Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K1.12
	Importance Rating	3.4

RPS

Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: Reactor/turbine pressure control system: Plant-Specific

Proposed Question: #1

A plant startup is in progress with the following:

- Reactor power is 35%.
- Annunciator F3-4-6, FIRST STAGE BOWL PRESS LOW, is in alarm.
- The Main Generator is synchronized to the grid.

Then, the Main Turbine trips due to a thrust bearing wear signal.

Which one of the following describes the response of the Reactor Protection System and High Pressure Coolant Injection (HPCI)?

The Reactor...

A. scrams and HPCI initiates.

B. scrams, but HPCI does NOT initiate.

C. does NOT scram, but HPCI initiates.

D. does NOT scram and HPCI does NOT initiate.

Proposed Answer: C

Explanation: HPCI initiates on a Turbine trip signal (unless the HPCI reset pushbuttons are being depressed, which is not indicated in the stem). The Reactor scrams on a Turbine trip signal only if first stage bowl pressure is above a value corresponding to approximately 45% Reactor power. Since Annunciator F3-4-6 is in alarm, this Reactor scram is currently bypassed, therefore the Reactor does not scram.

Note: The question meets the K/A because it tests the cause and effect relationship between the Reactor/turbine pressure control system (Main Turbine trips due to a thrust bearing wear signal, which is a function of MHC, and Turbine first stage bowl pressure is sensed by MHC < F3-4-6 setpoint) and RPS (whether the Reactor scrams or not). NUREG 1123 includes tasks related to the Main Turbine trips under system 241000, Reactor/Turbine Pressure Regulating System.

A. Plausible – The Reactor does not scram on the Turbine trip signal because first stage bowl pressure is low enough to bypass this scram signal.
B. Plausible – The Reactor does not scram on the Turbine trip signal because first stage bowl pressure is low enough to bypass this scram signal. HPCI initiates on the Turbine trip even with first stage bowl pressure low.
D. Plausible – HPCI initiates on the Turbine trip even with first stage bowl pressure low.

Technical Reference(s): N1-OP-40, N1-OP-16, ARP F3-4-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-8

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 #	264000 K1.01
	Importance Rating	3.8

EDGs

Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: A.C. electrical distribution

Proposed Question: #2

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

(1) Powerboard (PB) 102 will be transferred from reserve power to Emergency Diesel Generator (EDG) 102 to support maintenance on breaker R-1012, Reserve Supply to PB 102.

(2) PB 102 will be transferred back from EDG 102 to reserve power.

Which one of the following describes the status of PB 102 during each of these transfers, in accordance with N1-OP-45, Emergency Diesel Generators?

	During evolution (1), PB 102...	During evolution (2), PB 102...
A.	must be momentarily de-energized.	must be momentarily de-energized.
B.	must be momentarily de-energized.	may remain continuously energized.
C.	may remain continuously energized.	must be momentarily de-energized.
D.	may remain continuously energized.	may remain continuously energized.

Proposed Answer: C

Explanation: Interlocks allow paralleling EDG 102 onto PB 102 while it is energized from reserve power and N1-OP-45 section E.4.0 provides the associated procedural guidance. Interlocks prevent paralleling reserve power onto PB 102 while it is energized from EDG 102. This interlock is reflected in the procedural guidance of N1-OP-45 section G.1.0. Therefore, during evolution (1), PB 102 may remain energized throughout the evolution, but in evolution (2), PB 102 must be momentarily de-energized.

A. Plausible – PB 102 must be momentarily de-energized when returning to reserve power, but not when being paralleled with EDG 102.
B. Plausible – PB 102 must be momentarily de-energized when returning to reserve power, but not when being paralleled with EDG 102.
D. Plausible – PB 102 may remain continuously energized when being paralleled with EDG 102, but must be momentarily de-energized when returning to reserve power.

Technical Reference(s): N1-OP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-5

Question Source: Bank – 2015 Cert #19

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K2.03
	Importance Rating	2.8

HPCI

Knowledge of electrical power supplies to the following: Initiation logic: BWR-2,3,4

Proposed Question: #3

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

- Feedwater level control is in 3-element mode.
- A Reactor scram occurs.
- Reactor water level shrinks to a low of 15".

Which one of the following describes the minimum electrical power supply(ies) required for Feedwater level control to shift to the HPCI mode?

A. Either RPS Bus 11 OR 12

B. Both RPS Bus 11 AND 12

C. Either Reactor Trip Bus 131 OR 141

D. Both Reactor Trip Bus 131 AND 141

Proposed Answer: A

Explanation: Either relay 11K47 (RPS Bus 11) or 12K47 (RPS Bus 12) must energize for HPCI initiation. Only one of the two power supplies is required.

B. Plausible – Either relay 11K47 (RPS Bus 11) or 12K47 (RPS Bus 12) must energize for HPCI initiation. Both power supplies must fail to prevent Feedwater level control from shifting into the HPCI mode.
C. Plausible – Reactor Trip Bus 131 AND 141 supply the scram relays, scram and SDV valves and feedwater heater NRVs, they would not affect the FWLC/HPCI logic.
D. Plausible – Reactor Trip Bus 131 AND 141 supply the scram relays, scram and SDV valves and feedwater heater NRVs, they would not affect the FWLC/HPCI logic.

Technical Reference(s): N1-OP-16; C-19859-C sheets 3, 6; C-23077-C sheet 4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-8

Question Source: Bank – 2010 Cert #23

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 #	400000 K2.02
	Importance Rating	2.9

Component Cooling Water

Knowledge of electrical power supplies to the following: CCW valves

Proposed Question: #4

Which one of the following identifies the electrical power supplies to the Containment Spray Raw Water discharge valves and intertie valves (93-25, 93-26, 93-27, 93-28, 93-71, 93-72, 93-73, 93-74)?

A. Battery Boards 11 and 12

B. DC Valve Boards 11 and 12

C. Powerboards 161A and 171A

D. Powerboards 161B and 171B

Proposed Answer: D

Explanation: Each Containment Spray Raw Water pump has a discharge valve and intertie valve that are used to route water either to the associated Containment Spray heat exchanger or a Core/Containment Spray injection header. These valves are AC motor-operated and powered from Powerboards 161B and 171B.

Note: The question fits under system 400000, Component Cooling Water, because this system designation includes all component cooling water systems (including DG cooling water, Emergency Service Water, Containment Spray Raw Water), not just those that are closed loop (such as RBCLC and TBCLC).

A. Plausible – These valves are AC motor-operated and powered from Powerboards 161B and 171B. Battery Boards 11 and 12 supply power for the Containment Spray pump discharge valves (80-15, 80-16, 80-35, 80-36).
B. Plausible – These valves are AC motor-operated and powered from Powerboards 161B and 171B. Plausible because the Containment Spray pump discharge valves (80-15, 80-16, 80-35, 80-36) are DC powered from a similar board.
C. Plausible – These valves are powered from the B side of Powerboards 161 and 171, not the A side.

Technical Reference(s): N1-OP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226001-RBO-4

Question Source: New

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 #	263000 K3.02
	Importance Rating	3.5

DC Electrical Distribution

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Components using D.C. control power (i.e. breakers)

Proposed Question: #5

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

- A loss of Battery Board 11 occurs.
- N1-SOP-47A.1, Loss of DC, is entered.

Which one of the following describes an affected load that can be restored by transferring control power to Battery Board 12?

A. All functions of ERV 111, 112, and 113 can be restored.

B. All functions of Powerboard 101 supply breakers, R1011 and R1014, can be restored.

C. Some functions of the Electric Fire pump can be restored, except its auto start remains defeated.

D. Some functions of Emergency Diesel Generator 102 can be restored, except output breaker R1022 must be operated locally.

Proposed Answer: D

Explanation: The loss of Battery Board 11 results in loss of 125 VDC control power to EDG 102. N1-SOP-47A.1 directs transferring this control power to Battery Board 12 per N1-OP-47A. This restores auto and manual start capability, however the output breaker must still be operated locally.

A. Plausible – Multiple functions for ERV 111, 112, and 113 remain inoperable, including F panel controls, automatic pressure relief, and the ADS function. Plausible because ERVs are one of the loads affected by the power loss.
B. Plausible – Powerboard 101 supply breakers are supplied from Battery Board 12, and are unaffected by the loss of Battery Board 11. Plausible because PB 101 supply breakers would be affected by a similar power loss (Battery Board 12).
C. Plausible – Diesel Fire pump (DFP) control circuits are DC powered, not the Electric Fire pump (EFP). EFP is not affected by BB11 loss. Indicating auto start feature remaining inoperable is similar to the actual affected control circuits of the DFP.

Technical Reference(s): N1-SOP-47A.1, C-19839-C Sh 7

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-8

Question Source: Bank – 2009 Cert #12

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 #	300000 K3.02
	Importance Rating	3.3

Instrument Air

Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following: Systems having pneumatic valves and controls

Proposed Question: #6

The Reactor is scrammed with the following:

- Emergency Condenser (EC) Loop 11 is being used for pressure control.
- EC Loop 12 is unavailable.
- Instrument air pressure is 0 psig.

Which one of the following describes the response of 60-17, EC Shell Side Makeup LCV 11, and 60-04, EC Makeup Tank Makeup LCV 11?

	60-17, EC Shell Side Makeup LCV 11	60-04, EC Makeup Tank Makeup LCV 11
A.	Fails open	Fails open
B.	Fails open	Fails closed
C.	Fails closed	Fails open
D.	Fails closed	Fails closed

Proposed Answer: B

Explanation: 60-17 fails open on loss of IA. 60-04 fails closed due to loss of IA.

A. Plausible – 60-04 fails closed due to loss of IA. Plausible that this valve would fail open such that makeup to the Makeup Tank was not lost.
C. Plausible – 60-17 fails closed due to loss of IA. Plausible that this valve would fail closed such that the shell side would not be overfilled. 60-04 fails closed due to loss of IA. Plausible that this valve would fail open such that makeup to the Makeup Tank was not lost.
D. Plausible – 60-17 fails closed due to loss of IA. Plausible that this valve would fail closed such that the shell side would not be overfilled.

Technical Reference(s): C-18017-C, N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278000-RBO-8

Question Source: Modified Bank – 2015 Cert #5

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 #	239002 K4.04
	Importance Rating	3.4

SRVs

Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Ensures even distribution of heat load to suppression pool, and adequate steam condensing

Proposed Question: #7

A steam leak in the Main Steam Tunnel has resulted in the following:

- The Reactor has scrammed.
- The Control Room Supervisor has directed Reactor pressure controlled 800-1000 psig using ERVs.

Which one of the following describes how ERVs should be operated, in accordance with the Emergency Operating Procedures?

A. Cycle through all ERVs sequentially, as needed, to verify they are all seated properly.

B. Cycle the same ERV repeatedly, as needed, to prevent thermal cycling of multiple ERV tailpipes.

C. Cycle through all ERVs sequentially, as needed, to distribute the heat load uniformly around the Torus.

D. Cycle the same ERV repeatedly, as needed, to maintain Torus heating as close as possible to the Torus Cooling injection point.

Proposed Answer: C

Explanation: ERVs discharges are designed to be equally spaced at points around the Torus. N1-EOP-2 contains direction on the use of ERVs for Reactor pressure control based on this design to ensure more equal heat load distribution around the Torus by opening ERVs in a preferred sequence, rather than just opening the same ERV repeatedly.

Note: The question meets the K/A by testing knowledge of ERV design features (discharge spacing around Torus, preferred opening sequence) that ensures equal distribution of heat load around the Torus, and thereby helps ensure adequate steam condensing by avoiding localized hot spots.

A. Plausible – The reason for operating ERVs sequentially is to distribute the heat load uniformly around the torus and equalize the number of actuations among the ERVs, NOT to ensure they are all seated. Plausible because cycling a leaking ERVs is the method used to attempt to reseat the valve.
B. Plausible – N1-EOP-2 contains a note that directs opening ERVs per a pre-defined sequence, vice opening the same ERV multiple times. Plausible because opening a new ERV will cause a larger thermal transient since its tailpipe is colder.
D. Plausible – N1-EOP-2 contains a note that directs opening ERVs per a pre-defined sequence, vice opening the same ERV multiple times. Plausible because Torus Cooling does discharge to one point in the Torus through the test return line, so this area will be cooled more directly.

Technical Reference(s): N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239002-RBO-10

Question Source: Bank – 2013 Cert #24

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K4.06
	Importance Rating	2.6

APRM / LPRM

Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Effects of detector aging on LPRM/APRM readings

Proposed Question: #8

Which one of the following describes the method used to adjust APRM readings to offset the effects of LPRM detector aging and the required range for APRM readings relative to core thermal power, in accordance with Reactor Engineering Procedures?

Note: Assume FPAPDR is less than or equal to 1.

__(1)__ is adjusted to achieve a reading that is __(2)__ of core thermal power.

	(1)	(2)
A.	Flux amplifier gain	+2%, -0%
B.	Flux amplifier gain	+0%, -2%
C.	Power supply voltage	+2%, -0%
D.	Power supply voltage	+0%, -2%

Proposed Answer: A

Explanation: APRMs are adjusted to offset the effects of LPRM aging by changing the flux amplifier gain. N1-REP-12 requires the APRM reading to be adjusted to +2.0%, -0.0% of core thermal power.

B. Plausible – The APRM reading is allowed to be slightly higher than core thermal power, not lower, as would be the case with a range of +0.0%, -2.0%.
C. Plausible – Flux amplifier gain, not power supply voltage, is the parameter adjusted. Plausible because changing power supply voltage would affect instrument behavior.
D. Plausible – Flux amplifier gain, not power supply voltage, is the parameter adjusted. Plausible because changing power supply voltage would affect instrument behavior. The APRM reading is allowed to be slightly higher than core thermal power, not lower, as would be the case with a range of +0.0%, -2.0%.

Technical Reference(s): N1-REP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-2

Question Source: Modified Bank – 2008 Cert #24

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 #	215003 K5.01
	Importance Rating	2.6

IRM

Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: Detector operation

Proposed Question: #9

A plant startup is in progress with the following:

- The following annunciators alarm:
  - F2-3-6, IRM 11-14
  - F3-4-4, ROD BLOCK
- IRMs 11-14 are all on range 4 and indicate stable as follows:

IRM	Indication
11	92
12	3
13	40
14	65

Which one of the following is required to clear the rod block?

Place IRM…

A. 11 on range 3.

B. 11 on range 5.

C. 12 on range 3.

D. 12 on range 5.

Proposed Answer: C

Explanation: N1-OP-43A, Plant Startup, directs maintaining IRM indication in a band of 25-75. IRMs cause a downscale rod block if they indicate below 7.5 and an upscale rod block if they indicate above 107.5. With the given indications, IRM 11 is indicating above the preferred range, but is not so high as to cause an upscale rod block. IRM 12 is both indicating below the preferred range and causing a downscale rod block. To clear the rod block, IRM 12 indication must be raised. This is accomplished by selecting IRM 12 to the lower range 3.

A. Plausible – IRM 11 should be taken to a different range because it is outside of the nominal 25-75 band directed by N1-OP-43A. However, IRM 11 is already below the upscale rod block setpoint of 107.5, therefore this action will not clear the rod block.
B. Plausible – IRM 11 should be taken to a different range because it is outside of the nominal 25-75 band directed by N1-OP-43A. However, IRM 11 is already below the upscale rod block setpoint of 107.5, therefore this action will not clear the rod block.
D. Plausible – IRM 12 is causing the rod block, however to raise its indication, it must be taken to a lower range, not a higher range.

Technical Reference(s): ARP F2-3-6, OP-38B, OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-10

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 #	205000 K5.02
	Importance Rating	2.8

Shutdown Cooling

Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Valve operation

Proposed Question: #10

The plant is shutdown with Shutdown Cooling loop 12 in service when the following occur:

- Shutdown Cooling pump 12 trips.
- Reactor pressure rises to 130 psig.
- Reactor coolant temperature rises to 355°F.

Which one of the following actions, if any, must be taken to start an alternate Shutdown Cooling pump?

A. The Shutdown Cooling Isolation Valves must be re-opened, only.

B. Reactor coolant temperature must be reduced below 350°F, only.

C. Nothing must be done; an alternate SDC pump may be started under present conditions.

D. Reactor coolant temperature must be reduced below 350°F AND the Shutdown Cooling Isolation Valves must be re-opened.

Proposed Answer: B

Explanation: Isolation Valves 38-01, SDC SYSTEM IN IV SC-11 (INSIDE) and 38-02, SDC SYSTEM OUT IV SC-12 (OUTSIDE) are interlocked so that only one valve can be opened when Reactor pressure is above 120 psig. Below 120 psig, both valves can be opened. However, there is no close signal to these valves if pressure rises back above 120 psig. There are no other isolation signals given in the stem of the question.

The SDC IVs will close on the following signals:

- Reactor Vessel Level Low-Low (≥ + 5")
- High Area Temperature (170°F T.S. Limit)
- Manual Isolation

Since no condition caused closure of the SDC IVs, they do not need to be re-opened. The SDC pumps have a 350°F start permissive, so Reactor coolant temperature must be lowered below that value.

Note: This question satisfies the KA by testing the applicant's knowledge of valve operation during abnormal system conditions, other than just the conditions required for auto closure.

A. Plausible – Since no condition caused closure of the SDC IVs, they do not need to be re-opened. SDC IVs have an open permissive interlock for <120 psig, but do not automatically closed >120 psig.
C. Plausible – The SDC pumps have a 350°F start permissive, so Reactor coolant temperature must be lowered below that value. This choice would be correct if Reactor coolant temperature were <350°F still.
D. Plausible – Since no condition caused closure of the SDC IVs, they do not need to be re-opened. SDC IVs have an open permissive interlock for <120 psig, but do not automatically closed >120 psig.

Technical Reference(s): N1-OP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: Bank – 2010 NRC #50

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 K6.03
	Importance Rating	3.8

ADS

Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: Nuclear boiler instrument system (level indication)

Proposed Question: #11

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

- The following Lo-Lo-Lo Reactor water level instruments fail upscale:
  - 36-05A
  - 36-05B

Then, a transient occurs and the conditions required to initiate ADS are met. These conditions are valid and sustained.

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

A. The primary and secondary valves will remain closed; the 111 second and 115.5 second timers will never start timing.

B. The primary and secondary valves will remain closed after both the 111 second and 115.5 second timers have timed out.

C. The secondary valves will be opened when the initiation signal has been present for 115.5 seconds; the primary valves remain closed.

D. The primary valves will be opened when the initiation signal has been present for 111 seconds; the secondary valves remain closed.

Proposed Answer: D

Explanation: Both one (out of two) Hi DW Press and one (out of two) Lo-Lo-Lo Reactor water level trip signals from both the Channel 11 and Channel 12 logic are required to start the ADS timers. 36-05A failing upscale removes one of the Channel 11 level inputs. 36-05B failing upscale removes one of the Channel 12 level inputs. With a valid and sustained Lo-Lo-Lo Reactor water level condition, instruments 36-05C and 36-05D will trip Channels 11 and 12, respectively. This will initiate the ADS timers and the primary valves will open after 111 seconds. The secondary ADS valves are a backup for the primary valves in case of failure. ADS logic is designed such that if the three primary valves (111, 112, 113) open, the three secondary valves (121, 122, 123) do NOT open.

A. Plausible – With a valid and sustained Lo-Lo-Lo Reactor water level condition, instruments 36-05C and 36-05D will trip Channels 11 and 12, respectively. This will initiate the ADS timers and the primary valves will open after 111 seconds. If two level instruments in the same Channel were upscale, this would be the correct answer.
B. Plausible – With a valid and sustained Lo-Lo-Lo Reactor water level condition, instruments 36-05C and 36-05D will trip Channels 11 and 12, respectively. This will initiate the ADS timers and the primary valves will open after 111 seconds. If two level instruments in the same Channel were upscale, no valves would open.
C. Plausible – With a valid and sustained Lo-Lo-Lo Reactor water level condition, instruments 36-05C and 36-05D will trip Channels 11 and 12, respectively. This will initiate the ADS timers and the primary valves will open after 111 seconds.

Technical Reference(s): C-19859-C sheets 2, 5, 18, and 18A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-5

Question Source: Modified Bank – 2013 NRC #39

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 #	259002 K6.01
	Importance Rating	3.2

Reactor Water Level Control

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

Proposed Question: #12

A plant startup is in progress with the following:

- Feedwater pump 11 is operating.
- Feedwater Bypass Flow Control Valve 11 is fully closed.
- Feedwater Main Flow Control Valve 11 is 25% open.
- Reactor water level is 72" and steady.
- Then, the Instrument Air supply is lost to just Feedwater Main Flow Control Valve 11.

Note: Assume zero leak-by on Feedwater Main Flow Control Valve 11.

Which one of the following describes the effect on Feedwater flow to the Reactor?

A. Rises

B. Stays the same

C. Lowers to zero

D. Lowers, but NOT to zero

Proposed Answer: B

Explanation: The main FCV fails as-is on loss of air. This keeps Feedwater flow to the Reactor the same as before the loss of air.

A. Plausible – Feedwater flow to the Reactor stays the same. Plausible because it would rise if the valve failed open on loss of air and many air operated valves are designed this way.
C. Plausible – Feedwater flow to the Reactor stays the same. Plausible because it would lower to zero if the valve failed closed on loss of air and many air operated valves are designed this way.
D. Plausible – Feedwater flow to the Reactor stays the same. Plausible because it would lower (but remain above zero) if the valve failed closed on loss of air but had a mechanical stop to maintain some flow, such as in some cooling water systems. Also plausible because Feedwater has the HPCI flow control mode at NMP1, which places extra importance on maintaining flow from Feedwater FCV 11.

Technical Reference(s): N1-OP-16, C-18005-C sheets 1 and 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-11

Question Source: Modified Bank – 2010 Cert #17

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 #	207000 A1.10
	Importance Rating	3.2

Isolation (Emergency) Condenser

Ability to predict and/or monitor changes in parameters associated with operating the ISOLATION (EMERGENCY) CONDENSER controls including: Primary side temperature: BWR-2,3

Proposed Question: #13

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

- Annunciator K1-3-2, EMER CONDENSER 111-112 INLET STEAM TEMP HIGH, alarms.
- Computer point F376, EMER CNDSR TU 11 TEMP F., is in alarm high.
- Operators have found NO other indications to suspect an Emergency Condenser (EC) tube leak or leaking Condensate Return valve.

Which one of the following identifies an Emergency Condenser (EC) system manipulation that may be performed in an attempt to correct these conditions, in accordance with ARP K1-3-2?

A. Lower EC keep-full system flow.

B. Raise EC keep-full system flow.

C. Lower EC shell side water level.

D. Raise EC shell side water level.

Proposed Answer: B

Explanation: High EC steam inlet temperature during normal operation with the EC in standby is indicative of a low tube side water level. To lower temperature, tube side water level is raised by raising EC keep-full flow rate.

A. Plausible – EC keep-full flow rate is the correct parameter to adjust, however it must be raised in order to raise level in the EC tubes and lower the alarming temperature.
C. Plausible – Improper EC tube side water level causes the given alarming temperature, not shell side water level. Plausible because shell side water is also in contact with the EC tubes.
D. Plausible – Improper EC tube side water level causes the given alarming temperature, not shell side water level. Plausible because shell side water is also in contact with the EC tubes.

Technical Reference(s): ARP K1-3-2, N1-OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-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 #	223002 A1.04
	Importance Rating	2.6

PCIS/Nuclear Steam Supply Shutoff

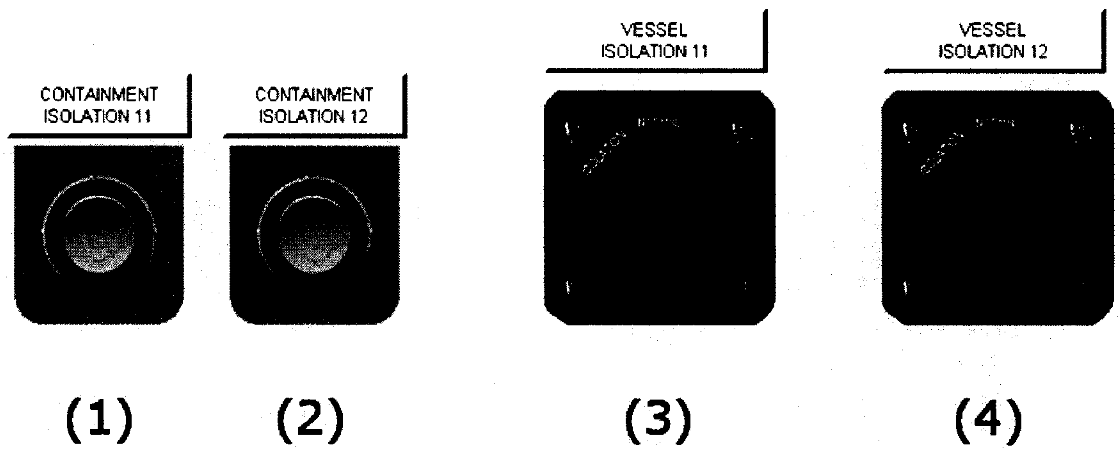
Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: Individual system relay status

Proposed Question: #14

The plant has experienced a Reactor Water Cleanup (RWCU) leak into the Reactor Building with the following:

- The Reactor has been scrammed.
- RWCU isolation valves have failed to automatically close.
- RWCU isolation valves have failed to close using the K-panel controls.
- The US has directed attempting to close the RWCU isolation valves using the isolation controls on E-console.

Given the following picture with controls labeled (1), (2), (3), and (4):



Which one of the following describes the manipulations required to initiate the manual isolation the US has requested?

A. Depress pushbuttons (1) and (2) to energize the associated isolation relays.

B. Depress pushbuttons (1) and (2) to de-energize the associated isolation relays.

C. Rotate control switches (3) and (4) to energize the associated isolation relays.

D. Rotate control switches (3) and (4) to de-energize the associated isolation relays.

Proposed Answer: D

Explanation: The RWCU isolation valves close as part of the Vessel Isolation, not the Containment Isolation. A Vessel Isolation is initiated by rotating the control switches labeled (3) and (4) to ISOLATION. This de-energizes the 4-11 relays in the RWCU isolation circuitry to cause an isolation.

A. Plausible – Pushbuttons (1) and (2) initiate a Containment Isolation, however RWCU isolates on a Vessel Isolation, not a Containment Isolation.
B. Plausible – Pushbuttons (1) and (2) initiate a Containment Isolation, however RWCU isolates on a Vessel Isolation, not a Containment Isolation.
C. Plausible – Control switches (3) and (4) are the correct controls, however they de-energize, not energize, the 4-11 relays in the RWCU isolation circuitry to cause an isolation.

Technical Reference(s): N1-SOP-40.2, C-19859-C sheet 12

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: New

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 #	1
	K/A #	209001 A2.03
	Importance Rating	3.4

LPCS

Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. failures

Proposed Question: #15

The plant is operating at 100% power when the following events occur:

- RPS Bus 11 de-energizes.
- N1-SOP-40.1, Loss of RPS, is entered.
- Ten (10) minutes later, RPS Bus 11 is re-energized from I&C Bus 130A, in accordance with N1-OP-40, Reactor Protection and ATWS Systems.

Which one of the following describes how the Core Spray automatic initiation logic is affected by this sequence of events, in accordance with N1-SOP-40.1?

Core Spray had half of the automatic initiation logic...

A. defeated. The initiation logic was restored to normal as soon as the bus was re-energized.

B. defeated. Further action is required in N1-SOP-40.1 to restore the initiation logic to normal.

C. satisfied. The initiation logic was restored to normal as soon as the bus was re-energized.

D. satisfied. Further action is required in N1-SOP-40.1 to restore the initiation logic to normal.

Proposed Answer: D

Explanation: Loss of RPS Bus 11 causes half of the Core Spray automatic initiation logic to be satisfied. After restoration of power to RPS Bus 11, the Operators must depress the high Drywell pressure reset pushbuttons before the initiation logic is restored to normal.

A. Plausible – Loss of RPS Bus 11 causes half of the Core Spray automatic initiation logic to be satisfied, not defeated. After restoration of power to RPS Bus 11, the Operators must depress the high Drywell pressure reset pushbuttons before the initiation logic is restored to normal. Plausible because RPS Bus 11 is the power supply to the Core Spray automatic initiation logic and some RPS-powered logic is energize-to-function.
B. Plausible – Loss of RPS Bus 11 causes half of the Core Spray automatic initiation logic to be satisfied, not defeated. Plausible because RPS Bus 11 is the power supply to the Core Spray automatic initiation logic and some RPS-powered logic is energize-to-function.
C. Plausible – After restoration of power to RPS Bus 11, the Operators must depress the high Drywell pressure reset pushbuttons before the initiation logic is restored to normal. Plausible because some signals do automatically reset up on power restoration.

Technical Reference(s): N1-SOP-40.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-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 #	1
	K/A #	215004 A2.01
	Importance Rating	2.7

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: Power supply degraded

Proposed Question: #16

A plant startup is in progress with the following:

- Control rod withdrawals are in progress.
- All IRMs are mid-scale on Range 9.
- SRM 11 is bypassed for calibration.

Then, the following occurs:

- The SRM 14 high voltage power supply is lost.
- NO other SRMs are affected by the power supply issue.

Which one of the following describes (1) the impact of this malfunction on the ability to continue with control rod withdrawals and (2) the ability to bypass SRM 14 while SRM 11 is still bypassed, in accordance with N1-OP-38A, Source Range Monitor?

	(1)	(2)
A.	Control rod withdrawal may continue without bypassing SRM 14.	SRM 14 can be bypassed while SRM 11 is still bypassed.
B.	Control rod withdrawal may continue without bypassing SRM 14.	SRM 14 CANNOT be bypassed while SRM 11 is still bypassed.
C.	SRM 14 must be bypassed to allow further control rod withdrawals.	SRM 14 can be bypassed while SRM 11 is still bypassed.
D.	SRM 14 must be bypassed to allow further control rod withdrawals.	SRM 14 CANNOT be bypassed while SRM 11 is still bypassed.

Proposed Answer: B

Explanation: Loss of high voltage power to SRM 14 causes an SRM INOP condition to be detected and provides an input to the rod block circuitry. However, all SRM rod blocks are bypassed with IRMs on Range 8 or higher in accordance with N1-OP-43A, section E.1, so no rod block actually occurs. In Accordance with N1-OP-38A section H, only 1 SRM can be bypassed at a time. Therefore, SRM 14 cannot be bypassed while SRM 11 is still bypassed.

A. Plausible – SRM 14 cannot be bypassed while SRM 11 is still bypassed. Plausible because some combinations of IRMs or APRMs can be simultaneously bypassed.
C. Plausible – Control rod withdrawal may continue. Plausible because if IRMs were below range 7, SRM 14 would need to be bypassed for control rod withdrawals to continue. SRM 14 cannot be bypassed while SRM 11 is still bypassed. Plausible because some combinations of IRMs or APRMs can be simultaneously bypassed.
D. Plausible – Control rod withdrawal may continue. Plausible because if IRMs were below range 7, SRM 14 would need to be bypassed for control rod withdrawals to continue.

Technical Reference(s): ARP F3-4-1, 1101-215000C01, C-22030-C sh 2, N2-OP-38A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Modified Bank – 2009 Cert #9

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 A3.01
	Importance Rating	2.8

UPS (AC/DC)

Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source

Proposed Question: #17

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

- UPS 162A is in service.
- Then, the UPS 162A inverter fails and the inverter output goes to zero.

Which one of the following describes the resulting status of RPS Bus 11?

RPS Bus 11 is...

A. powered by UPS 162B.

B. powered by the bypass transformer from Powerboard 16B.

C. powered by the bypass transformer from Battery Board 11.

D. de-energized and must be manually transferred to I&C Bus 130A.

Proposed Answer: B

Explanation: With an inverter failure, the UPS will automatically swap to the bypass transformer with no interruption in power. The bypass transformer is powered from Powerboard 16B.

A. Plausible – UPS 162B is available to supply RPS Bus 11, however transfer to this supply is manual, not automatic.
C. Plausible – The bypass transformer is powered from Powerboard 16B, not Battery Board 11. Battery Board 11 is the DC source to UPS 162A, but requires the inverter to be functioning to supply RPS Bus 11.
D. Plausible – An inverter failure causes the UPS output to transfer automatically and with no interruption in power to the bypass transformer. I&C Bus 130A is available to supply RPS Bus 11 if needed.

Technical Reference(s): N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-5

Question Source: Bank – 2009 NRC #48

Question History: 2009 NRC #48

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 A3.04
	Importance Rating	3.0

SGTS

Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: System temperature

Proposed Question: #18

The plant is operating at 100% power with both trains of Reactor Building Emergency Ventilation (RBEVS) in standby.

Given the following components:

(1) 10 KW heater at the RBEVS system inlet
(2) 1 KW heaters at the RBEVS charcoal filter inlets

Which one of the following describes the operation of these heaters?

A. Both (1) and (2) are continuously de-energized until the system initiates.

B. (1) is continuously de-energized until the system initiates. (2) cycles based on temperature.

C. (1) cycles based on temperature. (2) are continuously de-energized until the system initiates.

D. Both (1) and (2) cycle based on temperature.

Proposed Answer: B

Explanation: The 10 KW heater at the RBEVS system inlet remains off while the system is in standby. The 10 KW heater automatically starts when system flow is detected. The 1 KW heaters at the charcoal filter inlets cycle to maintain temperature at the charcoal filters approximately 165°F. These heaters perform this function even when the system is in standby.

A. Plausible – The 1 KW heaters at the charcoal filter inlets cycle to maintain temperature at the charcoal filters approximately 165°F, even when the system is in standby. Plausible since the 10 KW heater is continuously de-energized until the system initiates.
C. Plausible – The 10 KW heater remains off until system flow is initiated. The 1 KW heaters at the charcoal filter inlets cycle to maintain temperature at the charcoal filters approximately 165°F, even when the system is in standby. Plausible if applicant confuses the operation of the two heaters.
D. Plausible – The 10 KW heater remains off until system flow is initiated. Plausible because the 1 KW heater cycles based on temperature.

Technical Reference(s): C-18013-C, C-19437-C sheet 7, C-19438-C sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-5

Question Source: New

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 #	1
	K/A #	262001 A4.05
	Importance Rating	3.3

AC Electrical Distribution

Ability to manually operate and/or monitor in the control room: Voltage, current, power, and frequency on A.C. buses

Proposed Question: #19

A loss of coolant accident has occurred with the following:

- A manual Reactor scram was inserted.
- Following the scram, Lines 1 and 4 de-energized.
- N1-SOP-33A.1, Loss of 115 KV, has been entered.
- EDG 102 load is 2450 KW.
- EDG 103 load is 2750 KW.

Which one of the following describes the status of EDG load, in accordance with N1-SOP-33A.1?

A. Both EDGs are below the load limit requiring operator action.

B. EDG 102 is below the load limit requiring operator action. EDG 103 is above the load limit requiring operator action, but below the emergency load limit.

C. Both EDGs are above the load limit requiring operator action, but below the emergency load limit.

D. Both EDGs are above the load limit requiring operator action. EDG 102 is below the emergency load limit, but EDG 103 is above the emergency load limit.

Proposed Answer: C

Explanation: N1-SOP-33A.1 requires operator action if EDG load exceeds 2300 KW and lists the EDG emergency load limit as 2845 KW. Both EDGs are loaded greater than 2300 KW, but less than 2845 KW.

Note: The question meets the K/A by testing ability to monitor AC bus power. EDG 102 and 103 loads are directly related to Powerboard 102 and 103 power.

A. Plausible – N1-SOP-33A.1 requires operator action if EDG load exceeds 2300 KW. Both EDGs are above this load. Plausible because both EDGs are below the emergency load limit.
B. Plausible – N1-SOP-33A.1 requires operator action if EDG load exceeds 2300 KW. EDG 102 is above this load. Plausible because the EDG loads are on either side of the continuous load limit of 2586 KW.
D. Plausible – EDG 103 is below the emergency load limit of 2845 KW. Plausible because EDG 103 is above the continuous load limit of 2586 KW.

Technical Reference(s): N1-SOP-33A.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-RBO-9

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 #	2
	Group #	1
	K/A #	211000 A4.07
	Importance Rating	3.6

SLC

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

Proposed Question: #20

A failure to scram has occurred with the following:

- The US has directed initiation of Liquid Poison pump 11.
- The Liquid Poison keylock switch is rotated to the SYS 11 position.

Which one of the following describes the resulting status of the LIQUID POISON EXPL VALVE CONTINUITY white lights and Annunciator K1-2-1, LIQ POISON EXPLOSIVE VALVE 11-12 CONTINUITY?

The LIQUID POISON EXPL VALVE CONTINUITY white lights are...

A. illuminated and Annunciator K1-2-1 is in alarm.

B. extinguished and Annunciator K1-2-1 is in alarm.

C. illuminated and Annunciator K1-2-1 is NOT alarm.

D. extinguished and Annunciator K1-2-1 is NOT in alarm.

Proposed Answer: B

Explanation: The white continuity lights are normally lit, but extinguish when the Liquid Poison keylock switch is taken to SYS 11. Annunciator K1-2-1, LIQ POISON EXPLOSIVE VALVE 11-12 CONTINUITY, is not normally in alarm, but alarms when the Liquid Poison keylock switch is taken to SYS 11.

A. Plausible – The white continuity lights are normally lit, but extinguish when the Liquid Poison keylock switch is taken to SYS 11. Plausible because on some other plants the lights remain lit until the switch is taken back to OFF.
C. Plausible – The white continuity lights are normally lit, but extinguish when the Liquid Poison keylock switch is taken to SYS 11. Plausible because on some other plants the lights remain lit until the switch is taken back to OFF. Annunciator K1-2-1, LIQ POISON EXPLOSIVE VALVE 11-12 CONTINUITY, is not normally in alarm, but alarms when the Liquid Poison keylock switch is taken to SYS 11. Plausible because on some other plants this alarm does not come in until the switch is taken back to OFF.
D. Plausible – Annunciator K1-2-1, LIQ POISON EXPLOSIVE VALVE 11-12 CONTINUITY, is not normally in alarm, but alarms when the Liquid Poison keylock switch is taken to SYS 11. Plausible because on some other plants this alarm does not come in until the switch is taken back to OFF.

Technical Reference(s): ARP K1-2-1, N1-EOP-HC Attachment 10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO-5

Question Source: Bank – 9/14 JAF NRC #19

Question History: 9/14 JAF NRC #19

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 #	212000 2.4.34
	Importance Rating	4.2

RPS

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: #21

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

- A significant fire develops in the Control Room.
- N1-SOP-21.2, Control Room Evacuation, is entered.
- The Reactor Mode Switch is placed in SHUTDOWN.
- All RPS Group Scram Solenoid white lights remain lit.
- NO control rod motion is observed.
- The Control Room is evacuated ***before*** any other operator actions are taken.

Which one of the following describes the required field actions to be performed to scram the Reactor, in accordance with N1-SOP-21.2?

A. Isolate and vent the scram air header.

B. Pull RPS group scram solenoid fuses.

C. Trip RPS Motor Generator Sets 131 and 141.

D. Trip RPS Uninterruptible Power Supplies 162 and 172.

Proposed Answer: C

Explanation: N1-SOP-21.2 normally requires scramming the Reactor and verifying control rod insertion from the Control Room prior to evacuation. In this event, with failure of the Mode Switch to insert control rods, alternate field actions are required to insert control rods. N1-SOP-21.2 requires a Reactor Operator to trip the RPS MG sets (131 and 141) from outside the Control Room.

A. Plausible – Pulling RPS fuses is a valid method for inserting control rods under normal failure to scram conditions, however it is not the method required by N1-SOP-21.2. Other actions in N1-SOP-21.2 do require pulling other fuses, such as ERV fuses.
B. Plausible – Isolating and venting the scram air header is a valid method for inserting control rods under normal failure to scram conditions, however it is not the method required by N1-SOP-21.2. Other actions in N1-SOP-21.2 do require isolating/venting, such as those for manual vessel isolation.
D. Plausible – N1-SOP-21.2 does require de-energizing a part of the RPS power distribution to insert control rods. However, it does so by tripping RPS MG sets, not the UPSs. Tripping UPS 162 and 172 would cause control rods to insert, but would also complicate the transient due to de-energizing other electrical loads.

Technical Reference(s): N1-SOP-21.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.2C01 EO-2

Question Source: Modified Bank – SSES LOC27 NRC #84

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 #	218000 2.4.11
	Importance Rating	4.0

ADS

Knowledge of abnormal condition procedures.

Proposed Question: #22

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

- ERV 111 inadvertently opens due to an erroneous ADS signal.
- NO other ERVs are affected.
- N1-SOP-1.4, Stuck Open ERV, is being executed.

Given the following methods to attempt closure of ERV 111 from N1-SOP-1.4:

(1) Pulling ERV 111 control power fuses (F15 and F30) in the Control Room.
(2) Pulling ERV 111 control power fuses (F1 and F2) in the Reactor Building.

Note: A portion of C-19859-C Sheet 24, ERV Valve #111, is provided on the next page.

Which one of the following describes which of these methods, if any, will close ERV 111 given this specific failure, in accordance with N1-SOP-1.4?

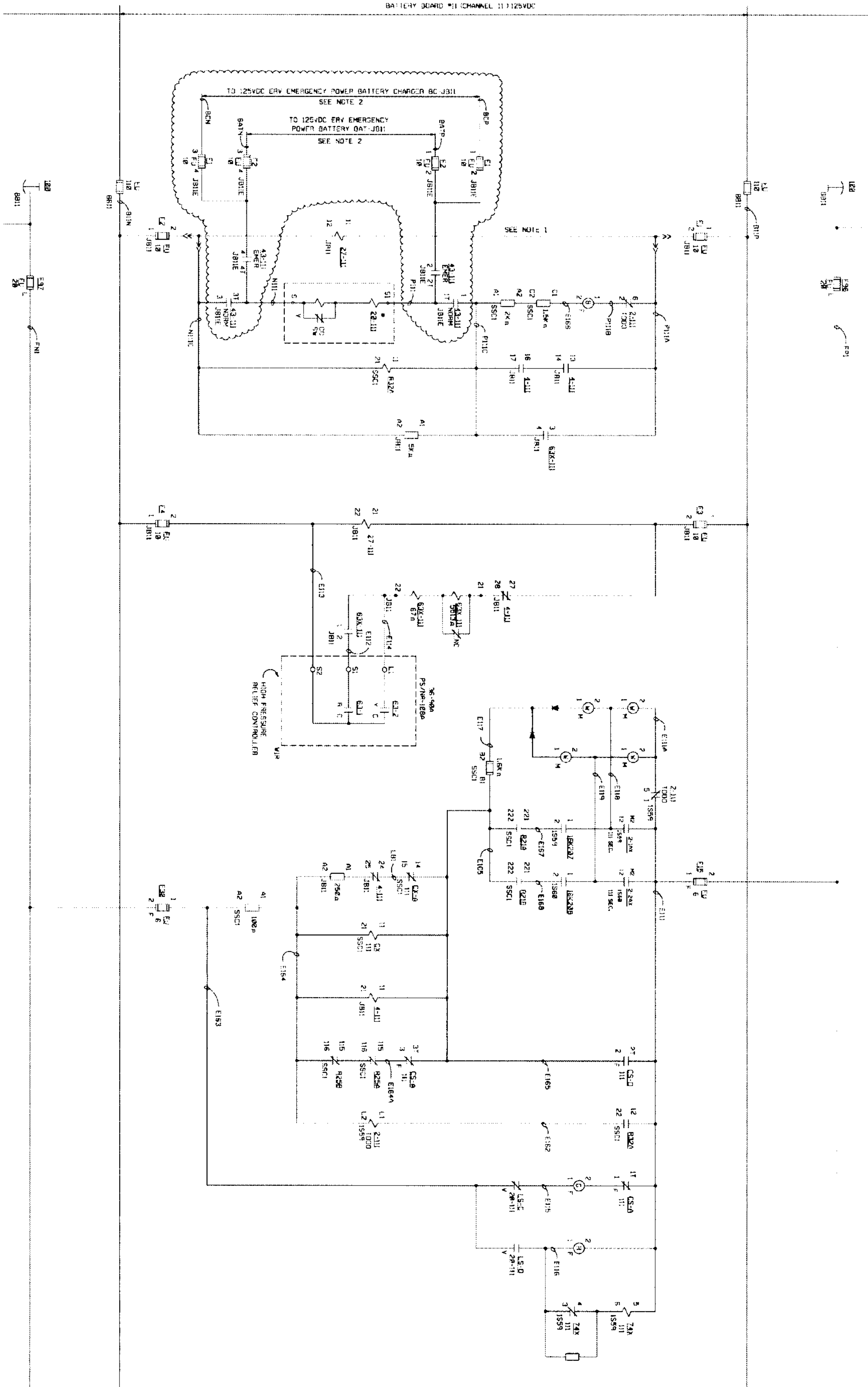
A. Neither of these methods will close ERV 111.

B. Method (1) will close ERV 111, but method (2) will NOT.

C. Method (2) will close ERV 111, but method (1) will NOT.

D. Either of these methods will close ERV 111.

BATTERY BOARD #1 (CHANNEL 1) 125VDC



CONTINUED ON SH.25 THIS DRAWING

Proposed Answer: D

Explanation: In the Control Room (method (1)), N1-SOP-1.4 directs pulling Fuses F15 and F30 in F panel for ERV 111. These two fuses take away power from the ADS portion of the ERV 111 logic. In the Reactor Building (method (2)), N1-SOP-1.4 directs pulling Fuses F1 and F2 in JB11 for ERV 111. These fuses do NOT specifically take power away from the ADS logic for ERV 111, but they do take away power from the ERV solenoid itself. Therefore, this method will close ERV 111 with the given failure.

A. Plausible – Both Methods (1) and (2) will close ERV 111 with the given failure.
B. Plausible – Method (2) will also close ERV 111 with the given failure.
C. Plausible – Method (1) will also close ERV 111 with the given failure.

Technical Reference(s): N1-SOP-1.4, C-19859-C sheet 24

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-10

Question Source: Modified Bank – 2015 NRC #15

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 #	400000 K6.07
	Importance Rating	2.7

Component Cooling Water

Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Breakers, relays, and disconnects

Proposed Question: #23

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

- Service Water pump 12 is operating.
- Service Water pump 11 is in standby.
- RBCLC pumps 11 and 13 are operating.
- RBCLC pump 12 is in standby.

Then, Breaker R113, PB-11 – Normal Supply Trans. 10, trips due to a sustained electrical fault.

Which one of the following identifies the resulting number of operating Service Water and RBCLC pumps?

	Number of Operating Service Water Pumps	Number of Operating RBCLC Pumps
A.	0	1
B.	0	2
C.	1	1
D.	1	2

Proposed Answer: C

Explanation: Trip of Breaker R113 due to a sustained electrical fault causes Powerboard 11 to de-energize. This also de-energizes downstream Powerboard 16A, which causes the loss of RBCLC pump 11. Service Water pump 12 remains operating (powered from unaffected Powerboard 12). RBCLC pump 13 remains operating (powered from unaffected Powerboard 16B). RBCLC pump 12 remains in standby, since there is no auto-start feature on this pump. This results in 1 operating Service Water pump and 1 operating RBCLC pump.

A. Plausible – Service Water pump 12 is unaffected by the trip of Breaker R113 and remains operating. Plausible because Service Water pump 12 is powered from a similar board.
B. Plausible – Service Water pump 12 is unaffected by the trip of Breaker R113 and remains operating. RBCLC pump 11 trips and RBCLC pump 12 does NOT auto-start. Plausible because Service Water pump 12 is powered from a similar board and many plants have RBCLC pump auto-starts.
D. Plausible – RBCLC pump 11 trips and RBCLC pump 12 does NOT auto-start. Plausible because many plants have RBCLC pump auto-starts.

Technical Reference(s): C-19409-C sheet 1b

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-4

Question Source: Bank – SYSID 88440

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 K4.03
	Importance Rating	3.8

Shutdown Cooling

Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: Low reactor water level: Plant-Specific

Proposed Question: #24

The plant is shutting down for a refueling outage with the following:

- Reactor pressure is 110 psig and slowly lowering.
- Shutdown Cooling has just been placed in service on Loop 12.
- Then, a Reactor coolant leak develops.
- Reactor water level is 0 inches and slowly lowering.

Which one of the following describes the status of Shutdown Cooling?

Shutdown Cooling...

A. remains in service.

B. isolates, but Shutdown Cooling pump 12 remains running.

C. pump 12 trips, but Shutdown Cooling remains un-isolated.

D. isolates and Shutdown Cooling pump 12 trips.

Proposed Answer: D

Explanation: Reactor water level less than +5" causes a Vessel Isolation. The Vessel Isolation closes the Shutdown Cooling isolation valves. The Vessel Isolation does NOT provide a direct signal to trip Shutdown Cooling pump 12. However, Shutdown Cooling pump 12 trips on interlock when the isolation valves leave the full open position.

A. Plausible – SDC isolates with Reactor water level less than +5". The isolation valves closing cause the pump to trip. This would be correct if Reactor water was slightly higher.
B. Plausible – The Vessel Isolation does NOT provide a direct signal to trip Shutdown Cooling pump 12. However, Shutdown Cooling pump 12 trips on interlock when the isolation valves leave the full open position.
C. Plausible – SDC isolates with Reactor water level less than +5". This choice is the SDC response to a high coolant temperature (>350°F).

Technical Reference(s): N1-OP-4 P&Ls 4 & 5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

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 #	215004 2.1.20
	Importance Rating	4.6

Source Range Monitor

Ability to interpret and execute procedure steps.

Proposed Question: #25

Which one of the following describes the criteria used to fully withdraw SRM and IRM detectors in accordance with N1-OP-43A, Plant Startup?

	SRMs are fully withdrawn as soon as all IRMs are...	IRMs are fully withdrawn as soon as...
A.	on-scale and overlap is verified.	all APRMs indicate ≥ 6%.
B.	on-scale and overlap is verified.	the Reactor Mode Switch is placed in RUN.
C.	on Range 8 or above.	all APRMs indicate ≥ 6%.
D.	on Range 8 or above.	the Reactor Mode Switch is placed in RUN.

Proposed Answer: D

Explanation: N1-OP-43A step E.2.18 requires fully withdrawing SRMs when all IRMs are on Range 8 or above. N1-OP-43A step E.4.10 requires fully withdrawing IRMs after the Reactor Mode Switch has been taken to RUN.

A. Plausible – SRM-IRM overlap is verified when IRMs are on Range 2. SRMs are not fully withdrawn until IRMs indicate much higher (Range 8). Plausible because overlap is confirmed as soon as the IRMs are on-scale, and after this point the SRMs are only redundant indication. APRMs being above the downscale setpoint (≥6%) is part of the criteria used to go to RUN, but not enough by itself to fully withdraw IRMs. Plausible because when APRMs are above 6%, IRM-APRM overlap is verified, which makes the IRMs redundant indication.
B. Plausible – SRM-IRM overlap is verified when IRMs are on Range 2. SRMs are not fully withdrawn until IRMs indicate much higher (Range 8). Plausible because overlap is confirmed as soon as the IRMs are on-scale, and after this point the SRMs are only redundant indication.
C. Plausible – APRMs being above the downscale setpoint (≥6%) is part of the criteria used to go to RUN, but not enough by itself to fully withdraw IRMs. Plausible because when APRMs are above 6%, IRM-APRM overlap is verified, which makes the IRMs redundant indication.

Technical Reference(s): N1-OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-10

Question Source: Modified Bank – JAF 4/14 NRC #20

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 #	206000 K1.04
	Importance Rating	3.6

HPCI

Knowledge of the physical connections and/or cause-effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: Reactor feedwater system: BWR-2,3,4

Proposed Question: #26

The plant is operating at 2% power during a startup with the following:

- Feedwater pump 12 is injecting to the Reactor with Feedwater flow control valve (FCV) 12 in AUTO.
- Feedwater pump 12 trips.
- The Reactor scrams on low Reactor water level.
- Feedwater pump 11 fails to start and CANNOT be manually started.
- Reactor water level is 30" and slowly lowering.
- Reactor pressure is manually lowered to 400 psig using Turbine Bypass Valves.

Which one of the following describes the response of Feedwater FCV 12?

Feedwater FCV 12...

A. automatically injects to the Reactor in HPCI mode.

B. automatically injects to the Reactor, but does NOT enter HPCI mode.

C. does NOT automatically inject to the Reactor and can only inject if locally pinned.

D. does NOT automatically inject to the Reactor, but can be aligned for injection by pulling HPCI fuses.

Proposed Answer: D

Explanation: With Reactor water level <53", Feedwater FCV 12 shifts to the HPCI mode of operation. With no high pressure Feedwater pump operating, the open permissive (~990 psig) for Feedwater FCV 12 is NOT met, therefore even when Reactor pressure is lowered to within the injection capacity of Feedwater Booster pumps, Feedwater FCV 12 will remain closed. N1-EOP-1 Attachment 26 provides a method for opening Feedwater FCV 12 by pulling HPCI fuses and then operating the controller in the Control Room.

A. Plausible – Feedwater FCV 12 does shift to the HPCI mode of operation, however with no operating high pressure Feedwater pump, it does not open to inject to the Reactor due to no open permissive signal based on Feedwater discharge pressure.
B. Plausible – With no high pressure Feedwater pump operating, the open permissive (~990 psig) for Feedwater FCV 12 is NOT met, however the valve still shifts to the HPCI mode of operation and remains closed.
C. Plausible – Feedwater FCV 12 does NOT inject automatically, however N1-EOP-1 Attachment 26 provides a method for injecting with the valve without needing to locally pin the valve.

Technical Reference(s): N1-OP-16, N1-EOP-2, N1-EOP-1 Attachment 26

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-5

Question Source: Bank – 2015 Cert #13

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 #	215001 K1.01
	Importance Rating	2.5

Traversing In-core Probe

Knowledge of the physical connections and/or cause-effect relationships between TRAVERSING IN-CORE PROBE and the following: Local power range monitors (Not-BWR1)

Proposed Question: #27

Which one of the following describes the relationship between the Neutron Monitoring and the Traversing In-core Probes (TIPs)?

TIPS are used for ___(1)___. Each TIP machine is installed with the capability to run traces for ___(2)___ of the in-core guide tubes.

	(1)	(2)
A.	adjustment of APRM gains	approximately 25%
B.	adjustment of APRM gains	100%
C.	calibration of LPRMs	approximately 25%
D.	calibration of LPRMs	100%

Proposed Answer: C

Explanation: The TIPs are used to calibrate LPRMs, which are the inputs to the APRMS. Each of the four TIP machines is installed with the ability to run traces along 8-9 of the 30 LPRM strings (~25-30%).

A. Plausible – APRM gains are adjusted from a heat balance, not a flux reading from the TIPs. Plausible because LPRMs and APRMs are closely related.
B. Plausible – APRM gains are adjusted from a heat balance, not a flux reading from the TIPs. Plausible because LPRMs and APRMs are closely related. Each of the four TIP machines is installed with the ability to run traces along 8-9 of the 30 LPRM strings (~25-30%). 100% is plausible if the system were designed with better redundancy.
D. Plausible – Each of the four TIP machines is installed with the ability to run traces along 8-9 of the 30 LPRM strings (~25-30%). 100% is plausible if the system were designed with better redundancy.

Technical Reference(s): N1-OP-39

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215001-RBO-8

Question Source: Bank – 2010 Cert #1

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 K2.02
	Importance Rating	3.2

Recirculation

Knowledge of electrical power supplies to the following: MG sets: Plant-Specific

Proposed Question: #28

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

- Reactor Recirculation pump (RRP) 12 is shut down for maintenance.
- Then, Powerboard 12 de-energizes due to a sustained electrical fault.

Which one of the following describes the required control of the Reactor and the resulting number of operating RRPs, in accordance with N1-SOP-1.3, Recirc Pump Trip at Power?

A. Continue Reactor operation at power with only two RRPs operating.

B. Continue Reactor operation at power with only three RRPs operating.

C. Manually scram the Reactor because only two RRPs are operating.

D. Manually scram the Reactor because only three RRPs are operating.

Proposed Answer: C

Explanation: The loss of Powerboard 12 leads to the trip of RRPs 14 and 15. This leaves only RRPs 11 and 13 operating. With less than three RRPs operating, N1-SOP-1.3 requires manually scramming the Reactor.

A. Plausible – N1-SOP-1.3 requires at least three RRPs operating to continue Reactor operation at power. Plausible because 2 RRPs are operating and still providing forced core circulation.
B. Plausible – The loss of Powerboard 12 leads to the trip of RRPs 14 and 15. This leaves only RRPs 11 and 13 operating. Plausible because N1-SOP-1.3 requires at least three RRPs operating to continue Reactor operation at power.
D. Plausible – The loss of Powerboard 12 leads to the trip of RRPs 14 and 15. This leaves only RRPs 11 and 13 operating. Plausible if believed that Powerboard 12 supplied RRPS 11 and 12 and at least 4 RRPs required to continue operation.

Technical Reference(s): C-19409-C sheet 1b, N1-SOP-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-4

Question Source: Modified Bank – SYSID 88308

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 #	233000 K3.08
	Importance Rating	2.9

Fuel Pool Cooling/Cleanup

Knowledge of the effect that a loss or malfunction of the FUEL POOL COOLING AND CLEAN-UP will have on following: Refueling operations

Proposed Question: #29

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

- Fuel movement is in progress in the Spent Fuel Pool in preparation for a refueling outage.
- An irradiated fuel assembly has just been loaded on the Refuel Bridge main hoist and raised to the full up position.

Then, a seismic event results in the following:

- The Refuel Bridge main hoist CANNOT be moved.
- The common discharge line from the Spent Fuel Pool Cooling pumps completely ruptures.

Which one of the following describes the Spent Fuel Pool water level response to this event and the availability of Spent Fuel Pool makeup?

Spent Fuel Pool water level ___(1)___. Spent Fuel Pool makeup water is ___(2)___.

	(1)	(2)
A.	lowers but maintains the fuel assembly fully covered	still available from the normal makeup source
B.	lowers but maintains the fuel assembly fully covered	NOT available from the normal makeup source
C.	lowers and partially uncovers the fuel assembly	available using hoses from Fire Water only
D.	lowers and partially uncovers the fuel assembly	available using hoses from Condensate Transfer, Demineralized Water, and Fire Water

Proposed Answer: A

Explanation: A complete rupture of the SFPC pumps common discharge line results in loss of water circulation back to the SFP. This will result in SFP water level dropping about 2" due to the height of the weirs. However, siphoning of SFP water back through the rupture is prevented by vacuum breakers installed in the discharge piping. The design of the Refuel Bridge main hoist also ensures this SFP water level will maintain the elevated fuel assembly covered with water. While the backup source of makeup water (from Condensate Transfer to the SFPC skimmer surge tanks) will be unavailable for SFP makeup due to the pipe break, the normal source of makeup water (from Condensate Transfer directly to the SFP) will still be available.

Note: The first half of the question meets the K/A by presenting a malfunction of SFPC (pipe rupture) and asking about the effect on a Refueling operation (covery of a fuel bundle stuck in full-up position).

B. Plausible – Normal makeup from Condensate Transfer to the SFP is still available through LCV 57-25. Plausible because the backup source of makeup is lost.
C. Plausible – SFP water level will lower about 2" to the top of the weirs, which will still maintain the fuel assembly fully covered with water. Normal makeup water is also available from Condensate Transfer and Demineralized Water. Plausible because flow from SFPC pumps is lost and level does lower.
D. Plausible – SFP water level will lower about 2" to the top of the weirs, which will still maintain the fuel assembly fully covered with water. Plausible because flow from SFPC pumps is lost and level does lower.

Technical Reference(s): C-18008-C, OP-6, 1101-233000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-233000-RBO-11

Question Source: Bank – 2013 NRC #10

Question History: 2013 NRC #10

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 K4.19
	Importance Rating	3.6

Reactor/Turbine Pressure Regulator

Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Steam bypass valve control

Proposed Question: #30

The plant is operating at 80% power with the EPR in control.

Which one of the following describes the response of the Turbine Control Valves (TCVs) and Turbine Bypass Valves (TBVs) if the BY-PASS OPENING JACK control switch is held in the OPEN position?

TCVs...

A. close as TBVs open.

B. open, then TBVs open.

C. and TBVs simultaneously open.

D. remain in the same position while TBVs open.

Proposed Answer: B

Explanation: At 80% power, the TCVs are partially open and the TBVs are closed. Holding the BY-PASS OPENING JACK control switch in the OPEN position raises the steam demand signal from the MHC pressure control unit. This causes the TCVs to further open until the flow limit is reached. Once the flow limit is reached, the control valve relay passes signal to the bypass valve relay, which causes the TBVs to begin opening.

A. Plausible – The Bypass Opening Jack opens TCVs further prior to opening TBVs. Plausible if only TBVs were opened by BOJM, and TCVs were still attempting to control Reactor pressure.
C. Plausible – The Bypass Opening Jack operates through the MHC pressure control unit. Therefore, TCVs are opened further first, and only after they reach a limit are the TBVs opened.
D. Plausible – The Bypass Opening Jack opens TCVs further prior to opening TBVs. Plausible if only TBVs were opened by BOJM, and TCVs were fixed by another portion of MHC.

Technical Reference(s): N1-OP-31, 1101-248000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-248000-RBO-2

Question Source: Bank – SYSID 36260

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 #	226001 K5.06
Importance Rating	2.6

RHR/LPCI: Containment Spray Mode

Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE: Vacuum breaker operation

Proposed Question: #31

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

- Drywell pressure is 16 psig and rising slowly.
- Torus pressure is 14 psig and rising slowly.
- Containment Sprays are manually initiated per N1-EOP-1 Attachment 17, Auto or Manual Initiation of Containment Spray.
- Containment Sprays are secured when required by N1-EOP-4, Primary Containment Control.

Which one of the following describes the operation of the Torus to Drywell vacuum breakers and the Reactor Building to Torus vacuum breakers during Containment Spray operation?

A. Both the Torus to Drywell vacuum breakers and the Reactor Building to Torus vacuum breakers will cycle.

B. Both the Torus to Drywell vacuum breakers and the Reactor Building to Torus vacuum breakers will remain closed.

C. The Torus to Drywell vacuum breakers will cycle, but the Reactor Building to Torus vacuum breakers will remain closed.

D. The Reactor Building to Torus vacuum breakers will cycle, but the Torus to Drywell vacuum breakers will remain closed.

Proposed Answer: C

Explanation: Operation of Containment Sprays during a loss of coolant accident causes Drywell pressure to lower below Torus pressure. The Torus to Drywell vacuum breakers cycle to prevent excessive pressure differential. N1-EOP-1 Attachment 17 and N1-EOP-4 require securing Containment Sprays when Drywell pressure drops below 3.5 psig. This prevents Containment pressure from lowering below Reactor Building pressure and cycling of the Reactor Building to Torus vacuum breakers.

A. Plausible – The Reactor Building to Torus vacuum breakers do not cycle because Containment Sprays are secured prior to Containment pressures lowering below Reactor Building pressure. Plausible since the Torus to Drywell vacuum breakers do cycle.
B. Plausible – The Torus to Drywell vacuum breakers cycle during Containment Spray operation because Drywell pressure lowers below Torus pressure. Plausible because the Reactor Building to Torus vacuum breakers remain closed.
D. Plausible – The Torus to Drywell vacuum breakers cycle during Containment Spray operation because Drywell pressure lowers below Torus pressure. The Reactor Building to Torus vacuum breakers do not cycle because Containment Sprays are secured prior to Containment pressures lowering below Reactor Building pressure. Plausible if the two concepts are confused.

Technical Reference(s): N1-EOP-1 Attachment 17, N1-EOP-4, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226001-RBO-12

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 #	2
	K/A #	290001 K6.01
	Importance Rating	3.5

Secondary Containment

Knowledge of the effect that a loss or malfunction of the following will have on the SECONDARY CONTAINMENT: Reactor building ventilation: Plant-Specific

Proposed Question: #32

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

- An un-isolable steam leak has developed in the Reactor Building.
- Reactor Building Ventilation Exhaust fan 11 is operating.
- Reactor Building Ventilation Exhaust fan 12 is in standby.
- Reactor Building differential pressure is -0.25" Hg and stable.
- Reactor Building Ventilation exhaust radiation is 0.7 mR/hr and stable.

Then, Reactor Building Ventilation Exhaust fan 11 trips on overcurrent.

Which one of the following describes the resulting Reactor Building differential pressure?

Reactor Building differential pressure…

A. becomes positive.

B. is maintained negative by Reactor Building Ventilation Exhaust fan 12.

C. is maintained negative by one Reactor Building Emergency Ventilation fan, only.

D. is maintained negative by both Reactor Building Emergency Ventilation fans.

Proposed Answer: A

Explanation: With trip of the running Reactor Building Ventilation Exhaust fan, the standby fan does NOT automatically start and must be manually started. Additionally, Reactor Building Emergency Ventilation does NOT automatically start on either the fan trip or the resulting change in Reactor Building differential pressure. With no fans exhausting air from the Reactor Building and a steam leak into the building, Reactor Building differential pressure becomes positive.

B. Plausible – The standby Reactor Building Ventilation Exhaust fan does NOT automatically start and must be manually started. Plausible if Reactor Building Ventilation Exhaust fan had an auto-start on fan 11 breaker position, flow, or differential pressure.
C. Plausible – Neither of the Reactor Building Emergency Ventilation fans automatically start. Plausible if RBEVS fan 11 had an auto-start on RBVS fan 11 breaker position, flow, or differential pressure. Also, the applicant has to know the RBEVS initiation setpoint on ventilation exhaust radiation and recognize the conditions in the stem do not meet that.
D. Plausible – Neither of the Reactor Building Emergency Ventilation fans automatically start. Plausible if RBEVS fans had an auto-start on RBVS fan 11 breaker position, flow, or differential pressure. Also, the applicant has to know the RBEVS initiation setpoint on ventilation exhaust radiation and recognize the conditions in the stem do not meet that.

Technical Reference(s): N1-OP-10, ARP L1-1-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288000-RBO-5

Question Source: New

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 #	272000 A1.02
	Importance Rating	2.9

Radiation Monitoring

Ability to predict and/or monitor changes in parameters associated with operating the RADIATION MONITORING SYSTEM controls including: Lights, alarms, and indications associated with surveillance testing

Proposed Question: #33

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

- Area Radiation Monitor (ARM) #23, CRD ACCUMULATOR AREA, is being tested.
- The ARM is placed in TRIP TEST and the TRIP CHECK ADJUST potentiometer is rotated clockwise such that the following indications are received:
  - Annunciator H1-4-8, Area Radiation Monitors, alarms.
  - The red HIGH alarm light for ARM #23 on J panel illuminates.

Which one of the following describes the response of these two indications as the TRIP CHECK ADJUST potentiometer is rotated counterclockwise to clear the simulated high alarm condition?

A. Both indications automatically clear.

B. The annunciator automatically clears, but the red light seals in until the RESET pushbutton is depressed.

C. The red light automatically clears, but the annunciator seals in until the RESET pushbutton is depressed.

D. Both indications seal in until the RESET pushbutton is depressed.

Proposed Answer: D

Explanation: The ARM high alarm seals in, such that even when the test signal is dialed down the red light and annunciator remain sealed in until the RESET pushbutton is manually depressed.

A. Plausible – As the potentiometer is dialed down, the ARM indication will lower below the high trip setpoint, however the light and alarm seal in. Plausible if there was no seal-in feature.
B. Plausible – As the potentiometer is dialed down, the ARM indication will lower below the high trip setpoint, however the light and alarm seal in. Plausible if there was no seal-in feature on the annunciator.
C. Plausible – As the potentiometer is dialed down, the ARM indication will lower below the high trip setpoint, however the light and alarm seal in. Plausible if there was no seal-in feature on the red light.

Technical Reference(s): N1-RTP-31, N1-OP-50A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-272000-RBO-5

Question Source: Bank – 2015 Cert #33

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 #	290003 A2.01
	Importance Rating	3.1

Control Room HVAC

Ability to (a) predict the impacts of the following on the CONTROL ROOM HVAC; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Initiation/reconfiguration

Proposed Question: #34

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

- Annunciator L1-4-2, CONTROL ROOM VENT. RAD. MON. OFF NORMAL, has alarmed.
- Control Room Ventilation Radiation Monitor 11 indicates 1.50 E 02 cpm and slowly rising.
- Control Room Ventilation Radiation Monitor 12 indicates 3.56 E 02 cpm and slowly rising.

Which one of the following describes the status of the Control Room Ventilation System and the required operator action?

	Control Room Ventilation Status	Required Action
A.	Normal Control Room Ventilation is operating.	Bypass Control Room Ventilation Radiation Monitor 12.
B.	Normal Control Room Ventilation is operating.	Place one train of Control Room Emergency Ventilation in service.
C.	Control Room Emergency Ventilation is operating.	Verify the Control Room Emergency Ventilation lineup.
D.	Control Room Emergency Ventilation is operating.	Shutdown Control Room Emergency Ventilation.

Proposed Answer: C

Explanation: The Control Room Ventilation System automatically swaps from the normal mode to the emergency mode if at least one of the two radiation monitors exceeds 168 cpm above background. The given conditions have one radiation monitors below this threshold (150 cpm) and one radiation monitor above this threshold (356 cpm). Therefore, Control Room Emergency Ventilation has initiated. Although one radiation monitor is below the initiation setpoint, its reading is elevated and trending up, which confirms a valid high radiation condition and rules out a single malfunctioning radiation monitor. Therefore, ARP L1-4-2 directs verification of CREVS initiation given the high rad level. If there was just one failed radiation monitor, ARP L1-4-2 and N1-OP-49 provide guidance to shutdown Control Room Emergency Ventilation.

A. Plausible – Control Room Ventilation swaps to the Emergency lineup with only one radiation monitor above the initiation setpoint. Plausible if both radiation monitors were required to be above the initiation setpoint for system response and if monitor 12 was erroneously high.
B. Plausible – Control Room Ventilation swaps to the Emergency lineup with only one radiation monitor above the initiation setpoint. Plausible if both radiation monitors were required to be above the initiation setpoint for system response. Plausible if both radiation monitors were required to be above the initiation setpoint for system response but pre-emptive manual intervention was required with a single radiation monitor high.
D. Plausible – Although only one of the radiation monitors is currently above the initiation setpoint, the other radiation monitor is well above normal background reading and is trending up. This indicates a valid high radiation condition, not just one failed radiation monitor. If there was just one failed radiation monitor, ARP L1-4-2 and N1-OP-49 provide guidance to shutdown Control Room Emergency Ventilation.

Technical Reference(s): ARP L1-4-2, N1-OP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288003-RBO-5

Question Source: Bank – 2009 Cert #33

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201003 A3.01
	Importance Rating	3.7

Control Rod and Drive Mechanism

Ability to monitor automatic operations of the CONTROL ROD AND DRIVE MECHANISM including: Control rod position

Proposed Question: #35

The plant is operating at 100% power when an automatic Reactor scram occurs.

Which one of the following describes:

(1) the rod position indication digits on the full core display ***prior*** to resetting the scram, and

(2) the rod position indication backlight on the full core display ***after*** resetting the scram?

A. (1) Blank digits indication
(2) Green-green backlight

B. (1) Blank digits indication
(2) NO backlight

C. (1) "00" digits indication
(2) Green-green backlight

D. (1) "00" digits indication
(2) NO backlight

Proposed Answer: A

Explanation: The un-reset Reactor scram signal applies CRD/Reactor pressure under the CRDM piston to hold the rod in beyond the 00 position. The number 51 reed switch keeps the green-green background lights energized as the control rod is inserted beyond the 00 position, which results in blank windows with the green-green background lights illuminated. When the operator resets the Reactor scram, the scram valves close and the control rod drive will settle on the collet fingers for position 00 and the 00 indication will illuminate. Another reed switch (switch 52) will illuminate the green-green background at position 00.

B. Plausible – A reed switch (switch 52) will illuminate the green-green background at position 00 once the scram is reset. Plausible because the control rods do settle to a slightly different position once the scram is reset.
C. Plausible – The un-reset scram signal applies CRD/Reactor pressure under the CRDM piston to hold the rod in beyond the 00 position, resulting in blank digits indication. Plausible because the rods are effectively at the 00 position.
D. Plausible – The un-reset scram signal applies CRD/Reactor pressure under the CRDM piston to hold the rod in beyond the 00 position, resulting in blank digits indication. Plausible because the rods are effectively at the 00 position. A reed switch (switch 52) will illuminate the green-green background at position 00 once the scram is reset. Plausible because the control rods do settle to a slightly different position once the scram is reset.

Technical Reference(s): N1-SOP-1, 1101-201002C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201002-RBO-5

Question Source: Bank – 2010 NRC #37

Question History: 2010 NRC #37

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 #	201002 A4.06
	Importance Rating	2.8

RMCS

Ability to manually operate and/or monitor in the control room: Rod select matrix power switch

Proposed Question: #36

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

- Troubleshooting is in progress on the Rod Select Matrix.
- Control rod 26-11 has been selected by depressing its Rod Select pushbutton.
- It is now desired to de-select control rod 26-11.

Given the following separate operator actions:

(1) Depress and release the control rod 18-19 Rod Select pushbutton.

(2) Cycle the CONTROL ROD POWER switch to OFF and then back to ON.

Which of these operator actions, if any, will result in control rod 26-11 being de-selected in accordance with N1-OP-5, Control Rod Drive?

A. (1) only

B. (2) only

C. Either (1) or (2)

D. Neither (1) Nor (2)

Proposed Answer: C

Explanation: There are two methods to de-select a control rod – either select a different control rod by depressing a different Rod Select pushbutton or turn off power to the rod select matrix by taking the CONTROL ROD POWER switch to OFF. These are the two methods presented in the stem.

A. Plausible – Both of these methods de-select the rod. Plausible because depressing the pushbutton does result in de-selecting the other rod, but so does cycling power.
B. Plausible – Both of these methods de-select the rod. Plausible that removing power momentarily would re-position relays to de-select the rod, but so does selecting another rod.
D. Plausible – Both of these methods de-select the rod. Plausible because depressing the same rod's pushbutton does not result in de-selecting the rod by design. Also plausible that just removing power momentarily would not re-position relays to de-select the rod.

Technical Reference(s): N1-OP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201002-RBO-5

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 #	201001 2.1.7
	Importance Rating	4.4

CRD Hydraulic

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

Proposed Question: #37

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

- F-panel annunciators have been lost.
- The following stable Control Rod Drive (CRD) parameters are observed during a walk-down of F-panel:
    - CONTROL ROD DRIVE PUMP 11 – 185 amps
    - CONTROL ROD DRIVE PUMP 12 – 0 amps
    - CRD CHARGING HDR – 14.5 x 10^2 psi
    - CR WTR FLO TO REACT – 12 x 10^3 lb/hr
    - CR DRIVE WATER FLOW – 0 gpm
    - DRIVE WTR / REACT DIFF PR – 260 psi
    - CR COOLING WTR FLO – 20 x 10^3 lb/hr
    - COOL WTR / REACT DIFF PR – 9 psi
    - CRD SYS TOTAL FLOW – 32 x 10^3 lb/hr

Which one of the following describes the status of the CRD system, in accordance with N1-OP-5, Control Rod Drive System?

The CRD...

A. system is operating properly.

B. flow control valve is malfunctioning.

C. drive water pressure control valve is mis-positioned.

D. cooling water pressure control valve is mis-positioned.

Proposed Answer: A

Explanation: All given values are normal for CRD system operation at 100% power. The flow control valve is operating properly as evidenced by proper total system flow (32×10^3 lb/hr ≈ 64 gpm; 64-66 gpm range acceptable per N1-OP-5). The drive water pressure control valve is operating properly as evidenced by proper drive water D/P (250-270 psi range acceptable per N1-OP-5). The cooling water pressure control valve is operating properly as evidence by proper cooling water flow (20×10^3 lb/hr ≈ 40 gpm; 28-44 gpm acceptable per N1-OP-5).

B. Plausible – The flow control valve is operating properly as evidenced by proper total system flow (32×10^3 lb/hr ≈ 64 gpm; 64-66 gpm range acceptable per N1-OP-5). Plausible if the applicant does not know the proper range for system flow.
C. Plausible – The drive water pressure control valve is operating properly as evidenced by proper drive water D/P (250-270 psi range acceptable per N1-OP-5). Plausible if the applicant does not know the proper range for drive water pressure/flow.
D. Plausible – The cooling water pressure control valve is operating properly as evidence by proper cooling water flow (20×10^3 lb/hr ≈ 40 gpm; 28-44 gpm acceptable per N1-OP-5). Plausible if the applicant does not know the proper range for cooling water pressure/flow.

Technical Reference(s): N1-OP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-5

Question Source: 2013 Cert #54

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 #	2
	K/A #	245000 A1.01
	Importance Rating	2.7

Main Turbine Generator and Auxiliaries

Ability to predict and/or monitor changes in parameters associated with operating the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS controls including: Generator megawatts

Proposed Question: #38

A plant startup is in progress with the following:

- Main Generator synchronization is in progress, in accordance with N1-OP-32, Generator.
- The first Generator output breaker, R915, has just been closed.

Which one of the following describes the next required action and the associated minimum parameter value to be established, in accordance with N1-OP-32?

Use the...

A. GOVERNOR switch to establish a minimum of 15 MWe.

B. GOVERNOR switch to establish a minimum of 115 MWe.

C. EXCITATION switch to establish a minimum of 25 MVARS TO BUS.

D. EXCITATION switch to establish a minimum of 75 MVARS TO BUS.

Proposed Answer: A

Explanation: Immediately after closing the first Generator output breaker, N1-OP-32 requires loading the Generator to a minimum of 15 MWe. This is accomplished by taking the GOVERNOR switch to RAISE.

B. Plausible – The GOVERNOR switch is the correct control, however the minimum required load is 15 MWe. 115 MWe is based on clearing the 113 MWe that activates the Power System Stablizer (PSS).
C. Plausible – The GOVERNOR switch is used to establish MWe prior to using the EXCITER RHEOSTAT switch to establish MVARS. 25 MVARS TO BUS is the correct minimum value in a subsequent step in N1-OP-32.
D. Plausible – The GOVERNOR switch is used to establish MWe prior to using the EXCITER RHEOSTAT switch to establish MVARS. 75 MVARS TO BUS is the high end of the range in a subsequent step in N1-OP-32.

Technical Reference(s): N1-OP-32

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245001-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 #	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

Proposed Question: #39

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

- The Fire Detection System senses a fire in Hazard C-2113, Powerboard 103 Room.
- The following Alarm Detection Zones are received at Main Fire Panel 2:
    - DX-2113A
    - DX-2113B
- The associated zone control switch on Main Fire Panel 2 is in ALARM ONLY.

Which one of the following describes the required response, in accordance with OP-NM-201-005, Firefighting?

A. Initiate CO_2 suppression with no further confirmation since this area is covered by cross-zone detection.

B. Initiate fire water suppression with no further confirmation since this area is covered by cross-zone detection.

C. Dispatch at least one member of the Fire Brigade to investigate. If the fire is confirmed and suppression is requested, then initiate CO_2 suppression.

D. Dispatch at least one member of the Fire Brigade to investigate. If the fire is confirmed and suppression is requested, then initiate fire water suppression.

Proposed Answer: C

Explanation: Powerboard 103 Room is protected by a CO_2 fire suppression system. OP-NM-201-005, Firefighting, section 4.1 gives the guidance for fire alarm response for a CO_2 system that has been placed in ALARM ONLY. The procedure requires dispatching the Fire Brigade Leader or at least one member of the Fire Brigade to confirm the alarm before taking further action.

A. Plausible – At least one member of the Fire Brigade must be dispatched to confirm the alarm before taking further action, even though this zone does have cross-zone detection.
B. Plausible – At least one member of the Fire Brigade must be dispatched to confirm the alarm before taking further action, even though this zone does have cross-zone detection. Powerboard 103 Room is protected by CO_2, not water. Plausible because there are multiple areas of the plant protected by water and not CO_2.
D. Plausible – Powerboard 103 Room is protected by CO_2, not water. Plausible because there are multiple areas of the plant protected by water and not CO_2.

Technical Reference(s): OP-NM-201-005, N1-OP-21C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-286000-RBO-10

Question Source: Bank – 2013 Cert #19

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 #	295038 EK1.03
	Importance Rating	2.8

High Off-site Release Rate

Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Meteorological effects on off-site release

Proposed Question: #40

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

- I&C reports that the wind speed and direction indications have been lost from all levels of the Meteorological Tower west of the Nine Mile Point site.

Which one of the following describes the impact of this loss, in accordance with N1-OP-64, Meteorological Monitoring?

A loss of Meteorological Tower indications effects the ability to __(1)__.

Alternate wind speed and direction indications are __(2)__ from installed instrumentation in the Unit 1 Control Room.

	(1)	(2)
A.	assess radiological releases	available
B.	assess radiological releases	NOT available
C.	predict adverse weather conditions	available
D.	predict adverse weather conditions	NOT available

Proposed Answer: A

Explanation: The purpose of the Meteorological Monitoring System is to accurately document local weather conditions. This data is used to assess both accidental and routine radiological releases. While the Met Tower west of NMP is the main/primary source of wind speed and direction used, a 90' Backup Tower east of Fitzpatrick is also available. Indications from this Backup Tower are provided on installed instrumentation in the NMP Unit 1 Control Room.

Note: The question meets the K/A by testing an operational implication (available instrumentation and basis for needing instrumentation) of meteorological effects on off-site release.

B. Plausible – The tower west of NMP is the primary source of information, but there are installed indications in the NMP1 Control Room for other meteorological data.
C. Plausible – While met data could be used to predict weather and N1-OP-64 contains guidance on how to respond to adverse weather conditions, the operational concern with loss of indications is the ability to obtain wind speed and direction for assessment of radiological releases.
D. Plausible – While met data could be used to predict weather and N1-OP-64 contains guidance on how to respond to adverse weather conditions, the operational concern with loss of indications is the ability to obtain wind speed and direction for assessment of radiological releases. The tower west of NMP is the primary source of information, but there are installed indications in the NMP1 Control Room for other meteorological data.

Technical Reference(s): N1-OP-64

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP64C01 EO-3

Question Source: Bank – 2015 Cert #39

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 #	295028 EK1.02
	Importance Rating	2.9

High Drywell Temperature

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification

Proposed Question: #41

Which one of the following describes the requirement in N1-EOP-4, Primary Containment Control, for performing an RPV Blowdown due to high Drywell temperature and the basis of this requirement?

N1-EOP-4 requires RPV Blowdown if Drywell temperature cannot be restored and maintained below the threshold of...

A. 300°F. This requirement is based on ADS qualification temperatures.

B. 300°F. This requirement is based on Recirc pump seal qualification temperatures.

C. 400°F. This requirement is based on ADS qualification temperatures.

D. 400°F. This requirement is based on Recirc pump seal qualification temperatures.

Proposed Answer: A

Explanation: N1-EOP-4 requires an RPV Blowdown if Drywell temperature cannot be restored and maintained below the threshold of 300°F. This temperature limit is based on the component qualification temperature for ERV solenoids (rounded down from 301°F).

B. Plausible – The basis is ADS qualification temperature, not Recirc pump seal qualification temperatures. Plausible because Recirc pump seals are affected by high Drywell temperature and temperature damage to Recirc pump seals is a concern addressed in PRA.
C. Plausible – The limit is 300°F, not 400°F. Plausible because 400°F is the upper limit on Drywell temperature indication and is part of the basis for multiple EOP limits (Detail A - Curve B and Table C, CSIL).
D. Plausible – The limit is 300°F, not 400°F. Plausible because 400°F is the upper limit on Drywell temperature indication and is part of the basis for multiple EOP limits (Detail A - Curve B and Table C, CSIL). The basis is ADS qualification temperature, not Recirc pump seal qualification temperatures. Plausible because Recirc pump seals are affected by high Drywell temperature and temperature damage to Recirc pump seals is a concern addressed in PRA.

Technical Reference(s): N1-EOP-4, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2015 Cert #45

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

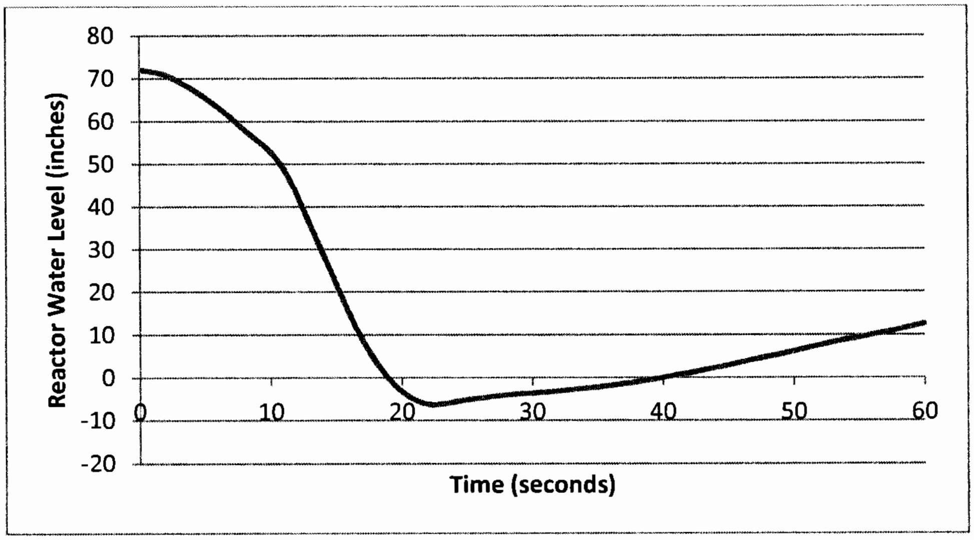
Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK2.09
	Importance Rating	3.3

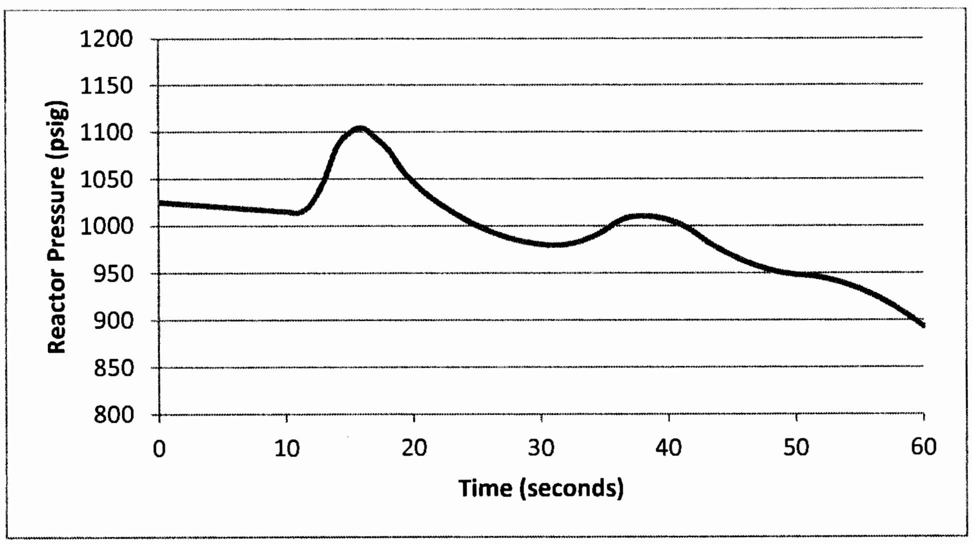
Reactor Low Water Level

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Recirculation system: Plant-Specific

Proposed Question: #42

The plant is operating at 50% power when a transient results in the following Reactor water level and pressure response:





Which one of the following describes the response of the Reactor Recirculation pumps?

Reactor Recirculation pumps…

A. receive a trip signal due to both Reactor water level and Reactor pressure.

B. receive a trip signal due to Reactor water level, only.

C. receive a trip signal due to Reactor pressure, only.

D. continue to operate through the transient.

Proposed Answer: B

Explanation: Multiple automatic actions occur as Reactor water level lowers. At 53" (Time = 10 seconds), an automatic Reactor scram occurs. At 5" (Time = 18 seconds), Core Spray logic is satisfied, which trips the Reactor Recirculation pumps (RRPs). Once Reactor water level has been <5" for 9 seconds (Time = 27 seconds), another RRP trip signal is received from the ATWS/RPT logic. Reactor pressure goes above 1080 psig, which causes a Reactor scram signal and EC initiation. Reactor pressure goes above 1100 psig, which causes ERVs to lift. However, Reactor pressure remains below the 1135 psig value that would provide a Recirc pump trip signal.

A. Plausible – Reactor pressure remains below the 1135 psig value that would provide a Recirc pump trip signal. Plausible because Reactor pressure does go above other thresholds for automatic actions.
C. Plausible – Reactor pressure remains below the 1135 psig value that would provide a Recirc pump trip signal. Plausible because Reactor pressure does go above other thresholds for automatic actions. Reactor water level goes below +5", which results in a Recirc pump trip signal. Plausible because Reactor water level stays above the -10" threshold which is used in ADS logic.
D. Plausible – Reactor water level goes below +5", which results in a Recirc pump trip signal. Plausible because Reactor water level stays above the -10" threshold which is used in ADS logic.

Technical Reference(s): N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-5

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 #	1
	Group #	1
	K/A #	295019 AK2.11
	Importance Rating	2.5

Partial or Complete Loss of Instrument Air

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

Proposed Question: #43

A plant startup is in progress with the following conditions:

- Reactor pressure is 300 psig.
- Main condenser vacuum is 20" Hg.
- Reactor Water Cleanup (RWCU) reject flow is established to Radwaste.
- RWCU reject flow rate is 7×10^4 lbm/hr.

Then, a complete loss of Instrument Air occurs.

Which one of the following describes the effect on RWCU reject flow rate to Radwaste?

RWCU reject flow to Radwaste…

A. rises.

B. lowers to zero.

C. remains the same.

D. lowers, but remains greater than zero.

Proposed Answer: B

Explanation: RWCU reject flow to Radwaste is established by opening 33-10, CLEANUP TO WASTE DISPOSAL BV, and then 33-165, CLEANUP TO COND & WASTE FLOW. Both of these valves are remotely operated from the Control Room. 33-10 is an AC motor-operated valve and 33-165 is an air-operated valve. 33-165 fails closed on a loss of Instrument Air. This causes RWCU reject flow to Radwaste to lower to zero.

A. Plausible – RWCU reject flow to Radwaste lowers to zero, not rises. Plausible because this could be the response if 33-165 failed open, not closed.
C. Plausible – RWCU reject flow to Radwaste lowers to zero, not remains the same. Plausible because this could be the response if 33-165 failed as-is, not closed.
D. Plausible – RWCU reject flow to Radwaste lowers completely to zero. Plausible because this could be the response if 33-10 and 33-11, CLEANUP TO CONDENSER BV, were the air-operated valves in the reject line and 33-11 failed open, or if 33-165 had a minimum pinned position for loss of IA, as some other air-operated valves do.

Technical Reference(s): N1-SOP-20.1, N1-OP-3, C-18009-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-204000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK2.01
	Importance Rating	4.4

Control Room Abandonment

Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Remote shutdown panel: Plant-Specific

Proposed Question: #44

The plant was operating at 100% power when a significant fire in the Control Room resulted in the following:

- A Control Room evacuation has been performed.
- All immediate actions of N1-SOP-21.2, Control Room Evacuation, have been performed.
- Operators are establishing a Reactor cool down from outside the Control Room.

Which one of the following describes the preferred system to be used to cool down the Reactor, in accordance with N1-SOP-21.2?

Cool down the Reactor by operating...

A. ERVs from the Remote Shutdown Panels.

B. ERVs from Reactor Building elevation 237'.

C. Emergency Condensers from the Remote Shutdown Panels.

D. Emergency Condensers from Reactor Building elevation 281'.

Proposed Answer: C

Explanation: N1-SOP-21.2 directs performing a Reactor cool down using Emergency Condensers from the Remote Shutdown Panels.

A. Plausible – ERVs may automatically cycle while operators are establishing control of Reactor pressure, but they are not the system used to conduct the Reactor cooldown.
B. Plausible – ERVs may automatically cycle while operators are establishing control of Reactor pressure, but they are not the system used to conduct the Reactor cooldown. N1-SOP-21.2 contains actions on Reactor Building elevation 237' to pull ERV fuses within 1 hour of event start.
D. Plausible – Emergency Condensers are the correct system, but they are controlled from the Remote Shutdown Panels. There is a procedure section in N1-OP-13 to allow manually initiating Emergency Condensers from Reactor Building elevation 281'.

Technical Reference(s): N1-SOP-21.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.2C01 EO-2

Question Source: Modified Bank – 2009 Cert #78

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 #	295004 AK3.01
	Importance Rating	2.6

Partial or Complete Loss of DC Power

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Load shedding: Plant-Specific

Proposed Question: #45

Which one of the following describes the reason for performing Battery load reductions during a Station Blackout?

A. Support a manual dead bus transfer to MG Set 167.

B. Maintain power to Reactor instrumentation, EC controls, and to start an EDG.

C. Avoid a loss of critical Battery Board loads due to breaker trips on over-current.

D. Maintain power to the Process Computer and Annunciators for the entire coping period.

Proposed Answer: B

Explanation: During a Station Blackout, Static Battery Chargers are lost and Batteries begin to discharge. Load reductions are performed to preserve Battery capacity, specifically for Reactor instrumentation, Emergency Condenser control, and EDG starting.

A. Plausible – MG Set 167 does operate from DC power during a Station Blackout, but is not transferred to any other buses. N1-SOP-33A.2 requires load shedding MG Set 167 within 2 hours of event start.
C. Plausible – Breakers on the SR Battery Boards have been bypassed. Fuses have been installed in their place in the back of the Battery Boards. Both the Negative and Positive legs to each load are fused. Plausible if believe lowering voltage will draw more current and cause trips.
D. Plausible – PPC and annunciators are de-energized when MG 167 is tripped 2 hours into the 4 hour coping period. Plausible because these loads are maintained for a longer period of time than most other loads.

Technical Reference(s): N1-OP-47A, N1-SOP-33A.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-10

Question Source: Bank – 2009 NRC #3

Question History: 2009 NRC #3

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 #	295021 AK3.04
	Importance Rating	3.3

Loss of Shutdown Cooling

Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Maximizing reactor water cleanup flow

Proposed Question: #46

The plant is shutdown with the following:

- The 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 (RWCU) is currently in service.
- Reactor coolant temperature is 125°F and rising slowly.

Which one of the following describes the required control of RWCU and the associated reason, in accordance with N1-SOP-6.1, Loss of SFP / Rx Cavity Level / Decay Heat Removal?

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

B. Isolate RWCU in anticipation of automatic isolation on high filter inlet temperature.

C. Maximize RWCU non-regenerative heat exchanger cooling water flow to assist in decay heat removal.

D. Maximize RWCU non-regenerative heat exchanger cooling water flow to prevent automatic isolation on high filter inlet temperature.

Proposed Answer: C

Explanation: If SDC cannot be restored for decay heat removal, N1-SOP-6.1 requires opening valve 70-85, which maximizes RBCLC cooling water flow to the RWCU non-regenerative heat exchanger to maximize the amount of decay heat removal that the RWCU system is supplying.

A. Plausible – RWCU is left in service with cooling water to the non-regenerative heat exchanger maximized. With cooling water maximized, RWCU has a net effect of removing heat from the Reactor coolant, even given the heat input from the RWCU pumps.
B. Plausible – RWCU is left in service with cooling water to the non-regenerative heat exchanger maximized. With normal cooling water still lined up to RWCU, high filter inlet temperature is not an issue even with SDC unavailable.
D. Plausible – The reason for maximizing cooling water flow is to assist in decay heat removal. Reactor coolant temperature is approaching the filter inlet temperature that would result in isolation. However, even without raising cooling water flow, normal RWCU heat removal mechanisms prevent automatic isolation due to high filter inlet temperature, even with SDC unavailable.

Technical Reference(s): N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP6.1C01 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 #	295005 AK3.06
	Importance Rating	3.3

Main Turbine Generator Trip

Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: Realignment of electrical distribution

Proposed Question: #47

The plant is operating at 25% power during a startup with the following:

- Powerboards 11 and 12 are energized from House Service Transformer T-10.
- Main Turbine vibrations are rising.
- An Operator manually trips the Main Turbine.

Which one of the following describes the response of Powerboards 11 and 12?

A. Fast transfer based on undervoltage signal.

B. Fast transfer based on Turbine/Generator trip signal.

C. Slow transfer based on undervoltage signal.

D. Slow transfer based on Turbine/Generator trip signal.

Proposed Answer: B

Explanation: At 25% power, a Main Turbine trip does NOT result in a Reactor scram, however it still does result in a Generator trip. Due to the Turbine and Generator trips, Powerboard 11 and 12 fast transfer is initiated. Powerboard 11 and 12 also have a slow transfer mechanism in the event of degraded voltage on the board. The fast transfer mechanism is preferable when normal voltage is present so that power is not interrupted to loads. The slow transfer mechanism is only necessary with degraded voltage conditions to ensure Powerboard voltage decays before connecting to the Reserve power source. Nothing in the stem of the question indicates degraded voltage, therefore the fast transfer will occur.

A. Plausible – Undervoltage will NOT occur since the Generator trip will directly cause fast transfer, which happens quick enough to preserve Powerboard voltage. Undervoltage is plausible if applicant believed only automatic Turbine trip signal led directly to fast transfer.
C. Plausible – Slow transfer will NOT occur since no degraded voltage conditions are present. Undervoltage will NOT occur since the Generator trip will directly cause fast transfer, which happens quick enough to preserve Powerboard voltage. Slow transfer and undervoltage are plausible if applicant believed only automatic Turbine trip signal led directly to fast transfer.
D. Plausible – Slow transfer will NOT occur since no degraded voltage conditions are present. Slow transfer is plausible if applicant believed only automatic Turbine trip signal led directly to fast transfer.

Technical Reference(s): N1-OP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-05
N1-245001-RBO-05

Question Source: Bank – 2013 NRC #4

Question History: 2013 NRC #4

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 #	700000 AA1.01
	Importance Rating	3.6

Generator Voltage and Electric Grid Disturbances

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

Proposed Question: #48

The plant is operating at 100% power when a grid disturbance causes the following:

- Annunciator A8-1-3, 115 KV BUS LOW VOLTAGE, alarms.
- Computer points F432, F433, F434 indicate 115 KV Bus phase voltages are 114.5 KV and slowly lowering.
- Annunciator A6-3-3, 345 KV SYS FREQUENCY HIGH-LOW, alarms.
- Grid frequency is 59.4 Hz and slowly lowering.
- The crew enters N1-SOP-33A.3, Major 115 KV Grid Disturbances, and N1-SOP-33B.1, Major 345 KV Grid Disturbances.

Which one of the following identifies the parameter threshold that, once dropped below, requires an immediate Reactor scram, in accordance with N1-SOP-33A.3 and/or N1-SOP-33B.1?

A. 115 KV Bus voltage of 114 KV

B. 115 KV Bus voltage of 109.3 KV

C. Grid frequency of 58.6 Hz

D. Grid frequency of 58.1 Hz

Proposed Answer: D

Explanation: N1-SOP-33B.1 requires an immediate Reactor scram if frequency variation is greater than ±1.9 Hz. Normal Generator/grid frequency is 60 Hz, therefore the threshold requiring an immediate Reactor scram is 58.1 Hz.

A. Plausible – N1-SOP-33A.3 requires LCO entry and possible tap changer manipulations if 115 KV Bus voltage reaches 114 KV, but does not require an immediate Reactor scram.
B. Plausible – N1-SOP-33A.3 requires declaring HPCI inoperable if 115 KV Bus voltage reaches 109.3 KV, but does not require an immediate Reactor scram.
C. Plausible – If Generator/grid frequency varies by ±1.4 Hz, then N1-SOP-33B.1 requires entering a 12 minute operating time clock. Given a normal frequency of 60 Hz, this occurs at 58.6 Hz. However, a Reactor scram is only required if the 12 minute clock expires without restoration of frequency, not an immediate Reactor scram.

Technical Reference(s): N1-SOP-33B.1, N1-SOP-33A.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33B.1C01 EO-2

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 #	1
	Group #	1
	K/A #	295006 AA1.02
	Importance Rating	3.9

SCRAM

Ability to operate and/or monitor the following as they apply to SCRAM: Reactor water level control system

Proposed Question: #49

The plant is operating at 80% power with the following conditions:

- Feedwater pump 13 is operating.
- Operators insert a manual Reactor scram.
- All control rods insert.
- Reactor water level is 86" and rising slowly.
- The Unit Supervisor orders entry into N1-SOP-1, Reactor Scram.

Which one of the following actions is required, in accordance with N1-SOP-1?

A. Secure all Feedwater pumps.

B. Secure Feedwater pump 13, only.

C. Close the Main Steam Isolation Valves.

D. Close Feedwater Isolation Valves 11 and 12.

Proposed Answer: A

Explanation: N1-SOP-1 contains an override that must be executed if Reactor water level exceeds 85" following a scram. This override requires securing all Feedwater pumps, not just the higher capacity Feedwater pump 13.

B. Plausible – The override of N1-SOP-1 requires securing all Feedwater pumps with Reactor water level above 85", not just the higher capacity Feedwater pump 13. Feedwater pump 13 would be the only Feedwater pump required to be secured if Reactor water level were less than 85".
C. Plausible – The override of N1-SOP-1 allows closure of the MSIVs, but does not require it. Closure of the MSIVs would only be necessary if Reactor water level was approaching the elevation of the Main Steam Lines. The bottom of the Main Steam Line nozzles is at least 3.5 feet higher than current water level.
D. Plausible – The override of N1-SOP-1 allows closure of the FWIVs, but does not require it. Closure of the FWIVs would only be necessary if earlier actions to terminate FW injection were unsuccessful (ex. leakby on FCVs, failure of FWP to trip).

Technical Reference(s): N1-SOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1C01 EO-2

Question Source: Bank – 2010 Cert #19

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 #	295030 EA1.01
	Importance Rating	3.6

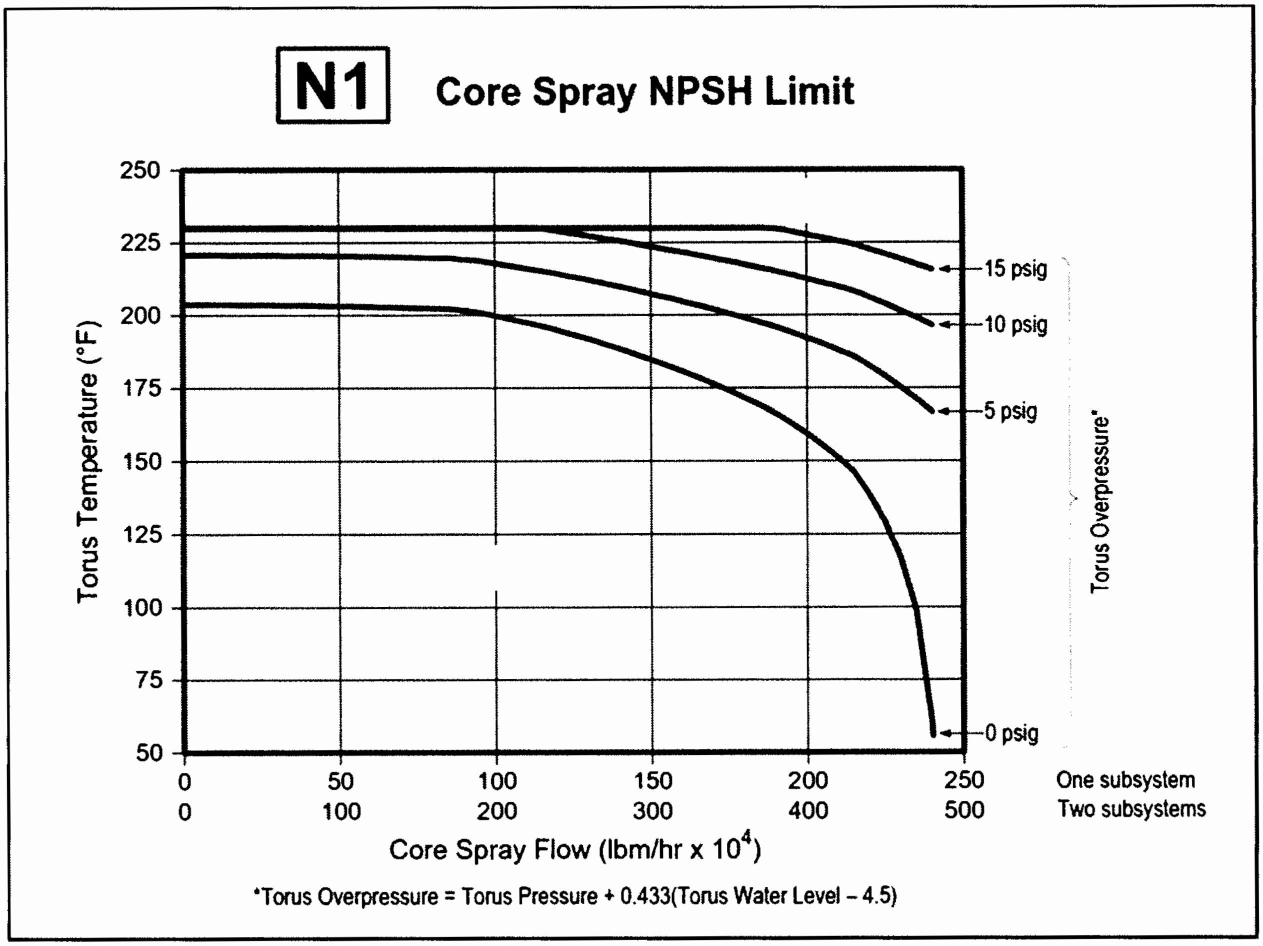
Low Suppression Pool Water Level

Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: ECCS systems (NPSH considerations): Plant-Specific

Proposed Question: #50

A loss of coolant accident has resulted in the following:

- The Reactor has been depressurized using Emergency Condensers and ERVs.
- Core Spray is injecting and maintaining Reactor water level.
- Containment Sprays have been utilized to lower Containment pressure.
- Torus water temperature is 175°F and stable.
- Torus water level is 8.5 feet and stable.
- Torus pressure is 2 psig and slowly rising.
- Drywell pressure is 3 psig and slowly rising.



Which one of the following states the maximum Core Spray flow (lbm/hr x 10^4) that may be used for Reactor injection while maintaining Core Spray within the NPSH limit?

A. 170

B. 230

C. 340

D. 460

Proposed Answer: C

Explanation: The given combination of Torus pressure and Torus water level result in a Torus overpressure of 3.7 psig (2 psig + 0.433(8.5-4.5)). Since this is less than 5 psig, the 0 psig curve must be used on the Core Spray NPSH Limit. With a Torus temperature of 175°F and all Core Spray subsystems available, this allows a maximum Core Spray flow of 340 lbm/hr x10^4.

A. Plausible – The given conditions allow a maximum Core Spray flow of 340 lbm/hr x10^4. 170 lbm/hr x10^4 would be the limit if only one Core Spray subsystem was available.
B. Plausible – The given conditions allow a maximum Core Spray flow of 340 lbm/hr x10^4. 230 lbm/hr x10^4 would be the limit if only one Core Spray subsystem was available and the 5 psig overpressure curve were applicable.
D. Plausible – The given conditions allow a maximum Core Spray flow of 340 lbm/hr x10^4. 460 lbm/hr x10^4 would be the limit if the 5 psig overpressure curve were applicable.

Technical Reference(s): N1-EOP-2, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Modified Bank - 2009 NRC #13

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 #	295025 EA2.02
	Importance Rating	4.2

High Reactor Pressure

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor power

Proposed Question: #51

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

- A spurious Main Generator trip occurs.
- Multiple control rods fail to fully insert.
- Turbine Bypass Valves are slow to operate.
- The Reactor power response causes a peak pressure of 1310 psig during the transient.

Which one of the following describes the impact of the Reactor power and pressure response?

A. All of the Electromatic Relief Valves (ERVs) open, but NONE of the Safety Valves open.

B. Some of the Safety Valves open, but NOT all of them.

C. All of the Safety Valves open, but the Reactor pressure Safety Limit is NOT violated.

D. All of the Safety Valves open and the Reactor pressure Safety Limit is violated.

Proposed Answer: C

Explanation: The ERVs all open, with setpoints ranging from 1090 to 1100 psig. The Safety Valves all open, with setpoints ranging from 1218 to 1254 psig. The Reactor pressure Safety Limit of 1375 psig is NOT violated.

Note: The question meets the K/A by providing a situation where a failure to scram results in high Reactor power, which causes a much higher than normal Reactor pressure spike, and requiring the candidates to interpret how this affects ERVs, safety valves, and the status of a safety limit.

A. Plausible – The Safety Valves all open, with setpoints ranging from 1218 to 1254 psig. This would be correct if peak Reactor pressure was <1218 psig.
B. Plausible – The Safety Valves all open, with setpoints ranging from 1218 to 1254 psig. This would be correct if peak Reactor pressure was ≥1218 psig but <1254 psig.
D. Plausible – The Reactor pressure Safety Limit of 1375 psig is NOT violated. This would be correct if peak Reactor pressure was >1375 psig.

Technical Reference(s): N1-OP-1, Technical Specification 2.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-101001-RBO-14

Question Source: Modified Bank – 2009 NRC #43

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AA2.04
	Importance Rating	3.4

Refueling Accidents

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Occurrence of fuel handling accident

Proposed Question: #52

A refueling outage is in progress with the following:

- Core shuffle phase II is in progress.
- A one cell section of the core is displayed on the following page.

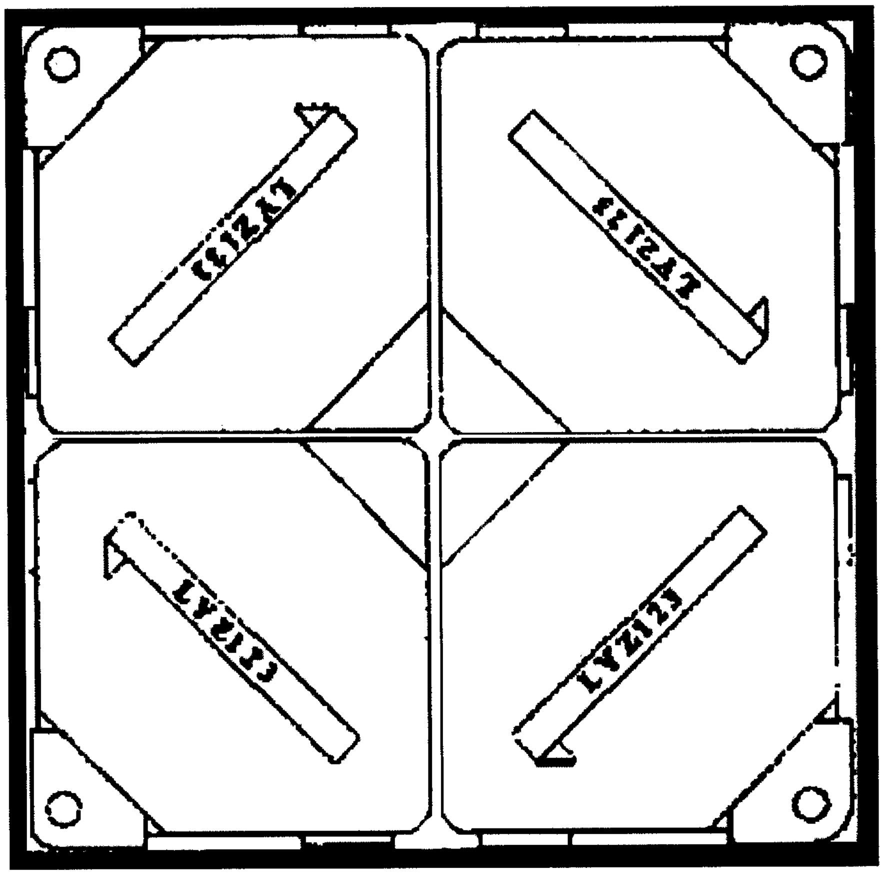
Which one of the following describes the status of this portion of the core?

A. The fuel is loaded correctly.

B. All fuel bundles must be re-positioned 90° clockwise.

C. All fuel bundles must be re-positioned 90° counterclockwise.

D. All fuel bundles must be re-positioned 180°.



Proposed Answer: D

Explanation: All four fuel bundles are oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell. To correct this issue, all fuel bundles need to be re-positioned 180°.

<u>Note:</u> The question fits evolution 295023, Refueling Accident, because UFSAR Chapter 15 (Safety Analysis) C.6.0 analyzes mis-oriented fuel bundle in core as one of the postulated refueling accidents.

A. Plausible – All four fuel bundles are oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell.
B. Plausible – All fuel bundles need to be re-positioned a full 180° to be correctly oriented.
C. Plausible – All fuel bundles need to be re-positioned a full 180° to be correctly oriented.

Technical Reference(s): N1-REP-17 Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: S-101002-RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EA2.01
	Importance Rating	4.1

Suppression Pool High Water Temperature

Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature

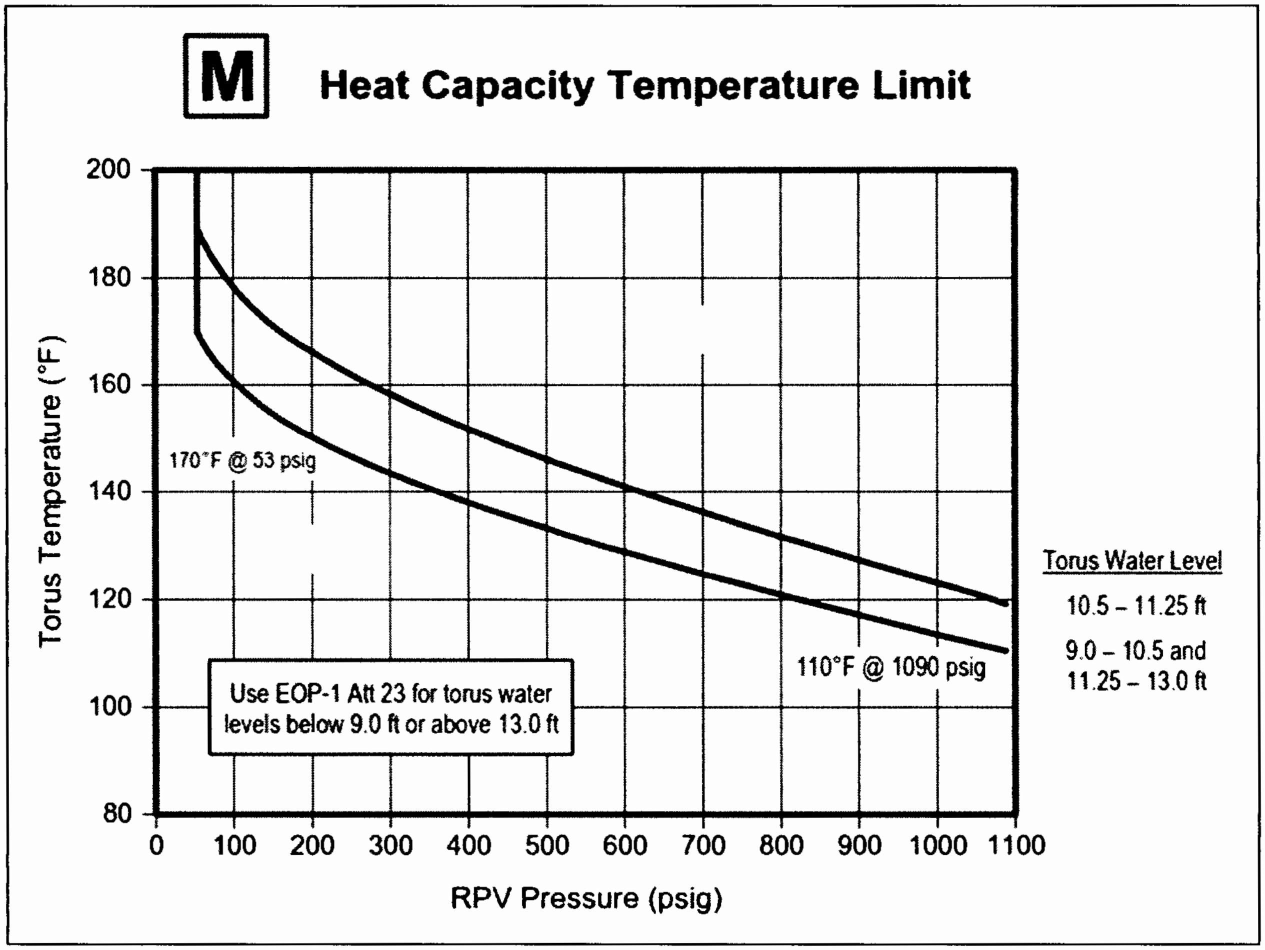
Proposed Question: #53

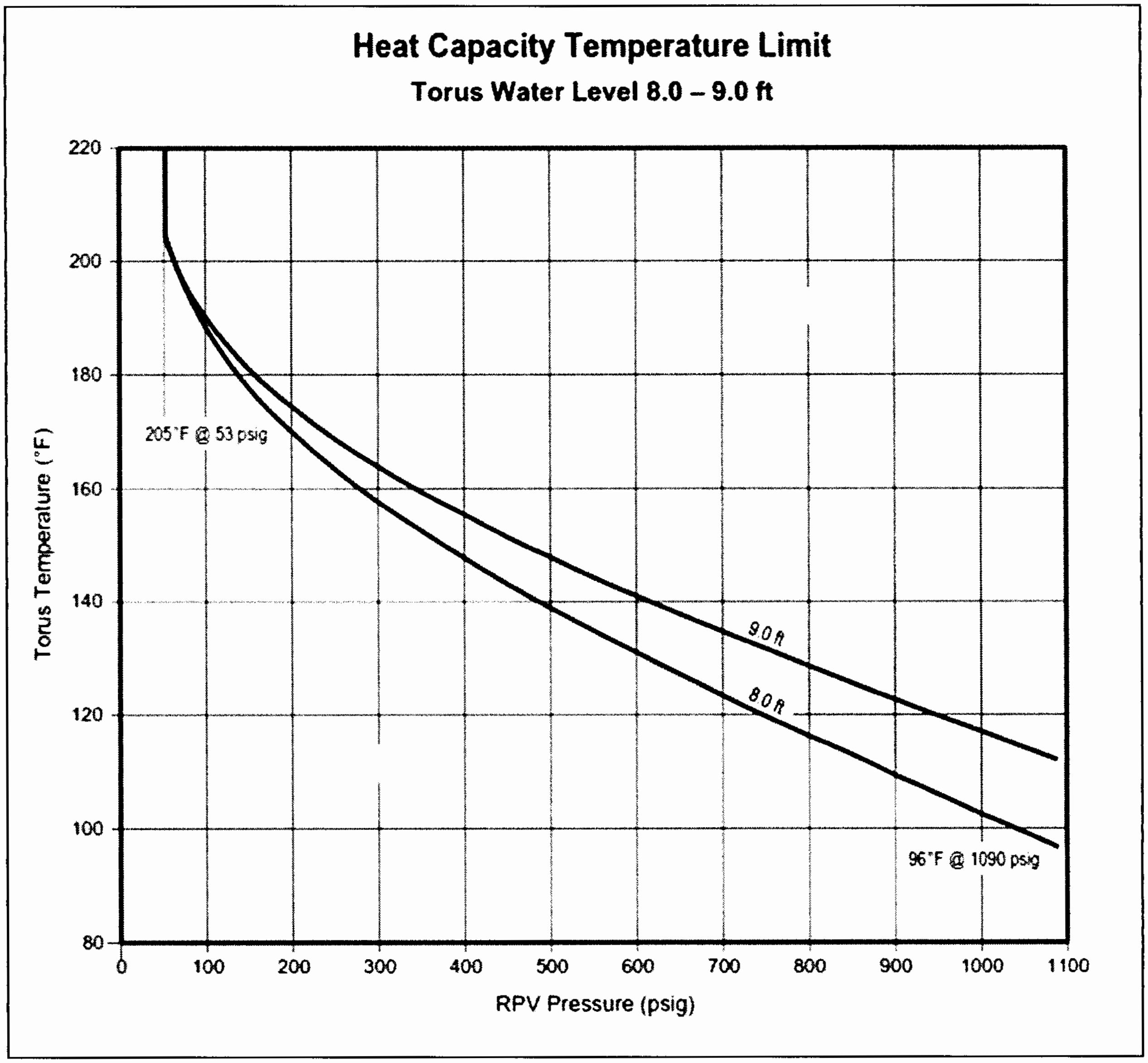
The plant has experienced a failure to scram with an initial power of 4%.

Note: The Heat Capacity Temperature Limit curves are provided on the following pages.

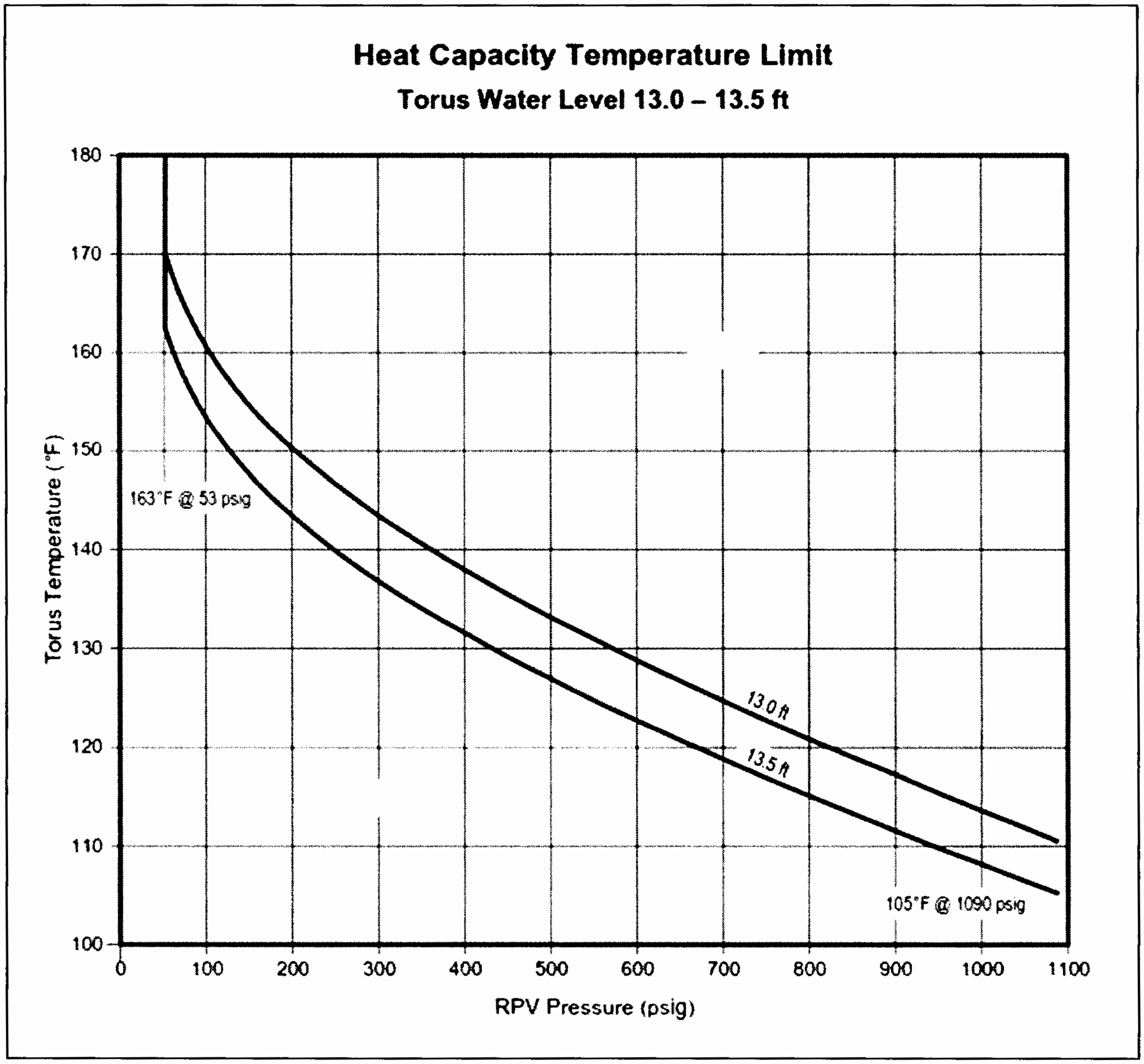
Which one of the following sets of conditions would require an RPV Blowdown due to exceeding the Heat Capacity Temperature Limit, in accordance with N1-EOP-4, Primary Containment Control?

A.	Reactor pressure	250 psig
	Torus temperature	158°F
	Torus water level	11.2'
B.	Reactor pressure	750 psig
	Torus temperature	112°F
	Torus water level	13.2'
C.	Reactor pressure	850 psig
	Torus temperature	125°F
	Torus water level	8.7'
D.	Reactor pressure	450 psig
	Torus temperature	132°F
	Torus water level	12.1'


M
Heat Capacity Temperature Limit
Torus Temperature (°F)
200
180
160
140
120
100
80
0
100
200
300
400
500
600
700
800
900
1000
1100
RPV Pressure (psig)
170°F @ 53 psig
110°F @ 1090 psig
Use EOP-1 Att 23 for torus water levels below 9.0 ft or above 13.0 ft
Torus Water Level
10.5 – 11.25 ft
9.0 – 10.5 and 11.25 – 13.0 ft



Heat Capacity Temperature Limit
Torus Water Level 8.0 – 9.0 ft
Torus Temperature (°F)
RPV Pressure (psig)
220
200
180
160
140
120
100
80
0
100
200
300
400
500
600
700
800
900
1000
1100
205°F @ 53 psig
9.0 ft
8.0 ft
96°F @ 1090 psig


Heat Capacity Temperature Limit
Torus Water Level 13.0 – 13.5 ft
Torus Temperature (°F)
RPV Pressure (psig)
180
170
160
150
140
130
120
110
100
0
100
200
300
400
500
600
700
800
900
1000
1100
163°F @ 53 psig
13.0 ft
13.5 ft
105°F @ 1090 psig

Proposed Answer: C

Explanation: This combination of Reactor pressure and Torus temperature is above the required Torus water level line (8.0'), therefore HCTL is violated and an RPV Blowdown is required.

Note: This question tests more than the applicants' ability to plot points on a graph because they must also select the applicable Torus water level limit curve for each data point (with 3 possible curves available) and interpret the acceptability of each data point versus the applicable limit (curves lack labeling as to which region is acceptable / not acceptable).

A. Plausible – This combination of Reactor pressure and Torus temperature is below the required Torus water level line (10.5-11.25').
B. Plausible – This combination of Reactor pressure and Torus temperature is below the required Torus water level line (13.5').
D. Plausible – This combination of Reactor pressure and Torus temperature is below the required Torus water level line (11.25-13.0').

Technical Reference(s): N1-EOP-4, N1-EOP-1 Attachment 23

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – SSES LOC27 NRC #63

Question History: SSES LOC27 NRC #63

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 #	295001 2.2.40
	Importance Rating	3.4

Partial or Complete Loss of Forced Core Flow Circulation

Ability to apply technical specifications for a system.

Proposed Question: #54

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

- Reactor Recirc pump (RRP) 11 trips.
- N1-SOP-1.3, Recirc Pump Trip at Power, is entered.
- Reactor power is 85% and stable.

Which one of the following describes the required notification to the Shift Manager, in accordance with N1-SOP-1.3 and Technical Specifications?

Notify the Shift Manager that...

A. APRMs are inoperable until the RRP 11 discharge valve is closed.

B. APRMs are inoperable until the RRP 11 control switch is green flagged.

C. the Partial Loop Operation power level is exceeded until the RRP 11 discharge valve is closed.

D. the Partial Loop Operation power level is exceeded until the RRP 11 control switch is green flagged.

Proposed Answer: A

Explanation: When RRP 11 trips, reverse flow occurs through the associated loop. This causes erroneous core flow indication, which causes the APRM flow biased scram setpoints to be non-conservative. There is no feature that automatically adjusts for this reverse flow in the APRM flow biased scram circuitry. N1-SOP-1.3 requires notifying the Shift Manager that APRMs are inoperable due to this condition. APRMs become operable again once the RRP 11 discharge valve is closed because the reverse flow condition is corrected.

B. Plausible – APRMs become operable again once the RRP 11 discharge valve is closed because the reverse flow condition is corrected. Plausible because N1-SOP-1.3 does have the Operator green flag the RRP 11 control switch, which could be used as a signal to correct for the reverse flow condition in the APRM flow biased scram circuitry.
C. Plausible – Technical Specification 3.1.7 contains restrictions on Partial Loop Operation power level, however with Reactor power at 85%, none of these restrictions are being violated. N1-SOP-1.3 does require closing the RRP 11 discharge valve, and this does affect the restrictions in Technical Specification 3.1.7.
D. Plausible – Technical Specification 3.1.7 contains restrictions on Partial Loop Operation power level, however with Reactor power at 85%, none of these restrictions are being violated. N1-SOP-1.3 does require green flagging the RRP 11 control switch and RRP 11 breaker position does affect the restrictions in Technical Specification 3.1.7.

Technical Reference(s): N1-SOP-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1.3C01 EO-2

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 #	1
	Group #	1
	K/A #	295018 2.2.44
	Importance Rating	4.2

Partial or Complete Loss of CCW

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: #55

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

- The RBCLC temperature controller, 70-23B, has malfunctioned and is providing a constant 50% demand signal while in automatic.
- An Operator has placed the controller in manual and toggled the readout to display "70-23B.V".
- The Operator now rotates the setpoint adjustment knob fully counterclockwise (↺).

Which one of the following describes the response of RBCLC flow rate through the RBCLC heat exchangers?

RBCLC flow rate through the RBCLC heat exchangers…

A. rises.

B. lowers to zero.

C. lowers, but does NOT go to zero.

D. does NOT change because the controller readout is set to the wrong mode.

Proposed Answer: C

Explanation: The RBCLC temperature controller has multiple operating modes. In some modes, such as "70-23B.P", the controller provides indication only and rotating the control knob will not have any effect on temperature control valve position. "70-23B.V" is the correct mode to allow the control knob to change temperature control valve position in manual. Rotating the knob counterclockwise causes the temperature control valve to close. Normally, going fully counterclockwise would provide a full close signal and cause flow rate to drop to zero. However, the RBCLC temperature control valve is provided with a mechanical stop which prevents the valve from going full closed in order to ensure minimum cooling during an accident.

A. Plausible – Taking the knob counterclockwise lowers flow. Clockwise operation would cause flow to rise in this situation.
B. Plausible – A mechanical stop prevents the RBCLC temperature control valve from going full closed, therefore flow rate does NOT drop to zero. This would be correct if this mechanical stop were not part of the temperature control valve design.
D. Plausible – "70-23B.V" is a correct display setting to enable a manual change to the RBCLC temperature control valve demand signal. Other display settings, such as "70-23B.P", would result in no change to RBCLC flow rate.

Technical Reference(s): N1-OP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-5

Question Source: Bank - 2013 Cert #53

Question History:

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 #	295003 2.2.36
	Importance Rating	3.1

Partial or Complete Loss of AC

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

Proposed Question: #56

Given the following separate events:

(1) Circulating Water pump 11 is tagged out for breaker inspection.
(2) 115 KV Line 4 is removed from service for maintenance.

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

A. Neither event

B. Event (1) only

C. Event (2) only

D. Events (1) and (2)

Proposed Answer: C

Explanation: Technical Specification 3.6.3 requires LCO entry for removing Line 4 from service. Circulating Water pumps are not required by any Technical Specification.

A. Plausible – Technical Specification 3.6.3 requires LCO entry for removing Line 4 from service. Plausible because 3 other emergency power sources (EDG 102, EDG 103, Line 1) remain operable and the loss of Line 4 is planned.
B. Plausible – Technical Specification 3.6.3 requires LCO entry for removing Line 4 from service. Plausible because 3 other emergency power sources (EDG 102, EDG 103, Line 1) remain operable and the loss of Line 4 is planned. Circulating Water pumps are not required by any Technical Specification. Plausible because Circulating Water pumps are a major piece of equipment important for plant operation and other cooling water pumps (ESW, Containment Spray Raw Water) would require LCO entry.
D. Plausible – Circulating Water pumps are not required by any Technical Specification. Plausible because Circulating Water pumps are a major piece of equipment important for plant operation and other cooling water pumps (ESW, Containment Spray Raw Water) would require LCO entry.

Technical Reference(s): Technical Specifications

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-14

Question Source: Bank – 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 #	1
	Group #	1
	K/A #	295024 EA1.11
	Importance Rating	4.2

High Drywell Pressure

Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Drywell spray: Mark-I&II

Proposed Question: #57

The plant is operating at 25% power with the following:

- ERV 112 inadvertently opened.
- The crew closed ERV 112 in accordance with N1-SOP-1.4, Stuck Open ERV.
- Containment Spray loop 111 is in the Torus Cooling lineup in accordance with N1-EOP-1, Attachment 16, Torus Cooling.
- Torus water temperature is 87°F and slowly lowering.

Then, a small loss of coolant accident results in the following:

- The Reactor is manually scrammed.
- The Control Room Supervisor re-enters N1-EOP-4, Primary Containment Control.
- Drywell pressure is 3.7 psig and slowly rising.
- Torus pressure is 2.0 psig and slowly rising.
- Drywell average temperature is 155°F and slowly rising.
- Reactor water level is 50" and slowly lowering.

Which one of the following describes the required control of Containment Spray, in accordance with N1-EOP-4?

A. Verify all four Containment Spray pump control switches in Pull-to-Lock.

B. Verify only Containment Spray pump 112, 121, and 122 control switches in Pull-to-Lock.

C. Start at least one additional Containment Spray pump and close 80-118, CONT SPRAY TORUS TEST TO TORUS FCV.

D. Start at least two additional Containment Spray pumps and leave 80-118, CONT SPRAY TORUS TEST TO TORUS FCV, open.

Proposed Answer: B

Explanation: Containment Spray has not automatically initiated because Reactor water level has not lowered below -10" with Drywell pressure greater than 3.5 psig. N1-EOP-4 does not currently require Containment Spray operation because Torus pressure has not reached 13 psig and Drywell average temperature is well below 300°F. Therefore, N1-EOP-4 requires locking out all Containment Spray pumps not being used for Torus Cooling. This requires placing Containment Spray pump 112, 121, and 122 control switches in Pull-to-Lock. Containment Spray loop 111 is maintained in the Torus Cooling lineup.

A. Plausible – Since Containment Spray loop 111 is initially in the Torus Cooling lineup, Containment Spray pump 111 control switch is not placed in Pull-to-Lock.
C. Plausible – Containment parameters do not yet require initiation of Containment Spray. With Containment Spray loop 111 in Torus Cooling, only one additional Containment Spray pump would need to be started for initiate Containment Spray and 80-118 would be closed.
D. Plausible – Containment parameters do not yet require initiation of Containment Spray. Two Containment Spray pumps are normally started upon manual initiation of Containment Spray.

Technical Reference(s): N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-1

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 #	295037 EK1.07
	Importance Rating	3.4

SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Shutdown margin

Proposed Question: #58

A failure to scram has occurred with the following:

- N1-EOP-3, Failure to Scram, is being executed.
- Reactor power is 10%.
- Control rods are being manually inserted.
- Boron is being injected.

Which one of the following conditions provides indication of sufficient negative reactivity to permit exiting N1-EOP-3, Failure to Scram, and entering N1-EOP-2, RPV Control?

A. All IRMs are fully inserted and indicating below Range 7.

B. All SRMs are fully inserted and indicating below 100 cps.

C. All control rods are fully inserted except 24 control rods at position 04.

D. All control rods are fully inserted except four control rods at position 12.

Proposed Answer: C

Explanation: Position 04 is the Maximum Subcritical Banked Withdrawal Position. It is the greatest rod position at which multiple rods can be withdrawn and still ensure the Reactor will stay shutdown under all conditions without boron. This allows exiting N1-EOP-3 and entering N1-EOP-2.

A. Plausible – All IRMs fully inserted and indicating below Range 7 indicates the Reactor is currently shutdown and is a threshold allowing Reactor cooldown in N1-EOP-3 if no boron has been injected. However, it does not allow exiting N1-EOP-3.
B. Plausible – All SRMs fully inserted and indicating below 100 cps indicates the Reactor is currently shutdown. However, this does not allow exiting N1-EOP-3.
D. Plausible – Four control rods at position 12 does not allow exiting N1-EOP-3 because it exceeds the Maximum Subcritical Banked Withdrawal Position and the Technical Specification shutdown margin assumptions (one rod at 48). Even though it is an identical number of withdrawn notches to the Technical Specification shutdown margin assumptions, it is not enough to exit N1-EOP-3.

Technical Reference(s): N1-EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO-2

Question Source: Modified Bank – 2008 Cert #59

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 #	2
	K/A #	295036 EK1.02
	Importance Rating	2.6

Secondary Containment High Sump/Area Water Level

Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Electrical ground/ circuit malfunction

Proposed Question: #59

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

- A seismic event occurs.
- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, alarms.
- Computer point B128, RBFDT 11 (NW) LVL, is in alarm HIGH.
- Computer point B129, RBFDT 13 (SW) LVL, is in alarm HIGH.
- Computer point F189, NW RB CORNER RM WTR LVL, is in alarm HIGH.

Note: A portion of ARP H2-2-1 is provided on the following page.

Which one of the following describes the ability to determine which areas have reached or exceeded the Maximum Safe Area Water Level from the given alarms, in accordance with N1-EOP-5, Secondary Containment Control?

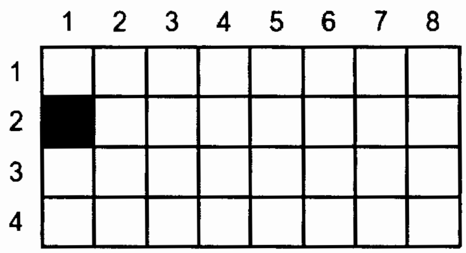
A. NO area has reached or exceeded the Maximum Safe Area Water Level.

B. One area has reached or exceeded the Maximum Safe Area Water Level, only.

C. Two areas have reached or exceeded the Maximum Safe Area Water Level.

D. It CANNOT be determined from this information if any area has reached or exceeded the Maximum Safe Area Water Level.

PANEL: H2	ANNUNCIATOR: 2-1



R BLDG FL DR
SUMPS 11-16
AREA WTR LVL
LEVEL HIGH

Computer Printout:			Device – Setpoint:	
B126	RBFDT 15 (NW) LVL	HIGH	LSE-104-09A	Elev 197'7"
B127	RBFDT 16 (NE) LVL	HIGH	LSE-104-10A	Elev 197'7"
B128	RBFDT 11 (NW) LVL	HIGH	LS-104-11A	Elev 197'11"
B129	RBFDT 13 (SW) LVL	HIGH	LS-104-12A	Elev 197'11"
B130	RBFDT 12 (NE) LVL	HIGH	LS-104-13A	Elev 197'11"
B131	RBFDT 14 (SE) LVL	HIGH	LS-104-14A	Elev 197'11"
F188	NE RB CORNER RM WTR LVL	HIGH		Elev 203'0"
F189	NW RB CORNER RM WTR LVL	HIGH		Elev 203'0"
F190	SE RB CORNER RM WTR LVL	HIGH		Elev 203'0"
F191	SW RB CORNER RM WTR LVL	HIGH		Elev 203'0"

Proposed Answer: B

Explanation: The Maximum Safe Area Water Level is 5' in the Reactor Building corner rooms (or elevation 203'). Water at this level is approaching the Core and/or Containment Spray pump motors, which threatens to make them inoperable due to electrical malfunction. This threatening of equipment necessary for safe shutdown of the plant is the basis for the Maximum Safe Area Water Level. All of the given computer points are associated with annunciator H2-2-1. Computer points B128 and B129 correspond to approximately floor level, which is below the Maximum Safe Area Water Level. Computer point B189 corresponds to elevation 203'. Therefore, computer point B189 indicates that NW Reactor Building corner room has reached the Maximum Safe Area Water Level.

Note: The question meets the K/A by testing the operational implication of electrical grounds / circuit malfunctions (the Maximum Safe Area Water Level) as it applies to Secondary Containment High Sump / Area Water Levels. The Maximum Safe Area Water Level is the operational implication of electrical grounds / circuit malfunctions with respect to Secondary Containment High Sump / Area Water Levels because the basis for the level is water reaching ECCS pump motors, which would cause electrical grounds / malfunctions and potential unavailability of critical plant equipment. The question requires knowledge of the Maximum Safe Area Water Level and how it is evaluated.

A. Plausible – Computer point B189 alarming high is enough information to determine that a Maximum Safe Area Water Level has been reached.
C. Plausible – Computer point B189 alarming high is enough information to determine that a Maximum Safe Area Water Level has been reached. Computer points B128 and B129 do not provide enough indication to determine a second area has reached the Maximum Safe Area Water Level.
D. Plausible – Computer point B189 alarming high is enough information to determine that a Maximum Safe Area Water Level has been reached.

Technical Reference(s): ARP H2-2-1, N1-EOP-5, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

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 #	295002 AK2.07
	Importance Rating	3.1

Loss of Main Condenser Vacuum

Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: Offgas system

Proposed Question: #60

A plant startup is in progress with the following:

- Reactor power is 18%.
- Turbine roll is about to commence.

Then, an explosion in the Offgas System results in the following:

- Annunciator H1-1-6, AIR EJECTOR HI-LO STM PR V CLOSED, alarms.
- Annunciator H1-2-6, AIR EJECTOR OFF GAS FLOW-PRESS TEMP HIGH, alarms.

Which one of the following identifies the response of Main Condenser vacuum and an automatic action that occurs?

	Main Condenser Vacuum	Automatic Action
A.	Remains stable	77-03, Stack Block Valve, closes
B.	Remains stable	76-12 and 76-13, Interstage Blocking Valves, close
C.	Degrades	77-03, Stack Block Valve, closes
D.	Degrades	76-12 and 76-13, Interstage Blocking Valves, close

Proposed Answer: D

Explanation: At 18% power, the Steam Jet Air Ejectors (SJAEs) are in service and the Mechanical Vacuum pump is secured. High temperature and pressure (H1-2-6) due to the Offgas explosion causes the SJAE Interstage Blocking valves (76-12 and 76-13) to close. This causes Main Condenser vacuum to degrade due to build-up of non-condensable gases.

A. Plausible – Main Condenser vacuum degrades due to closure of the SJAE Interstage Blocking valves. If power were lower such that the Mechanical Vacuum pump were still in service, Main Condenser vacuum would remain stable. The Stack Block Valve closes on high Offgas radiation, but not on high temperature/pressure.
B. Plausible – Main Condenser vacuum degrades due to closure of the SJAE Interstage Blocking valves. If power were lower such that the Mechanical Vacuum pump were still in service, Main Condenser vacuum would remain stable.
C. Plausible – The Stack Block Valve closes on high Offgas radiation, but not on high temperature/pressure.

Technical Reference(s): N1-OP-25, N1-SOP-25.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-271000-RBO-5

Question Source: Bank – Vision SYSID 51103

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 EK3.06
	Importance Rating	3.1

High Containment Hydrogen Concentration

Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Operation of wet well vent

Proposed Question: #61

A loss of coolant accident has resulted in the following:

- Hydrogen concentrations in the Containment require venting.
- Torus water level is 21 feet 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 N1-EOP-4, Primary Containment Control?

Vent from the...

A. Torus to minimize cycling of the Torus-to-Drywell vacuum breakers.

B. Drywell because Torus water level is too high for venting from the Torus.

C. Torus to better scrub fission products from the Containment atmosphere before release.

D. Drywell to more quickly reduce the risk of hydrogen ignition by electrical equipment operation.

Proposed Answer: C

Explanation: N1-EOP-4 Detail Z1 requires venting from the Torus as long as Torus water level is below 27 feet. This is to scrub the Containment atmosphere through the Torus water volume prior to release to lower the radioactive release.

A. Plausible – Torus venting is required, but the reason is to scrub the atmosphere. Lowering Torus pressure first does also prevent the need for vacuum breakers to cycle.
B. Plausible – If Torus water level were ≥27', then Drywell venting would be required.
D. Plausible – There is greater risk of hydrogen ignition in the Drywell due to presence of electrical equipment, however venting is required to be from the Torus to lower the radioactive release.

Technical Reference(s): N1-EOP-4, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2015 Cert #61

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 #	295008 AA1.08
	Importance Rating	3.5

High Reactor Water Level

Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL: Feedwater system

Proposed Question: #62

The plant is operating at 80% power with the following:

- A manual Reactor scram is inserted due to high Turbine vibration.
- Feedwater pumps 11 and 12 start and inject in the HPCI mode.
- Operators reset the HPCI signal in accordance with N1-SOP-1, Reactor Scram.
- Operators establish injection with both Feedwater flow control valves 11 and 12 in MAN.
- Feedwater pumps 11 and 12 trip on high Reactor water level.
- Reactor water level is now 100" and slowly lowering with Reactor Water Cleanup reject in service.
- NO further Operator action is taken with the Feedwater system.

Which one of the following describes the response of the Feedwater system as Reactor water level continues to lower?

A. Feedwater pump 11 automatically restarts when Reactor water level reaches 72". Feedwater pump 12 automatically restarts if Reactor water level reaches 65".

B. Feedwater pump 12 automatically restarts when Reactor water level reaches 72". Feedwater pump 11 automatically restarts if Reactor water level reaches 65".

C. Feedwater pumps 11 and 12 automatically restart when Reactor water reaches 53".

D. NEITHER Feedwater pump 11 NOR 12 will automatically restart.

Proposed Answer: C

Explanation: With flow control valves open, Feedwater pumps 11 and 12 automatically trip when Reactor water level exceeds 95". As Reactor water level lowers, Feedwater pumps 11 and 12 automatically restart when Reactor water level reaches 53". No action is required to reset the high Reactor water level trip. Feedwater flow control valves also shift back into the HPCI mode when Reactor water level reaches 53". Feedwater flow control valves 11 and 12 will then attempt to maintain Reactor water level at 65" and 72", respectively.

A. Plausible – 72" and 65" are the Reactor water levels at which the Feedwater flow control valves will attempt to maintain Reactor water level, but these levels do not restart a tripped Feedwater pump.
B. Plausible – 72" and 65" are the Reactor water levels at which the Feedwater flow control valves will attempt to maintain Reactor water level, but these levels do not restart a tripped Feedwater pump.
D. Plausible – Feedwater pumps 11 and 12 both restart automatically at 53". The HPCI signal is manually reset by momentarily depressing pushbuttons, not some other action that maintains HPCI overridden. Additionally, the high Reactor water level trip does not prevent restart of Feedwater pumps if Reactor water level lowers to 53".

Technical Reference(s): N1-SOP-1, N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-9

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 #	2
	K/A #	295007 AA2.01
	Importance Rating	4.1

High Reactor Pressure

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure

Proposed Question: #63

The plant is operating at 100% power when EPR sensed pressure fails downscale.

Which one of the following describes the plant response to this malfunction?

A. The Reactor scrams on high pressure.

B. The Reactor scrams on a low pressure MSIV isolation.

C. The Reactor remains at power. The MPR controls pressure at a new, lower value.

D. The Reactor remains at power. The MPR controls pressure at a new, higher value.

Proposed Answer: D

Explanation: The EPR will attempt to close TCVs, resulting in rising Reactor pressure. As MPR pressure error becomes greater than EPR pressure error, the MPR will assume control of TCVs, limiting the pressure rise. The MPR will control Reactor pressure at a new higher value and the Reactor will remain at power.

A. Plausible – Reactor pressure does rise, however the EPR and MPR are setup such that this transient results in the MPR controlling Reactor pressure prior to receipt of an automatic Reactor scram.
B. Plausible – This EPR malfunction will cause Reactor pressure to rise, not lower. Failure of EPR set pressure downscale would result in Reactor pressure lowering until MSIVs closed.
C. Plausible – This EPR malfunction will cause Reactor pressure to rise, not lower. Failure of EPR set pressure downscale would result in Reactor pressure lowering, but the MPR would not take over control of Reactor pressure as it does with rising Reactor pressure.

Technical Reference(s): N1-OP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-248000-RBO-11

Question Source: Bank – 2009 NRC #12

Question History: 2009 NRC #12

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 2.4.45
	Importance Rating	4.1

Loss of CRD Pumps

Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: #64

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

Time (hh:mm)	Event
00:00	• CRD pump 11 is in service. • CRD pump 12 is in standby.
00:01	• Annunciator F3-2-2, CONTROL ROD DRIVE PUMP 11 SUCT PRESS LOW, alarms. • Annunciator F3-1-2, CONTROL ROD DRIVE PUMP 11 TRIP-VIB, alarms. • Annunciator F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, alarms.
00:05	• Annunciator F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, alarms. • Control rod 22-19 accumulator pressure is 990 psig and slowly lowering.

Which one of the following describes the required operator action(s) and the reason for the action(s), in accordance with N1-SOP-5.1, Loss of Control Rod Drive?

A. Immediately scram the Reactor to prevent spurious control rod drifts.

B. Immediately scram the Reactor to ensure the scram function is maintained.

C. Start a CRD pump by time 00:25 and then insert a control rod one notch, to prevent spurious control rod drifts.

D. Start a CRD pump by time 00:25 and then insert a control rod one notch, to ensure the scram function is maintained.

Proposed Answer: D

Explanation: This combination of alarms indicates that CRD pump 11 has tripped, N1-SOP-5.1 entry is required, and one accumulator is alarming on low pressure. With Reactor power at 100%, Reactor pressure is greater than 900 psig. With Reactor pressure greater than 900 psig, 20 minutes is allowed to restart a CRD pump before a Reactor scram is required. Once the CRD pump is started, at least one control rod must be inserted at least one notch to demonstrate function has been restored. If these conditions are not met within 20 minutes, a Reactor scram is then required. The basis for these requirements to restore CRD hydraulic pressure is to ensure the scram function is maintained.

A. Plausible – Since Reactor pressure is >900 psig, 20 minutes is allowed to restore a CRD pump prior to requiring a Reactor scram. If Reactor pressure was <900 psig, then an immediate Reactor scram would be required. Preventing spurious control rod drifts is the concern on a related CRD malfunction (low scram air header pressure).
B. Plausible – Since Reactor pressure is >900 psig, 20 minutes is allowed to restore a CRD pump prior to requiring a Reactor scram. If Reactor pressure was <900 psig, then an immediate Reactor scram would be required. Ensuring the scram function is maintained is the correct basis for scramming if <900 psig.
C. Plausible – Spurious control rod drifts is not the basis for restoring CRD hydraulic pressure. This is the basis for a related CRD malfunction (low scram air header pressure).

Technical Reference(s): N1-SOP-5.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP5.1C01 EO-2

Question Source: Bank – 2010 NRC #73

Question History: 2010 NRC #73

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 #	295035 EK1.01
	Importance Rating	3.9

Secondary Containment High Differential Pressure

Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment integrity

Proposed Question: #65

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

- Both the Reactor Building Inner Lift Door and the Outer Swing Door are closed.
- An operator is tasked with opening the Outer Swing Door.
- Prior to opening the Outer Swing Door, the operator verifies closed the Inner Lift Door and notes that the seal pressure is 0.5 psig.

Which one of the following describes the impact of this seal pressure on Secondary Containment and the required actions, in accordance with N1-OP-52, Reactor Building Track Bay Doors?

A. Secondary Containment is met and the Outer Swing Door may be opened.

B. Secondary Containment is met but the Outer Swing Door must be maintained closed.

C. Secondary Containment is NOT met and the Outer Swing Door Cremone bolts must be locked.

D. Secondary Containment is NOT met and RBEVS must be started to maximize Reactor Building negative pressure.

Proposed Answer: B

Explanation: Secondary Containment only requires one of the two Reactor Building Track Bay doors to be closed at a time. The Inner Door seal pressure is below the minimum required of 6 psig, therefore the Inner Door cannot be relied upon to provide Secondary Containment. However, the Outer Door has not yet been opened. The Outer Door still being closed is enough to satisfy Secondary Containment requirements. However, the Outer Door must be maintained closed to avoid losing Secondary Containment with the given Inner Door seal pressure.

Note: The question meets the K/A by testing the operational implication of maintaining Secondary Containment integrity (proper operation of Reactor Building doors, including required configuration with degraded seal pressure) as it applies to Secondary Containment High Differential Pressure (improper control of doors would result in high Reactor Building D/P). While the question does not specifically give a high Secondary Containment differential pressure in the stem, it meets the K/A by testing an operational implication of maintaining Secondary Containment integrity (control of Reactor Building door positions and seal pressure) as it relates to ***avoiding*** high Secondary Containment differential pressure (approaching N1-EOP-5 entry on D/P).

A. Plausible – While the Inner Door has some positive seal pressure, it is below the minimum requirement of 6 psig to ensure Secondary Containment integrity. Therefore, the Outer Door must not be opened.
C. Plausible – Although the Inner Door seal pressure is not high enough for the Inner Door to ensure Secondary Containment, the Outer Door being closed currently maintains Secondary Containment. Plausible if applicant believes both doors must be closed with adequate seal pressure to maintain Secondary Containment. The Cremone bolts provide a locking mechanism for the Outer Door, which would be locked. This would be a correct answer if seal pressure were noted after the Outer Door was open.
D. Plausible – Although the Inner Door seal pressure is not high enough for the Inner Door to ensure Secondary Containment, the Outer Door being closed currently maintains Secondary Containment. Plausible if applicant believes both doors must be closed with adequate seal pressure to maintain Secondary Containment. RBEVS would be started if Secondary Containment negative pressure were specifically challenged. This would be a correct answer if seal pressure were noted after the Outer Door was open and building D/P were degrading.

Technical Reference(s): N1-OP-52

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-290001-RBO-9

Question Source: Bank – Vision SYSID 88431

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.26
	Importance Rating	3.4

Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

Proposed Question: #66

Which one of the following describes the correct sequence and flow path for initially replacing air in the Main Generator with hydrogen during a plant startup?

A. H_2 in through the upper header, air vented through the lower header.

B. H_2 in through the lower header, air vented through the upper header.

C. CO_2 in through the upper header, air vented through the lower header. H_2 is then admitted through the lower header and the CO_2 is vented through the upper header.

D. CO_2 in through the lower header, air vented through the upper header. H_2 is then admitted through the upper header and the CO_2 is vented through the lower header.

Proposed Answer: D

Explanation: CO_2 is admitted through the lower header (heavier than Air). The CO_2 will fill the generator and push the Air out the upper header. H_2 is then admitted through the upper header (lighter than CO_2), CO_2 is vented through the lower header.

A. Plausible – Hydrogen is admitted through the upper header, however CO_2 must be used as a buffer to prevent having an explosive mixture of H_2 and Air.
B. Plausible – Air is vented through the upper header, however CO_2 must be used as a buffer to prevent having an explosive mixture of H_2 and Air.
C. Plausible – CO_2 is admitted through the lower header, not the upper header (heavier than Air). The CO_2 will fill the generator and push the Air out the upper header, not the lower header. H_2 is then admitted through the upper header (lighter than CO_2), CO_2 is vented through the lower header.

Technical Reference(s): N1-OP-7 Section B.

Proposed references to be provided to applicants during examination: None

Learning Objective: 252000 RBO-10 (SYSID 65603)

Question Source: Bank – 98622

Question History:

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 #	2.1.1
	Importance Rating	3.8

Knowledge of conduct of operations requirements.

Proposed Question: #67

Given the following:

- You are a licensed Reactor Operator.
- You have been on vacation for the previous 14 days.
- Today was your first shift back on watch.
- You worked the normal day shift from 0600 to 1800.
- You then were required to stay a total of four (4) hours past end of shift due to an on-coming Operator calling in sick.
- You were originally scheduled to cover day shift tomorrow from 0600-1800.

Which one of the following identifies the earliest time you can return to work tomorrow without receiving a waiver, in accordance with LS-AA-119, Fatigue Management and Work Hour Limits?

A. 0600

B. 0700

C. 0800

D. 1000

Proposed Answer: C

Explanation: LS-AA-119 Section 3.4 requires "a 10-hour break between the previous work period, or an 8-hour break between the previous work period when a break of less than 10 hours was necessary to accommodate a crew's scheduled transition between work schedules or shifts". Since this situation is not related to a crew's scheduled shift transition, the individual needs a minimum 10 hour break before returning to work. The individual worked 4 hours past 1800, so they left work at 2200. Ten hours later is 0800.

A. Plausible – 0600 is only 8 hours after the individual left work and would not provide the required 10 hour break. This would be acceptable only if the break were required to accommodate a crew's scheduled transition between shifts.
B. Plausible – 0700 is only 9 hours after the individual left work and would not provide the required 10 hour break. This would be correct if the requirement were 9 hours.
D. Plausible – 0800 satisfies the requirement, therefore 1000 is not the earliest allowed time. This would be correct if the requirement were 12 hours.

Technical Reference(s): LS-AA-119

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-ADMROJ15

Question Source: Bank – 2013 Cert #66

Question History:

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 #	2.2.1
	Importance Rating	4.5

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: #68

The plant is shutdown with the following:

- Pre-startup checks are being performed per N1-OP-43A, Plant Startup.
- The following SRM indications are observed:

SRM	Count Rate	Position	Bypassed?
11	2 cps	Fully inserted	Yes
12	110 cps	Fully inserted	No
13	10 cps	Fully inserted	No
14	50 cps	Fully inserted	No

Which one of the following describes the status of the SRMs, in accordance with N1-OP-43A and N1-OP-38A, Source Range Monitor (SRM)?

These SRM indications are...

A. satisfactory to commence Reactor startup.

B. NOT satisfactory to commence Reactor startup. The number of satisfactory SRMs is one less than required.

C. NOT satisfactory to commence Reactor startup. The number of satisfactory SRMs is two less than required.

D. NOT satisfactory to commence Reactor startup. The number of satisfactory SRMs is three less than required.

Proposed Answer: A

Explanation: N1-OP-43A and Technical Specification 3.5.1 require at least 3 SRMs indicating greater than 3 cps. SRMs 12, 13, and 14 meet this requirement, therefore SRMs support commencing Reactor startup.

Note: The question meets the K/A by testing knowledge related to the ability to perform pre-startup procedures for the facility (N1-OP-43A / N1-OP-38A pre-start checks for SRMs). Note that while K/A goes on to give a specific example of what could meet "Ability to perform pre-startup procedures for the facility", it is not all-inclusive and does not preclude testing something that is not "controls associated with plant equipment that could affect reactivity" (SRMs are an indication of reactivity used to aid in use of other controls, not directly a control of reactivity).

B. Plausible – SRMs are SAT for commencing Reactor startup. SRM A does not have the required number of counts and is bypassed, but only 3 SRMs are required. 1 less than required would be the answer if all 4 SRMs were required.
C. Plausible – SRMs are SAT for commencing Reactor startup. 2 less than required would be the answer if the minimum count rate was 100 cps (which is the setpoint for the SRM downscale rod block with detectors not fully inserted).
D. Plausible – SRMs are SAT for commencing Reactor startup. 3 less than required would be the answer if the minimum count rate was 100 cps and all 4 SRMs were required.

Technical Reference(s): N1-OP-43A, N1-OP-38A, Technical Specification 3.5.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-10

Question Source: Bank – JAF 9/14 NRC #68

Question History: JAF 9/14 NRC #68

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.42
	Importance Rating	3.9

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: #69

The plant is operating at 100% power when the following events occur:

- Fuels reports that a revised MCPR limit must be enforced.
- With the revised limit enforced, a 3D Monicore case shows MFLCPR is 1.03.

Which one of the following describes the required action with regards to the MCPR thermal limit, if any, in accordance with Technical Specifications?

A. No action required because MCPR is SAT.

B. Initiate action within 15 minutes to restore MCPR to within the prescribed limit.

C. Initiate a shutdown within 1 hour.

D. Immediately scram the Reactor.

Proposed Answer: B

Explanation: Actual MCPR is below the MCPR limit, as evidenced by the MFLCPR ratio being greater than 1, therefore MCPR is UNSAT. This requires entry into Technical Specification 3.1.7.c, which requires "action shall be initiated within 15 minutes to restore operation to within the prescribed limit. If all the operating MCPRs are not returned to within the prescribed limit within two (2) hours, reactor power reductions shall be initiated at a rate not less than 10% per hour until MCPR is within the prescribed limit."

A. Plausible – Actual MCPR is below the MCPR limit, as evidenced by the MFLCPR ratio being greater than 1, therefore MCPR is UNSAT. Plausible if candidate misunderstands the required value of MFLCPR to ensure MCPR is SAT.
C. Plausible – Action is required within 15 minutes. Plausible because lowering power will be required and many Technical Specifications do require such a shutdown.
D. Plausible – There is no requirement for an immediate Reactor scram. Plausible because fast action is required (15 minutes) and some other fast action Technical Specifications do require an immediate scram in some situations (such as high Torus water temperature).

Technical Reference(s): N1-OP-31 P&L #17, Technical Specification 3.1.7

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245000-RBO-14

Question Source: Bank – 2009 Cert #37

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

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

Ability to control radiation releases.

Proposed Question: #70

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

- A plant transient is in progress.
- Annunciator H1-1-8, STACK GAS MONITORS HIGH RADIATION, is in alarm.
- The running Turbine Building Ventilation fans have tripped.
- The Shift Manager declares an ALERT condition based on off-site release rates.

Which one of the following operator actions is required in accordance with N1-EOP-6, Radioactivity Release Control, and why?

A. Restart the Turbine Building Ventilation system to direct any radioactivity release through an elevated, monitored path.

B. Restart the Turbine Building Ventilation system to minimize transferring contamination from the Turbine Building to the Reactor Building.

C. Verify the Turbine Building Ventilation system isolated to minimize the overall radiological release from the Turbine Building.

D. Verify the Turbine Building Ventilation system isolated to minimize transferring contamination from the Turbine Building to the Reactor Building.

Proposed Answer: A

Explanation: The basis document for N1-EOP-6 states the Turbine Building Ventilation is restarted to prevent an unmonitored ground release. N1-EOP-6 is entered when the ALERT condition based on off-site release rates is exceeded. The Turbine Building Ventilation system maintains a negative pressure in the Turbine Building to ensure releases from or through systems that pass through secondary containment are captured for release through the plant stack.

B. Plausible – Turbine Building Ventilation is restarted to prevent an unmonitored ground release, not to control transfer of contamination from the Turbine Building to the Reactor Building. Plausible because if the Turbine Building is allowed to be at a more positive pressure (ventilation isolated), then leakage from the Turbine Building to the Reactor Building would increase, which would introduce more contamination into the Reactor Building.
C. Plausible – Turbine Building Ventilation must be restarted. Plausible because isolating Turbine Building Ventilation does slow the overall release to the environment.
D. Plausible – Turbine Building Ventilation must be restarted. Plausible because isolating Turbine Building Ventilation does slow the overall release to the environment, which may also slow the amount of contamination that is introduced to the Reactor Building through the supply ventilation intake.

Technical Reference(s): N1-EOP-6, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288002-RBO-12

Question Source: Bank – 2010 NRC #56

Question History: 2010 NRC #56

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

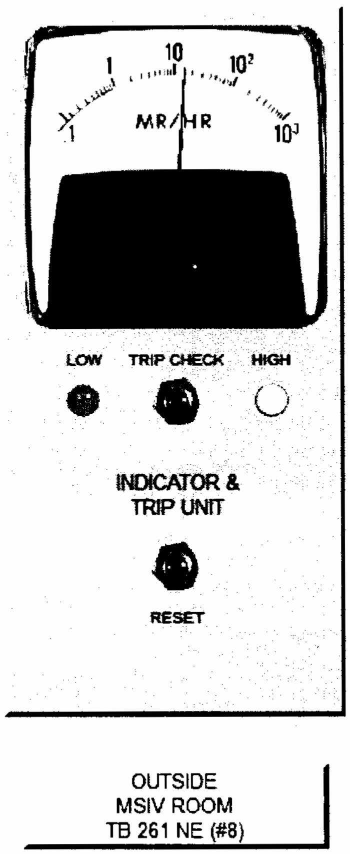
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.5
	Importance Rating	2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #71

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

- Annunciator H1-4-8, Area Radiation Monitors, alarms and is acknowledged.
- The following area radiation monitor (ARM) indication is observed and validated:



- NO other ARMs are in alarm.

Which one of the following describes (1) the need for Emergency Operating Procedure (EOP) entry based on this indication and (2) how Annunciator H1-4-8 will behave if another ARM in the same building reaches its Hi setpoint?

(1) EOP entry is...	(2) Annunciator H1-4-8 will...

A.	required.	re-flash.
B.	required.	NOT re-flash.
C.	NOT required.	re-flash.
D.	NOT required.	NOT re-flash.

Proposed Answer: C

Explanation: Annunciator H1-4-8 receives input from multiple ARMs, some of which require N1-EOP-5 entry and some that do not. The given ARM is for the Outer MSIV Room, which does not require N1-EOP-5 entry. This ARM input to Annunciator H1-4-8 will allow re-flash of the annunciator if another ARM in the same building reaches its hi setpoint.

A. Plausible – N1-EOP-5 entry is not required. Plausible because many of the ARM inputs to this annunciator would require entry and the Outer MSIV Room is located close to the Secondary Containment.
B. Plausible – N1-EOP-5 entry is not required. Plausible because many of the ARM inputs to this annunciator would require entry and the Outer MSIV Room is located close to the Secondary Containment. The annunciator will re-flash. Plausible because some of the ARM inputs to this annunciator would not allow re-flash (RSSB ARMs).
D. Plausible – The annunciator will re-flash. Plausible because some of the ARM inputs to this annunciator would not allow re-flash (RSSB ARMs).

Technical Reference(s): ARP H1-4-8, N1-EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-272000-RBO-5

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 #	3
	Group #	
	K/A #	2.4.25
	Importance Rating	3.3

Knowledge of fire protection procedures.

Proposed Question: #72

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

Time (minutes)	Event
0	• A fire alarm is received for Reactor Building 281 East. • A Fire Brigade member in the area immediately confirms the presence of a fire. • N1-SOP-21.1, Fire in Plant, is entered.
15	• The Fire Brigade reports that the fire is still in progress and NOT under control.
20	• 80-118, CONTAINMENT SPRAY TEST TO TORUS F.C.V., spuriously opens.
25	• Attempts to close 80-118 from the Control Room are unsuccessful.
30	• The Shift Manager determines that the fire endangers Safe Shutdown capability.

Which one of the following describes the earliest time a manual Reactor scram is required, in accordance with N1-SOP-21.1?

A. 15 minutes

B. 20 minutes

C. 25 minutes

D. 30 minutes

Proposed Answer: A

Explanation: N1-SOP-21.1 requires a manual Reactor scram if any of the following conditions exist due to a fire in an area included in Table 21.1-1 (which includes RB 281 East):

- Spurious valve operation
- Loss of equipment control
- Fire NOT under control within 15 minutes
- Fire endangers Safe Shutdown capability

At time 15 minutes, the fire is NOT under control and has been in progress for 15 minutes, therefore a manual Reactor scram is required.

B. Plausible – At time 20 minutes, there is indication of spurious valve control, which would require a manual Reactor scram. However, a scram was already required at time 15 minutes.
C. Plausible – At time 25 minutes, there is indication of loss of equipment control, which would require a manual Reactor scram. However, a scram was already required at time 15 minutes.
D. Plausible – At time 30 minutes, the fire is determined to be endangering Safe Shutdown capability, which would require a manual Reactor scram. However, a scram was already required at time 15 minutes.

Technical Reference(s): N1-SOP-21.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.1C01 EO-2

Question Source: Bank – 2013 NRC #19

Question History: 2013 NRC #19

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 #	2.4.21
	Importance Rating	4.0

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.

Proposed Question: #73

A loss of coolant accident has resulted in the following:

- Reactor water level is -94" and slowly lowering.
- Four (4) ERVs are open.
- Core Spray loop 12 is the only source of injection to the Reactor.
- Core Spray loop 12 is injecting at 190 x 10^4 lbm/hr.

Which one of the following describes the status of core cooling, in accordance with the Emergency Operating Procedures?

Adequate core cooling (ACC) is...

A. NOT presently assured.

B. presently assured. ACC will be first lost if Reactor water level reaches -109".

C. presently assured. ACC will be first lost if Reactor water level reaches -121".

D. presently assured. ACC will be maintained regardless of Reactor water level if Core Spray flow remains at the present value.

Proposed Answer: B

Explanation: The preferred method of ACC, Core Submergence, is not currently available because Reactor water level is below -84". However, ACC is still assured by Steam Cooling, which requires Reactor water level to be above -109". If Reactor water level reaches -109", this method of ACC will be lost. ACC by Spray Cooling is not available because only one Core Spray loop has adequate flow (>180 x 10^4 lbm/hr). Therefore, if Reactor water level reaches -109", ACC will no longer be assured.

A. Plausible – Since Reactor water level is greater than -109", ACC is asssured by Steam Cooling. Plausible because the preferred method of ACC, Core Submergence, is not currently available because Reactor water level is below -84".
C. Plausible – ACC will first be lost if Reactor water level reaches -109". -121" is the Reactor water level historically associated with Steam Cooling Without Injection, however this level has recently been revised out of the EOPs at NMP1.
D. Plausible – ACC will first be lost if Reactor water level reaches -109". ACC by Spray Cooling only exists if both Core Spray loops are injecting at greater than or equal to 180 x 10^4 lbm/hr. If both Core Spray loops were injecting at least this much water, then ACC would be maintained regardless of Reactor water level.

Technical Reference(s): N1-EOP-2, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Bank – JAF 9/12 NRC #22

Question History: JAF 9/12 NRC #22

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 #	2.1.8
	Importance Rating	3.4

Ability to coordinate personnel activities outside the control room.

Proposed Question: #74

The plant is operating at 90% power with the following:

- Reactor Recirc Motor Generator (RRMG) Set 11 is in local lock.
- Only two Reactor Operators are available on shift.
- Both Reactor Operators are currently in the Control Room.
- Speed must be lowered on RRMG Set 11 using local manual control per N1-OP-1, Nuclear Steam Supply System (NSSS).

Which one of the following describes the required control of this evolution, in accordance with N1-OP-1?

The speed adjustment...

A. must be performed by a Licensed Operator, and one of the available Reactor Operators may leave the Control Room.

B. must be performed by a Licensed Operator, but neither of the available Reactor Operators may leave the Control Room.

C. may be performed by a Non-Licensed Operator, as long as continuous communication is maintained with a Licensed Operator in the Control Room.

D. may be performed by a Non-Licensed Operator, as long as all actions at RRMG Set 11 get a concurrent verification from another Non-Licensed Operator.

Proposed Answer: A

Explanation: Changing speed on RRMG Set 11 directly affects reactivity and must be performed by a Licensed Operator. Only one Reactor Operator is required to stay in the Control Room, therefore one of the available Reactor Operators may exit the Control Room and go to RRMG Set 11 in the Turbine Building.

B. Plausible – Only one Reactor Operator is required to stay in the Control Room, therefore one of the available Reactor Operators may exit the Control Room and go to RRMG Set 11 in the Turbine Building. Plausible because there are usually two Reactor Operators in the Control Room, especially during a reactivity manipulation.
C. Plausible – Changing speed on RRMG Set 11 directly affects reactivity and must be performed by a Licensed Operator. Plausible because N1-OP-1 section H.5 does require continuous communication with the Control Room during this evolution, which would provide oversight from a Licensed Operator.
D. Plausible – Changing speed on RRMG Set 11 directly affects reactivity and must be performed by a Licensed Operator. Plausible because concurrent verification does provide a higher than normal degree of operating rigor and the manipulation is in the plant, which normally would allow conduct by a Non-Licensed Operator.

Technical Reference(s): N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-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 #	3
	Group #	
	K/A #	2.2.18
	Importance Rating	2.6

Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Proposed Question: #75

The plant is shutdown for a refueling outage 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.

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.

Note: The question meets the K/A by giving a desired maintenance activity during shutdown conditions and testing the process for managing this activity (use of clearance process to allow maintenance to have personnel protection and ability to manipulate disconnect switch).

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 a caution tag only may not be used for personnel protection. Plausible because a component with a caution 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.

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

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – 2013 NRC #71

Question History: 2013 NRC #71

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 #	295028 EA2.03
	Importance Rating	3.9

High Drywell Temperature

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level

Proposed Question: #76

The plant was operating at 100% power when a coolant leak in the Drywell resulted in the following:

- Containment Spray is in service.
- Drywell pressure is 9 psig and slowly lowering.
- Drywell temperature on elevation 319' is 320°F and slowly lowering.
- Torus pressure is 11 psig and slowly lowering.
- Reactor pressure is 500 psig and slowly lowering.
- Reactor water level indicates:
  - 0" and stable on Lo-Lo-Lo indicators.
  - 4' and stable on Wide Range indicators.
  - 0" and stable on GEMAC indicators.
  - 0" and stable on Rosemount indicators.
- Fuel zone level indicators are inoperable.

<u>**Note:**</u> See the following pages for a portion of EOP Detail A and the Containment Spray Initiation Limit curve.

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

A. Secure Containment Sprays due to the current position on the Containment Spray Initiation Limit curve.

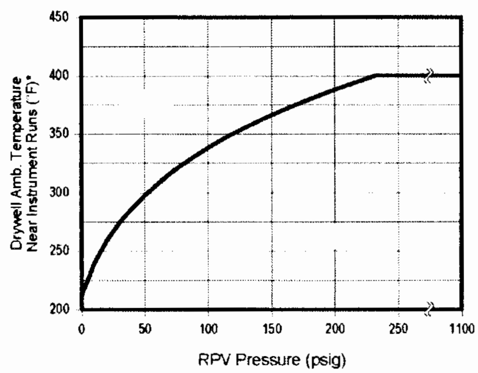
B. Remain in N1-EOP-2, RPV Control, and control Reactor water level using the Lo-Lo-Lo indicators.

C. Remain in N1-EOP-2, RPV Control, and control Reactor water level using the GEMACs.

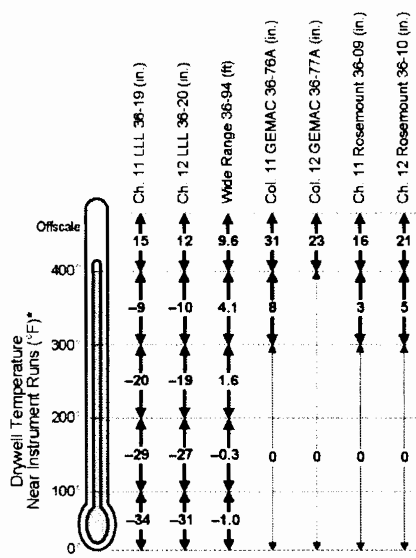
D. Exit N1-EOP-2, RPV Control, and enter N1-EOP-7, RPV Flooding.

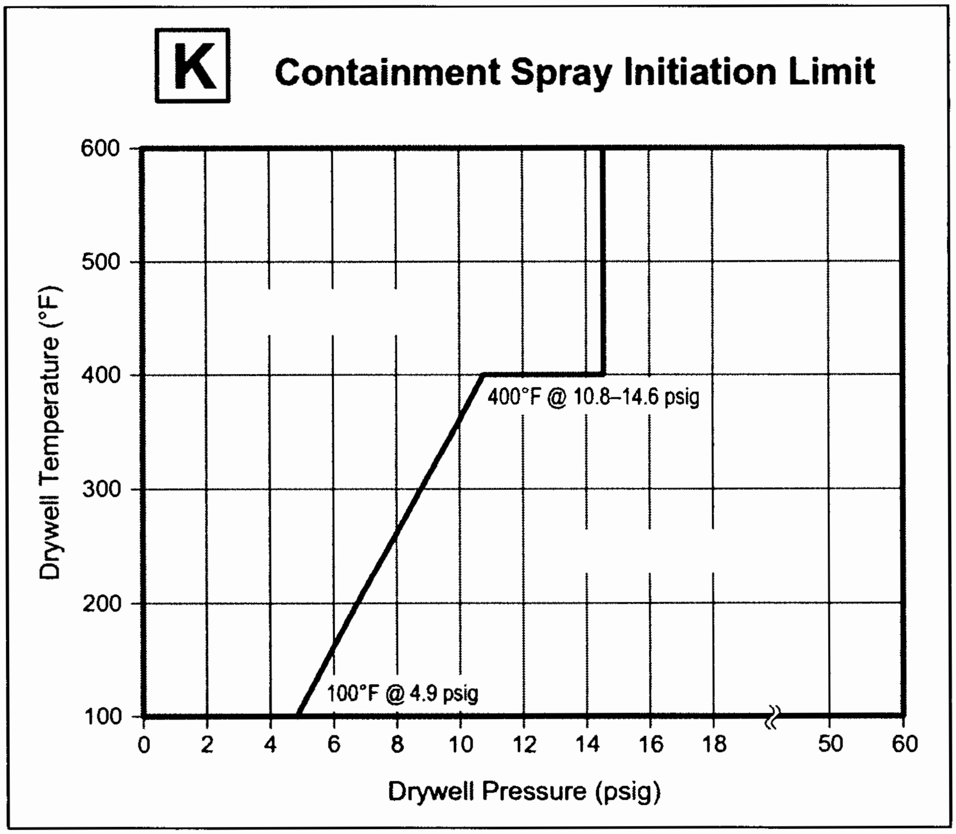
A RPV Water Level Instrument Restrictions

B RPV Saturation Temperature



C Variable Leg Tap Indicated Levels





K
Containment Spray Initiation Limit
Drywell Temperature (°F)
600
500
400
300
200
100
400°F @ 10.8–14.6 psig
100°F @ 4.9 psig
0
2
4
6
8
10
12
14
16
18
50
60
Drywell Pressure (psig)

Proposed Answer: B

Explanation: The elevated Drywell temperature and low Reactor water level requires use of Detail A to determine usability of Reactor water level instrumentation. Wide Range, GEMAC, and Rosemount level indicators are indicating at their Variable Leg Tap Indicated Levels and are therefore not usable for Reactor water level control. However, Lo-Lo-Lo indicators are still above their Variable Leg Tap Indicated Level of -10" for 320°F Drywell temperature. With Fuel Zone indicators inoperable, this means only Lo-Lo-Lo indicators are available for control of Reactor water level. With Lo-Lo-Lo indicators available, N1-EOP-2 remains the correct procedure to implement.

A. Plausible – Although the bad region of the CSIL curve has been entered, there is no direction to secure Containment Sprays that are already in service.
C. Plausible – GEMACs are indicating at or below their Variable Leg Tap Indicated Levels of 0" and 7" and therefore cannot be used to remain in N1-EOP-2 and control Reactor water level.
D. Plausible – Since Lo-Lo-Lo indicators are still usable, it is neither required nor allowed to exit N1-EOP-2 and enter N1-EOP-7.

Technical Reference(s): N1-EOP-2, N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO #2

Question Source: Bank – 2013 Cert #92

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 #	295001 AA2.06
	Importance Rating	3.3

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: Nuclear boiler instrumentation

Proposed Question: #77

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

- A spurious signal causes all Reactor Recirculation pumps to trip.
- The Reactor is manually scrammed.
- NO Reactor Recirculation pump is able to be re-started.

Five hours later, the following conditions exist:

- Reactor pressure is 90 psig and slowly lowering.
- Reactor water level is being controlled in a band of 53" to 95".
- Shutdown Cooling is being placed in service in accordance with N1-OP-43C, Plant Shutdown, and N1-OP-4, Shutdown Cooling.

Which one of the following describes (1) the required control of Reactor water level and (2) the required control of the Recirculation loops, in accordance with N1-OP-43C and N1-OP-4?

	(1) Reactor water level must be controlled...		(2) Required Control of Recirculation Loops
A.	in the current band.		The suction and discharge valves of at least two Recirculation loops must remain full open.
B.	in the current band.		All Recirculation pump suction valves, or all discharge and discharge bypass valves, must be closed.
C.	higher than the current band.		The suction and discharge valves of at least two Recirculation loops must remain full open.
D.	higher than the current band.		All Recirculation pump suction valves, or all discharge and discharge bypass valves, must be closed.

Proposed Answer: D

Explanation: With no Recirculation pumps in service, Reactor vessel thermal stratification and maintaining adequate Reactor water level indication are concerns addressed by both N1-OP-43C and N1-OP-4. Since no Recirculation pumps can be placed in service, thermal stratification will be prevented and adequate Reactor water level indication will be maintained by placing Shutdown Cooling in service, raising Reactor water level to the level of the Main Steam Lines (which is above the top end of the current control band of 95"), and closing either all the Recirculation loop suction valves or all the Recirculation loop discharge and discharge bypass valves.

Note: The question meets SRO-level guidelines because the applicant must assess multiple plant conditions (Reactor water level, lack of operating RRPs, need for Shutdown Cooling), determine the need to perform an off-normal subsection of N1-OP-4 (section H.1.0 versus E.3.0), and understand the impact of this assessment on Reactor water level control and Recirc loop control. Additionally, the question does not screen as an RO question per the NUREG-1021 ES-401 Attachment 2 Figure 2.

Note: The question meets the K/A because it tests the ability to determine/interpret Nuclear Boiler Instrumentation (interpret current Reactor water level indication vs. required level; determine required control of Recirc loop valves with SDC in service, which relates to ensuring Reactor water level indication remains valid and thermal stratification does not occur, which would make coolant temperature indications not representative of coolant temperature in the core) with a loss of Recirculation flow.

A. Plausible – The current level control band is typical per N1-SOP-1, N1-EOP-2, and N1-OP-43C, however level must be raised to the level of the Main Steam Lines (which is above the top end of the current control band of 95") due to the inability to restart a Recirculation loop. N1-OP-43C normally requires maintaining the suction and discharge valves of at least two Recirculation loops full open. However, all Recirculation loops must be isolated in this situation to prevent Shutdown Cooling flow from short-cycling the core.
B. Plausible – The current level control band is typical per N1-SOP-1, N1-EOP-2, and N1-OP-43C, however level must be raised to the level of the Main Steam Lines (which is above the top end of the current control band of 95") due to the inability to restart a Recirculation loop.
C. Plausible – N1-OP-43C normally requires maintaining the suction and discharge valves of at least two Recirculation loops full open. However, all Recirculation loops must be isolated in this situation to prevent Shutdown Cooling flow from short-cycling the core.

Technical Reference(s): N1-OP-43C, N1-OP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-10

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 #	295030 EA2.01
	Importance Rating	4.2

Low Suppression Pool Water Level

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool level

Proposed Question: #78

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

- A seismic event occurs.
- An un-isolable leak develops from the Torus.
- N1-EOP-4, Primary Containment Control, and N1-EOP-2, RPV Control, are entered.
- The Reactor is scrammed.
- Reactor pressure is 900 psig and stable on Turbine Bypass Valves (TBVs).
- Torus makeup is in service in accordance with N1-EOP-1 Attachment 18, Raw Water to Torus Makeup.
- Torus water level is 7.5' and slowly lowering.

Which one of the following describes the required control of Reactor pressure, in accordance with the Emergency Operating Procedures?

Entry into N1-EOP-8, RPV Blowdown, is...

A. required. Rapidly depressurize the Reactor by opening ERVs.

B. required. Rapidly depressurize the Reactor using Blowdown Systems other than ERVs.

C. NOT required. Reactor pressure may be lowered with a cooldown rate of greater than 100°F/hr.

D. NOT required. Reactor pressure may be lowered, but the cooldown rate must be maintained less than 100°F/hr.

Proposed Answer: B

Explanation: With Torus makeup in service from the highest capacity source (Containment Spray Raw Water) and Torus water level 7.5' and lowering, Torus level cannot be maintained above 8'. Therefore, N1-EOP-4 requires entering N1-EOP-8, RPV Blowdown. With Torus water level less than 8', N1-EOP-8 uses alternate Blowdown Systems, such as Turbine Bypass Valves and Emergency Condensers, to rapidly depressurize the Reactor and does NOT allow opening ERVs.

A. Plausible – N1-EOP-8 normally requires opening 4 ERVs, however not with Torus water level less than 8'.
C. Plausible – With Torus makeup in service from the highest capacity source (Containment Spray Raw Water) and Torus water level 7.5' and lowering, Torus level cannot be maintained above 8'. Therefore, N1-EOP-4 requires entering N1-EOP-8, RPV Blowdown. If Torus water level were greater than 8' with the potential to be maintained above 8', then N1-EOP-8 entry would not yet be required and N1-EOP-2 would allow exceeding the normal 100oF/hr cooldown rate limit due to the potential for an RPV Blowdown.
D. Plausible – With Torus makeup in service from the highest capacity source (Containment Spray Raw Water) and Torus water level 7.5' and lowering, Torus level cannot be maintained above 8'. Therefore, N1-EOP-4 requires entering N1-EOP-8, RPV Blowdown.

Technical Reference(s): N1-EOP-4, N1-EOP-8

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 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 2.1.23
	Importance Rating	4.4

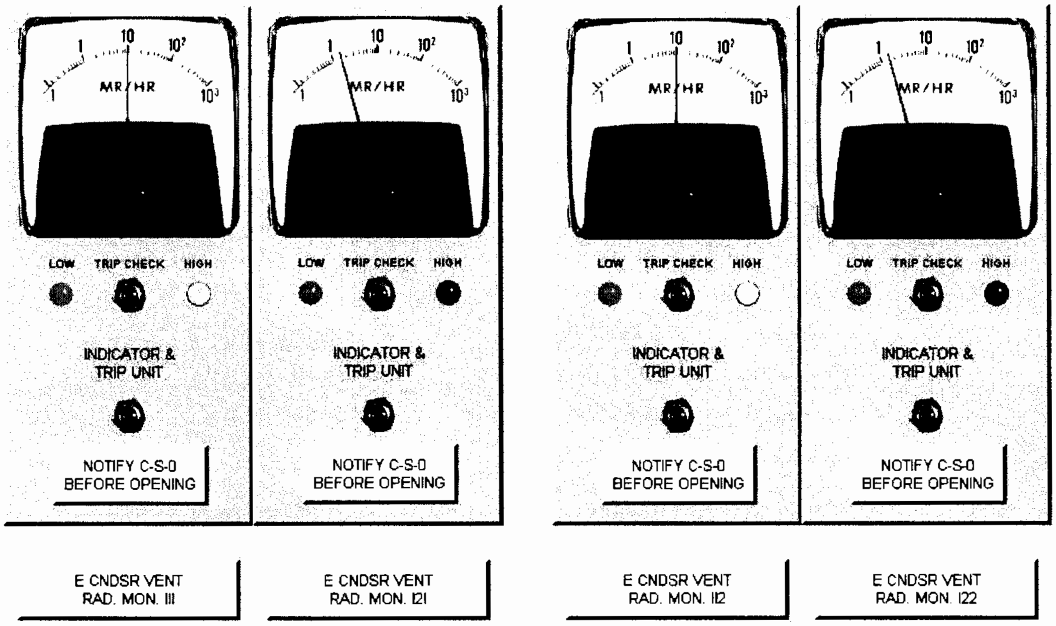
High Off-site Release Rate

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: #79

The plant has scrammed from 100% power with the following:

- The MSIVs are closed and CANNOT be re-opened.
- Emergency Condenser (EC) 12 is out of service for maintenance.
- Emergency Condenser 11 has been placed in service.
- Annunciator K1-1-2, EMER COND VENT 11 RAD MONITOR, is in alarm.
- The following indications are present for the EC vent radiation monitors:



Which one of the following describes the impact of these indications on operation of EC 11, in accordance with ARP K1-1-2?

A. These are valid indications of an EC 11 tube leak. Direct isolating EC 11.

B. One or more indications have failed. Direct RP to monitor dose rates at EC piping.

C. These indications are expected for an in service EC. Direct continued use of EC 11 unless vent radiation levels exceed a threshold of 30 mr/hr.

D. These indications are higher than expected for an in service EC, but do NOT indicate an EC 11 tube leak. Direct Chemistry to obtain an EC 11 shell side sample and maintain EC 11 in service unless vent radiation levels exceed a threshold of 30 mr/hr.

Proposed Answer: A

Explanation: The given indications show two EC vent radiation monitors in alarm high (111 and 112 – both associated with EC 11) and two EC vent radiation monitors slightly elevated above normal readings but below the high alarm setpoint (121 and 122 – both associated with EC 12). ARP K1-1-2 provides specific guidance on how to interpret EC vent radiation monitors alarms. With **both** EC 11 vent radiation monitors (111 and 112) in alarm high, this is classified as a valid EC tube leak and ARP K1-1-2 requires isolating EC 11 prior to any additional sampling.

Note: The question meets the K/A by presenting indications of an Emergency Condenser tube leak, which causes a direct release path from a primary system to the environment, and a higher than normal off-site release rate. The question requires the ability to perform a specific procedure, ARP K1-1-2.

B. Plausible – Two indications are markedly higher than the other two, however this is because the two high indications are on EC 11 while the two lower indications are on EC 12. ARP K1-1-2 does direct RP monitoring of EC piping dose rates in the event of monitor inoperability to satisfy ODCM requirements.
C. Plausible – A slight rise in EC vent radiation monitor reads may be expected during the EC is in service due to circulation of Reactor coolant through the EC tubes, however it is NOT expected that the vent radiation monitors would exceed the high alarm setpoint. 30 mr/hr is the EC vent radiation level requiring declaration of an Alert and entering N1-EOP-6, Radioactivity Release Control.
D. Plausible – These indications are higher than expected for an in service EC, but ARP K1-1-2 provides specific guidance that they are enough to deem this a valid EC tube leak. EC 11 must be isolated. ARP K1-1-2 does direct Chemistry sampling/assessment if a tube leak is suspected but not yet validated. 30 mr/hr is the EC vent radiation level requiring declaration of an Alert and entering N1-EOP-6, Radioactivity Release Control.

Technical Reference(s): ARP K1-1-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-10

Question Source: Bank – 2015 NRC #87

Question History: 2015 NRC #87

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 #	295025 2.4.8
	Importance Rating	4.5

High Reactor Pressure

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question: #80

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

- A Station Blackout occurs.
- All control rods insert.
- N1-SOP-33A.2, Station Blackout, is entered.
- N1-EOP-2, RPV Control, is entered due to low Reactor water level and high Reactor pressure.
- Both Emergency Condensers automatically initiate.

Which one of the following describes the required control of Reactor pressure?

A. Cycle the Emergency Condensers to stabilize Reactor pressure near the current value.

B. Cycle the Emergency Condensers to establish a cooldown, but maintain the rate less than 100°F/hr.

C. Maintain one Emergency Condenser in continuous service and secure the other Emergency Condenser.

D. Maintain both Emergency Condensers in continuous service.

Proposed Answer: D

Explanation: EOPs are higher tiered documents than SOPs, and therefore when a conflict exists between the direction in EOPs and SOPs, the EOP direction normally is followed. However, Reactor pressure control during a Station Blackout is a special case in which the SOP direction is followed over the normal EOP direction. Under normal N1-EOP-2 direction, the Emergency Condensers would need to be cycled in and out of service to maintain the cooldown rate less than 100°F/hr. However, since N1-SOP-33A.2 is the overriding document in this special case, the Emergency Condensers are required to be maintained in continuous service, even though the normal EOP cooldown rate is expected to be violated.

Note: The question meets the K/A by presenting a specific situation where Reactor pressure is initially higher than required (initial Reactor pressure of ~1000 psig, whereas Station Blackout requires much lower pressure to be established) and where coordination of varying SOP and EOP direction is critical to correct control of the plant.

A. Plausible – N1-SOP-33A.2 requires maintaining both ECs in continuous service. Plausible because there are conditions where Reactor pressure is stabilized (such as initial ATWS response) and stabilizing pressure would assist in stable level control.
B. Plausible – N1-SOP-33A.2 requires maintaining both ECs in continuous service. Plausible because this is the normal direction in N1-EOP-2
C. Plausible – N1-SOP-33A.2 requires maintaining both ECs in continuous service. Plausible because this would lower the cooldown rate while still conserving some amount of power.

Technical Reference(s): N1-SOP-33A.2, N1-EOP-2, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33A.2C01 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 #	295026 2.2.25
	Importance Rating	4.2

Suppression Pool High Water Temperature

Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

Proposed Question: #81

The plant is operating at 100% power with no surveillance testing in progress.

Which one of the following describes:

(1) the lowest Torus water temperature that would require a Reactor scram, in accordance with Technical Specifications, and

(2) the **Technical Specification** basis for restrictions on Torus water temperature?

	Temperature	Associated Basis
A.	110°F	Ensure post-accident system temperature and pressure remain within design limits.
B.	110°F	Ensure adequate net positive suction head for ECCS pumps.
C.	120°F	Ensure post-accident system temperature and pressure remain within design limits.
D.	120°F	Ensure adequate net positive suction head for ECCS pumps.

Proposed Answer: A

Explanation: Technical Specification 3.3.2.e requires a Reactor scram when Torus water temperature reaches 110°F. The higher 120°F limit requires depressurizing the Reactor. The basis for the Torus water temperature limits in Technical Specification 3.3.2 is to maintain post-accident system temperature and pressure within FSAR design limits.

B. Plausible – The basis for the Torus water temperature limits in Technical Specification 3.3.2 is to maintain post-accident system temperature and pressure within FSAR design limits. ECCS pump NPSH is affected by Torus water temperature and is addressed in the EOPs, but not the specific basis in Technical Specifications.
C. Plausible – A 120°F limit exists in TS 3.3.2.f which requires depressurizing the Reactor, however a lower limit of 110°F requires a Reactor scram per TS 3.3.2.e.
D. Plausible – A 120°F limit exists in TS 3.3.2.f which requires depressurizing the Reactor, however a lower limit of 110°F requires a Reactor scram per TS 3.3.2.e. The basis for the Torus water temperature limits in Technical Specification 3.3.2 is to maintain post-accident system temperature and pressure within FSAR design limits. ECCS pump NPSH is affected by Torus water temperature and addressed in the EOPs, but not the specific basis in Technical Specifications.

Technical Reference(s): Technical Specification 3.3.2 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 2.2.36
	Importance Rating	4.2

Partial or Complete Loss of DC Power

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

Proposed Question: #82

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

- Static Battery Charger (SBC) 161A is unavailable due to maintenance.
- SBC 161B trips due to a sustained electrical fault.
- Electrical Maintenance is troubleshooting the condition and reports the expected restoration time is unknown.

Which one of the following describes the required action(s), in accordance with N1-OP-47A, 125 VDC Power System, and Technical Specifications?

A. Restore either battery charger to service or place the plant in a cold shutdown condition within 24 hours.

B. Place MG 167 in Battery Charger mode and continue plant operations until the SBC is restored to service.

C. Restore either battery charger to service or place the plant in a cold shutdown condition within 10 hours.

D. Restore either battery charger to service within 24 hours or then reduce Reactor pressure to less than 110 psig within 10 hours.

Proposed Answer: D

Explanation: The loss of SBC 161B with SBC 161A means no battery charger is available to Battery Board 11. Per OP-47A P&L 7, this is treated as a loss of the battery, requiring a 24 hour LCO for return of a SBC. If no SBC is returned within the 24 hours, TS 3.1.5 is entered, requiring reactor coolant pressure be reduced to 110 psig or less and reactor coolant temperature be reduced to saturation temperature or less within 10 hours.

__Note:__ The question meets SRO-level guidelines because it requires understanding of Technical Specification interpretation and application of required actions.

A. Plausible – A battery charger must be returned to service within 24 hours per T.S. 3.6.3.h or take the action required by T.S. 3.1.5, which requires reactor coolant pressure be reduced to 110 psig or less and reactor coolant temperature be reduced to saturation temperature or less within 10 hours.
B. Plausible – Although MG 167 can be used as a battery charger, it is not safety related and cannot be used to avoid entry into TS 3.6.3.h LCO.
C. Plausible – TS 3.1.5 requires reactor coolant pressure be reduced to 110 psig or less and reactor coolant temperature be reduced to saturation temperature or less within 10 hours. However, 24 hours are allowed by TS 3.6.3.h to restore a battery charger before entering the 10 hour requirement in TS 3.1.5.

Technical Reference(s): N1-OP-47A, Technical Specifications 3.1.5 and 3.6.3

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.5 and 3.6.3

Learning Objective: N1-263000-RBO-14

Question Source: Bank – 2010 NRC #77

Question History: 2010 NRC #77

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 #	295035 EA2.02
	Importance Rating	4.1

Secondary Containment High Differential Pressure

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Off-site release rate: Plant-Specific

Proposed Question: #83

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

- A seismic event occurs.
- An un-isolable Reactor Water Cleanup steam leak develops in the Reactor Building.
- Reactor Building Ventilation automatically isolates due to high exhaust radiation levels.
- Both Reactor Building Emergency Ventilation fans fail to start.
- Reactor Building differential pressure is +0.05 inches.
- Area Radiation Monitor #21, Cleanup Pump Area – RB 261, indicates 25 mr/hr and stable.
- Reactor Building Ventilation exhaust radiation indicates 6 mr/hr and stable.
- The Shift Manager (SM) has declared an Unusual Event due to the seismic event.
- Radiation Protection personnel have NOT yet begun on any offsite surveys in support of dose assessment.
- The SM is filling out the following section on 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*)		

Which one of the following identifies the letter to circle in this section, in accordance with EP-CE-114-100-F-05, NMP Notification Fact Sheet – Part 1?

A. A – No release

B. B – Release below federal limits (ODCM)

C. C – Release above federal limits (ODCM)

D. D – Unmonitored release requiring evaluation

Proposed Answer: D

Explanation: The Reactor Building functions as the Secondary Containment, however it is NOT designed for zero leakage. Negative pressure is required to prevent an unmonitored release from the Reactor Building. With positive Reactor Building pressure from loss of ventilation and a steam leak, air is leaking from the Reactor Building into the outside environment. This is an unmonitored release that requires additional evaluation (i.e. field surveys and dose assessment) in order to quantify.

A. Plausible – Although the release rate is not currently known, a release is in progress because the Reactor Building pressure is greater than outside pressure and the Reactor Building is NOT designed to be air tight.
B. Plausible – There is a release in progress, however it is through an unmonitored release pathway. Therefore, more evaluation is required to determine the status of the release with respect to limits.
C. Plausible – There is a release in progress, however it is through an unmonitored release pathway. Therefore, more evaluation is required to determine the status of the release with respect to limits.

Technical Reference(s): EP-CE-114-100-F-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP6C01 EO-3

Question Source: Modified Bank - SYSID 109122

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 #	2
	K/A #	295015 2.4.35
	Importance Rating	4.0

Incomplete SCRAM

Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.

Proposed Question: #84

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

- Multiple methods in N1-EOP-3.1, Alternate Control Rod Insertion, have been utilized.
- N1-EOP-3.1 Attachment 1, Scram Control Rods Electrically, has been performed.
- The Restoration section of N1-EOP-3.1 Attachment 1 has NOT been performed.
- N1-EOP-3.1 Attachment 3, Driving Control Rods Using Reactor Manual Control System, has been performed.
- 44-167 (CRD 12), Charging Water Header Blocking Valve, was closed to provide more drive pressure.
- The US has directed use of N1-EOP-3.1 Attachment 4, Scram Control Rods by Repeated Manual Scram Signals.

Which one of the following describes the required control of RPS fuses and 44-167 to allow use of repeated manual scram signals to insert control rods, in accordance with N1-EOP-3.1?

A. RPS fuses must be re-installed in the ON position per N1-EOP-3.1 Attachment 1 Restoration section. 44-167 must be re-opened per N1-EOP-3.1 Attachment 3 or 4.

B. 44-167 must be re-opened per N1-EOP-3.1 Attachment 3 or 4. RPS fuses do NOT need to be re-installed in the ON position.

C. RPS fuses must be re-installed in the ON position per N1-EOP-3.1 Attachment 1 Restoration section. 44-167 does NOT need to be re-opened.

D. RPS fuses do NOT need to be re-installed in the ON position and 44-167 does NOT need to be re-opened.

Proposed Answer: A

Explanation: N1-EOP-3.1 Attachment 1 has pulled RPS fuses and re-installed them in the OFF position. This de-energizes the scram group solenoids, which makes resetting the Reactor scram impossible. N1-EOP-3.1 Attachment 3 installs RPS jumpers to allow resetting the Reactor scram, however this does not actually allow resetting the Reactor scram with the RPS fuses still installed in OFF. To perform N1-EOP-3.1 Attachment 4, the Reactor scram must be able to be reset. Therefore, the RPS fuses must be re-installed in the ON position. This will require the US to also direct performance of N1-EOP-3.1 Attachment 1 Restoration section. Additionally, N1-EOP-3.1 Attachment 4 requires 44-167 to be re-opened to allow HCUs to be re-pressurized.

Note: The question meets SRO-level guidelines because it requires the applicant to assess multiple plant conditions, as well as previously performed procedure sections, to determine the necessity of performing additional procedure sub-sections in N1-EOP-3.1. The question meets the K/A by testing knowledge of local auxiliary operator tasks during an incomplete scram (specific field actions in N1-EOP-3.1) and the resultant operational effects (how these field actions affect use of another procedure section performed inside the Control Room).

B. Incorrect – RPS fuses must also be re-installed in the ON position. Plausible because N1-EOP-3.1 Attachments 3 and 4 install RPS jumpers, however this does not fully compensate for the RPS fuses being installed in OFF.
C. Incorrect – 44-167 must be re-opened. Plausible because the scram can be reset with 44-167 closed, it just won't have the desired effect on the HCUs.
D. Incorrect – RPS fuses must be re-installed in the ON position and 44-167 must be re-opened. Plausible because N1-EOP-3.1 Attachments 3 and 4 install RPS jumpers, however this does not fully compensate for the RPS fuses being installed in OFF. Also plausible because the scram can be reset with 44-167 closed, it just won't have the desired effect on the HCUs. Also plausible because not all attachments would require these actions (eg. Attachments 6 & 7 do not require these both of these actions to be successful).

Technical Reference(s): N1-EOP-3.1, C-19859-C Sheet 4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-12

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 #	2
	K/A #	295009 AA2.01
	Importance Rating	4.2

Low Reactor Water Level

Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water level

Proposed Question: #85

N1-EOP-2, RPV Control, contains the following step:

IF	THEN
RPV Blowdown (EOP-8) is anticipated	Rapidly depressurize the RPV using EC and the main turbine bypass valves. ⚠ **Hi/Lo – Lo/Lo Rosemounts may be unreliable following rapid depressurization below 500 psig.** ☛ OK to exceed 100°F/hr cooldown.

Given the following parameters approaching limits that will require RPV Blowdown:

(1) Low Torus water level
(2) Low Reactor water level

Which one of the following identifies the ability to anticipate RPV Blowdown and rapidly depressurize the RPV per this step, in accordance with OP-NM-101-111-1001, Transient Mitigation Guidelines?

	Low Torus Water Level	Low Reactor Water Level
A.	Rapid depressurization allowed	Rapid depressurization allowed
B.	Rapid depressurization allowed	Rapid depressurization NOT allowed
C.	Rapid depressurization NOT allowed	Rapid depressurization allowed
D.	Rapid depressurization NOT allowed	Rapid depressurization NOT allowed

Proposed Answer: B

Explanation: OP-NM-101-111-1001 provides guidance on how to implement the anticipatory Blowdown in N1-EOP-2. OP-NM-101-111-1001 specifically allows use of anticipatory Blowdown in the case of Torus water level lowering and specifically prohibits use of anticipatory Blowdown in the case of Reactor water level lowering. Low Reactor water level is called out as the lone case where anticipatory Blowdown is not appropriate since inventory conservation is the key to mitigating low Reactor water level.

A. Plausible – Anticipatory Blowdown is NOT authorized for low Reactor water level, but is for high Torus water level.
C. Plausible – Anticipatory Blowdown is NOT authorized for low Reactor water level, but is for high Torus water level.
D. Plausible – Anticipatory Blowdown is NOT authorized for low Reactor water level, but is for high Torus water level.

Technical Reference(s): N1-EOP-2, NER-1M-095, OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Bank – 2015 Cert #83

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 #	2
	Group #	1
	K/A #	300000 A2.01
	Importance Rating	2.8

Instrument Air

Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Air dryer and filter malfunctions

Proposed Question: #86

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

- Annunciator L1-3-7, INST AIR SYSTEM, is in alarm.
- Computer point A193, IA DRYERS OR FILTERS, is in alarm due to both high differential pressure across the filters and low pressure downstream of the filters.
- Annunciator F3-3-2, CRD CONTROL AIR PRESSURE HI-LO, is in alarm.
- Scram Air Header pressure indicates 65 psig and slowly lowering.
- Instrument Air Header pressure indicates 100 psig and stable in the Control Room.

Which one of the following describes the procedure to be entered and the required action?

	Procedure to Be Entered	Required Action
A.	N1-SOP-5.1, Loss of Control Rod Drive	Scram the Reactor now.
B.	N1-SOP-5.1, Loss of Control Rod Drive	Scram the Reactor if Scram Air Header pressure reaches 60 psig.
C.	N1-SOP-20.1, Instrument Air Failure	Scram the Reactor now.
D.	N1-SOP-20.1, Instrument Air Failure	Scram the Reactor if Scram Air Header pressure reaches 60 psig.

Proposed Answer: D

Explanation: N1-SOP-5.1 covers multiple different CRD system malfunctions, however lowering Instrument Air and Scram Air Header pressure requires entry into N1-SOP-20.1, not N1-SOP-5.1. N1-SOP-20.1 does not require a scram until Scram Air Header pressure lower to 60 psig. 70 psig is the scram threshold for Instrument Air Header pressure.

A. Plausible – Lowering Instrument Air and Scram Air Header pressure requires entry into N1-SOP-20.1, not N1-SOP-5.1. Plausible because N1-SOP-5.1 covers multiple different CRD system malfunctions, just not low Scram Air Header pressure.
B. Plausible – Lowering Instrument Air and Scram Air Header pressure requires entry into N1-SOP-20.1, not N1-SOP-5.1. Plausible because N1-SOP-5.1 covers multiple different CRD system malfunctions, just not low Scram Air Header pressure.
C. Plausible – N1-SOP-20.1 does not require a scram until Scram Air Header pressure lower to 60 psig. 70 psig is the scram threshold for Instrument Air Header pressure.

Technical Reference(s): N1-SOP-20.1, ARP F3-3-2, N1-SOP-5.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-10

Question Source: Bank – JAF 9/14 NRC #77

Question History: JAF 9/14 NRC #77

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 #	259002 A2.03
	Importance Rating	3.7

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 reactor water level input

Proposed Question: #87

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

- Feedwater pump 13 is running with its flow control valve in AUTO.
- Feedwater level control is selected to Level Column 12.
- Reactor water level transmitter 36-76, #11 Narrow Range GEMAC, fails upscale.

Which one of the following describes the need for LCO entry in Technical Specifications (TS) 3.1.8, High Pressure Coolant Injection, and 3.6.2, Protective Instrumentation, due to this failure?

	TS 3.1.8 LCO application is…	TS 3.6.2 LCO application is…
A.	required.	required.
B.	required.	NOT required.
C.	NOT required.	required.
D.	NOT required.	NOT required.

Proposed Answer: B

Explanation: #11 Narrow Range GEMAC is required to be operable for proper functioning of Feedwater flow control valve #11 in the HPCI mode. With this level transmitter failed high, Feedwater flow control valve #11 will remain closed when required to open for HPCI. This requires entry into a 15 day LCO in Technical Specification 3.1.8. Technical Specification 3.6.2 is impacted by other narrow range Reactor water level transmitters, such as the Hi/Lo Rosemounts (36-03 A-D) for the HPCI initiation logic, but is NOT affected by failure of #11 Narrow Range GEMAC (36-76). Therefore, LCO entry is NOT required in Technical Specification 3.6.2.

Note: The question meets both parts of the K/A by requiring the applicant to understand the effect of a failed Reactor water level instrument on the Feedwater Level Control system (and the Reactor Protection System) and using this understanding to determine the impact on Technical Specifications (which is requisite to correcting, controlling, and mitigating the consequences). The question meets SRO level guidelines by requiring closed-book knowledge of "below the line" Technical Specification information (TS 3.6.2 Table requirements) and Technical Specification bases information (ex. TS 3.1.8 bases that affect operability – need for 3420 gpm from both electric Feedwater pumps).

A. Plausible – Technical Specification 3.6.2 is impacted by other narrow range Reactor water level transmitters, such as the Hi/Lo Rosemounts (36-03 A-D) for the HPCI initiation logic, but is NOT affected by failure of #11 Narrow Range GEMAC (36-76). Therefore, LCO entry is NOT required in Technical Specification 3.6.2.
C. Plausible – #11 Narrow Range GEMAC is required to be operable for proper functioning of Feedwater flow control valve #11 in the HPCI mode. With this level transmitter failed high, Feedwater flow control valve #11 will remain closed when required to open for HPCI. This requires entry into a 15 day LCO in Technical Specification 3.1.8. Technical Specification 3.6.2 is impacted by other narrow range Reactor water level transmitters, such as the Hi/Lo Rosemounts (36-03 A-D) for the HPCI initiation logic, but is NOT affected by failure of #11 Narrow Range GEMAC (36-76). Therefore, LCO entry is NOT required in Technical Specification 3.6.2.
D. Plausible – #11 Narrow Range GEMAC is required to be operable for proper functioning of Feedwater flow control valve #11 in the HPCI mode. With this level transmitter failed high, Feedwater flow control valve #11 will remain closed when required to open for HPCI. This requires entry into a 15 day LCO in Technical Specification 3.1.8.

Technical Reference(s): Technical Specifications 3.1.8 and 3.6.2, P&ID C-18015-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-14

Question Source: Modified Bank – SYSID 59606

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 #	400000 2.1.28
	Importance Rating	4.1

Component Cooling Water

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

Proposed Question: #88

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

- N1-ST-Q13, Emergency Service Water Pump Operability Test, is in progress.
- Then, Emergency Service Water pump 11 trips due to a fault in the motor windings.

Which one of the following describes the Technical Specification impact, if any, in accordance with N1-OP-18, Service Water System, and Technical Specifications?

A. NO LCO entry is required.

B. LCO entry is required in Technical Specification 3.4.4, Emergency Ventilation.

C. LCO entry is required in Technical Specification 3.4.5, Control Room Ventilation.

D. LCO entry is required in Technical Specification 3.1.6, Control Rod Drive Pump Coolant Injection.

Proposed Answer: C

Explanation: There is no Technical Specification (TS) specifically addressing Emergency Service Water. However, N1-OP-18 Precaution and Limitation #2 states, "Loss of one train of Emergency Service Water while in the power operating condition requires entry into TS 3.4.5.e CREVS LCO, which is the most limiting LCO for a system supported by ESW."

Note: The question meets the K/A by requiring knowledge of ESW system function/purpose (systems supported by ESW) and the associated impact on TS implementation (which supported system has the most limiting LCO and is declared inoperable when ESW cannot perform its function). This is SRO-level knowledge because it cannot be answered solely by knowing ≤1 hour limits, "above the line" information, or safety limits, plus it requires knowledge of TS bases/interpretation provided in N1-OP-18.

A. Plausible – While there is no TS specifically addressing Emergency Service Water, N1-OP-18 requires entering an LCO in TS 3.4.5 due to the effect on CREVS operability.
B. Plausible – TS 3.4.5 is applicable, not TS 3.4.4.
D. Plausible – TS 3.4.5 is applicable, not TS 3.1.6. CRD pumps are cooled by RBCLC, which is cooled by ESW if SW is lost. However, ESW operability is not required to meet TS 3.1.6.

Technical Reference(s): N1-OP-18

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-276000-RBO-14

Question Source: Modified Bank – 2013 Cert #79

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	239002 2.4.41
	Importance Rating	4.6

SRVs

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #89

The plant is operating at 100% power when spurious isolation of both Main Steam lines results in the following:

- The Reactor Mode Switch has been placed in SHUTDOWN.
- The manual scram pushbuttons have been depressed.
- ARI has been initiated.
- Multiple control rods remain withdrawn.
- APRMs are unavailable.
- One Emergency Cooling loop is in service.
- The other Emergency Cooling loop has failed to initiate.
- One ERV is open.
- Reactor pressure is 1080 psig and stable.
- Reactor water level is 72" and stable.

Which one of the following describes the required control of Reactor water level, in accordance with the Emergency Operating Procedures, and the highest Emergency Action Level (EAL) that is met or exceeded?

Reactor water level...

A. must be lowered. The highest EAL met or exceeded is an Alert.

B. must be lowered. The highest EAL met or exceeded is a Site Area Emergency.

C. may be maintained at the current level. The highest EAL met or exceeded is an Alert.

D. may be maintained at the current level. The highest EAL met or exceeded is a Site Area Emergency.

Proposed Answer: B

Explanation: One loop of Emergency Cooling can remove heat equivalent to approximately 3-6% of Reactor power. By design, the heat removal rate is $19x10^7$ Btu/hr, which equates to 55 MW or 3% of Reactor power. Per in-service testing of Emergency Condensers, the actual heat removal rate was closer to 6% of Reactor power. One ERV is designed to pass 540,910 lb/hr at 1120 psig, which equates to approximately 7.5% of Reactor power. Combined, one Emergency Cooling loop and one ERV are removing heat equivalent to approximately 10.5-13.5% of Reactor power. Since Reactor pressure is steady, this is the approximate value of Reactor power.

N1-EOP-3, Failure to Scram, entry is required. Steps L-5 and L-9 require intentionally lowering Reactor water level since Reactor power is above 6% and Reactor water level is currently above -41".

Alert SA3.1 is NOT met since an automatic scram (MSIV position) failed to shut down the Reactor but manual action (Mode Switch in SHUTDOWN) has NOT shut down the Reactor. Site Area Emergency SS3.1 is met since manual action has NOT shut down the Reactor as indicated by Reactor power >6%.

A. Plausible – The highest EAL met is SAE SS3.1, not an Alert such as SA3.1. EAL SA3.1 is plausible if the applicant fails to determine the approximate power level based on the conditions given.
C. Plausible – Reactor water level must be lowered since Reactor power can be determined to be >6% due to operation of ECs and ERVs. The highest EAL met is SAE SS3.1, not an Alert such as SA3.1. EAL SA3.1 and maintaining water level are both plausible if the applicant fails to determine the approximate power level based on the conditions given.
D. Plausible – Reactor water level must be lowered since Reactor power can be determined to be >6% due to operation of ECs and ERVs.

Technical Reference(s): N1-EOP-3, EAL Matrix

Proposed references to be provided to applicants during examination: EAL Matrix

Learning Objective: 1101-EOP-3C01 EO-5

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 #	1
	K/A #	218000 A2.04
	Importance Rating	4.2

ADS

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS failure to initiate

Proposed Question: #90

A loss of coolant accident has occurred with the following conditions:

- Reactor water level is -15" and slowly lowering.
- Drywell pressure is 8 psig and slowly rising.
- The crew has taken NO action with respect to ADS.
- The ADS timer has timed out.
- The crew notices the automatic ADS initiation and that only ERV 113 opens.

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

A. Open two additional ERVs, in accordance with N1-EOP-8, RPV Blowdown.

B. Open three additional ERVs, in accordance with N1-EOP-8, RPV Blowdown.

C. Depress the ADS Reset pushbuttons, in accordance with N1-EOP-2, RPV Control.

D. Place the ADS Inhibit Keylock Bypass switches in BYPASS, in accordance with N1-EOP-2, RPV Control.

Proposed Answer: D

Explanation: The current combination of Reactor water level (<-10") and Drywell pressure (>3.5 psig) satisfy the ADS logic, therefore the ADS actuation is appropriate. However, N1-EOP-2 requires overriding ADS and only manually initiating an RPV Blowdown if Reactor water level cannot be restored and maintained >-84". Since Reactor water level is still well above -84", and RPV Blowdown is not yet required by N1-EOP-2. N1-EOP-2 requires bypassing ADS by placing the ADS Inhibit Keylock Bypass switches in BYPASS.

Note: The question meets the K/A by testing the aspect of the K/A statement requiring the highest cognitive level (the (b) part). Based on the nature of the K/A, the (a) part could not be directly tested without lowering the plausibility of two distractors (predicting the impact on ADS of "ADS failure to initiate" is rather obvious and low level for an SRO question).

A. Plausible – Although the ADS logic is satisfied by the current combination of Reactor water level and Drywell pressure, N1-EOP-2 requires bypassing ADS by placing the ADS Inhibit Keylock Bypass switches in BYPASS. Two ERVs have failed to open with the ADS actuation, however bypass is still required.
B. Plausible – Although the ADS logic is satisfied by the current combination of Reactor water level and Drywell pressure, N1-EOP-2 requires bypassing ADS by placing the ADS Inhibit Keylock Bypass switches in BYPASS. Three additional ERVs would need to be opened if an RPV Blowdown were required.
C. Plausible – Depressing the ADS Reset pushbuttons would temporarily result in closing the ERV and is allowed by ARP F2-3-1. However, N1-EOP-2 requires bypassing ADS by placing the ADS Inhibit Keylock Bypass switches in BYPASS.

Technical Reference(s): N1-EOP-2, N1-EOP-8, ARP F2-3-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-12

Question Source: Modified Bank – 2008 NRC #87

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 A2.02
	Importance Rating	2.9

RWM

Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of steam flow input: P-Spec(Not-BWR6)

Proposed Question: #91

A Reactor startup is in progress with the following:

- Only the first 8 control rods have been withdrawn.
- The steam flow input to the Rod Worth Minimizer (RWM) has failed downscale.
- The last startup performed with an inoperable Rod Worth Minimizer was on January 15, 2016.
- Today is March 1, 2017.

Which one of the following describes the impact of the loss of steam flow input on the RWM and the plant startup in accordance with N1-OP-37 and Technical Specifications?

The RWM will…

A. enforce the required rod blocks. The RWM may be bypassed once Reactor power reaches 10% of rated.

B. enforce the required rod blocks. The RWM must be left in service until the steam flow input is restored.

C. NOT enforce the required rod blocks. The startup must be placed on hold with the last known rod pattern maintained until the RWM can be restored.

D. NOT enforce the required rod blocks. The startup may continue only if a second qualified individual is stationed to verify compliance with the approved rod withdrawal sequence.

Proposed Answer: A

Explanation: Both steam flow and feedwater flow provide inputs to the RWM as indirect measurements of Reactor power.

- On decreasing power, either the steam flow input or the feedwater flow input will trip the low power setpoint (approximately 10% reactor power) to enable the RWM.
- On increasing power, both steam flow and feedwater flow inputs are required to disable the RWM above the low power setpoint (LPSP).

With the steam flow input failed low, the RWM will be enabled and fully operable to enforce the required rod blocks at the present low Reactor power level. The RWM will NOT automatically disable above the LPSP. Technical Specification 3.1.1 requires the RWM to be operable to enforce rod blocks below 10% of rated Reactor power, but does not require automatic enabling/disabling at 10% of rated Reactor power. Technical Specification 3.1.1 allows the RWM to be bypassed once at or above 10% of rated Reactor power.

B. Plausible – TS 3.1.1 does not require the automatic enabling/disabling feature of the RWM to be available. The RWM may be bypassed as soon as at or above 10% of rated Reactor power. Plausible because bypassing the RWM will prevent it from automatically coming back into service in the future when Reactor power goes below 10%. Also plausible because the RWM may be left in service, it is just not required to be left in service.
C. Plausible – Loss of the steam flow input affects the ability of the RWM to automatically enable/disable, but does not prevent the RWM from monitoring the rod pattern and enforcing rod blocks as needed. Additionally, TS 3.1.1 does not require the automatic enabling/disabling feature of the RWM to be available. Plausible because other RWM input losses would prevent proper enforcing of rod blocks. Also plausible because the 2nd half would be correct if a startup with an inoperable RWM had been performed this calendar year.
D. Plausible – Loss of the steam flow input affects the ability of the RWM to automatically enable/disable, but does not prevent the RWM from monitoring the rod pattern and enforcing rod blocks as needed. Additionally, TS 3.1.1 does not require the automatic enabling/disabling feature of the RWM to be available. Plausible because other RWM input losses would prevent proper enforcing of rod blocks. Also plausible because the 2nd half would be correct if the RWM was inoperable.

Technical Reference(s): N1-OP-37, Technical Specification 3.1.1

Proposed references to be provided to applicants during examination: Technical Specification 3.1.1 w/o bases

Learning Objective: N1-201003-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 #	2
	K/A #	271000 2.2.22
	Importance Rating	4.7

Off-gas

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #92

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

- Annunciator H1-2-7, OFF GAS RAD MON 11-12 FILTER DP SAMPLE FLOW, alarms.
- Computer point C139, OFFGAS SAMPLER FLOW, indicates LOW.
- Investigation reveals that the alarm is valid and sample flow is completely blocked.

Note: A portion of drawing C-18010-C, Main Condenser Air Removal & Off Gas System P&I Diagram, is provided on the following page.

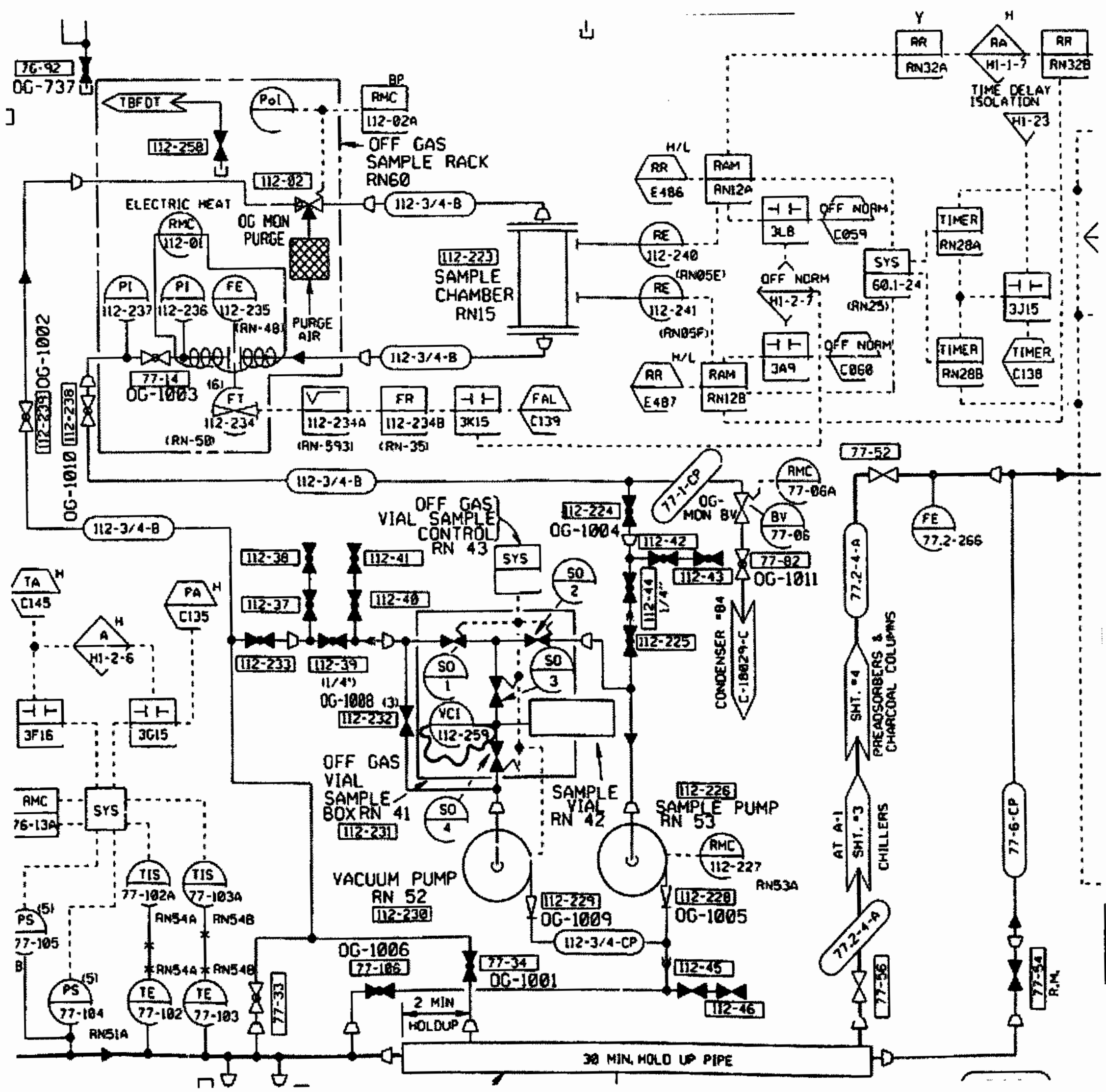
Considering the following two items:

- Technical Specification 3.6.2, Protective Instrumentation
- Offsite Dose Calculation Manual 3.6.14, Radioactive Effluent Instrumentation

Which one of the following describes the impact of this failure on compliance with these Technical Specification and Offsite Dose Calculation Manual sections?

	Technical Specification 3.6.2	Offsite Dose Calculation Manual 3.6.14
A.	LCO entry required	DLCO entry required
B.	LCO entry required	DLCO entry NOT required
C.	LCO entry NOT required	DLCO entry required
D.	LCO entry NOT required	DLCO entry NOT required

From C-18010-C, Main Condenser Air Removal & Off Gas System P&I Diagram:



Proposed Answer: C

Explanation: The given conditions make both Offgas radiation monitors RN-12A and RN-12B inoperable, since they both require flow through this sample path for operability. Both TS 3.6.2 and ODCM 3.6.14 contain requirements for radiation monitors with protective functions. TS 3.6.2 contains requirements for both Vacuum Pump isolation and Emergency Ventilation initiation, however neither of these functions are affected by operability of RN-12A and RN-12B, so TS 3.6.2 LCO entry is NOT required. ODCM 3.6.14 does require both of these radiation monitors to be operable, therefore DLCO 3.6.14 entry is required.

Note: The question meets SRO level guidelines by requiring analysis/knowledge of "below the line" Technical Specification information and Technical Specification bases information (which protective functions are required by which Technical Specification and how low flow condition affects operability).

Note: The question meets the K/A by testing ability to determine/interpret availability of Offgas radiation monitors, which are important to safety due to their function in plant operation and required operability per the ODCM.

A. Plausible – TS 3.6.2 contains requirements for both Vacuum Pump isolation and Emergency Ventilation initiation, however neither of these functions are affected by operability of RN-12A and RN-12B, so TS 3.6.2 LCO entry is NOT required.
B. Plausible – TS 3.6.2 contains requirements for both Vacuum Pump isolation and Emergency Ventilation initiation, however neither of these functions are affected by operability of RN-12A and RN-12B, so TS 3.6.2 LCO entry is NOT required. ODCM 3.6.14 does require both RN-12A and RN-12B to be operable, therefore DLCO 3.6.14 entry is required.
D. Plausible – ODCM 3.6.14 does require both RN-12A and RN-12B to be operable, therefore DLCO 3.6.14 entry is required.

Technical Reference(s): ARP H1-2-7, C-18010-C, Technical Specification 3.6.2, ODCM Section 3.6.14

Proposed references to be provided to applicants during examination: Technical Specification 3.6.2 (with allowable values removed and without bases), ODCM Section 3.6.14 (without bases)

Learning Objective: N1-271000-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 #	2
	K/A #	286000 A2.11
	Importance Rating	3.2

Fire Protection

Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Pump trips: Plant-Specific

Proposed Question: #93

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

- The Electric Fire Pump is being operated per N1-PM-M9, Fire Protection System – Monthly Operation of Fire Pumps.
- The Electric Fire Pump trips due to a sustained electrical fault on the motor.
- The Diesel Fire Pump has been verified operable.

Which one of the following describes the associated requirement for fire pump operability, in accordance with OP-NM-201-105, Compensatory Measures For Inoperable Fire Protection Systems and Components?

A. Restore the Electric Fire Pump to operable status within a maximum of 24 hours, or provide an alternate backup pump or supply.

B. Restore the Electric Fire Pump to operable status within a maximum of 7 days, or provide an alternate backup pump or supply.

C. Restore the Electric Fire Pump to operable status within a maximum of 31 days, or provide an alternate backup pump or supply.

D. There is no time limit for restoration of the Electric Fire Pump to operable status as long as the Diesel Fire Pump remains operable.

Proposed Answer: B

Explanation: NFPA 805 requires two high-pressure pumps, each with a capacity of 2500 gpm (i.e. both the Electric Fire Pump and the Diesel Fire Pump). OP-NM-201-105, Compensatory Measures For Inoperable Fire Protection Systems and Components, Section 4.3.1 requires, with an inoperable fire pump, it must be restored to operable within a maximum of 7 days, or provide an alternate backup pump or supply.

A. Plausible – 24 hours is the time limit if both the Electric and Diesel Fire Pumps are inoperable.
C. Plausible – 31 days is the required surveillance frequency for running the Electric Fire Pump, however only 7 days are allowed to restore operability.
D. Plausible – NFPA 805 requires both the Electric Fire Pump and the Diesel Fire Pump to normally be operable. No credit is given for the low-capacity Jockey Pumps.

Technical Reference(s): NFPA 805 Appendix H, section H.4, OP-NM-201-105 section 4.3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-286000-RBO-14

Question Source: Modified Bank – 2015 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 #	2.1.32
Importance Rating	4.0

Ability to explain and apply all system limits and precautions.

Proposed Question: #94

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

- Condenser Inlet temperature is 35°F.
- Circulating Water Tempering is in progress.
- The Circulating Water system has been operating in normal flow for six hours.
- Computer Point C685, CIRC WTR TEMP RISE, indicates the actual ΔT between the Condenser Inlet and the Plant Discharge is 37°F.

Which one of the following describes the requirements for Condenser ΔT?

Condenser ΔT is...

A. above the limit for these conditions. Lower Reactor power at least one percent to meet the Technical Specification limit.

B. above the limit for these conditions. Lower reactor power at least one percent to meet the State Pollutant Discharge Elimination System limits.

C. allowed as high as 47.2°F for these conditions, but must be returned to below 35°F within one hour to remain within the Technical Specification limit.

D. allowed as high as 47.2°F for these conditions, but must be returned to below 35°F within one hour to remain within the State Pollutant Discharge Elimination System limits.

Proposed Answer: B

Explanation: In accordance with N1-OP-19 P&L #1, the maximum ΔT allowable between the Condenser inlet and discharge tunnel is 35°F (SPDES Permit Limit). During the winter season when inlet icing may occur, additional temperature limits may apply as follows:

- The ΔT limitation may be exceeded by 35% or (12.25°F), (for a Maximum ΔT limit of 47.2°F) for no more than one hour during each reverse flow or return to normal operation.

However since there has been no reverse flow for six hours, the 35°F limitation may not be exceeded.

N1-OP-19 section H.7.0 directs reducing Reactor power at least 1% in response to violation of SPDES limits.

Note: Per NUREG 1021, this Tier 3 question is required to maintain focus on plant-wide generic knowledge and abilities and not become an extension of Tier 2, "Plant Systems." Yet the K/A also requires testing a ***system*** precaution and limitation. In order to best balance these requirements, a precaution and limitation is being tested that:

(1) Involves a system that is not explicitly tested in Tier 2, and
(2) Fits with other generic K/As (ex. 2.2.38).

A. Plausible – Limits on Condenser ΔT are contained in the State Pollutant Discharge Elimination System (SPDES) permit. Tech Specs contain Radioactive discharge limits, but not thermal discharge limits.
C. Plausible – Since there has been no reverse flow for six hours, the 35°F ΔT limitation may not be exceeded. Limits on Condenser ΔT are contained in the State Pollutant Discharge Elimination System (SPDES) permit. Tech Specs contain Radioactive discharge limits, but not thermal discharge limits.
D. Plausible – Since there has been no reverse flow for six hours, the 35°F ΔT limitation may not be exceeded.

Technical Reference(s): N1-OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-275000-RBO-9

Question Source: Bank – 2010 Cert #100

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.40
	Importance Rating	4.5

Ability to apply technical specifications for a system.

Proposed Question: #95

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

- It is discovered that a surveillance has not been completed on time for a Technical Specification required system.
- The surveillance frequency is 7 days.
- The surveillance was last performed 11 days ago.
- A risk evaluation has been performed and the associated impact is being managed.

Which one of the following describes the status of the system, in accordance with Technical Specifications?

A. The associated LCO must be declared NOT met at this time.

B. Complete the surveillance within a maximum of 24 hours from the time of discovery or then the associated LCO must be declared NOT met.

C. Complete the surveillance within a maximum of 2 days from the time of discovery or then the associated LCO must be declared NOT met.

D. Complete the surveillance within a maximum of 7 days from the time of discovery or then the associated LCO must be declared NOT met.

Proposed Answer: D

Explanation: Surveillance Requirement (SR) 4.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, 7 days), whichever is greater. Therefore up to 7 days are allowed to perform the missed surveillance before being required to declare the LCO not met.

A. Plausible – SR 4.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met.
B. Plausible – SR 4.0.3 allows the longer of 24 hours or the specified frequency (7 days).
C. Plausible – SR 4.0.3 allows the longer of 24 hours or the specified frequency (7 days). 2 days is based on the grace period (25% of 7 days) allowed by SR 3.0.2.

Technical Reference(s): Technical Specification Surveillance Requirement 4.0.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-TECHSPEC-CE-05

Question Source: Bank – JAF 9/14 NRC #99

Question History: JAF 9/14 NRC #99

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 #	2.3.13
	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.

Proposed Question: #96

Refueling is in progress with the following:

- An irradiated fuel bundle is grappled and is being removed from the Reactor core.
- The Refuel Bridge team observes that Spent Fuel Pool and Reactor cavity water level is lowering.
- Spent Fuel Pool and Reactor cavity water level has lowered approximately one foot below the normal level.
- The Refuel Bridge radiation monitors indicate 25 mr/hr and slowly rising.
- The Control Room enters N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal.
- The irradiated fuel bundle is currently half way out of the Reactor core.

Which one of the following describes the required direction to be given in accordance with N1-SOP-6.1?

Direct evacuation of...

A. all personnel from the Refuel Floor, leaving the irradiated fuel bundle in the current position.

B. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to transfer the irradiated fuel bundle to the Spent Fuel Pool.

C. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to lower the irradiated fuel bundle back into the original Reactor core location.

D. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to halt fuel movement and leave the irradiated fuel bundle in the current position.

Proposed Answer: C

Explanation: N1-SOP-6.1 entry is required due to reported or observed loss of SFP/Reactor cavity inventory. A conditional override step requires evacuation of all personnel from the Refuel Floor if either irradiated fuel is uncovered or the Refuel Bridge high range radiation monitor alarms. Neither of these conditions are met, as evidenced by water level being one foot below normal level (therefore fuel bundle in SFP are covered by over 20 feet of water still) and Refuel Bridge radiation monitors indicating 20 mr/hr and rising (well below the 50 mr/hr and 1000 mr/hr nominal high alarm setpoints). A subsequent step in N1-SOP-6.1 requires evacuating non-essential personnel from the Refuel Floor and returning any core component being transferred to the nearest storage location in the SFP or Reactor core. Since the irradiated fuel bundle is only half way out of the Reactor core, the nearest storage location is back in the Reactor core.

A. Plausible – If conditions were worse, such that either irradiated fuel was uncovered or the Refuel Bridge high radiation alarm sounded, immediate evacuation of the Refuel Floor would be the required action in N1-SOP-6.1.
B. Plausible – N1-SOP-6.1 requires evacuating non-essential personnel from the Refuel Floor and returning any core component being transferred to the nearest storage location in the SFP or Reactor core. Since the irradiated fuel bundle is only half way out of the Reactor core, the nearest storage location is back in the Reactor core. If the irradiated fuel bundle had already been transferred closer to the SFP, that would be the correct storage location to continue to. Additionally, going to the SFP instead of the Reactor core does eliminate a positive reactivity addition, which is desirable in certain situations.
D. Plausible – N1-SOP-6.1 requires evacuating non-essential personnel from the Refuel Floor and returning any core component being transferred to the nearest storage location in the SFP or Reactor core. Since the irradiated fuel bundle is only half way out of the Reactor core, the nearest storage location is back in the Reactor core. Leaving the irradiated fuel bundle in the current position does keep it covered with more water than going to the SFP and eliminates a positive reactivity addition associated with re-inserting it into the Reactor core.

Technical Reference(s): N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP6.1CO1 EO-2

Question Source: Bank – 2015 NRC #92

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.30
	Importance Rating	4.1

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Proposed Question: #97

Which one of the following describes when initial Protective Action Recommendations (PARs) are made to the County and State, in accordance with EP-CE-114-100, Emergency Notifications?

Initial PARs are required to be made to the County and State within a maximum time limit of...

A. 15 minutes after declaration of a General Emergency.

B. 30 minutes after declaration of a General Emergency.

C. 15 minutes after the offsite release rate exceeds the Technical Specification limit.

D. 30 minutes after the offsite release rate exceeds the Technical Specification limit.

Proposed Answer: A

Explanation: EP-CE-114-100, Emergency Notifications, requires initial notifications to the State Watch Center and County Emergency Operations Centers be made within 15 minutes of emergency classification for any change in the station's PARs. Therefore, upon declaration of a General Emergency, the station has up to 15 minutes to make the required notifications regarding the specific PARs.

B. Plausible –PAR notifications are required to be made within a maximum of 15 minutes after declaration of a General Emergency. 30 minutes is based on the time for follow-up notifications to the County and State.
C. Plausible –PAR notifications are required to be made within a maximum of 15 minutes after declaration of a General Emergency, not 15 minutes after the offsite release rate exceeds the Technical Specification limit.
D. Plausible –PAR notifications are required to be made within a maximum of 15 minutes after declaration of a General Emergency, not 15 minutes after the offsite release rate exceeds the Technical Specification limit. 30 minutes is based on the time for follow-up notifications to the County and State.

Technical Reference(s): EP-CE-114-100

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPL001-TO-01

Question Source: Bank – JAF 4/14 NRC #97

Question History: JAF 4/14 NRC #97

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments:

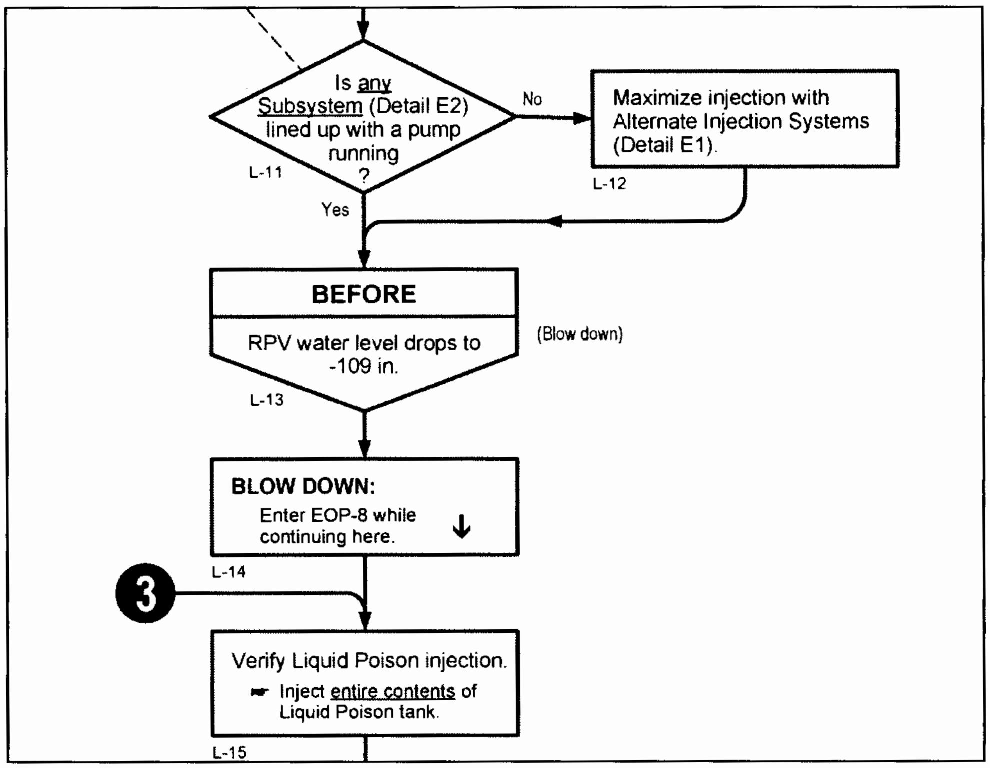
Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	
K/A #	2.3.14
Importance Rating	3.8

Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: #98

Given the following series of steps from N1-EOP-2, RPV Control:



Which one of the following describes the basis for injecting Liquid Poison per Step L-15 of N1-EOP-2?

A. Provide additional injection to the Reactor vessel.

B. Provide additional negative reactivity to the Reactor core.

C. Minimize radioactive release by buffering Suppression Pool pH.

D. Minimize radioactive release by plating out fission products inside the Reactor vessel.

Proposed Answer: C

Explanation: Design basis analyses credit Liquid Poison injection for limiting the radiological dose following loss of coolant accidents involving core damage. Radiation induced reactions are predicted to convert large fractions of dissolved ionic iodine into elemental iodine and organic iodides which can escape into the containment atmosphere. The rate of these reactions is strongly dependent on suppression pool pH. If the pH is maintained greater than 7, very little of the dissolved iodine will be converted to volatile forms and most of the iodine fission products will be retained in the suppression pool. Over time, the pH in the torus will tend to decrease due to the addition of acidic chemicals. The sodium pentaborate solution used in the Liquid Poison system is derived from a strong base and therefore raises suppression pool pH.

A. Plausible – Establishing Liquid Poison in Step L-15 is not as a means for level control but as a means to control radiological dose following a loss of coolant accident involving core damage. Since Liquid Poison is identified as an Alternate Injection System it would likely be started to augment RPV injection in a different step of the Level branch, before RPV water level reaches the top of the active fuel (Element L-16). This is a plausible distractor for those candidates that do not recognize the radiological impact from Liquid Poison injection once the TAF has been reached.
B. Plausible – Injecting Liquid Poison does insert negative reactivity and is the basis for injection in N1-EOP-3. However, the basis for Step L-15 is to control radioactive release by buffering Suppression Pool pH, not inserting negative reactivity.
D. Plausible – Plating out of boron and plating out of fission products inside the Reactor vessel following an accident are both concepts discussed in the EOP Bases, but are not the specific bases for Step L-15.

Technical Reference(s): N1-EOP-2, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Bank – 2009 NRC #98

Question History: 2009 NRC #98

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments:

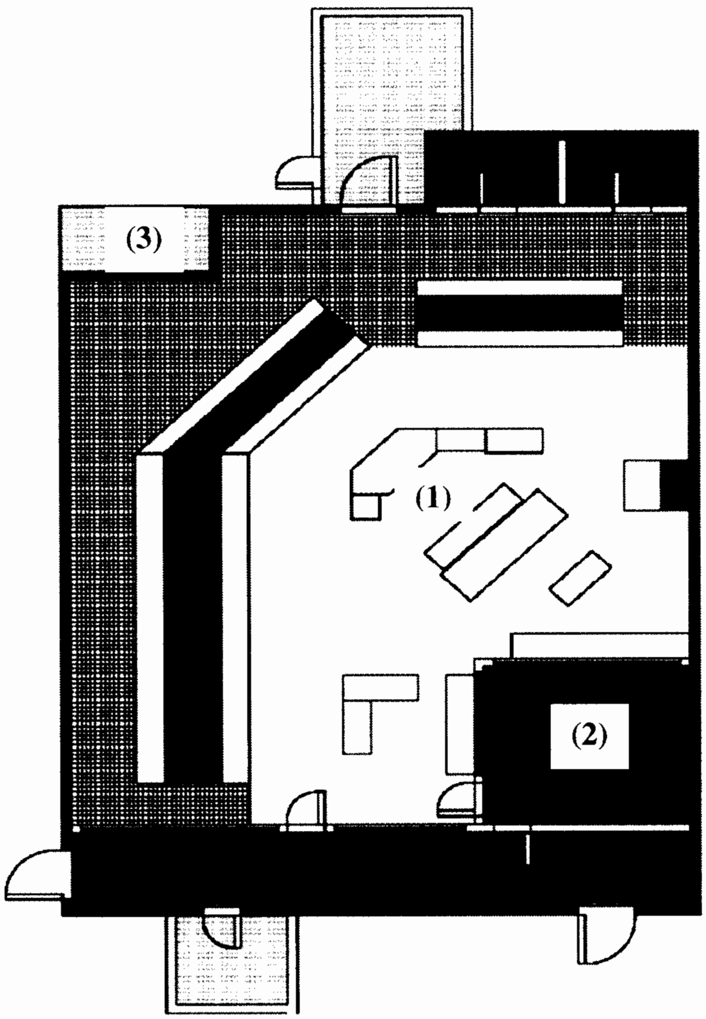
Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.4
	Importance Rating	3.8

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: #99

The plant is operating at 100% power.

Given the following map of the Main Control Room, with three locations marked (1), (2), and (3):



Which one of the following identifies which of these areas Unit Supervisor may access, without relief or special authorization, in accordance with OP-NM-103-102, Watch-Standby Practices at Nine Mile Point?

A. (1) only

B. (1) and (2) only

C. (1) and (3) only

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

Proposed Answer: B

Explanation: Location (1) is within the at-the-controls area and accessible to the Unit Supervisor. Location (2) is in the Shift Manager's office and accessible to the Unit Supervisor. Location (3) is in the stairwell to the Auxiliary Control Room and NOT accessible to the Unit Supervisor.

Note: The question meets the K/A by testing an aspect of an SROs individual licensed operator responsibilities related to shift staffing (physical limits of where they can be when fulfilling the role of Unit Supervisor). Additionally, there is specific OE at Nine Mile Point regarding deficiencies in understanding of these responsibilities.

A. Plausible – Location (2) is in the Shift Manager's office. This location is off limits to the At-The-Controls Reactor Operator, but the Unit Supervisor is allowed here.
C. Plausible – Location (2) is in the Shift Manager's office. This location is off limits to the At-The-Controls Reactor Operator, but the Unit Supervisor is allowed here. Location (3) is in the stairwell to the Auxiliary Control Room and NOT accessible to the Unit Supervisor, but the Balance of Plant Reactor Operator is allowed to go there.
D. Plausible – Location (3) is in the stairwell to the Auxiliary Control Room and NOT accessible to the Unit Supervisor, but the Balance of Plant Reactor Operator is allowed to go there..

Technical Reference(s): OP-NM-103-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – SSES LOC26R NRC #66

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 #	2.2.23
	Importance Rating	4.6

Ability to track Technical Specification limiting conditions for operations.

Proposed Question: #100

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

- Current time is 0800 on March 1, 2017.
- Chemistry reports the following data from their weekly isotopic Reactor water sample surveillance:
  - 10.55 μCi/gm Total Iodine
  - 2.23 μCi/gm Dose Equivalent Iodine-131

Which one of the following identifies the latest time that the plant is required to be placed in cold shutdown if actions to correct this condition are unsuccessful, in accordance with Technical Specification 3.2.4, Reactor Coolant Specific Activity?

A. 2000 on March 1, 2017

B. 2000 on March 2, 2017

C. 2000 on March 3, 2017

D. 2000 on March 4, 2017

Proposed Answer: D

Explanation: With Reactor coolant specific activity >0.2 μCi/gm and ≤4.0 μCi/gm Dose Equivalent I-131, Technical Specification 3.2.4.b must be entered. This requires restoring activity ≤0.2 μCi/gm Dose Equivalent I-131 within 48 hours (by 0800 on March 3, 2017). If actions to correct this condition are unsuccessful, Technical Specification 3.2.4.c is then entered. This requires placing the plant in hot shutdown within 12 hours (2000 on March 3, 2017) and then in cold shutdown within the following 24 hours (2000 on March 4, 2017).

A. Plausible – Cold shutdown would not be required until 2000 on March 4, 2017. 2000 on March 1, 2017 is the time hot shutdown would be required if activity were >4.0 μCi/gm Dose Equivalent I-131.
B. Plausible – Cold shutdown would not be required until 2000 on March 4, 2017. 2000 on March 2, 2017 is the time cold shutdown would be required if activity were >4.0 μCi/gm Dose Equivalent I-131.
C. Plausible – Cold shutdown would not be required until 2000 on March 4, 2017. 2000 on March 3, 2017 is the time hot shutdown is required for the given activity.

Technical Reference(s): Technical Specification 3.2.4

Proposed references to be provided to applicants during examination: Technical Specification 3.2.4

Learning Objective: N1-101001-RBO-14

Question Source: Modified Bank – 2008 NRC #82

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments: