operability of the Hydrogen Recombiner heaters, therefore the first event does not require Technical Specification condition entry. Plausible because Technical Specifications used to require operability of this component, and other functions of the Hydrogen Recombiner system are still required for Technical Specifications.
D. Plausible – Technical Specifications no longer require operability of the Hydrogen Recombiner heaters, therefore the first event does not require Technical Specification condition entry. Plausible because Technical Specifications used to require operability of this component, and other functions of the Hydrogen Recombiner system are still required for Technical Specifications.

Technical Reference(s): N2-OP-100A, Technical Specifications

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-14

Question Source: Modified Bank – NMP1 2013 Cert #2

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	3
Group #	
K/A #	2.2.13
Importance Rating	4.1

Knowledge of tagging and clearance procedures.

Question: #69

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

- Electrical Maintenance is preparing to perform an inspection on a lighting panel.
- During the inspection, it is desired for the Electricians to be able to both:
  - Open the panel's disconnect switch for personnel protection, and
  - Close the panel's disconnect switch for periodic verifications.
- It is also desired for the Tagout to be continuously hung during the activity, such that repeated tag clearing and re-hanging is NOT required.

Which one of the following describes a tagging arrangement that will allow this maintenance activity, in accordance with OP-CE-109-101, Clearance and Tagging?

Tag the panel's disconnect switch with...

A. a Danger Tag, only.

B. an Information Tag, only.

C. a Special Condition Tag and lock, only.

D. both a Danger Tag and a Special Condition Tag simultaneously.

Proposed Answer: C

Explanation: Components tagged with a Special Condition Tag may be manipulated by the Tagout Holder, or a person under the direction of the Tagout Holder, without removing the Special Condition Tag. These components may be used for personnel protection.

A. Plausible – A component tagged with a danger tag may not be manipulated without removing the tag. Plausible because the danger tag provides the personnel protection that is desired.
B. Plausible – A component tagged with an information tag only may not be used for personnel protection. Plausible because a component with an information tag may be manipulated as requested.
D. Plausible – The presence of the Danger Tag prevents the ability to manipulate the component without lifting the tag. Plausible because the Special Condition Tag allows manipulation while the Danger Tag provides personnel protection. However, these are not allowed to hang simultaneously on the same component.

Technical Reference(s): OP-CE-109-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2017 NRC #75

Question History: NMP1 2017 NRC #75

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.11
	Importance Rating	3.8

Ability to control radiation releases.

Question: #70

A loss of coolant accident has resulted in the following:

- Primary Containment venting is required due to approaching the Primary Containment Pressure Limit (PCPL).
- Suppression Pool water level is 210' and stable.

Which one of the following describes the path required to be used to vent the Containment and the reason why, in accordance with N2-EOP-PC, Primary Containment Control?

Vent from the...

A. Drywell to more quickly reduce Containment pressure.

B. Suppression Chamber to minimize cycling of the Suppression Chamber-to-Drywell vacuum breakers.

C. Drywell because Suppression Pool water level is too high for venting from the Suppression Chamber.

D. Suppression Chamber to better scrub fission products from the Containment atmosphere before release.

Proposed Answer: D

Explanation: N2-EOP-PC requires venting from the Suppression Chamber as long as Suppression Pool water level is below 217'. This is to scrub the Containment atmosphere through the Suppression Pool water volume prior to release to lower the radioactive release.

A. Plausible – Venting should be from the Suppression Chamber. Plausible because Drywell pressure is expected to be higher than Suppression Chamber pressure under LOCA conditions, which would result in a faster pressure reduction if the Drywell were vented.
B. Plausible – The reason is not to minimize cycling of vacuum breakers. Plausible because lowering Suppression Chamber first would minimize cycling of vacuum breakers.
C. Plausible – Venting should be from the Suppression Chamber. Plausible because Suppression Pool water level is elevated, and if it were above 217', then Drywell venting would be required.

Technical Reference(s): N2-EOP-PC, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – JAF 2016 NRC #64

Question History: JAF 2016 NRC #64

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.4
	Importance Rating	3.2

Knowledge of radiation exposure limits under normal or emergency conditions.

Question: #71

The plant is shutdown. Conditions are as follows:

- A 31 year old Operator is entering the Drywell for a job.
- General area dose rate is 3 Rem/hr.
- The Operator's TEDE for the year is 1610 mRem.
- The Operator's lifetime exposure is 25 Rem.
- No dose extensions have been obtained.
- It will take 75 minutes to complete the job.

Which one of the following describes the dose gained in order to complete the job, in accordance with RP-AA-203, Exposure Control and Authorization?

The operator's dose...

A. remains within the normal dose control level and does NOT require an extension.

B. requires an extension beyond the normal dose control level, but does NOT exceed the Exelon annual administrative limit.

C. exceeds the Exelon annual administrative limit, but NOT the Federal annual limit.

D. exceeds both the Exelon annual administrative limit and the Federal annual limit.

Proposed Answer: D

Explanation: The normal administrative dose control level without any extensions is 2000 mRem/yr TEDE. The Exelon annual administrative limit and the Federal annual limit are both 5000 mRem. The total dose the worker will have received after completing the job will be 1610+3000(75/60)=5360 mRem. This exceeds the Exelon annual administrative limit and the Federal limit.

A. Plausible – The dose exceeds the normal control level of 2000 mRem/yr. Plausible because the Operator's initial dose is below this level.
B. Plausible – The dose exceeds the Exelon administrative limit of 5000 mRem/yr. Plausible because the Operator's initial dose is well below this level.
C. Plausible – The dose exceeds the federal limit of 5000 mRem/yr. Plausible because the Operator's initial dose is well below this level.

Technical Reference(s): RP-AA-203

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – 2015 NRC #71

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.6
	Importance Rating	3.7

Knowledge of EOP mitigation strategies.

Question: #72

The plant has experienced a failure to scram with the following:

- Multiple control rods are stuck at position 48.
- N2-EOP-C5, Failure to Scram, is being executed.
- Reactor water level has been intentionally lowered.
- Reactor water level is now to be maintained between -39" and 100".
- All injection systems are available.

Which one of the following describes the allowed use of injection systems based on location, in accordance with N2-EOP-C5?

With the exception of SLC, RCIC, and CRD, using systems that inject outside the core shroud is ___(1)___ and using systems that inject inside the core shroud is ___(2)___.

	(1)	(2)
A.	allowed	allowed
B.	allowed	NOT allowed
C.	NOT allowed	allowed
D.	NOT allowed	NOT allowed

Proposed Answer: B

Explanation: N2-EOP-C5, Failure to Scram, contains a Reactor water level mitigation strategy designed to control Reactor power. One of the elements of this strategy is to limit the use of RPV injection sources that inject inside the core shroud. This strategy is implemented through the N2-EOP-C5 step that states, "Using only Preferred ATWS Systems (Detail G), restore and maintain RPV water level between -39 in. (Fig Z) and the level you lowered it to." Detail G contains systems that inject outside the core shroud and does not contain systems such as LPCS and HPCS that inject inside the core shroud. This strategy prevents injecting cold water directly into the core region, which could cause power excursions.

A. Plausible – Injecting with systems inside the core shroud is not allowed. Plausible because if other injection systems were not available and Reactor water level control was challenged, then injecting with systems inside the core shroud would be allowed after first performing an RPV blowdown.
C. Plausible – Injecting with systems outside the core shroud is allowed. Plausible because even these systems are terminated while Reactor water level is intentionally lowered, but they become allowed again once level is controlled in the lower band. Injecting with systems inside the core shroud is not allowed. Plausible because if other injection systems were not available and Reactor water level control was challenged, then injecting with systems inside the core shroud would be allowed after first performing an RPV blowdown.
D. Plausible – Injecting with systems outside the core shroud is allowed. Plausible because even these systems are terminated while Reactor water level is intentionally lowered, but they become allowed again once level is controlled in the lower band.

Technical Reference(s): N2-EOP-C5, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Bank – JAF 9/12 NRC #75

Question History: JAF 9/12 NRC #75

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Q #73 withheld Fm
Public Disclosure due
to Security Info Content

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.39
	Importance Rating	3.9

Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Question: #74

The plant is operating at 50% power when an EHC Pressure Regulator malfunction results in Reactor pressure slowly rising from 1010 psig to 1040 psig.

Which one of the following describes (1) the maximum Reactor pressure permitted and (2) the required completion time for restoring Reactor pressure, in accordance with Technical Specifications?

A. (1) 1020 psig
(2) 15 minutes

B. (1) 1020 psig
(2) 1 hour

C. (1) 1035 psig
(2) 15 minutes

D. (1) 1035 psig
(2) 1 hour

Proposed Answer: C

Explanation: Technical Specification 3.4.12 requires Reactor pressure to be ≤1035 psig. Condition A allows 15 minutes to restore pressure below this value.

A. Plausible – Maximum pressure is 1035 psig. 1020 psig is plausible because it is the maximum normal pressure at 100% power and close to 1035 psig.
B. Plausible – Maximum pressure is 1035 psig. 1020 psig is plausible because it is the maximum normal pressure at 100% power and close to 1035 psig. 15 minutes is the required time. 1 hour is plausible because many Technical Specification completion times are 1 hour and it is the maximum an RO would be tested on.
D. Plausible – 1 hour is plausible because many Technical Specification completion times are 1 hour and it is the maximum an RO would be tested on.

Technical Reference(s): Technical Specification 3.4.12

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-101001-RBO-14

Question Source: Bank – 2009 NRC #100

Question History: 2009 NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.18
	Importance Rating	3.3

Knowledge of the specific bases for EOPs.

Question: #75

Which one of the following describes the basis for maintaining Suppression Pool temperature below the Heat Capacity Temperature Limit, in accordance with N2-EOP-PC, Primary Containment Control?

Ensure that...

A. ECCS pump Net Positive Suction Head limits are NOT violated.

B. ECCS suction piping design temperature limits are NOT exceeded.

C. Suppression Chamber pressure stays within the Primary Containment Pressure Limit.

D. Suppression Chamber pressure stays within the Pressure Suppression Pressure Limit.

Proposed Answer: C

Explanation: The Heat Capacity Temperature Limit is the highest Suppression Pool temperature from which an RPV Blowdown will not raise:

- Suppression Chamber temperature above the design value (270°F), or
- Suppression Chamber pressure above the Primary Containment Pressure Limit,

before the rate of energy transfer from the RPV to the Primary Containment is within the capacity of the Primary Containment vent. The temperature at which Suppression Chamber pressure equals the Primary Containment Pressure Limit is limiting at NMP2.

A. Plausible – High Suppression Pool water temperature does reduce available ECCS pump NPSH, but this is not the basis behind HCTL.
B. Plausible – High Suppression Pool water temperature does raise ECCS suction temperature, and part of the generic basis behind HCTL is Suppression Pool temperature exceeding a mechanical design value.
D. Plausible – PSP is a limit on Suppression Chamber pressure, similar to PCPL; but it is a lower limit and not the basis behind HCTL.

Technical Reference(s): NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – NMP1 2015 Cert #64

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295019 AA2.01
	Importance Rating	3.6

Partial or Complete Loss of Instrument Air

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure

Question: #76

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

- A significant Instrument Air leak has occurred.
- N2-SOP-19, Loss of Instrument Air, is being executed.
- 2IAS-PI194 (RB el. 261'), Inst Air Rcvr 2IAS-TK3 Pressure, indicates 72 psig and stable.
- 2IAS-PI101 (P851), Instrument Air Pressure, indicates 72 psig and stable.

Which one of the following describes the need for a Reactor scram and the operability of the outboard MSIVs based on the given air pressures, in accordance with N2-SOP-19?

	Reactor scram	Outboard MSIVs
A.	Required	Operable
B.	Required	Inoperable
C.	NOT required	Operable
D.	NOT required	Inoperable

Proposed Answer: B

Explanation: N2-SOP-19 includes the following condition steps:

IF	THEN
2IAS-PI101 (P851), Instrument air pressure, lowers to < 85 psig,	Verify 2IAS-AOV171 auto closes <u>AND</u> dispatch an Operator to monitor 2RDS-PI133 & 2IAS-PI194 (RB el. 261').
2IAS-PI194 (RB el. 261'), Inst air rcvr 2IAS-TK3 pressure, lowers to < 74 psig,	**SCRAM** the Reactor per N2-SOP-101C <u>AND</u> declare the OUTBOARD MSIV's inoperable.
2RDS-PI133 (RB el. 261'), scram air header pressure, lowers to < 60 psig,	**SCRAM** the Reactor per N2-SOP-101C.
2IAS-PI101 (P851), Instrument air pressure, lowers to < 70 psig,	

Reactor Scram required <u>ONLY</u> if control rod(s) are withdrawn.

Due to air pressure on 2IAS-PI194 (RB el. 261') being below 74 psig, N2-SOP-19 requires both entering N2-SOP-101C to scram the Reactor and declaring the outboard MSIVs inoperable.

<u>Note:</u> The question meets SRO only requirements because the candidate must assess plant conditions (two given IA pressures) and then use this assessment to determine if an additional procedure is yet required to be implemented (N2-SOP-101C). This cannot be answered solely by knowing systems knowledge, immediate operator actions, SOP entry conditions, or overall mitigative strategy. Additional SRO knowledge is tested in the operability determination of the MSIVs.

A. Plausible – The outboard MSIVs must be declared inoperable. Plausible because air pressure on 2IAS-PI101 (P851) is above its 70 psig action threshold.
C. Plausible – A Reactor scram is required. The outboard MSIVs must be declared inoperable. Plausible because air pressure on 2IAS-PI101 (P851) is above its 70 psig action threshold.
D. Plausible – A Reactor scram is required. Plausible because air pressure on 2IAS-PI101 (P851) is above its 70 psig action threshold.

Technical Reference(s): N2-SOP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP19C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 EA2.04
	Importance Rating	4.5

High Off-site Release Rate

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Source of off-site release

Question: #77

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

- A seismic event occurs.
- Significant damage has occurred to the Spent Fuel Pool.
- The Refuel Floor has been evacuated.
- The Reactor Building has isolated based on high airborne radiation levels.
- GTS Trains A and B are operating and maintaining the RB DP negative.
- The Shift Manager has declared a Site Area Emergency based on Main Stack Effluent radiation levels.
- Main Stack Effluent radiation levels are continuing to rise and are about to reach the General Emergency level.
- Operators have verified that the source of the release is from the damaged Spent Fuel Pool only.

Which one of the following describes how the source of the release is classified and the need for entry into N2-EOP-C2, RPV Blowdown, in accordance with N2-EOP-RR, Radioactivity Release Control, based on this release from the Spent Fuel Pool?

	Source of Release	Entry into N2-EOP-C2, RPV Blowdown
A.	Primary system	Required
B.	Primary system	NOT required
C.	NOT a primary system	Required
D.	NOT a primary system	NOT required

Proposed Answer: D

Explanation: The given release is caused by damage to the Spent Fuel Pool, which does not meet the EOP definition of a primary system (not connected directly to the RPV and reducing RPV pressure will not decrease release rate). N2-EOP-RR entry is still required based on the release being above the Emergency Plan Alert level. This entry is not contingent on the source of the release. However, the requirements in N2-EOP-RR to transition to N2-EOP-RPV and N2-EOP-C2 are based on the release being both un-isolable and from a primary system. Therefore, since the release is from a non-primary system, even if the offsite release rate reaches the General Emergency level, entry into N2-EOP-C2 will not be required.

Note: The question meets the K/A because it presents a high offsite release rate and requires interpreting the source of the release to determine the proper execution of the radioactivity release control emergency operating procedure.

Note: The question meets SRO only guidelines because it requires the candidate to assess plant conditions (source of leak, status of release rate) and then use this assessment to determine if an additional procedure is yet required to be implemented (N2-EOP-C2) based on a diagnostic step / decision point in N2-EOP-RR. This cannot be answered solely by knowing systems knowledge, immediate operator actions, EOP entry conditions, or overall mitigative strategy.

A. Plausible – The release is NOT from a primary system. Plausible because the release is from a source that contains irradiated fuel. Entry into N2-EOP-C2 is not required. Plausible because, with release rate about to exceed the General Emergency level, entry into N2-EOP-C2 would be required if the source of the release was from a different system, such as HPCI, RCIC, or RWCU.
B. Plausible – The release is NOT from a primary system. Plausible because the release is from a source that contains irradiated fuel.
C. Plausible – Entry into N2-EOP-C2 is not required. Plausible because, with release rate about to exceed the General Emergency level, entry into N2-EOP-C2 would be required if the source of the release was from a different system, such as HPCI, RCIC, or RWCU.

Technical Reference(s): N2-EOP-RR, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRRC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.03
	Importance Rating	3.5

Loss of Shutdown Cooling

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water level

Question: #78

The plant is shutdown with the following:

- Shutdown Cooling is in service.
- Both Recirculation pumps are out of service for maintenance and unavailable for start.

Then, a leak in the RHR system occurs:

- The leak is isolated.
- Shutdown Cooling CANNOT be restored.
- Reactor water level is 170" and stable.

Which one of the following describes the required direction to be given regarding Reactor water level in accordance with N2-SOP-31, Loss of Shutdown Cooling?

Direct Reactor water level to be...

A. maintained at the current value.

B. raised to a minimum of 202.3".

C. raised to a minimum of 227.0".

D. raised to a minimum of 245.0".

Proposed Answer: C

Explanation: N2-SOP-31 contains the following step to ensure adequate natural circulation is established in the Reactor vessel:

IF	THEN
RPV level is less than 227 inches with NO Recirc Pump running,	Within one hour, raise RPV level to 227 - 243 inches on shutdown range.

Note: The question meets SRO only guidelines because it requires the candidate to assess plant conditions (current Reactor water level, status of SDC and Recirc) and then use this assessment to determine if an additional procedure / section is yet required to be implemented (diagnostic override in N2-SOP-31, steps in N2-OP-101C) based on a diagnostic step / decision point in N2-SOP-31. This cannot be answered solely by knowing systems knowledge, immediate operator actions, SOP/EOP entry conditions, or overall mitigative strategy.

A. Plausible – Reactor water level must be raised to establish adequate natural circulation. Plausible because Reactor water level is currently in the allowable band of N2-EOP-RPV (159.3" to 202.3") and maintaining it at the current value minimizes introduction of cold water to the Reactor vessel in conditions where thermal stratification is a concern.
B. Plausible – The minimum value is 227.0". 202.3" is plausible because it is the high end of the N2-EOP-RPV band.
D. Plausible – The minimum value is 227.0". 245.0" is plausible because it is near the high end of the N2-SOP-31 required band.

Technical Reference(s): N2-SOP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP31C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295025 2.2.25
	Importance Rating	4.2

High Reactor Pressure

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question: #79

The plant is at 100% power. Conditions are as follows:

- A Turbine Control malfunction has occurred.
- Reactor pressure has risen to 1038 psig and stabilized.

Which one of the following describes the Technical Specification implication of this condition and the basis for the Technical Specification limit on Reactor pressure?

Reactor pressure __(1)__ the Technical Specification limit. The basis for the Technical Specification limit on Reactor pressure is to ensure __(2)__.

	(1)	(2)
A.	has exceeded	operation is maintained within the initial conditions of the plant accident analyses
B.	has exceeded	core shroud stresses are maintained within limits while operating at power
C.	remains below	operation is maintained within the initial conditions of the plant accident analyses
D.	remains below	core shroud stresses are maintained within limits while operating at power

Proposed Answer: A

Explanation: Technical Specification 3.4.12 requires Reactor pressure to be maintained ≤1035 psig. With Reactor pressure at 1038 psig, this limit has been exceeded. The basis for this limit is to ensure actual plant operation is maintained within the initial condition assumptions of various plant accident analyses, in particular the overpressure protection analysis.

B. Plausible – Higher Reactor pressure would cause higher steam flow and higher stresses in the core shroud during power operation, however this is NOT the basis for TS 3.4.12.
C. Plausible – The given Reactor pressure is below the N2-EOP-RPV entry condition of 1052 psig, however it is above the TS 3.4.12 limitation.
D. Plausible – The given Reactor pressure is below the N2-EOP-RPV entry condition of 1052 psig, however it is above the TS 3.4.12 limitation. Higher Reactor pressure would cause higher steam flow and higher stresses in the core shroud during power operation, however this is NOT the basis for TS 3.4.12.

Technical Reference(s): TS 3.4.12 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-101001-RBO-14

Question Source: Bank – 2015 NRC #80

Question History: 2015 NRC #80

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295003 2.4.41
	Importance Rating	4.6

Partial or Complete Loss of AC Power

Knowledge of the emergency action level thresholds and classifications.

Question: #80

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

- The Division 3 EDG is inoperable and unavailable due to maintenance.

The following sequence then occurs:

Time (hh:mm)	Condition
00:00	• A complete loss of offsite power results in a full load reject and Reactor scram. • Both available EDGs start and energize their respective switchgears. • Power control states that offsite power will be restored at 02:00.
00:10	• The Division 2 EDG trips on overspeed. • Reactor water level is 140" and slowly rising with RCIC injecting.
00:30	A fuel oil leak in the Division 1 EDG room causes a large fire and loss of the Division 1 EDG.
01:00	The fire is extinguished.
01:30	The Division 2 EDG is started and Division 2 switchgear is energized.

Which one of the following describes the emergency action levels resulting from these events, in accordance with EPIP-EPP-02-EAL and EP-CE-111, Emergency Classification and Protective Action Recommendations?

Note: Apply all required time limits.

The first EAL exceeded was an ___(1)___.

The highest EAL exceeded was a(n) ___(2)___.

A. (1) Unusual Event at 00:00
(2) Alert

B. (1) Unusual Event at 00:00
(2) Site Area Emergency

C. (1) Alert at 00:25
(2) Alert

D. (1) Alert at 00:25
(2) Site Area Emergency

Proposed Answer: B

Explanation: At time 00:00, Unusual Event SU1.1 was exceeded due to loss of all offsite power to Division 1 and 2 switchgear. At time 00:10, the loss of the Division 2 EDG, combined with the inoperability of the Division 3 EDG, started a clock for an Alert to be met at time 00:25. At time 00:45, the highest EAL exceeded is Site Area Emergency SS1.1 due to loss of all offsite and onsite power to Division 1 and 2 switchgear.

A. Plausible – Alerts SA1.1 and HA2.1 are exceeded at time 00:45, however the higher SAE is also exceeded.
C. Plausible – An Alert is met at time 00:25, however an Unusual Event was met earlier at time 00:00. Alerts SA1.1 and HA2.1 are exceeded at time 00:45, however the higher SAE is also exceeded.
D. Plausible – An Alert is met at time 00:25, however an Unusual Event was met earlier at time 00:00.

Technical Reference(s): EPIP-EPP-02-EAL, EP-CE-111

Proposed references to be provided to applicants during examination: EPIP-EPP-02-EAL (remove EALs referencing I-131 – one UE and a portion of the fission product barrier matrix)

Learning Objective: N2-EAL-UP-CE-1.09

Question Source: Bank – 2012 NRC #79

Question History: 2012 NRC #79

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295001 2.1.25
	Importance Rating	4.2

Partial or Complete Loss of Forced Core Flow Circulation

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Question: #81

The plant is operating at 40% power with the following:

- Single Recirculation loop operation exists following the trip of Recirculation pump B.
- Loop A jet pump diffuser to lower plenum D/Ps have been determined to be UNSAT.
- Jet pump loop flow versus flow control valve position is being verified using N2-OSP-LOG-D001 Figure 11-1.
- FCV A position is 58%.
- Loop A JP Flowrate is 60 Mlb/hr.

Which one of the following identifies the need for Technical Specification condition entry based on the operating point on N2-OSP-LOG-D001 Figure 11-1?

Based on the operating point on N2-OSP-LOG-D001 Figure 11-1, ...

A. NO Technical Specification condition entry is required.

B. Technical Specification 3.4.1, Recirculation Loops Operating, condition entry is required.

C. Technical Specification 3.4.2, Flow Control Valves (FCVs), condition entry is required.

D. Technical Specification 3.4.3, Jet Pumps, condition entry is required.

Predicted Jet Pump Loop Flow With Loop A Operating

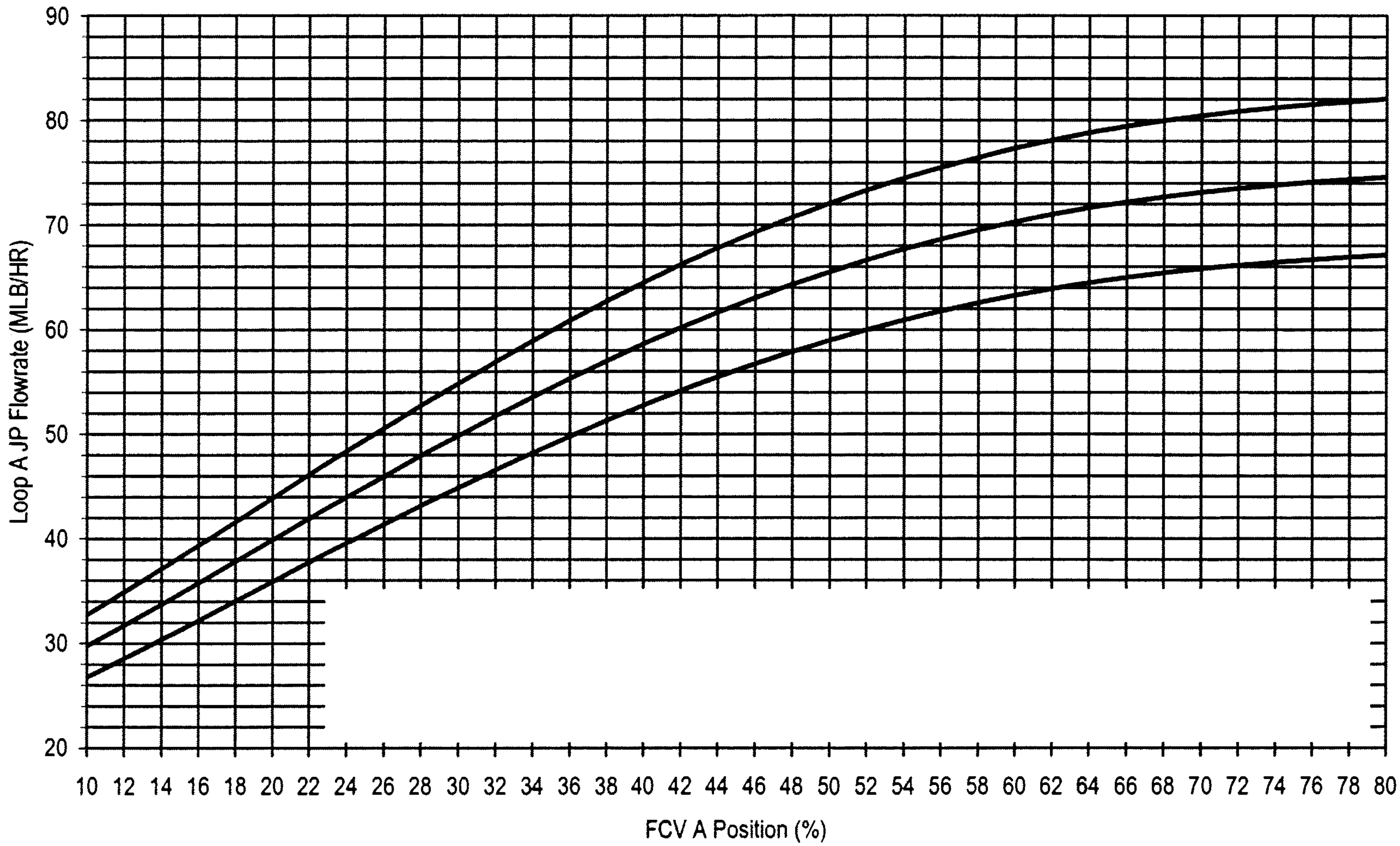


FIGURE 11-1

Proposed Answer: D

Explanation: Technical Specification Surveillance Requirement 3.4.3.1.a requires jet pump loop flow versus flow control valve position to be within ±10% from established patterns. When in single loop operation on Recirculation loop A, N2-OSP-LOG D001 figure 11-1 gives the established pattern. The middle curve is predicted flow. The top curve is 10% higher than predicted flow. The bottom curve is 10 less than predicted flow. Therefore, to satisfy Technical Specifications, the actual operating point must be between the top and bottom curves. The given data places operation slightly below the bottom curve, therefore Surveillance Requirement 3.4.3.1.a is not met and Technical Specification 3.4.3 Condition A must be entered.

Note: The question meets SRO level guidelines because it requires knowledge of SRs and/or knowledge of TS bases to know that this portion of N2-OSP-LOG-D001 is related to SR 3.4.3.1.a and not an SR in TS 3.4.1 or 3.4.2. This cannot be answered solely based on ≤1 hr actions, "above the line" information, or safety limits.

A. Plausible – Technical Specification 3.4.3 Condition A must be entered. Plausible because this would be correct if the given flowrate were slightly higher or the given FCV position were slightly lower. Also plausible if candidate does not understand required operating region in this figure (related notes have been removed).
B. Plausible – Technical Specification 3.4.3 Condition A must be entered. Plausible because this Figure is based on FCV position and candidate may believe this Surveillance Requirement relates to Technical Specification 3.4.2, which deals with FCV operability.
C. Plausible – Technical Specification 3.4.3 Condition A must be entered. Plausible because the surveillance is being performed due to recently entering single loop operation and candidate may believe this Surveillance Requirement relates to Technical Specification 3.4.1, which deal with Recirc loop operation.

Technical Reference(s): N2-OP-29 section H.6.0, Technical Specification Surveillance Requirement 3.4.3.1, N2-OSP-LOG-D001 Attachment 11

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295031 2.1.7
	Importance Rating	4.7

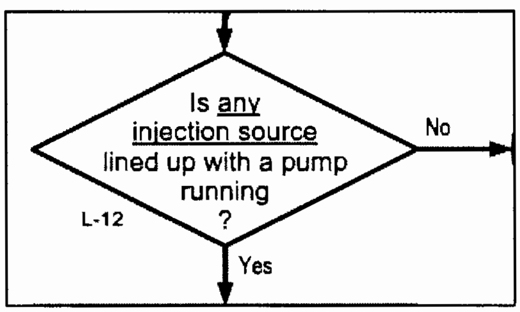
Reactor Low Water Level

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

Question: #82

The plant was operating at 100% power when a transient resulted in the following:

- Actual reactor water level is -38" and slowly lowering.
- Reactor pressure is 850 psig and slowly lowering with an SRV open.
- NO Reactor injection sources are available.
- N2-EOP-RPV, RPV Control, has been executed to the following step in the alternate level control leg:



Which one of the following describes how Reactor pressure is now required to be controlled in accordance with the Emergency Operating Procedures?

A. Stabilize Reactor pressure around the current value.

B. Lower Reactor pressure. Do NOT exceed a cool-down rate of 100°F/hr.

C. Rapidly lower Reactor pressure. The cool-down rate is allowed to exceed 100°F/hr.

D. Close the SRV and allow Reactor pressure to rise until SRVs automatically actuate.

Proposed Answer: A

Explanation: With no injection sources available and Reactor water level -38" and lowering, N2-EOP-RPV must be exited and N2-EOP-C3, Steam Cooling, must be entered. N2-EOP-C3 requires stabilizing Reactor pressure, even though it is currently above the capacity of many low pressure injection systems. This is done to ensure the Minimum Zero Injection RPV Water Level calculation assumptions remain valid and Reactor inventory loss is minimized.

B. Plausible – Lowering Reactor pressure would allow quicker injection with low pressure systems when they become available, however N2-EOP-C3 requires stabilizing Reactor pressure.
C. Plausible – Lowering Reactor pressure would allow quicker injection with low pressure systems when they become available, however N2-EOP-C3 requires stabilizing Reactor pressure.
D. Plausible – The SRV will need to be at least temporarily secured to prevent Reactor pressure from lowering too much more. However N2-EOP-C3 directs stabilizing Reactor pressure, not allowing it to rise significantly, such as would be required to reach the Reactor pressure at which SRVs automatically actuate.

Technical Reference(s): N2-EOP-RPV, N2-EOP-C3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC3C01 EO-2

Question Source: Bank – NMP1 2015 NRC #82

Question History: NMP1 2015 NRC #82

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295013 AA2.01
	Importance Rating	4.0

High Suppression Pool Temperature

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression pool temperature

Question: #83

The plant is operating at 10% power with the following:

Time (hh:mm)	Condition
12:00	• RCIC is started per N2-OSP-ICS-R002, RCIC System Flow Test. • Suppression Pool average water temperature is 80.1°F and rising slowly.
12:10	• Suppression Pool average water temperature is 85.1°F and rising slowly.
12:20	• Suppression Pool average water temperature is 90.1°F and rising slowly.
12:30	• Suppression Pool average water temperature is 95.1°F and rising slowly.
12:40	• Suppression Pool average water temperature is 100.1°F and rising slowly.
12:50	• RCIC is secured. • Suppression Pool average water temperature is 105.1°F and rising slowly.

Which one of the following describes the required application of Technical Specification (TS) 3.6.2.1, Suppression Pool Average Temperature?

TS 3.6.2.1 Condition entry is first required at time __(1)__.

To avoid the need to enter TS 3.6.2.1 Condition B, Suppression Pool average water temperature must be restored to ≤ 90°F by the latest time of __(2)__ the next day.

A. (1) 12:20
   (2) 12:20

B. (1) 12:20
   (2) 12:50

C. (1) 12:50
   (2) 12:20

D. (1) 12:50
   (2) 12:50

Proposed Answer: D

Explanation: Under normal conditions, TS 3.6.2.1 Condition A would be entered at time 12:20 when Suppression Pool average water temperature first exceeds 90°F. However, the given RCIC surveillance qualifies as testing that adds heat to the Suppression Pool. Therefore, TS 3.6.2.1 Condition A does not need to be entered at time 12:20 because this testing is still in progress. At time 12:50, at least one of two TS 3.6.2.1 Condition entries is required. If RCIC is still operating when Suppression Pool average water temperature exceeds 105°F, then TS 3.6.2.1 Condition C is entered and requires immediately suspending the test. Once RCIC is secured, TS 3.6.2.1 Condition A is entered. Required Action A.2 then gives 24 hours from the time of entry to restore temperature to ≤ 90°F (12:50 the next day). There is no requirement to retroactively apply Condition A to the first time Suppression Pool average water temperature exceeded 90°F.

Note: The question meets SRO level guidelines because it requires application of TS required actions (given TS 3.6.2, must determine when Condition A first applies based on multiple conditions, and then apply the completion time appropriately). This cannot be answered solely based on ≤1 hr actions, "above the line" information, or safety limits.

A. Plausible – TS 3.6.2.1 Condition entry is first required at 12:50, not 12:20. Plausible because 12:20 would be correct if testing that adds heat to the Suppression Pool was not in progress. TS 3.6.2.1 Condition B entry would not be required until 12:50 the next day, not 12:20. Plausible because 12:20 is 24 hours after Suppression Pool average water temperature first exceeded 90°F.
B. Plausible – TS 3.6.2.1 Condition entry is first required at 12:50, not 12:20. Plausible because 12:20 would be correct if testing that adds heat to the Suppression Pool was not in progress.
C. Plausible – TS 3.6.2.1 Condition B entry would not be required until 12:50 the next day, not 12:20. Plausible because 12:20 is 24 hours after Suppression Pool average water temperature first exceeded 90°F.

Technical Reference(s): Technical Specification 3.6.2.1

Proposed references to be provided to applicants during examination: Technical Specification 3.6.2.1

Learning Objective: N2-223001-RBO-14

Question Source: Modified Bank – Vision SYSID 33531

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295029 2.4.8
	Importance Rating	4.5

High Suppression Pool Water Level

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Question: #84

A plant startup is in progress with the following:

- An SRV has inadvertently opened.
- N2-SOP-34, Stuck Open Safety Relief Valve, is being executed.
- Suppression Pool average water temperature is 83°F and rising slowly.
- Suppression Pool water level has just exceeded 201'.

Which one of the following sets of actions is required?

A. Enter N2-EOP-PC, Primary Containment Control. Continue performing N2-SOP-34. In the event of a conflict between the procedures, N2-SOP-34 is the overriding document.

B. Enter N2-EOP-PC, Primary Containment Control. Continue performing N2-SOP-34. In the event of a conflict between the procedures, N2-EOP-PC is the overriding document.

C. Exit N2-SOP-34 and enter N2-EOP-PC, Primary Containment Control. N2-SOP-34 is re-entered at the step in-progress after exiting N2-EOP-PC.

D. Exit N2-SOP-34 and enter N2-EOP-PC, Primary Containment Control. N2-SOP-34 entry conditions are re-evaluated after exiting N2-EOP-PC.

Proposed Answer: B

Explanation: N2-EOP-PC is entered due to Suppression Pool water level above 201'. There is no requirement to exit SOPs when EOPs are entered. In fact, both procedures are executed concurrently. The EOPs are higher-tiered documents than the SOPs, therefore in the event of a conflict, the EOP must be followed.

A. Plausible – N2-SOP-34 was entered first and is event-specific, however the EOP is a higher-tiered document.
C. Plausible – N2-EOP-PC is a higher-tiered document, however there is no requirement to exit the SOP.
D. Plausible – N2-EOP-PC is a higher-tiered document, however there is no requirement to exit the SOP.

Technical Reference(s): NER-2M-039, section 3.1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOP00C01 TO #1

Question Source: Modified Bank – 2015 NRC #82

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295035 2.1.19
	Importance Rating	3.8

Secondary Containment High Differential Pressure

Ability to use plant computers to evaluate system or component status.

Question: #85

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

- An un-isolable leak develops from the RCIC system into the Reactor Building.
- The Reactor Building has isolated based on high airborne radiation levels
- GTS Trains A and B are operating.
- The following page shows a computer screen image during the event.
- The Shift Manager has declared a Site Area Emergency.
- NO field surveys have been completed yet for dose assessment (on-site or off-site).
- The Shift Manager is filling out the following section of the NMP Notification Fact Sheet – Part 1:

5.	**Release of radioactive Materials due to the classified event:**		
	A. No release		
	B. Release **below** federal limits (ODCM),	□ To atmosphere	□ To Water
	C. Release **above** federal limits (ODCM),	□ To atmosphere	□ To Water
	D. Unmonitored release requiring evaluation. (*Reason documented in Section 7)*		

For the above Part 1 Fact Sheet, which one of the following describes which letter to circle on Block 5?

A. A – No release

B. B – Release below federal limits (ODCM)

C. C – Release above federal limits (ODCM)

D. D – Unmonitored release requiring evaluation

[illegible] PAGE 1 OF 1   GROUP POINT DISPLAY SERVICES   04/25/17 08:53:47

[illegible]OUP NUMBER: 12 CSO STATUS DISPLAY II ** DO NOT ERASE**

[illegible]OINT NO.	DESCRIPTION	ALARM STATE	IS NOW	WAS LAST	HIGH LIMIT	LOW LIMIT	UNITS
[illegible]MSLA02	SUPPRESSION POOL LEVEL	↓	199.78	199.78	200.58	199.50	FEET
[illegible]MSPA01	DW LOOP A PRESS EL 283FT	↓	.33	.33	1.20	0	PSIG
[illegible]MSTU05	AVG DRYWELL TEMP CH A+B	↓	111.42	111.36	150.00	.00	DEG F
[illegible]MSTU03	AVERAGE SUPPR CHMB TEMP	↓	88.82	88.84	0	0	DEG F
[illegible]MSPA05	CONTMT DW PRESS	↑	15.03	15.03	15.45	14.20	PSIA
[illegible]MSPA06	SUPPR CHAM PRESS ABSOL	↓	14.64	14.64	15.45	14.20	PSIA
[illegible]TSPA02	RB IN/OUT D/P PDTEB	↑	.60	.57	-.25	-.90	IN WG
[illegible]FR[illegible]A01	DW FLR DRN LEAK RATE	↓	.28	0	5.00	0	GPM
[illegible]ER[illegible]A01	DW EQUIP DR LEAK RATE	↓	2.71	2.70	5.00	0	GPM
[illegible]HPA01	GEN GAS PRESSURE	↓	76.26	76.11	85.00	50.00	PSIG
[illegible]RTA01	RB VENT SUPPLY AIR TEMP	↑	75.00	0	92.00	60.00	DEG F
					0	0	
[illegible]TA19	SFP WTR TEMP	↓	107.28	107.28	150.00	60.00	DEG F
					0	0	
					0	0	
[illegible]TPA01	TB INSIDE/OUTSIDE D/P	↓	-.28	-.28	0	-.63	IN WG
[illegible]WSTA30	CLG TWR FLUME WTR TEMP	↓	76.91	80.82	90.00	40.00	DEG F
[illegible]WSLA05	CLG TWR BASIN WTR LEVEL	↓	22.25	20.91	80.00	10.00	INCHES
[illegible]WSFA04	CLG TWR BLOWDOWN FLOW	↑	7.97	8.12	0	0	KGPM
					0	0	GPDM

Proposed Answer: D

Explanation: An unmonitored release is in progress because Reactor Building D/P is positive with elevated airborne contamination levels. The Reactor Building is not a zero-leakage containment, therefore with positive pressure, contaminated air is leaking from the Reactor Building into the atmosphere and is not being monitored by installed radiation detectors. Field surveys will be required before the release can be considered monitored.

A. Plausible – Since both GTS Trains A and B are both running and designed to filter out airborne contamination, the candidate may determine that filtered release could be considered no release.
B. Plausible – A release is in progress due to the event, however with the Reactor Building D/P positive, it must be classified as an unmonitored release requiring evaluation.
C. Plausible – A release is in progress due to the event, however with the Reactor Building D/P positive, it must be classified as an unmonitored release requiring evaluation.

Technical Reference(s): EP-CE-114-100-F-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRRC01 EO-2

Question Source: Modified Bank – 2015 NRC #77

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	212000 A2.16
	Importance Rating	4.1

RPS

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Changing mode switch position

Question: #86

A plant startup is in progress with the following:

- The Reactor Mode Switch has just been transferred to RUN.
- The following annunciators are in alarm:
  - 603112, RPS A CONT & STOP V CLOSURE BYPASSED
  - 603116, RPS A MSIV CLOSURE TRIP BYPASSED
  - 603412, RPS B CONT & STOP V CLOSURE BYPASSED
  - 603416, RPS B MSIV CLOSURE TRIP BYPASSED
- The following associated computer points are also all in alarm:
  - RPSBC01 RPS A1 CV/SV CLSR BYP
  - RPSBC02 RPS A2 CV/SV CLSR BYP
  - RPSBC05 RPS A1 MSIV CLSR TR BYP
  - RPSBC06 RPS A2 MSIV CLSR TR BYP
  - RPSBC03 RPS B1 CV/SV CLSR BYP
  - RPSBC04 RPS B2 CV/SV CLSR BYP
  - RPSBC07 RPS B1 MSIV CLSR TR BYP
  - RPSBC08 RPS B2 MSIV CLSR TR BYP

Which one of the following describes the implication of these indications on Technical Specification (TS) 3.3.1.1, Reactor Protection System (RPS) Instrumentation?

TS 3.3.1.1 Condition…

A. entry is NOT required.

B. A must be entered, only.

C. A and B must be entered, only.

D. A, B, and C must be entered.

Proposed Answer: D

Explanation: All of the given annunciators are expected to be in alarm during a plant startup prior to transferring the Reactor Mode Switch to RUN. When the Reactor Mode Switch is transferred to RUN, annunciators 603116 and 603416 should immediately clear, indicating that the MSIV closure scram is un-bypassed, as required by TS Table 3.3.1.1-1 Function 5. With these annunciators remaining in alarm, and supported by the associated computer points, the RPS scram based on MSIV position is not maintained. This requires entering TS 3.3.1.1 Conditions A, B, and C.

A. Plausible – TS 3.3.1.1 entry is required because the MSIV closure scram is not operable. Plausible because these annunciators are expected prior to transferring the Reactor Mode Switch to RUN and two of the annunciators are still expected to be in alarm.
B. Plausible – TS 3.3.1.1 Conditions B and C also must be entered because the RPS scram based on MSIV position is not maintained. Plausible because this would be the correct answer if the failure were limited to MSIV position limit switches in only one channel.
C. Plausible – TS 3.3.1.1 Condition C also must be entered because the RPS scram based on MSIV position is not maintained. Plausible because this would be the correct answer if the failure were limited to MSIV position limit switches in both channels.

Technical Reference(s): ARPs 603112, 603116, 603412, 603416, Technical Specification 3.3.1.1

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 (with Table 3.3.1.1-1 applicable modes and allowable values blocked out)

Learning Objective: N2-212000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	209002 A2.05
	Importance Rating	2.9

HPCS

Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: D.C. electrical failure: BWR-5,6

Question: #87

The plant is operating at 100% power when Division III 125 VDC panel 2CES*IPNL414 de-energizes due to a sustained electrical fault.

Which one of the following describes the impact of this loss on the High Pressure Core Spray (CSH) system and the need to report this event to the NRC under 10CFR50.72?

CSH...

A. automatically initiates. This event must be reported to the NRC under 10CFR50.72.

B. automatically initiates. This event does NOT need to be reported to the NRC under 10CFR50.72.

C. automatic initiation is defeated. This event must be reported to the NRC under 10CFR50.72.

D. automatic initiation is defeated. This event does NOT need to be reported to the NRC under 10CFR50.72.

Proposed Answer: C

Explanation: The loss of 2CES*IPNL414 results in loss of DC electrical power to the CSH relay logic. The system is designed such that this prevents CSH from automatically initiating if needed. CSH is a single train system. Therefore, this unplanned inoperability must be reported to the NRC under 10CFR50.72.

A. Plausible – This event takes away power to CSH logic, however the system is designed such that this does not cause automatic initiation. Plausible because some logic systems are arranged such that the system actuates on loss of power (de-energize to function).
B. Plausible – This event takes away power to CSH logic, however the system is designed such that this does not cause automatic initiation. Plausible because some logic systems are arranged such that the system actuates on loss of power (de-energize to function). CSH is a single train system. Therefore, this unplanned inoperability must be reported to the NRC under 10CFR50.72. Plausible because planned CSH inoperability does not need to be reported and loss of a train of many other systems would not require a report either.
D. Plausible – CSH is a single train system. Therefore, this unplanned inoperability must be reported to the NRC under 10CFR50.72. Plausible because planned CSH inoperability does not need to be reported and loss of a train of many other systems would not require a report either.

Technical Reference(s): N2-SOP-04 Attachment 4, N2-OP-33 D.4.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209002-RBO-14

Question Source: Modified Bank – Vision SYSID 100213

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	400000 2.2.37
	Importance Rating	4.6

Component Cooling Water

Ability to determine operability and / or availability of safety related equipment.

Question: #88

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

- The running Service Water pump configuration is:
  - Division I: A, C, E
  - Division II: B, D
- Service Water supply header temperature has been rising steadily over the past 24 hours.
- Local temperatures taken at 2SWP*TT31A and B indicate 84.2°F.

Then, the following occurs:

- 2SWP*P1E trips on motor electrical fault.
- 2SWP*P1F is started and five (5) Service Water pumps are now back in service.

Which one of the following describes the most restrictive Technical Specification Condition entry that applies, if any, in accordance with Technical Specification 3.7.1, SW System and UHS?

A. Condition C

B. Condition E

C. Condition G

D. Condition entry is NOT currently required.

Proposed Answer: C

Explanation: N2-OP-11 P&L 28 discusses various instrument uncertainties that affect interpretation of Service Water supply header temperature indications regarding operability in TS 3.7.1. Of the three given indications, local temperatures taken at 2SWP*TT31A and B are considered the most reliable. With these local temperature indications > 84°F, both Service Water subsystems must be declared inoperable. This requires entering TS 3.7.1 Conditions C and G. Condition G is the most restrictive.

A. Plausible – Condition C must be entered, but Condition G must be entered also and is more restrictive. Plausible because this would be the correct answer if the given conditions only required declaring one Service Water subsystem inoperable.
B. Plausible – Condition E is entered momentarily when only 4 Service Water pumps are in operation, but is less restrictive than Condition G. Plausible because Condition E is entered and could be the most restrictive if temperatures did not require declaring Service Water subsystems inoperable.
D. Plausible – Conditions C, E, and G must be entered. Plausible because this would be the correct answer if the local temperature indications were < 82.93°F.

Technical Reference(s): N2-OP-11 D.28.0, Technical Specification 3.7.1

Proposed references to be provided to applicants during examination: Technical Specification 3.7.1

Learning Objective: N2-276000-RBO-14

Question Source: Modified Bank # - 2009 NRC #88

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 2.1.23
	Importance Rating	4.4

SLC

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question: #89

The plant has experienced a failure to scram with the following:

- Reactor power after the scram was 6% and Reactor water level was 120".
- Reactor water level was intentionally lowered and is being controlled at approximately 80" with Feedwater.
- Initial Standby Liquid Control (SLC) tank level was 2000 gallons.
- Control rod insertion is in progress using RMCS.
- Multiple control rods remain fully withdrawn.
- Reactor pressure is 950 psig and stable on Turbine Bypass Valves.
- Boron is being injected.
- Reactor power is downscale on the IRMs and APRMs.
- Current SLC tank level is 1400 gallons.

Which one of the following describes the ability to restore Reactor water level to a band of 159.3" to 202.3" and to commence a Reactor cooldown, in accordance with N2-EOP-C5, Failure to Scram?

Reactor water level may...

A. be restored and a Reactor cooldown may be commenced.

B. be restored, but a Reactor cooldown may NOT be commenced.

C. NOT be restored, but a Reactor cooldown may be commenced.

D. NOT be restored and a Reactor cooldown may NOT be commenced.

Proposed Answer: D

Explanation: A significant amount of boron has been injected to the Reactor, but the Cold Shutdown Boron Weight has not been injected yet ($\geq$ 850 gallons). Therefore, a Reactor cooldown is not yet allowed. Since Reactor water level has been intentionally lowered and N2-EOP-C5 is not yet able to be exited (multiple control rods still fully withdrawn), Reactor water level is not allowed to be restored to the band of 159.3" to 202.3".

A. Plausible – Reactor water level may not be restored until N2-EOP-C5 is exited. Plausible because enough boron has been injected that other BWR designs would be allowed to restore Reactor water level (Hot Shutdown Boron Weight). A Reactor cooldown may not be commenced because boron has been injected, but not enough for the Cold Shutdown Boron Weight. Plausible because if no boron had been injected, cooldown would be allowed with IRMs downscale. Also plausible because most of the Cold Shutdown Boron Weight has been injected.
B. Plausible – Reactor water level may not be restored until N2-EOP-C5 is exited. Plausible because enough boron has been injected that other BWR designs would be allowed to restore Reactor water level (Hot Shutdown Boron Weight).
C. Plausible – A Reactor cooldown may not be commenced because boron has been injected, but not enough for the Cold Shutdown Boron Weight. Plausible because if no boron had been injected, cooldown would be allowed with IRMs downscale. Also plausible because most of the Cold Shutdown Boron Weight has been injected.

Technical Reference(s): N2-EOP-C5, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Modified Bank – 2012 Cert SRO #12

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215004 A2.04
	Importance Rating	3.7

Source Range Monitor

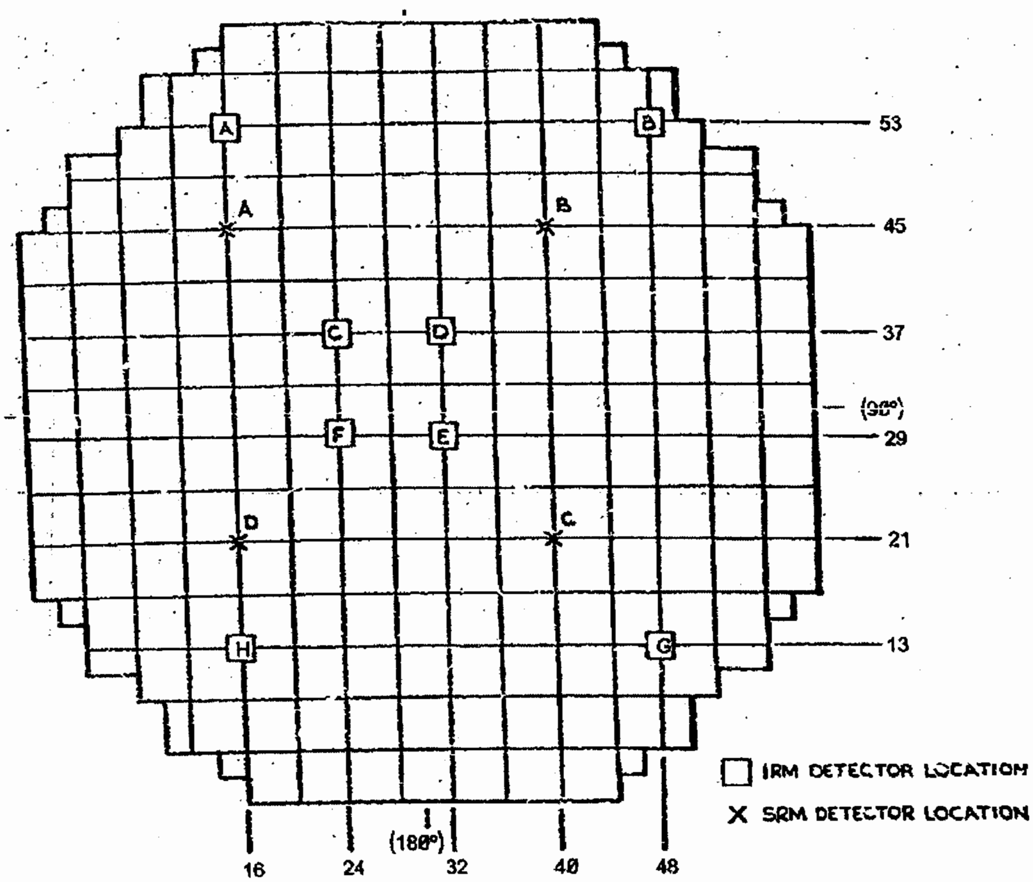
Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale and downscale trips

Question: #90

The plant is in Mode 5 with the following:

- Core shuffle is in progress.
- SRM C is inoperable and bypassed due to an upscale failure.
- SRM A fails downscale.

Given the following core map:



Which one of the following identifies the number of core quadrants in which irradiated fuel bundles may be moved, if any, in accordance with Technical Specifications?

A. 0

B. 2

C. 3

D. 4

Proposed Answer: A

Explanation: Movement of irradiated fuel bundles qualifies as a core alteration. Technical Specification Surveillance Requirement 3.3.1.2.2 requires an operable SRM is both the core quadrant where a core alteration is taking place and an adjacent quadrant. Surveillance Requirement 3.3.1.2.4 requires an SRM to have ≥3.0 cps with signal to noise ratio ≥2:1 or >1.3 cps with signal to noise ratio ≥5:1 to be considered operable. SRM A is inoperable based on this requirement. With SRMs A and C inoperable, no core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant.

Note: The question meets the K/A by giving SRMs that have failed upscale and downscale (and thus exceeding trip setpoints), requiring the candidate to determine the effect that this has on the number of operable SRM detectors, and then using procedures (Technical Specifications) to control refueling accordingly.

B. Plausible – Two SRMs are operable (B and D), however they are in opposite core quadrants. No core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant. Plausible because another combination of inoperable SRMs (such as A and B) would result in 2 quadrants being the correct answer.
C. Plausible – Two SRMs are operable (B and D), however they are in opposite core quadrants. No core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant. Plausible because the failure of SRM A without SRM C being bypassed would make 3 quadrants the correct answer.
D. Plausible – Two SRMs are operable (B and D), however they are in opposite core quadrants. No core quadrant has both an operable SRM in the quadrant and adjacent to the quadrant. Therefore, irradiated fuel bundles are not allowed to be moved in any core quadrant. Plausible because all 4 quadrants have either an operable SRM in the quadrant or 1 operable SRMs in each of the adjacent quadrants.

Technical Reference(s): Technical Specification 3.3.1.2

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.2, including Table 3.3.1.2-1

Learning Objective: N2-215002-RBO-14

Question Source: Modified Bank – 2009 NRC #90

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	204000 A2.08
	Importance Rating	3.1

RWCU

Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RWCU pump seal failure

Question: #91

The plant is operating at 20% power with the following:

- Reactor Water Cleanup pump A is in service when the pump seal catastrophically fails.
- Annunciator 602309, RWCU PUMP ROOM A TEMPERATURE HIGH, alarms.
- Annunciator 601157, REACTOR BLDG GENERAL AREAS TEMP HIGH, alarms.
- On Control Room panels 632 and 642, Reactor Water Cleanup Pump Room A temperature switches 2LDS*TRSH3A(1) and 2LDS*TRSH3B(1) are alarming high at 210°F and slowly lowering.
- NO other areas have elevated temperatures.
- WCS*MOV112, CLEANUP SUCT OUTBOARD ISOL VLV, has closed automatically.
- WCS*MOV102, CLEANUP SUCT INBOARD ISOL VLV, has failed to close both automatically and manually.
- N2-EOP-SC, Secondary Containment Control, is being executed.

Given the following possible actions from N2-EOP-SC:

(1) Operate unit coolers and HVR in affected areas.
(2) Shut down the Reactor per N2-OP-101C, Plant Shutdown.
(3) Enter N2-EOP-RPV, RPV Control, and scram the Reactor per N2-SOP-101C, Reactor Scram.
(4) Enter N2-EOP-C2, RPV Blowdown, and open seven (7) ADS valves.

Note: A portion of N2-EOP-6.28, Determining Reactor Building Temperatures, Table 3 is provided on the following page.

Which one of the following describes which of these actions is(are) required, in accordance with N2-EOP-SC?

A. (1) only

B. (1) and (2) only

C. (1) and (3) only

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

Table 3, **Reactor Bldg Areas**

2LDS*TRSH1A & 2LDS*TRSH1B

For MSIV Isolations, not for Reactor Building Areas

2LDS*TRSH2A & 2LDS*TRSH2B

(1) RCIC Pump Room

(2) RB Pipe Chase 206'

These channels isolate RCIC only

2LDS*TRSH3A & 2LDS*TRSH3B

(1) WCS A Pump Room

(2) WCS B Pump Room

(3) WCS HX Room

These channels isolate WCS only

2LDS*TRSH4A & 2LDS*TRSH4B

(1) RHS A Pump Room		(5) N240' **P632**	(5) S240' **P642**
(2) RHS B Pump Room		(6) N215' **P632**	(6) S215' **P642**
(3) N289' **P632**	(3) S289' **P642**	(7) N206' **P632**	(7) S206' **P642**
(4) N261' **P632**	(4) S261' **P642**		

NOTE:
Channels 3 – 7 are general areas, each division is a separate area on the same elevation, North or South

These channels isolate RCIC & RHS

2LDS*TRSH5A & 2LDS*TRSH5B

(1) SFC HX Room B

(2) Pipe Chase 266'

(3) Pipe Chase 292'

(4) Pipe Chase 305'

These channels isolate WCS & RCIC & RHS

Maximum Safe Values are defined as 212°F for all "Areas" and 135°F when access is required for support of EOP actions.

Each **BOLD BLOCK** is a different "Area" as defined by EOP-SC/RR
There are 17 different "Areas"

P632	**Isolations**	**P642**
	RCIC Isolations	
2ICS*MOV121 ☐		2ICS*MOV170 ☐
		2ICS*MOV128 ☐
	WCS Isolations	
2WCS*MOV112 ☐		2WCS*MOV102 ☐
	RHS Isolations	
2RHS*MOV113 ☐		2RHS*MOV112 ☐
2RHS*MOV40A ☐		2RHS*MOV40B ☐
2RHS*MOV67A ☐		2RHS*MOV67B ☐
2RHS*MOV104 ☐		

Proposed Answer: A

Explanation: The given conditions show an isolated primary system leak that has resulted in multiple high temperature alarms and indications near the Maximum Safe Value of 212°F. Per N2-EOP-6.28 Table 3, all of the high temperature indications are confined to a single Reactor Building General Area (one bold surrounded block). With the primary system discharge isolated and a single General Area just below Max Safe, N2-EOP-SC requires operating unit coolers and HVR in affected areas (step SC-3), but Reactor shutdown, scram, and blowdown are not required.

B. Plausible – Reactor shutdown is not required. Plausible because if a two Reactor Building General Area experience temperatures above Max Safe, then a Reactor shutdown becomes required by N2-EOP-SC.
C. Plausible – Reactor scram is not required. Plausible because if the leak were not isolated (WCS*MOV112 also failed to close), then a Reactor scram would be required.
D. Plausible – Reactor scram is not required. Plausible because if the leak were not isolated (WCS*MOV112 also failed to close), then a Reactor scram would be required. RPV blowdown is not required. Plausible because if the leak were not isolated (WCS*MOV112 also failed to close) and two Reactor Building General Areas experienced temperature above Max Safe, then a RPV blowdown would be required.

Technical Reference(s): N2-EOP-SC, N2-EOP-6.28

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	201006 2.1.32
	Importance Rating	4.0

RWM

Ability to explain and apply system limits and precautions.

Question: #92

A plant startup is in progress with the following:

- Today is 6/14/17.
- The Rod Worth Minimizer (RWM) has become inoperable.
- All attempts to re-initialize the RWM have failed.
- The RWM has been bypassed.
- Only the first eight (8) control rods of the startup sequence have been withdrawn.
- The last startup performed with an inoperable RWM was on 3/9/17.

Which one of the following describes the permissible control rod movement, in accordance with N2-OP-95A, Rod Worth Minimizer System, and Technical Specifications?

Control rods may…

A. NOT be withdrawn unless the RWM is restored to operable status.

B. be withdrawn with no additional restrictions.

C. be withdrawn, but only if an additional verifier is stationed.

D. be withdrawn, but only if a risk evaluation is performed and the associated impact is managed.

Proposed Answer: A

Explanation: N2-OP-95A Section H.1.0 and Technical Specification 3.3.2.1 Condition C provide limitations on performing a Reactor startup with an inoperable RWM. Since less than 12 control rods have been withdrawn and a startup with an inoperable RWM has been performed in the last calendar year, the only permissible control rod movement is to insert rods using a Reactor scram. Control rod withdrawal is NOT allowed.

B. Plausible – Control rods may not be withdrawn. Plausible because this would be the correct answer if Reactor power were >10%.
C. Plausible – Control rods may not be withdrawn. Plausible because this would be the correct answer if at least 12 control rods were withdrawn or a startup with an inoperable RWM had not been performed in the last calendar year.
B. Plausible – Control rods may not be withdrawn. Plausible because rod withdrawal is allowed in certain circumstances, based on risk analysis, and this wording is similar to that used in SR 3.0.3 for allowing operation with a missed surveillance.

Technical Reference(s): N2-OP-95A, Technical Specification 3.3.2.1 Condition C

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-201006-RBO-14

Question Source: Bank – Vision SYSID 33187

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	286000 2.4.11
	Importance Rating	4.2

Fire Protection

Knowledge of abnormal condition procedures.

Question: #93

The plant is at 100% power with the following:

- Annunciator 849213, TROUBLE DIESEL FIRE PUMP, alarms.
- Investigation reveals that all DC power to the Diesel Fire Pump starting motor is lost due to electrical failures.

Which one of the following describes the impact of this fault on the Fire Protection system, in accordance with N2-OP-43, Fire Protection Water, and the impact on USAR requirements for Fire Protection?

The Diesel Fire Pump is...

A. inoperable but available for manual start. Restore operability within a maximum of 24 hours or provide an alternate fire suppression source.

B. inoperable but available for manual start. Restore operability within a maximum of 7 days or provide an alternate fire suppression source.

C. inoperable and unavailable. Restore operability within a maximum of 24 hours or provide an alternate fire suppression source.

D. inoperable and unavailable. Restore operability within a maximum of 7 days or provide an alternate fire suppression source.

Proposed Answer: D

Explanation: Two sets of batteries supply DC power to the Diesel Fire Pump starting motor. If both of these power supplies are lost, the Diesel Fire Pump will not automatically start and cannot be manually started per N2-OP-43, even using the local start section of the procedure. Therefore, the Diesel Fire Pump is both inoperable and unavailable. With the Diesel Fire Pump inoperable, the requirements of USAR Section 9.A.3.6.2.6 are not met, and the required action is to restore operability within 7 days or provide an alternate fire suppression source.

A. Plausible – The Diesel Fire Pump is inoperable due to loss of ability to auto-start. It is also unavailable because the manual starting methods in N2-OP-43 also require DC power to the starting motor. Plausible that another method could be available to manually start, such as with the NMP1 Diesel Fire Pump with no control power (manual override of solenoids). 24 hours is the maximum time limit if both the Electric and Diesel Fire Pumps are inoperable.
B. Plausible – The Diesel Fire Pump is inoperable due to loss of ability to auto-start. It is also unavailable because the manual starting methods in N2-OP-43 also require DC power to the starting motor. Plausible that another method could be available to manually start, such as with the NMP1 Diesel Fire Pump with no control power (manual override of solenoids).
C. Plausible – 24 hours is the maximum time limit if both the Electric and Diesel Fire Pumps are inoperable.

Technical Reference(s): N2-OP-43, USAR Section 9A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-286000-RBO-14

Question Source: Bank – 2014 Cert #93

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.34
	Importance Rating	3.5

Knowledge of primary and secondary plant chemistry limits.

Question: #94

Which one of the following identifies a chemistry parameter and associated threshold used to define Loss of the Fuel Clad Barrier, in accordance with the Emergency Action Levels?

Dose equivalent primary coolant activity of...

A. Iodine-131; 0.2 μCi/gm

B. Iodine-131; 300 μCi/gm

C. Cesium-137; 0.2 μCi/gm

D. Cesium-137; 300 μCi/gm

Proposed Answer: B

Explanation: The Emergency Action Levels define Loss of the Fuel Clad barrier as dose equivalent primary coolant activity of I-131 above the threshold of 300 μCi/gm.

A. Plausible – The threshold is 300 μCi/gm. Plausible because 0.2 μCi/gm is used in Technical Specification 3.4.8.
C. Plausible – The isotope is I-131. Plausible because Cs-137 is a significant radioisotope referenced in ODCM. The threshold is 300 μCi/gm. Plausible because 0.2 μCi/gm is used in Technical Specification 3.4.8.
D. Plausible – The isotope is I-131. Plausible because Cs-137 is a significant radioisotope referenced in ODCM.

Technical Reference(s): Emergency Action Levels

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-204000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.42
	Importance Rating	4.6

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question: #95

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

- Standby Liquid Control (SLC) A has just been restored to service following maintenance.
- The following data is obtained for the sodium pentaborate solution in SLC storage tank 2SLS*TK1:
  - Temperature is 72°F.
  - Net volume is 2034 gallons.
  - Concentration is 13.2% by weight.
  - Enrichment is 93 atom % B-10.

Which one of the following describes the status of the SLC system, in accordance with Technical Specifications?

The SLC system is...

A. fully operable.

B. inoperable. Restore operability or be in Mode 3 within a maximum of 12 hours.

C. inoperable. Restore operability or be in Mode 3 within a maximum of 20 hours.

D. inoperable. Restore operability or be in Mode 3 within a maximum of 7 days and 12 hours.

Proposed Answer: C

Explanation: All of the given parameters are satisfactory except concentration is below the lower limit of 13.6% in Technical Specification Figure 3.1.7-1. Since this is a common tank for both SLC subsystems, they are both inoperable. This requires entering Condition B, which gives 8 hours to restore operability before entering Condition C, which gives another 12 hours to reach Mode 3. This makes a total of 20 hours to either restore operability or reach Mode 3.

A. Plausible – All of the given parameters are satisfactory except concentration is below the lower limit of 13.6% in Technical Specification Figure 3.1.7-1.
B. Plausible – This is the answer if Condition C is incorrectly entered (without first entering Condition B).
D. Plausible – This is the answer if Condition A is entered instead of Condition B (declaring only one subsystem inoperable instead of both).

Technical Reference(s): Technical Specification 3.1.7

Proposed references to be provided to applicants during examination: Technical Specification 3.1.7

Learning Objective: N2-211000-RBO-14

Question Source: Bank – NMP1 2010 NRC #95

Question History: NMP1 2010 NRC #95

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.15
	Importance Rating	3.1

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Question: #96

Which one of the following radiation monitors may be used to provide an initial estimate of core damage during an accident, in accordance with EPIP-EPP-09, Determination of Core Damage Under Accident Conditions and/or N2-EOP-C4, RPV Flooding?

A. Off-gas radiation monitor

B. Drywell radiation monitor

C. Main Stack radiation monitor

D. Main Steam Line radiation monitor

Proposed Answer: B

Explanation: EPIP-EPP-09/N2-EOP-C4 Table "V" uses the Drywell radiation level to provide an initial estimate of core damage under accident conditions.

Note: This question tests appropriate SRO-level knowledge under 10CFR55.43(b)(4) by presenting a situation with extreme radiation hazards (a LOCA with fuel damage) and testing knowledge of how to analyze conditions to assess the magnitude of core damage (which radiation monitors are used for assessment). This assessment is required within the Emergency Plan and would be required to select the appropriate operational response during a beyond design-basis event. This is similar in nature to the example item in the Guidance for SRO-only Questions, Section II.D bullet #3. Drywell radiation level can be used to estimate core damage more quickly than coolant activity level.

A. Plausible – The Off-gas radiation monitors are used to determine entry into N2-SOP-17, Fuel Failure, but not for core damage estimation in EPIP-EPP-09.
C. Plausible – The Main Stack radiation monitor are used during an accident to estimate off-site release rate, but not for core damage estimation in EPIP-EPP-09.
D. Plausible – The Main Steam Line radiation monitors are used to determine entry into N2-SOP-17, Fuel Failure, but not for core damage estimation in EPIP-EPP-09.

Technical Reference(s): EPIP-EPP-09 Attachment 1, N2-EOP-C4 Table "V"

Proposed references to be provided to applicants during examination: None

Learning Objective: EPIP-EPP-09-TO-01

Question Source: Bank – 2015 NRC #100

Question History: 2015 NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.1
	Importance Rating	4.8

Knowledge of EOP entry conditions and immediate action steps.

Question: #97

The plant is in Mode 4 during an outage with the following:

- A planned system draining evolution to Reactor Building floor drain sump tank 2A is in progress.
- Annunciator 851453, REACTOR BLDG FLOOR DRAIN SYSTEM TROUBLE, alarms unexpectedly due to this evolution.
- Radwaste reports that annunciator 513113, RX BLDG FL DR SUMP TK2A LEVEL HIGH/HIGH, is also unexpectedly in alarm.
- An operator in the field reports that the floor drain sump has NOT yet overflowed.

Entry into N2-EOP-SC, Secondary Containment Control, is...

A. required.

B. NOT required because the plant is in Mode 4.

C. NOT required because an entry condition is NOT met or exceeded.

D. NOT required because the high floor drain water level is part of a planned evolution.

Proposed Answer: A

Explanation: OP-NM-101-111-1001 provides the administrative guidance for exceeding an EOP entry condition due to a planned evolution. The procedure requires entering the EOP, verifying no emergency exists, and then exiting the EOP. An N2-EOP-SC entry condition is exceeded in this situation based on a Reactor Building floor drain sump water level above the high-high alarm, therefore N2-EOP-SC entry is required. There is no administrative guidance for not entering EOPs while in Modes 4 or 5, as there is at some other plants.

Note: This question meets SRO-level guidelines by testing knowledge of administrative procedures that control implementation of the Emergency Operating Procedures. This is in alignment with the third bullet in NUREG 1021 ES-401 Attachment 2 Section II.E.

B. Plausible – N2-EOP-SC entry is required. Plausible because some other plants do not require EOP entry while in Modes 4 or 5. This is similar to having different EAL matrices for various plant modes.
C. Plausible – N2-EOP-SC entry is required. Plausible because the floor drain high-high alarm is received, but the sump has not overflowed and caused a high general area water level, which is another entry condition.
D. Plausible – N2-EOP-SC entry is required. Plausible because EOP entry due to a planned evolution is a special case. OP-NM-101-111-1001 allows not making station announcements and management notifications based on the EOP entry, but does require entering the EOP. Some other plants have specific administrative guidance that allows not entering the EOP in such a situation.

Technical Reference(s): ARP 851453, N2-EOP-SC, OP-NM-101-111-1001 section 4.5.A, NER-2M-039 section 3.1.8

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Modified Bank – JAF 4/14 NRC #80

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.18
	Importance Rating	4.0

Knowledge of the specific bases for EOPs.

Question: #98

A LOCA has occurred. Conditions are as follows:

- No Reactor injection sources are currently available.
- N2-EOP-C3, Steam Cooling, is being executed.
- Reactor water level is -35 inches (actual) and lowering slowly.

Then five minutes later...

- A Control Rod Drive pump is restored and is now injecting to the Reactor.
- Reactor water level is -42 inches (actual) and continuing to lower.

Which one of the following describes the status of adequate core cooling and the required action, in accordance with the Emergency Operating Procedures?

	Status of Adequate Core Cooling	Required Action
A.	Assured	Return to N2-EOP-RPV, RPV Control.
B.	Assured	Remain in N2-EOP-C3, Steam Cooling.
C.	NOT assured	Enter N2-EOP-C2, RPV Blowdown.
D.	NOT assured	Enter the Severe Accident Procedures.

Proposed Answer: C

Explanation: Initially, adequate core cooling (ACC) was assured because there was no RPV injection and Reactor water level was above -58 inches. However, once CRD began injecting, adequate core cooling is not assured between -39 and -58 inches. With an RPV injection source injecting and Reactor water not able to be restored/maintained above -39 inches, N2-EOP-C3 requires entering N2-EOP-C2, RPV Blowdown.

A. Plausible – ACC is not currently assured. Plausible because ACC was assured until the CRD pump began injecting.
B. Plausible – ACC is not currently assured. Plausible because ACC was assured until the CRD pump began injecting.
D. Plausible – N2-EOP-C2 must be entered, not Severe Accident Procedures. Plausible because if core cooling cannot be restored after the Blowdown, the Severe Accident Procedures would then be entered from N2-EOP-RPV.

Technical Reference(s): N2-EOP-C3 Step 1, NER-2M-039 Section 2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC3C01 EO-2

Question Source: Bank – 2015 Cert #99

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.6
	Importance Rating	3.6

Knowledge of the process for making changes to procedures.

Question: #99

A surveillance test is in progress with the following:

- An Equipment Operator reports that there is an error in the next step.
- The step requires manipulating a valve, but the component ID is incorrect.
- A Reactor Operator has reviewed the step and concurs with the Equipment Operator.

Which one of the following describes the actions necessary to complete this surveillance, in accordance with HU-AA-104-101, Procedure Use and Adherence?

A. Complete the step using the correct component ID, then submit a Procedure Performance Improvement System (PPIS) item to correct the step before the next use.

B. Complete the step using the correct component ID, then submit a Temporary Change (TC) to correct the step before the next use.

C. Stop the surveillance, process a Procedure Performance Improvement System (PPIS) item to correct the step, then continue once the change has been made.

D. Stop the surveillance, process a Temporary Change (TC) to correct the step, then continue once the change has been made.

Proposed Answer: D

Explanation: HU-AA-104-101 step 3.1.8 requires the Equipment Operator to stop the surveillance because the step cannot be performed as written. Step 3.2.1 requires the job supervisor (SRO) to revise the procedure prior to continuing. The necessary change is beyond the scope of the PPIS process and requires the Temporary Change process in AD-AA-101 to be implemented before continuing on with the surveillance.

A. Plausible – The step must be fixed using a Temporary Change before the surveillance can be continued. Plausible because the procedure identifies the correct valve, but only has an error in the component ID, and because a licensed operator has concurred with the Equipment Operator. The necessary change is beyond the scope of the PPIS process. Plausible because the PPIS process is a valid system for making different types of changes to procedures.
B. Plausible – The step must be fixed using a Temporary Change before the surveillance can be continued. Plausible because the procedure identifies the correct valve, but only has an error in the component ID, and because a licensed operator has concurred with the Equipment Operator.
C. Plausible – The necessary change is beyond the scope of the PPIS process. Plausible because the PPIS process is a valid system for making different types of changes to procedures.

Technical Reference(s): HU-AA-104-101, AD-AA-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – Limerick 09-1 NRC SRO #20

Question History: Limerick 09-1 NRC SRO #20

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.13
	Importance Rating	3.8

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Question: #100

The plant is in Mode 5 with a core shuffle about to begin.

Per N2-FHP-13.3, Core Shuffle, which one of the following describes (1) the need for a Specific Radiation Work Permit (RWP) for conducting the Core Shuffle and (2) the requirements for personnel access to the Drywell during the Core Shuffle?

A Specific RWP is ___(1)___ for conducting the Core Shuffle.

Drywell access is ___(2)___.

A. (1) required
(2) NOT allowed during the Core Shuffle.

B. (1) required
(2) allowed during the Core Shuffle, provided that personnel do NOT go above elevation 288'

C. (1) NOT required
(2) NOT allowed during the Core Shuffle.

D. (1) NOT required
(2) allowed during the Core Shuffle, provided that personnel do NOT go above elevation 288'

Proposed Answer: B

Explanation: N2-FHP-13.3 requires all fuel movement to be done under the guidance of a Specific RWP. N2-FHP-13.3 allows Drywell access as long as personnel do not go above elevation 288'.

A. Plausible – N2-FHP-13.3 allows Drywell access as long as personnel do not go above elevation 288'. Plausible because Drywell access is restricted at the higher elevations.
C. Plausible – N2-FHP-13.3 requires all fuel movement to be done under the guidance of a Specific RWP. Plausible because radiation levels on the Refuel Bridge are not expected to be very high. N2-FHP-13.3 allows Drywell access as long as personnel do not go above elevation 288'. Plausible because Drywell access is restricted at the higher elevations.
D. Plausible – N2-FHP-13.3 requires all fuel movement to be done under the guidance of a Specific RWP. Plausible because radiation levels on the Refuel Bridge are not expected to be very high.

Technical Reference(s): N2-FHP-13.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-234000-RBO-10

Question Source: Bank – 2012 NRC #100

Question History: 2012 NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(7)

Comments: