Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 K1.01
	Importance Rating	3.6

Source Range Monitor

Knowledge of the physical connections and/or cause-effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: Reactor protection system

Proposed Question: #1

A plant startup is in progress. Conditions are as follows:

- The Reactor Mode Switch is in STARTUP.
- The non-coincidence shorting links are installed.
- All IRMs are indicating mid-scale on Range 5.
- All SRMs indicate between 1000 and 5000 cps.

Then, SRM C fails upscale.

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

	Rod Blocks	RPS
A.	A rod OUT block is received, only.	No scram occurs.
B.	A rod OUT block is received, only.	A full scram occurs.
C.	A rod IN and OUT block is received.	No scram occurs.
D.	A rod IN and OUT block is received.	A full scram occurs.

Proposed Answer: A

Explanation: With IRMs on less than range 8, the SRM upscale rod out block is enabled. With the non-coincidence shorting links installed, the SRM scram function is bypassed. When SRM C fails upscale, a rod out block is generated. No rod in block is generated and no full scram is generated.

B. Plausible – This would be true if the shorting links were removed.
C. Plausible – Based on candidate misconception that SRMs provide a rod in block
D. Plausible – This would be true if the shorting links were removed and that SRMs provided a rod in block.

Technical Reference(s): N2-OP-92 Sections D.4.0 and SRM Lesson Plan Objective 5.

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215002-RBO-05

Question Source: Modified – SYSID 33378

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 K1.02
	Importance Rating	3.2

Component Cooling Water

Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: Loads cooled by CCWS

Proposed Question: #2

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

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

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

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

Proposed Answer: B

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-208000-RBO-8

Question Source: Modified Bank – System ID 33365

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 #	212000 K2.01
	Importance Rating	3.2

RPS

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

Proposed Question: #3

Which one of the following power supply failures would result in loss of power to RPS MG Set 2RPM-MG1B?

A. 2NJS-US1

B. 2NJS-US4

C. 2NJS-US5

D. 2NJS-US6

Proposed Answer: D

Explanation: 2NJS-US6 supplies power to 2NHS-MCC009 which supplies power to the RPS MG Set 2RPM-MG1B drive motor.

A. Plausible – This is the alternate AC power supply to RPS A scram solenoids.
B. Plausible – This is the alternate AC power supply to RPS B scram solenoids.
C. Plausible – This is the power supply to RPS MG Set 2RPM-MG1A drive motor.

Technical Reference(s): RPS Big Note, EE-001AL

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-4

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 #	262001 K2.01
	Importance Rating	3.3

AC Electrical Distribution

Knowledge of electrical power supplies to the following: Off-site sources of power

Proposed Question: #4

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

- Division 3 is lined up to Line 5.
- An offsite power surge causes Annunciator 852536, 4KV BUS NNS 017 SPLY ACB 17-2 AUTO TRIP / FTC to alarm.

Which one of the following describes the plant AC bus(es) that immediately de-energize(s)?

A. 2ENS*SWG103 (Div. 2)

B. 2ENS*SWG101 and 2ENS*SWG102 (Div. 1 & 3)

C. 2ENS*SWG101 (Div. 1) and 2NNS-SWG014 (Stub Bus)

D. 2ENS*SWG103 (Div. 2) and 2NNS-SWG015 (Stub Bus)

Proposed Answer: A

Explanation: NNS-SWG017 is the normal supply for this bus from offsite power Line 6.

B. Plausible – 2ENS*SWG101 and 2ENS*SWG102 (Div 1 & 3) are normally supplied from NNS-SWG016 from offsite power Line 5.
C. Plausible – 2ENS*SWG101 (Div. 1) is normally supplied from NNS-SWG016 and NNS-SWG014 (Stub Bus) is normally supplied from the NPS-SWG003.
D. Plausible – Although 2ENS*SWG103 (Div. 2) is powered from NNS-SWG017, NNS-SWG015 (Stub Bus) is normally supplied from NPS-SWG003.

Technical Reference(s): ARP 852536

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264000-RBO-3

Question Source: Bank – 2012 NRC #4

Question History: 2012 NRC #4

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 #	264000 K3.02
	Importance Rating	3.9

EDGs

Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: A.C. electrical distribution

Proposed Question: #5

A loss of offsite power and loss of coolant accident has occurred. Conditions are as follows:

- Drywell pressure is 5 psig and slowly rising.
- All EDGs have started and are powering their respective busses.

Then, the following conditions occur:

- A lube oil leak causes 2EGS*EG2 lube oil pressure to lower to 5 psig.
- A governor failure causes 2EGS*EG3 speed to rise at 10 rpm per second.

Which one of the following describes the resulting status of 2ENS*SWG102 and 2ENS*SWG103 ten (10) seconds later?

	2ENS*SWG102	2ENS*SWG103
A.	Energized	Energized
B.	Energized	De-energized
C.	De-energized	Energized
D.	De-energized	De-energized

Proposed Answer: B

Explanation: 2EGS*EG2 and 2EGS*EG3 are initially powering 2ENS*SWG102 and 2ENS*SWG103, respectively, due to a loss of offsite power. Additionally, the EDGs are running with a LOCA signal (Drywell pressure > 1.68 psig), therefore numerous trips are bypassed. 2EGS*EG2 lube oil pressure is below the normal trip setpoint of 16 psig, however this trip is bypassed due to the LOCA signal. Therefore, 2EGS*EG2 remains running and powering 2ENS*SWG102. 2EGS*EG3 overspeed trip setpoint is 660 rpm and this trip is NOT bypassed by the LOCA signal. Therefore, 2EGS*EG3 trips after 6 seconds and 2ENS*SWG103 de-energizes.

A. Plausible – 2ENS*SWG103 de-energizes because 2EGS*EG3 trips on overspeed.
C. Plausible – 2ENS*SWG102 remains energized because the 2EGS*EG2 low lube oil pressure trip is bypassed due to the LOCA signal. 2ENS*SWG103 de-energizes because 2EGS*EG3 trips on overspeed.
D. Plausible – 2ENS*SWG102 remains energized because the 2EGS*EG2 low lube oil pressure trip is bypassed due to the LOCA signal.

Technical Reference(s): ARP 852225 and 852303

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-11, N2-264002-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

02072014: Changed “after 10 seconds” to “ten (10) seconds later”. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K3.07
	Importance Rating	3.2

APRM / LPRM

Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: Rod block monitor: Plant-Specific

Proposed Question: #6

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

- Annunciator 603202, APRM TRIP SYSTEM UPSCALE / INOPERABLE, alarms.
- Computer point NMPBC14, APRM CHANNEL 1 INOP, is received.
- Operators bypass APRM 1.

Which one of the following describes the resulting status of Rod Block Monitor (RBM) A?

RBM A...

A. is inoperable and enforces a control rod block.

B. automatically transferred to an alternate APRM input.

C. is unaffected because it does NOT receive an input from APRM 1.

D. is bypassed until it is manually transferred to an alternate APRM input.

Proposed Answer: B

Explanation: RBM A normally receives a Simulated Thermal Power input from APRM 1. When APRM 1 is bypassed, RBM A automatically receives an alternate Simulated Thermal Power input from APRM 3.

A. Plausible – RBM A automatically transferred to APRM 3 or 4 and therefore remains operable and does not enforce a rod block.
C. Plausible – RBM A does receive an input from APRM 1. It is RBM B that does not receive an input from APRM 1.
D. Plausible – The RBM is neither bypassed by the same joystick as its primary APRM input nor automatically bypassed with an INOP APRM input.

Technical Reference(s): RBM Lesson Plan, Objective 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-2

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 #	203000 K4.07
	Importance Rating	3.7

RHR/LPCI: Injection Mode

Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Emergency generator load sequencing

Proposed Question: #7

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

- A loss of ALL offsite power has just occurred.
- Concurrently, a steam leak in the Drywell has resulted in a Drywell pressure of 2.5 psig and rising rapidly.
- All Emergency Diesel Generators (EDGs) automatically start.
- All EDG output breakers close simultaneously

Which one of the following describes the approximate time for the sequencing of the low pressure ECCS pumps onto the 4.16KV emergency switchgears after the EDG output breakers are closed?

	RHR A & B	CSL & RHR C
A.	1 second	6 seconds
B.	1 second	10 seconds
C.	5 seconds	6 seconds
D.	5 seconds	10 seconds

Proposed Answer: A

Explanation: Due to the presence of combined LOOP and LOCA (Drywell pressure > 1.68 psig) signals, RHS*P1A and RHS*P1B start 1 second after the EDG output breakers are closed. CSL*P1 and RHS*P1C start 5 seconds later (6 seconds from closure of the EDG output breaker).

B. Plausible – CSL*P1 and RHS*P1C start 6 seconds after EDG output breaker closure. 10 seconds is the normal start time with no LOOP.
C. Plausible – RHS*P1A and RHS*P1B start 1 second after EDG output breaker closure. 5 seconds is the normal start time with no LOOP.
D. Plausible – RHS*P1A and RHS*P1B start 1 second after EDG output breaker closure. 5 seconds is the normal start time with no LOOP. CSL*P1 and RHS*P1C start 6 seconds after EDG output breaker closure. 10 seconds is the normal start time with no LOOP.

Technical Reference(s): UFSAR Tables 8.3-1 and 8.3-2, ESK-5ENS21, Sheet 2, ESK-5ENS22 (Sheet 2)

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264000-RBO-5

Question Source: Bank – SYSID 32383

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 #	215003 K4.05
	Importance Rating	2.9

IRM

Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Changing detector position

Proposed Question: #8

A plant startup is being conducted in accordance with N2-OP-101A. Conditions are as follows:

- All SRMs are reading between 5 x 10E4 and 9 x 10E4 cps.
- All IRMs are mid-scale on range 1.
- The operator has selected all the SRMs and the IRMs for withdraw.

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

A. IRM Downscale rod block

B. SRM Downscale rod block

C. IRM Detector NOT Fully Inserted rod block

D. SRM Detector NOT Fully Inserted rod block

Proposed Answer: C

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

A. Plausible – Although IRMs will eventually go downscale, this rod block is bypassed with the IRMs on Range 1.
B. Plausible – Although SRMs will move downscale, this will take time as the detectors withdraw. Immediately upon start of withdraw, IRMs will cause a rod block on detector position. Therefore SRM downscale would not be the first protective action of those listed.
D. Plausible – SRM detectors not being fully inserted only results in a rod block if the detectors also read < 100 cps. With the given initial count rates, if would take significant time for the detectors to move enough for counts to lower to 100 cps. Immediately upon start of withdraw, IRMs will cause a rod block on detector position. Therefore SRM detector position would not be the first protective action of those listed.

Technical Reference(s): N2-OP-96 Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-5

Question Source: Bank – 2010 NRC #10

Question History: 2010 NRC #10

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 #	1
	K/A #	300000 K5.13
	Importance Rating	2.9

Instrument Air

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

Proposed Question: #9

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

- Annunciator 603306, CRD SCRAM VALVE PILOT AIR HDR PRESS HIGH/LOW, alarms.
- 2IAS-TK3, RB Air Receiver, pressure is 120 psig and stable.
- An Operator reports the Scram Air Header pressure is 63 psig and stable.
- NO control rods are drifting.

Which one of the following statements describes the action that is to be attempted to restore Scram Air Header pressure, per N2-SOP-19, Loss of Instrument Air?

A. Verify IAS Compressors are loaded and bypass IAS Dryers.

B. Swap Scram Air Header Supply Filters and Pressure Control Valves.

C. Bypass Scram Air Header Supply Filters and Pressure Control Valves.

D. Verify IAS Compressors are loaded and isolate Service Air Header.

Proposed Answer: B

Explanation: Per N2-SOP-19, it is required to swap Scram Air Header Supply Filters and Pressure Control Valves.

A. Plausible – This would be true for Dryer Valve malfunction.
C. Plausible – This is not procedural. Scram air header is required to be filtered and regulated at 70 - 75 psig.
D. Plausible – This would be true if IA Header Pressure was below 85 psig, due to a Service Air Header rupture.

Technical Reference(s): N2-SOP-19 Attachment 3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-RBO-10

Question Source: Bank – 2012 NRC #11

Question History: 2012 NRC #11

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 #	259002 K5.01
	Importance Rating	3.1

Reactor Water Level Control

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: GEMAC/Foxboro/Bailey controller operation: Plant-Specific

Proposed Question: #10

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

- Reactor water level is 183" and stable.
- The Feedwater master controller, 2FWS-HIC1600, is in AUTO.
- The individual Feedwater level control valve controllers, 2FWS-HIC1010A and 2FWS-HIC1010B, are in AUTO.

Then, Reactor power is lowered to 90% and plant conditions stabilize.

Which one of the following describes the effect of depressing the Feedwater master controller, 2FWS-HIC1600, MAN pushbutton at this time?

Feedwater flow will ___(1)___. Feedwater flow may be adjusted by depressing the ___(2)___ OPEN/CLOSE pushbuttons.

A. (1) rapidly rise
(2) 2FWS-HIC1600

B. (1) rapidly rise
(2) 2FWS-HIC1010A or 2FWS-HIC1010B

C. (1) remain near the current value
(2) 2FWS-HIC1600

D. (1) remain near the current value
(2) 2FWS-HIC1010A or 2FWS-HIC1010B

Proposed Answer: C

Explanation: Although Reactor power has been lowered by 10%, 2FWS-HIC1600 manual signal automatically tracks the current Feedwater flow. This feature allows a smooth transfer while placing the controller in MAN without any need for controller nulling. A small mismatch between steam and feed flow at the time of transfer may necessitate manual adjustments. In the given conditions, this would be performed by depressing the OPEN/CLOSE pushbuttons on 2FWS-HIC1600. The OPEN/CLOSE pushbuttons on 2FWS-HIC1010A or 2FWS-HIC1010B will not cause a flow change while those controllers are still in AUTO.

A. Plausible – Feedwater flow will remain near the current value, because 2FWS-HIC1600 manual signal automatically tracks the auto signal.
B. Plausible – Feedwater flow will remain near the current value, because 2FWS-HIC1600 manual signal automatically tracks the auto signal. The OPEN/CLOSE pushbuttons on 2FWS-HIC1010A or 2FWS-HIC1010B will not cause a flow change while those controllers are still in AUTO.
D. Plausible – The OPEN/CLOSE pushbuttons on 2FWS-HIC1010A or 2FWS-HIC1010B will not cause a flow change while those controllers are still in AUTO.

Technical Reference(s): N2-OP-3 Section F.8.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-259002-RBO-5

Question Source: Modified – SYSID 105205

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 #	223002 K6.02
	Importance Rating	3.0

PCIS/Nuclear Steam Supply Shutoff

Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF: D.C. electrical distribution

Proposed Question: #11

Which one of the following power supply losses would cause a closure of the Group 8 and 9 INBOARD isolation valves?

A. 2BYS-SWG001A

B. 2BYS-SWG001B

C. 2BYS*SWG002A

D. 2BYS*SWG002B

Proposed Answer: D

Explanation: Loss of 2BYS*SWG002B causes a portion of the PCIS logic to de-energize. The specific logic causes closure of just the inboard Group 8 and 9 isolation valves.

A. Plausible – Loss of 2BYS-SWG001A causes a loss of position indication to Division 1 Group 8 CCP valves, however they would not cause an actual isolation.
B. Plausible – Loss of 2BYS-SWG001B causes a loss of position indication to Division 2 Group 8 CCP valves, however they would not cause an actual isolation.
C. Plausible – Loss of 2BYS*SWG002A causes closure of Group 8 and 9 outboard isolation valves.

Technical Reference(s): N2-SOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223002-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 #	262002 K6.02
	Importance Rating	2.8

UPS (AC/DC)

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

Proposed Question: #12

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

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

With regards to UPS3A/B, which one of the following describes the effect of these electrical losses?

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

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

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

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

Proposed Answer: C

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

A. Plausible – UPS-3A automatically receives power from the maintenance power supply, 2NJS-US5, therefore it does not de-energize.
B. Plausible – UPS-3B remains energized from the normal AC power source, 2NJS-US4, which is unaffected by the other electrical losses.
D. Plausible – UPS-3B remains energized from the normal AC power source, 2NJS-US4, which is unaffected by the other electrical losses.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262002-RBO-4

Question Source: 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 #	218000 A1.04
	Importance Rating	4.1

ADS

Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: Reactor pressure

Proposed Question: #13

The plant has experienced a failure to scram. Conditions are as follows:

- The DIV I ADS AUTOMATIC INITIATION DISABLE switch has been taken to ON.
- The DIV II ADS AUTOMATIC INITIATION DISABLE switch could not be operated and is stuck in OFF.

Then...

- 2CEC*PNL601 and 603 are terminated and prevented and Reactor water level is lowered to 5 inches.
- Reactor pressure is 920 psig and remains stable on the Turbine Bypass Valves.

Which one of the following describes the plant response two minutes later?

Reactor pressure is...

A. stable and still being controlled on the Turbine Bypass Valves.

B. rapidly lowering with only the 'A' ADS solenoids energized.

C. rapidly lowering with only the 'B' ADS solenoids energized.

D. rapidly lowering with both the 'A' and 'B' ADS solenoids energized.

Proposed Answer: C

Explanation: With the Division II ADS Automatic Initiation Disable switch in OFF, the Division II ADS system is capable of automatically blowing down the RPV with the Division II ADS Solenoids (B solenoids). Since PNL601 has been terminated and prevented and level is less than Level 1, both RHS A and B are running. With these conditions met, after 105 seconds the Division II ADS system will actuate and rapidly lower Reactor pressure with the B solenoids.

A. Plausible – Division II ADS system is still available to blowdown the Reactor, therefore Reactor pressure will be rapidly lowering and Turbine Bypass Valves will no longer have control of Reactor pressure.
B. Plausible – Since the Division I ADS Initiation Switch is in ON, the Division I ADS system cannot automatically blow down the RPV so the A solenoids are not energized.
D. Plausible – Since the Division I ADS Initiation Switch is in ON, the Division I ADS system cannot automatically blow down the RPV so the A solenoids are not energized.

Technical Reference(s): N2-OP-34 Section B.2.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-5

Question Source: Bank – 2012 Audit #5

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

02072014: Added the word “being” to distracter A. Moved “two minutes later” to right before the question mark. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A1.07
	Importance Rating	4.3

SLC

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: Reactor power

Proposed Question: #14

The plant has experienced a failure to scram. Conditions are as follows:

- Reactor power is 8%
- At time = 0 minutes, Standby Liquid Control (SLC) Pump 1A and 1B keylock switches are placed in RUN.
- Initial SLC tank level was 4600 gallons.

Which one of the following describes the earliest approximate time range at which enough boron will be injected to the Reactor to ensure that the Reactor will be shutdown under all conditions, in accordance with the EOP Bases and N2-EOP-C5, Failure to Scram?

A. 15 to 20 minutes

B. 25 to 30 minutes

C. 35 to 40 minutes

D. 65 to 75 minutes

Proposed Answer: C

Explanation: With both SLC pumps injecting, the flow rate is approximately 88 gpm. Enough boron will be injected to ensure that the Reactor will be shutdown under all conditions per EOP-C5 once SLC tank level lowers to 1450 gallons (Cold Shutdown Boron Weight). The time required for this amount of boron injection can be calculated by:

$$Time = \frac{4600\ gallons - 1450\ gallons}{88\ gpm} = 35.8\ minutes$$

Therefore, the earliest time range is 35-40 minutes.

Note: This question matches the K/A because it requires the candidate to predict the effect of a given SLC control manipulation on Reactor shutdown criteria. Reactor shutdown criteria directly relates to actual and future Reactor power.

A. Plausible – Enough boron is not injected until ~35 minutes. This time range could be calculated if 1450 gallons was used as the total amount of boron solution to inject with a flow rate of 88 gpm.
B. Plausible – Enough boron is not injected until ~35 minutes. This time range could be calculated if ~2300 gallons (roughly the Hot Shutdown Boron Weight) was used as the total amount of boron solution to inject with a flow rate of 88 gpm.
D. Plausible – Enough boron was injected ~35 minutes, therefore this is not the earliest time range. This time range could be calculated if a flow rate of 44 gpm was used (single SLC pump injection rate).

Technical Reference(s): N2-EOP-C5, N2-OP-36A section F.1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-211000-RBO-10

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 #	205000 A2.09
	Importance Rating	3.6

Shutdown Cooling

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

Proposed Question: #15

The plant is in Mode 3 with RHS B in Shutdown Cooling (SDC). Conditions are as follows:

- A transient occurs.
- Reactor water level drops to a low of 130 inches.
- Reactor pressure rises to a high of 115 psig.

After the transient, conditions are as follows:

- Reactor water level is 150 inches and very slowly rising.
- Reactor pressure is 115 psig and stable.

Which one of the following describes the status of SDC and, if necessary, the action required to mitigate the transient?

SDC...

A. remains in service.

B. has isolated, but the isolation may be reset under the current conditions.

C. has isolated. Reactor pressure must be lowered before the isolation can be reset.

D. has isolated. Reactor water level must be raised before the isolation can be reset.

Proposed Answer: D

Explanation: SDC will isolate on a high Reactor pressure of 128 psig (nominal, 148 psig by TS) or a low Reactor water level of 159.3 inches (nominal, 157.8 inches by TS). In the given conditions, SDC isolated due to low Reactor water level, and still has an active isolation signal due to continued low Reactor water level. Therefore, Reactor water level must be raised before resetting the isolation.

A. Plausible – SDC isolates because Reactor water level went below 159.3 inches.
B. Plausible – SDC isolation cannot be reset because Reactor water level is still below 159.3 inches.
C. Plausible – Reactor pressure is below 128 psig, therefore it is not causing a SDC isolation signal.

Technical Reference(s): TS Table 3.3.6.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-5

Question Source: Modified – SYSID 33683

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

02072014: Changed "Current" to "After the transient". Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209002 A2.02
	Importance Rating	3.6

HPCS

Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Pump trips: BWR-5,6

Proposed Question: #16

The plant is at 100% power when the CSH water leg pump (WTR LEG PMP 2) trips on motor electric fault.

Which one of the following describes the operational status of CSH and an appropriate mitigating action for the pump trip, in accordance with N2-OP-33, High Pressure Core Spray System?

	CSH Operational Status	Mitigating Action
A.	Operable	Line up a Condensate Transfer pump to directly pressurize the CSH piping.
B.	Operable	Verify Condensate Storage Tank 1B level is ≥ 47 feet.
C.	Inoperable	Line up a Condensate Transfer pump to directly pressurize the CSH piping.
D.	Inoperable	Verify Condensate Storage Tank 1B level is ≥ 47 feet.

Proposed Answer: D

Explanation: With the water leg pump 2 out of service, the HPCS system is inoperable. Per N2-OP-33, in order for it to remain available, CST Tank 1B level must remain above 47 feet.

A. Plausible – Trip of WTR LEG PMP 2 makes HPCS inoperable. Condensate Transfer pumps should not be used to keep HPCS piping pressurized due to potential to over-pressurize the CNS piping upon HPCS initiation.
B. Plausible – Trip of WTR LEG PMP 2 makes HPCS inoperable.
C. Plausible – Condensate Transfer pumps should not be used to keep HPCS piping pressurized due to potential to over-pressurize the CNS piping upon HPCS initiation.

Technical Reference(s): N2-OP-33 Sections D.1.1, D.1.6, and H.10

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209002-RBO-10

Question Source: Modified – SYSID 33319

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

02072014: Rearranged stem by deleting the bullet and making one sentence at the beginning of the stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 A3.01
	Importance Rating	3.6

LPCS

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: Valve operation

Proposed Question: #17

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

- Drywell pressure is 5 psig and rising.
- Reactor pressure is 600 psig and lowering.
- Annunciator 601416, LPCS SYSTEM MOTOR OVERLOAD, has alarmed.
- Computer points show 2CSL*MOV104, LPCS INJ VALVE, has a motor overload condition.

Which one of the following is correct with regards to the operation of 2CSL*MOV104?

As Reactor pressure lowers to the LPCS injection setpoint, 2CSL*MOV104 will...

A. automatically open. The valve is then throttleable from 2CEC*PNL601.

B. automatically open. The valve will NOT be throttleable from 2CEC*PNL601.

C. not automatically open and it can only be manually opened locally at the valve.

D. not automatically open, however it can be manually opened using the control switch on 2CEC*PNL601.

Proposed Answer: A

Explanation: The motor overload feature for the LPCS injection valve is designed to allow the valve to automatically open if a LOCA signal is present and the DP interlock is met.

B. Plausible – The motor overload condition could have been designed to allow fulfillment of safety function by opening, but then protect motor by not allowing throttling.
C. Plausible – Most valve motor overload conditions will cause the associated valve to not operate electrically and the proper course of action would then be to locally operate the valve. However the LPCS injection valve has a special motor overload feature which will allow it to automatically operate regardless of the motor overload condition.
D. Plausible – If a LOCA signal was not present, the LPCS injection valve would not automatically open. However with a motor overload condition, the valve could still be operated with the control switch on 2CEC*PNL601.

Technical Reference(s): ARP 601416

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 A3.01
	Importance Rating	3.2

DC Electrical Distribution

Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights

Proposed Question: #18

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

- Annunciator 852108, DIV I EMER BUS BYS 002A 125VDC SYSTEM TROUBLE, is in alarm.
- The next page of the exam gives the indications that are present at the rear of 2CEC*PNL852:

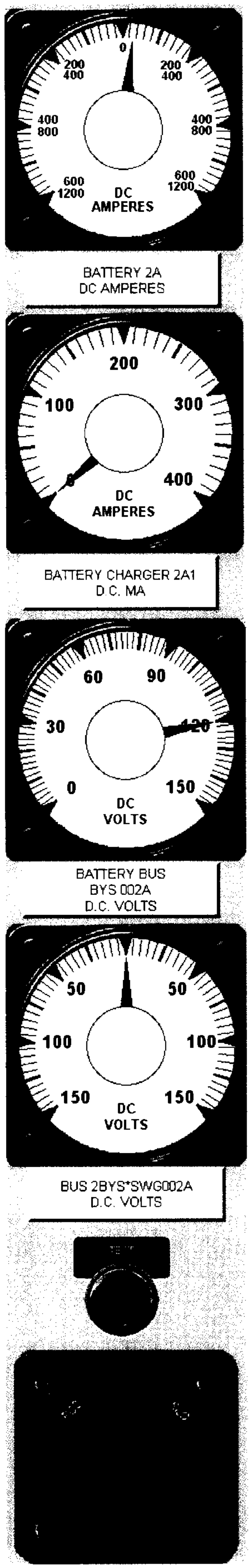
Which one of the following describes these indications?

A. A loss of battery has occurred.

B. Battery Cell Reversal is occurring.

C. A loss of battery charger has occurred.

D. A loss of 2BYS*SWG002A has occurred.


0
200 400
200 400
400 800
400 800
600 1200
600 1200
DC AMPERES
BATTERY 2A
DC AMPERES
200
100
300
0
400
DC AMPERES
BATTERY CHARGER 2A1
D.C. MA
60
90
30
120
0
150
DC VOLTS
BATTERY BUS
BYS 002A
D.C. VOLTS
50
50
100
100
150
150
DC VOLTS
BUS 2BYS*SWG002A
D.C. VOLTS

Proposed Answer: C

Explanation: The given indications show that battery charger output is zero amps and the battery is discharging. This indicates that a loss of the battery charger has occurred.

A. Plausible – A loss of battery would be indicated by 0 amps on the Battery Amp meter
B. Plausible – Cell reversal would occur at <105 VDC per P&L 12.0.
D. Plausible – The meter labeled BUS 2BYS*SWG002A D.C. VOLTS is the Ground Meter. This meter normally reads 0 unless grounds are being measured. This is a common misconception for licensed operators.

Technical Reference(s): ARP 852108

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 A4.06
	Importance Rating	3.3

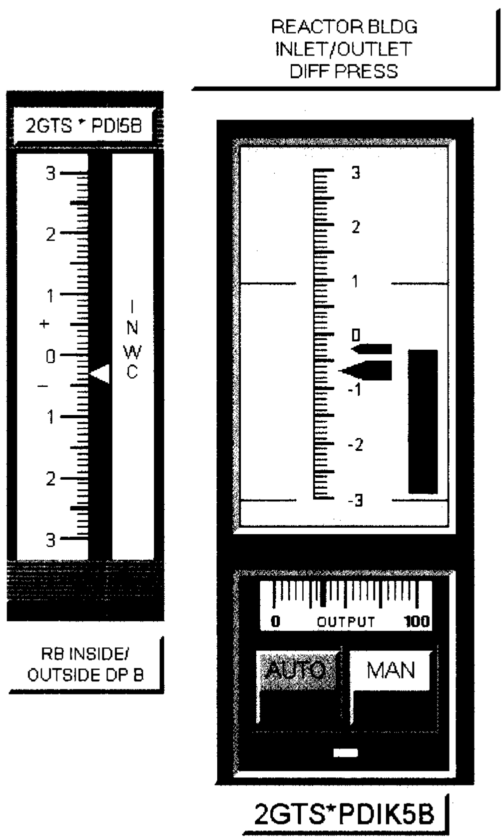
SGTS

Ability to manually operate and/or monitor in the control room: Reactor building differential pressure

Proposed Question: #19

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

- GTS Train B has been manually started and Reactor Building Ventilation has been secured due to turbulent atmospheric conditions.
- The Shift Manager has directed 2GTS*PV5B, FAN RECIRC VALVE, be placed in manual control due to inability of PV5B to automatically maintain Reactor Building differential pressure (RB D/P).
- Indications for RB D/P and 2GTS*PV5B are as follows:



Which one of the following describes the status of RB D/P and the action necessary to correct it?

RB D/P should be maintained __(1)__ negative. Adjust the slider on 2GTS*PDIK5B to further __(2)__ 2GTS*PV5B.

	(1)	(2)
A.	less	close
B.	less	open
C.	more	close
D.	more	open

Proposed Answer: C

Explanation: Per N2-OP-61B, Section H.6.0, with RB D/P less negative than -0.35 in WG action must be taken to make RB D/P more negative. With the PV5B controller in manual, in order to make RB D/P more negative, you must close down on PV5B. In order to close down on PV5B, the slider on the controller must be taken to the left.

A. Plausible – RB D/P is not negative enough.
B. Plausible – RB D/P is not negative enough.
D. Plausible – This would be true if the candidate did not understand how PV5B works. By shutting PV5B, you make RB D/P more negative, by opening PV5B, you make RB D/P less negative.

Technical Reference(s): N2-OP-61B, Section H.6.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-261000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 A4.09
	Importance Rating	3.7

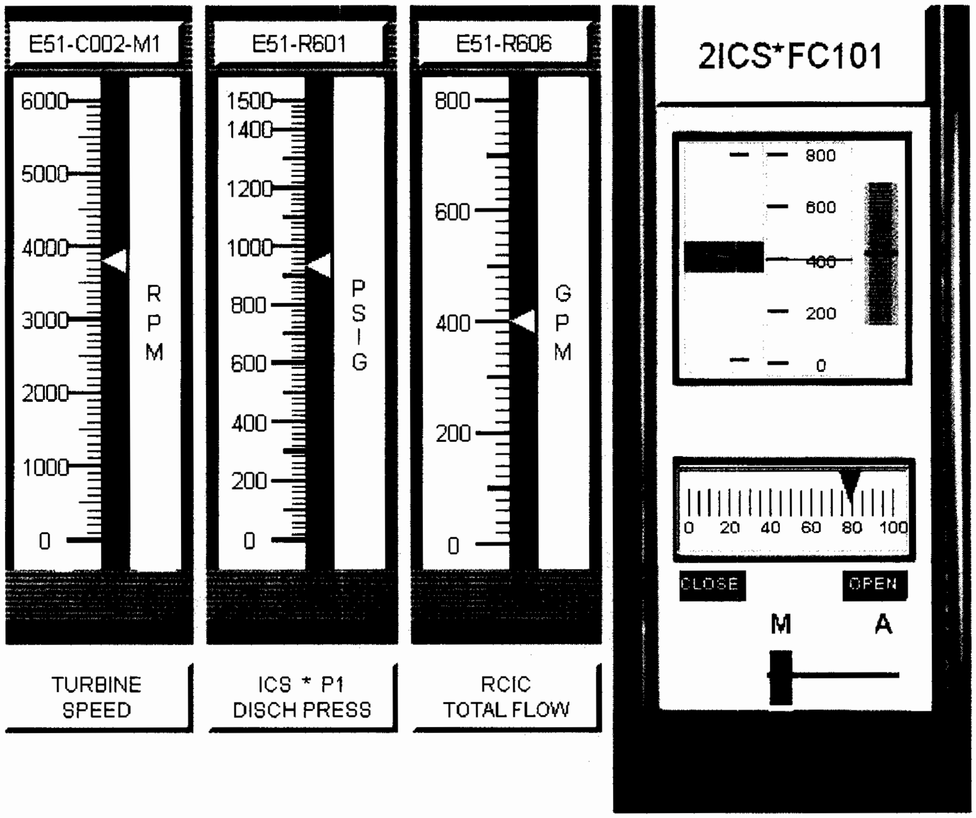
RCIC

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

Proposed Question: #20

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

- Reactor water level is 180 inches and stable with only RCIC injecting.
- Reactor pressure is 800 psig and slowly rising.
- The status of RCIC is shown below:



If Reactor pressure continued to rise, which one of the following describes the response of RCIC pump flow and the control(s) to be used to adjust RCIC pump discharge pressure while in this configuration?

	RCIC Pump Flow	**Control(s) to Adjust RCIC Pump Discharge Pressure**
A.	Lowers	2ICS*FC101 thumbwheel
B.	Lowers	2ICS*FC101 OPEN/CLOSE pushbuttons
C.	Remains approximately the same	2ICS*FC101 thumbwheel
D.	Remains approximately the same	2ICS*FC101 OPEN/CLOSE pushbuttons

Proposed Answer: B

Explanation: The given indications show the RCIC flow controller, 2ICS*FC101, in manual control. In manual, RCIC turbine speed will be held constant, whereas in auto, RCIC flow would be held constant. This will cause pump flow to lower as Reactor pressure rises. With the flow controller in manual, the OPEN/CLOSE pushbuttons are used to control system parameters, not the thumbwheel.

> **Note:** The question avoids overlap with the GFE by testing specific operating characteristics of the RCIC system based on status of the system's controls. Specifically, operators must have knowledge of how the RCIC Control System responds to plant parameter changes when operating the system in MANUAL (i.e. speed control) vice AUTOMATIC (flow control).

A. Plausible – The thumbwheel will not cause any system parameter changes while in manual, but would be used if in auto.
C. Plausible – If the controller were in auto, flow would be held relatively constant. However in manual, flow will lower as Reactor pressure rises. The thumbwheel will not cause any system parameter changes while in manual, but would be used if in auto.
D. Plausible – If the controller were in auto, flow would be held relatively constant. However in manual, flow will lower as Reactor pressure rises.

Technical Reference(s): N2-OP-35 Section F.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-217000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

TRH 02072014 – Revised heading of first column and adjusted choices/explanations accordingly, based on NRC comment.

02132014: Reviewed question, no comments. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 2.4.9
	Importance Rating	3.8

EDGs

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

Proposed Question: #21

The plant has experienced a LOCA and a complete loss of offsite power. Conditions are as follows:

- The Division 2 EDG failed to automatically start and CANNOT be started.
- Reactor water level is being controlled 160 to 200 inches using LPCS.
- Reactor pressure is 100 psig and slowly lowering.
- Drywell pressure is 1.5 psig and slowly lowering.
- The crew has exited N2-EOP-RPV, RPV Control.
- It is desired to cool down the plant to Cold Shutdown.

Which one of the following identifies the lineup to be used to cool down the plant to Cold Shutdown?

A. Place normal Shutdown Cooling in service using RHR pump A per N2-OP-31, Residual Heat Removal.

B. Place normal Shutdown Cooling in service using RHR pump B per N2-OP-31, Residual Heat Removal.

C. Place alternate Shutdown Cooling (Preferred Lineup) in service using RHR pump A per N2-SOP-31, Loss of Shutdown Cooling.

D. Place alternate Shutdown Cooling (Preferred Lineup) in service using RHR pump B per N2-SOP-31, Loss of Shutdown Cooling.

Proposed Answer: C

Explanation: N2-OP-31 P&L 22 references use of N2-SOP-31, Loss of Shutdown Cooling, if normal Shutdown Cooling cannot be established. N2-SOP-31 directs use of alternate Shutdown Cooling preferred lineup if you are in Mode 3 and attempting to cool down. RHR pump B is unavailable since 2ENS*SWG103 is de-energized. Therefore RHR pump A must be used for alternate Shutdown Cooling (preferred lineup).

A. Plausible – With a loss of offsite power and Division 2 EDG not available, there is no power to open 2RHS*MOV112, the SDC suction line isolation valve inside Primary Containment. Additionally, under LOCA conditions, it is not possible to enter Primary Containment to manually open the valve.
B. Plausible – With a loss of offsite power and 2EGS*EG3 not available, there is no power to RHR pump B.
D. Plausible – With a loss of offsite power and 2EGS*EG3 not available, there is no power to RHR pump B.

Technical Reference(s): N2-OP-31 P&L 22.0 and N2-SOP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-8

Question Source: Modified – 2010 Audit #22

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

02072014: Added the word "automatically" to first bullet. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 2.4.6
	Importance Rating	3.7

PCIS/Nuclear Steam Supply Shutoff

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: #22

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

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

Then...

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

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

A. Shutdown and isolate RCIC immediately.

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

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

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

Proposed Answer: B

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

A. Plausible – This would be correct if RCIC was not necessary to maintain adequate core cooling. Since RCIC is needed to maintain adequate core cooling, it should be left in service and have the high temperature isolation bypassed.
C. Plausible – The high temperature isolation should be bypassed to allow continued RCIC operation for adequate core cooling.
D. Plausible – The high temperature isolation should be bypassed to allow continued RCIC operation for adequate core cooling. Additionally, with no other injection systems available, if RCIC trips/isolates, Steam Cooling would be performed, not RPV Blowdown.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-217000-RBO-12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

02072014: Removed the word “Verify” from Answer B. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K4.02
	Importance Rating	3.0

SLC

Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Component and system testing

Proposed Question: #23

The Standby Liquid Control System (SLS) quarterly operability test on SLS pump A is about to be performed.

Which one of the following identifies how the automatic Reactor Water Cleanup System (WCS) isolation is avoided during this test?

A. WCS isolation signal is bypassed when using the SLS pump TEST switch to start the SLS pump.

B. WCS isolation bypass switches are placed in the BYPASS position prior to starting the SLS pump.

C. WCS system is shutdown and containment isolation valves closed prior to starting the SLS pump.

D. WCS containment isolation valve power supply breakers are opened prior to starting the SLS pump.

Proposed Answer: A

Explanation: Use of the TEST Switch for either pump bypasses the interlocks that open the MOV1A/B valves, fire the squibs, and isolate the respective divisional WCS isolation valve. This allows operation of the pump to circulate the contents of the test tank for surveillance testing.

B. Plausible – There are no bypass switches for this situation.
C. Plausible – WCS remains in operation.
D. Plausible – No breakers are opened.

Technical Reference(s): N2-OSP-SLS-Q001 and SLS GE Prints Sheet 4

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-211000-RBO-5

Question Source: Bank – 2009 NRC #21

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

02072014: Added the word "automatic" to question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K3.07
	Importance Rating	3.8

RPS

Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: Reactor power (thermal heat flux)

Proposed Question: #24

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

- The Mode Switch is placed in SHUTDOWN.
- A malfunction occurs such that the RPS B Scram Solenoids remain energized.
- All other RPS functions respond as designed.

Which one of the following gives (1) the value of Reactor power 20 seconds after the malfunction and (2) the action required?

	(1) Rx Power 20 seconds after malfunction	(2) Required Action
A.	100%	Initiate ARI per N2-SOP-101C, Reactor Scram
B.	100%	Actuate the MANUAL SCRAM PUSHBUTTONS per N2-SOP-101C, Reactor Scram.
C.	Downscale (<4%)	Enter N2-EOP-RPV, RPV Control, and stabilize/control Reactor water level and pressure.
D.	Downscale (<4%)	Enter N2-EOP-RPV, RPV Control. Exit N2-EOP-RPV and enter N2-EOP-C5, Failure to Scram

Proposed Answer: C

With a failure of the RPS B Scram Solenoids to de-energize, the Scram Pilot Valves (SOV139's) could not reposition themselves to vent off their associated HCUs and insert the control rods. However the backup scram valves are still functional and are energize to function valves. Since the backup scram valves would still operate, the scram air header would depressurize which will cause all rods to insert and power to go <4% (downscale). Since all rods would go in, there would be no ATWS and so normal pressure/level control would be used per N2-EOP-RPV.

A. Plausible – This action would be taken if the rods did not insert and the manual scram pushbuttons did not deenergize the RPS solenoids
B. Plausible – Although the operator could operate the manual scram pushbuttons, reactor power would not be 100%, it would be downscale after 20 seconds.
D. Plausible – Even though power would be <4%, if not all rods went in, entry into N2-EOP-C5 would be required.

Technical Reference(s): RPS Lesson Plan

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

02072014: Revised the 2nd bullet. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 A4.01
	Importance Rating	4.4

Safety Relief Valves

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

Proposed Question: #25

A loss of high pressure injection has occurred. Conditions are as follows:

- ADS automatically initiated.
- Reactor pressure is 80 psig and stable.
- Reactor water level is 180 inches and stable.

Which one of the following identifies the status of the Control Room P601 and P628/P631 RED indicating lights for SRV137?

	P601	**P628/P631**
A.	On	On
B.	On	Off
C.	Off	On
D.	Off	Off

Proposed Answer: C

Explanation: SRV137 is one of the ADS valves, so it would have opened upon ADS initiation. Therefore, the red solenoid lights on P628/631 will be on. However, Reactor pressure is too low (<400 psig) to cause the red acoustic monitor light on P601 to be on.

A. Plausible – This is the expected condition at higher Reactor pressures (>400 psig).
B. Plausible – This would be correct if the valve opened by relief or safety mode.
D. Plausible – This would be correct for a non-ADS valve that remained closed.

Technical Reference(s): N2-OP-34

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-5

Question Source: Bank – 2012 Audit #18

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Proposed Answer: C

Explanation: Breakers 11-1, 13-6, and 13-10 are interlocked such that if all three breakers are closed for greater than 15 seconds, Breaker 13-10 automatically opens. Once the Operator closed Breaker 13-10, all three breakers are closed. By 20 seconds later, Breaker 13-10 will automatically open and the other two breakers will remain closed.

A. Plausible – Both breakers are closed in this scenario, but after 15 seconds, Breaker 13-10 automatically opens.
B. Plausible – All three breakers are not allowed to remain closed together longer than 15 seconds, however it is Breaker 13-10, not Breaker 13-6, that automatically opens.
D. Plausible – All three breakers are not allowed to remain closed together longer than 15 seconds, however only Breaker 13-10 opens, not both 13-6 and 13-10,

Technical Reference(s): N2-OP-71B Section H.6.0 Note

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262001-RBO-5

Question Source: Bank – SYSID 103807

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

TRH 02192014 – Resampled K/A based on NRC comment. Replaced with Bank Question.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 K1.06
	Importance Rating	2.6

Main Turbine Generator and Auxiliary Systems

Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: Component cooling water systems

Proposed Question: #27

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

- 2CCS-TV104, TB CLOSED LOOP COOLING TEMPERATURE CONTROL VALVE, is stuck and cannot move.

Subsequently, Service Water supply temperature lowers 5°F.

Which one of the following describes the response of Main Turbine Lube Oil temperature and the Main Generator leads temperature?

	Main Turbine Lube Oil Temperature	Main Generator Leads Temperature
A.	Remains approximately the same	Remains approximately the same
B.	Remains approximately the same	Lowers
C.	Lowers	Remains approximately the same
D.	Lowers	Lowers

Proposed Answer: B

Explanation: Service Water supplies the cooling water to the TBCLC heat exchangers and TBCLC supplies the cooling water to the Main Turbine Lube Oil and Main Generator Leads coolers. With Service Water temperature lowering and 2CCS-TV104 remaining in the initial position, TBCLC supply temperature will lower. The Main Turbine Lube Oil coolers have a dedicated temperature control valve that will adjust to maintain Lube Oil temperature approximately constant. The Main Generator Leads coolers do NOT have a dedicated temperature control valve, therefore Leads temperature will lower along with the TBCLC supply temperature.

A. Plausible – The Main Generator Leads coolers do NOT have a dedicated temperature control valve, therefore Leads temperature will lower along with the TBCLC supply temperature.
C. Plausible – The Main Turbine Lube Oil coolers have a dedicated temperature control valve that will adjust to maintain Lube Oil temperature approximately constant. The Main Generator Leads coolers do NOT have a dedicated temperature control valve, therefore Leads temperature will lower along with the TBCLC supply temperature.
D. Plausible – The Main Turbine Lube Oil coolers have a dedicated temperature control valve that will adjust to maintain Lube Oil temperature approximately constant.

Technical Reference(s): P&ID 14B and 14E

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-274000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

02072014: Revised second bullet by deleting "Due to changing lake conditions" and adding "Subsequently". Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	286000 K2.02
	Importance Rating	2.9

Fire Protection

Knowledge of electrical power supplies to the following: Pumps

Proposed Question: #28

The plant is at 100% power when 2NNS-SWG012 de-energizes.

Which one of the following describes the impact on the Fire Protection system?

A. The Electric Fire pump motor loses power.

B. The Diesel Fire pump starting circuit loses power.

C. Both Pressure Maintenance pumps lose power. The Electric Fire pump auto-starts.

D. One Pressure Maintenance pump loses power, but the other maintains system pressure.

Proposed Answer: A

Explanation: 2NNS-SWG012 is the power supply to the Electric Fire pump motor.

B. Plausible – The Diesel Fire pump start circuit is not powered from 2NNS-SWG012, it is powered from 2SCA-PNL102 which is powered from US1.
C. Plausible – The Pressure Maintenance pumps are powered from 2NHS-MCC015, which does not receive power from 2NNS-SWG012.
D. Plausible – The Pressure Maintenance pumps are both powered from 2NHS-MCC015, which does not receive power from 2NNS-SWG012.

Technical Reference(s): N2-OP-43-ELU

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-286000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202002 K3.04
	Importance Rating	2.9

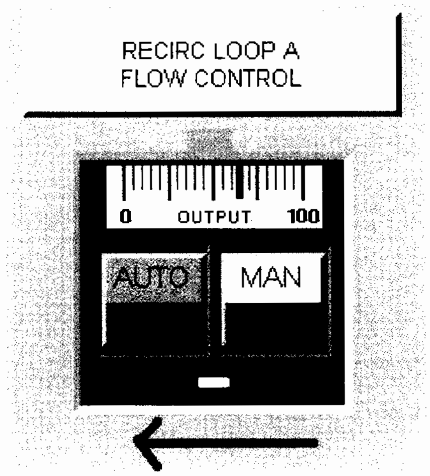
Recirculation Flow Control

Knowledge of the effect that a loss or malfunction of the RECIRCULATION FLOW CONTROL SYSTEM will have on the following: Reactor/turbine pressure regulation system

Proposed Question: #29

The plant is at 80% power. Conditions are as follows:

- A malfunction occurs in the M/A station for Recirculation Flow Control Valve 2RCS*HYV17A.
- The failure exactly mimics the slider being in the slow detent position in the direction indicated by the arrow below.
- The malfunction occurs for approximately 15 seconds and then clears.



Which one of the following describes how the EHC system controls Reactor pressure in response to this malfunction?

EHC...

A. adjusts Turbine Control Valves and stabilizes Reactor pressure at the initial value.

B. adjusts Turbine Control Valves and stabilizes Reactor pressure slightly higher than the initial value.

C. adjusts Turbine Control Valves and stabilizes Reactor pressure slightly lower than the initial value.

D. controls Reactor pressure at the initial value with no significant Turbine Control Valve adjustment because 2RCS*HYV17A locks up.

Proposed Answer: C

Explanation: This failure gives a close signal to 2RCS*HYV17A. Since the failure is the same as the M/A station being in the slow detent position, none of the FCV lockup signals will cause a lockup. The loop flow controller will throttle the FCV further closed. This will lower Recirculation flow and Reactor power. Reactor pressure will begin to lower and EHC will throttle Turbine Control Valves further closed to prevent a major drop in Reactor pressure. However, the EHC system controls Reactor pressure proportionally to Turbine steam flow. As Turbine steam flow lowers, Reactor pressure will be controlled at a lower value.

A. Plausible – The EHC setpoint will remain constant, however the control characteristics of the system are such that at lower steam flow, Reactor pressure will be controlled at a lower value.
B. Plausible – The given failure shows the loop flow controller with a close signal, not an open signal. Reactor pressure would rise if the FCV moved in the open direction.
D. Plausible – The FCV does have numerous lockup signals, however none that would prevent valve motion in this circumstance.

Technical Reference(s): N2-202002-RBO-5, N2-248000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

02072014: Deleted “right” from distracter A. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	204000 K4.04
	Importance Rating	3.5

RWCU

Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: System isolation upon receipt of isolation signals

Proposed Question: #30

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

- Annunciator 602319, RWCU FILTER DEMIN INLET TEMP HI-HI, alarms.
- The running Reactor Water Cleanup pump trips.
- 2WCS*MOV112, CLEANUP SUCT OUTBOARD ISOL VLV, closes.
- 2WCS*MOV200, CLEANUP RETURN ISOL VLV THROTTLE, remains open.

Which one of the following describes this RWCU response?

High non-regenerative heat exchanger outlet temperature....

A. directly caused the pump to trip, which caused MOV112 to close.

B. directly caused MOV112 to shut and MOV200 should have closed.

C. directly caused MOV112 to shut and the system responded properly.

D. should not have caused MOV112 to shut and the system responded improperly.

Proposed Answer: C

Explanation: Annunciator 602319 corresponds to a RWCU filter demin inlet temperature of 140°F. This causes closure of MOV112 to protect the resin. The closure of MOV112 trips the running WCS pump.

A. Plausible – The high temperature condition causes the valve isolation, which in turn causes the pump trip.
B. Plausible – MOV200 does not receive a closure signal from these conditions.
D. Plausible – The given annunciator should cause an isolation.

Technical Reference(s): ARP 602319

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-204000-RBO-5

Question Source: Bank – SYSID 33352

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 #	271000 K5.04
	Importance Rating	2.9

Offgas

Knowledge of the operational implications of the following concepts as they apply to OFFGAS SYSTEM: Hydrogen concentration measurement

Proposed Question: #31

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

- The Offgas system is in dual train operation.
- 2OFG-AT16A, H2 ANALYZER FOR RECOMBINER TRAIN "A", has been declared inoperable.

Which one of the following describes the operational implications (if any) for the inoperable H2 Monitor per N2-OP-42, Offgas System?

Operation with Offgas Train A may...

A. continue with no compensatory actions.

B. NOT continue until the associated H2 Monitor is operable.

C. NOT continue until H2 grab samples are taken and analyzed for Train A.

D. continue, however Train A H2 concentrations should be calculated using the common H2 concentration read on 2OFG-AT115, H2 ANALYZER FOR OFG SYSTEM A AND B.

Proposed Answer: D

Explanation: N2-OP-42, Section C.3.0 gives information regarding the operational implications for a H2 monitor out of service in dual train operation. When one H2 monitor is inoperable, the H2 concentration for that train can be calculated using the common H2 concentration and associated system flows. Use of Grab Samples or other direct reading methods to determine H2 concentrations may not provide a representative value of the actual H2 concentration throughout the Offgas System

A. Plausible – Operation can continue with the H2 monitor out of service, however additional actions are required to determine H2 concentrations for that train.
B. Plausible – P&L 9.0 requires the Offgas train to be shutdown when its associated H2 monitor is inoperable, however it allows one week of operation prior to shutting down the train.
C. Plausible – If the candidate is unaware of the one week requirement for allowable operations before requiring shutdown, then they may consider this a correct answer.

Technical Reference(s): N2-OP-42, Section C.3.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-271000-RBO-9 and 10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	230000 K6.01
	Importance Rating	3.3

RHR/LPCI: Torus/Pool Spray Mode

Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE: A.C. electrical

Proposed Question: #32

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

- RHR is in a normal standby lineup.
- An electrical fault causes 2EHS*MCC303 to de-energize and remain de-energized.

Which one of the following identifies the RHR lineup(s) that is(are) UNAVAILABLE unless an MOV is manually opened in the field?

(1) Loop B Suppression Pool Cooling
(2) Loop B Suppression Chamber Spray
(3) Loop B Low Pressure Coolant Injection

A. (1) only

B. (2) only

C. (1) and (2) only

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

Proposed Answer: D

Explanation: 2EHS*MCC303 provides power to numerous RHR Loop B MOVs, including MOV33B (needed for Suppression Chamber spray), MOV38B (needed for Suppression Pool cooling), and MOV24B (needed for LPCI Injection). Therefore, with 2EHS*MCC303 de-energized, RHR Loop B is unavailable for Suppression Pool cooling, Suppression Chamber spray, and LPCI Injection without manually opening an MOV in the field.

A. Plausible – 2EHS*MCC303 provides power to Loop B RHR MOVs including valves needed for suppression pool cooling
B. Plausible – 2EHS*MCC303 provides power to Loop B RHR MOVs including valves needed for suppression chamber spray.
C. Plausible – 2EHS*MCC303 provides power to Loop B RHR MOVs including valves needed for suppression pool cooling and spray.

Technical Reference(s): N2-OP-31 ELU

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

02072014: Rearranged question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 A1.01
	Importance Rating	3.6

RHR/LPCI: Containment Spray Mode

Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: Containment/drywell pressure

Proposed Question: #33

A loss of coolant accident has occurred. Conditions are as follows:

- RHR Loop A is spraying the Suppression Chamber.
- Suppression Chamber pressure is 10 psig and rising slowly.
- Drywell pressure is 12 psig and rising slowly.

RHR Loop A is subsequently aligned to spray both the Suppression Chamber and the Drywell.

In accordance with N2-EOP-PC, which one of the following describes (1) the response of Drywell pressure once Drywell spray is in service, and (2) the limit before which Drywell spray is REQUIRED to be secured?

Drywell pressure will lower ___(1)___.
Drywell spray is REQUIRED to be secured before Drywell pressure drops to ___(2)___.

A. (1) below Suppression Chamber pressure.
   (2) 0 psig.

B. (1) below Suppression Chamber pressure.
   (2) 1.68 psig.

C. (1) but remain above Suppression Chamber pressure.
   (2) 0 psig.

D. (1) but remain above Suppression Chamber pressure.
   (2) 1.68 psig.

Proposed Answer: A

Explanation: A loss of coolant accident drives non-condensable gases from the Drywell airspace to the Suppression Chamber airspace. When Drywell Spray is placed in service, the steam in the Drywell airspace condenses, lowering Drywell pressure. Drywell pressure will drop below Suppression Chamber pressure (by ~0.25 psid) before vacuum breakers will lift to return non-condensable gases from the Suppression Chamber to the Drywell. N2-EOP-PC step PCP-3 requires securing Drywell spray before Drywell pressure drops to 0 psig.

B. Plausible – It is allowable to secure Drywell sprays at or before 1.68 psig (N2-EOP-PC entry condition), but not required until before 0 psig.
C. Plausible – Drywell pressure is normally above Suppression Chamber pressure, but lowers below it during post-LOCA Drywell spray operation based on Containment design.
D. Plausible – Drywell pressure is normally above Suppression Chamber pressure, but lowers below it during post-LOCA Drywell spray operation based on Containment design. It is allowable to secure Drywell sprays at or before 1.68 psig (N2-EOP-PC entry condition), but not required until before 0 psig.

Technical Reference(s): N2-EOP-PC and Primary Containment Lesson Plan

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-10 and 12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

02072014: Rearranged question stem. Deleted the word “expected” from stem. Changed “should” to “will” in the first statement. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 A2.23
	Importance Rating	3.2

Recirculation

Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve closures

Proposed Question: #34

The plant is at 100% power with one OPRM inoperable, when the following occurs:

- 2RCS*MOV18A, RECIRC PMP 1A DISCH VLV, spuriously closes approximately 50% and then trips on motor overload.
- The crew enters N2-SOP-29, Sudden Reduction in Core Flow.
- Operators determine the plant is operating within the OPRM Dependent Stability Region and no core oscillations are occurring.

Which one of the following describes the response of Recirculation pump 1A and the required operator action in accordance with N2-SOP-29?

	Response of Recirculation Pump 1A	Required Action Per N2-SOP-29
A.	Trips	Scram the Reactor
B.	Trips	Insert the first four CRAM rods
C.	Remains operating	Scram the Reactor
D.	Remains operating	Insert the first four CRAM rods

Proposed Answer: B

Explanation: Recirculation pump 1A will automatically trip when RCS*MOV18A is less than 90% open. N2-SOP-29 would require a manual Reactor scram if core oscillations were observed, core flow and power entered the Scram Region, or if < 3 OPRMs were operable with core flow and power in the OPRM Dependent Stability Region. Since these conditions are NOT met, N2-SOP-29 requires inserting the first four CRAM rods.

A. Plausible – A scram is NOT required because core flow and power remain outside the Scram Region, no core oscillations are observed, and ≥ 3 OPRMs are operable with core flow and power in the OPRM Dependent Stability Region.
C. Plausible – Recirculation pump 1A does not immediately trip when the discharge valve comes off the full open position, however it does trip once the valve is ≥ 10% closed.
D. Plausible – Recirculation pump 1A does not immediately trip when the discharge valve comes off the full open position, however it does trip once the valve is ≥ 10% closed.

Technical Reference(s): N2-OP-29 Section D.6.0, N2-SOP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

02072014: Rearranged stem. Deleted the word "expected" from the stem and header. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 A3.03
	Importance Rating	3.3

Reactor Feedwater

Ability to monitor automatic operations of the REACTOR FEEDWATER SYSTEM including: System flow

Proposed Question: #35

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

- Over the course of one minute, 2FWS-FT1A, Feedwater Flow Transmitter A, fails downscale.

Which one of the following describes the automatic plant response to this failure?

Actual Feedwater flow will...

A. remain approximately constant because the Feedwater level control system bypasses the failed signal.

B. remain approximately constant because the Feedwater level control valves lock up due to the failed signal.

C. rise but Reactor water level will stay below level 8.

D. rise and Reactor water level will eventually exceed level 8.

Proposed Answer: D

Explanation: When 2FWS-FT1A fails downscale, the Feedwater level control system sees Feedwater flow as much lower than Main Steam flow and anticipates lowering Reactor water level. The system compensates by raising actual Feedwater system flow. The Feedwater level control system is biased such that this failure will cause a rise of approximately 25 inches in Reactor water level at rated power. This level rise will exceed level 8 (202.3 inches).

A. Plausible – There is no automatic bypass of this failed Feedwater flow signal, such as there is in other NMP 2 systems (i.e. APRMs, LPRMs, RWM, etc.) and many advanced digital Feedwater level control systems.
B. Plausible – There is no automatic lock up of the flow control valves on this failure, such as there is on a direct loss of control signal to the flow control valves.
C. Plausible – Reactor water level will exceed level 8 due to the biasing of the three element Feedwater level control system.

Technical Reference(s): N2-259002-RBO-11 Section B.2.b

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-259002-RBO-11

Question Source: Bank – SYSID 32082

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	223001 A4.12
	Importance Rating	3.5

Primary Containment and Auxiliaries

Ability to manually operate and/or monitor in the control room: Drywell coolers/chillers

Proposed Question: #36

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

- Line 5 de-energizes due to a fault.
- The Division 1 Diesel, 2EGS*EG1, re-energizes 2ENS*SWG101.

Which one of the following describes the operation of the Drywell Coolers during this transient?

The Drywell Coolers...

A. remain running throughout the transient.

B. trip, but then automatically re-start once 2ENS*SWG101 is re-energized.

C. trip and remain tripped once 2ENS*SWG101 is re-energized. The Division 1 CCP supply valves to the Drywell Coolers must be re-opened to restart the Drywell Coolers.

D. trip and remain tripped once 2ENS*SWG101 is re-energized. The Division 1 CCP supply valves to the Drywell Coolers do NOT need to be re-opened to restart the Drywell Coolers.

Proposed Answer: D

Explanation: Although loss of Line 5 does not cause a loss of power to the Drywell Coolers themselves, the loss does cause the fans to trip due to loss of position indication on the associated CCP isolation valves. Once 2ENS*SWG101 is re-energized by the diesel, this condition clears and the Drywell Coolers can be manually re-started. N2-SOP-03 Attachment 1 provides the guidance to manually re-start the Drywell Coolers. Although the Drywell Coolers trip due to the loss of CCP supply valve position indication, the valves do not actually shut.

A. Plausible – The Drywell Coolers have power throughout the transient, but trip due to loss of position indication for the associated CCP isolation valves.
B. Plausible – Once 2ENS*SWG101 is re-energized, conditions allow operation of Drywell Coolers, however they do not automatically re-start.
C. Plausible – The drywell coolers trip because the circuitry "thinks" the CCP valves are shut due to the loss of position indication, however the valves do not actually shut.

Technical Reference(s): N2-SOP-03 Attachment 1 Section 1.1.1.

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223004-RBO-5

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 #	268000 2.1.7
Importance Rating	4.4

Radwaste

Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: #37

The plant has been in Mode 3 for the last 24 hours. Conditions are as follows:

- Drywell floor and equipment drain tank levels are as given below.
- No Drywell floor or equipment drain tank pump downs have occurred during this time period.

Time	Drywell Floor Drain Tank	Drywell Equipment Drain Tank
15 minutes ago	100 gallons	150 gallons
Now	125 gallons	240 gallons

Which one of the following describes:

(1) the approximate value of Unidentified Leak Rate during this time period, and

(2) whether this value is above or below the Technical Specification limit?

A. (1) 1.7 gpm
(2) Below the Technical Specification limit

B. (1) 1.7 gpm
(2) Above the Technical Specification limit

C. (1) 6.0 gpm
(2) Below the Technical Specification limit

D. (1) 6.0 gpm
(2) Above the Technical Specification limit

Proposed Answer: A

Explanation: Unidentified Leak Rate is calculated using Drywell Floor Drain Tank levels. For the given 15 minute period, the leak rate into the Drywell Floor Drain Tank was approximately 1.7 gpm ((125 gallons – 100 gallons) / 15 minutes). This is below the Technical Specification limit of 5 gpm.

Note: The question satisfies the generic K/A by requiring a calculation using plant data (Ability to evaluate plant performance…based on…instrument interpretation) and then requiring a judgment to be made regarding compliance with operating restrictions based on this calculation (Ability to…make operational judgments based on…instrument interpretation).

B. Plausible – The Technical Specification limit is 5 gpm, therefore the 1.7 gpm leak rate is below the limit.
C. Plausible – This leak rate is calculated erroneously from the Drywell Equipment Drain Tank levels, which is identified leakage, not unidentified.
D. Plausible – This leak rate is calculated erroneously from the Drywell Equipment Drain Tank levels, which is identified leakage, not unidentified.

Technical Reference(s): Technical Specification 3.4.5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-291000-RBO-14

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 #	239001 2.2.44
	Importance Rating	4.2

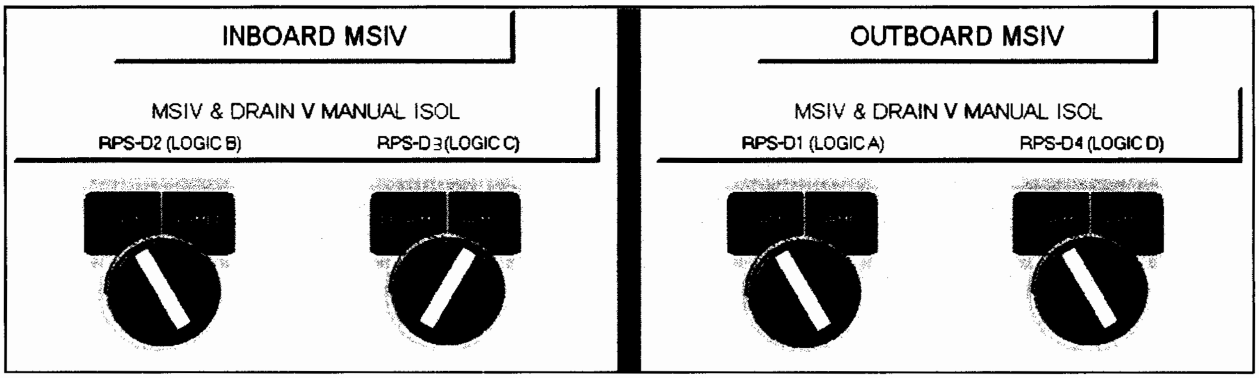
Main and Reheat Steam

Equipment Control: 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: #38

The plant has scrammed from 100% power due to a Turbine Trip. Conditions are as follows:

- The crew has entered N2-SOP-83, Primary Containment Isolation Failure, in an attempt to close all MSIVs.
- LOGIC C PUSHBUTTON has been rotated and depressed.



Which one of the following describes (1) the status of the MSIVs and (2) the actions required to close ONLY the MSIVs per N2-SOP-83?

A. (1) Only the Inboard MSIVs are Closed
(2) Rotate and depress LOGIC B pushbutton

B. (1) Only the Inboard MSIVs are Closed
(2) Rotate and depress LOGIC D pushbutton

C. (1) All MSIVs are Open
(2) Rotate and depress LOGIC B pushbutton

D. (1) All MSIVs are Open
(2) Rotate and depress LOGIC D pushbutton

Proposed Answer: D

Explanation: Per N2-SOP-83, Section 5.11, in order to isolate only the MSIVs, either pushbuttons for A and B -or- C and D need to be depressed. Since Pushbutton C is already rotated and depressed, then D would be the only additional pushbutton needed. One pushbutton cannot isolate the MSIVs. The MSIVs are either all open or all shut based on the isolation pushbuttons.

A. Plausible – This distracter takes advantage of a common misconception for candidates. Manual isolation of other PCIS components can be accomplished using one pushbutton on 2CEC*PNL602, however the MSIVs require the use of two pushbuttons. Additionally, LOGIC D needs to be depressed, not LOGIC B.
B. Plausible – This distracter takes advantage of a common misconception for candidates. Manual isolation of other PCIS components can be accomplished using one pushbutton on 2CEC*PNL602, however the MSIV's require the use of two pushbuttons.
C. Plausible – This would be true if LOGIC A was already rotated and depressed.

Technical Reference(s): N2-SOP-83

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-239001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AK1.02
	Importance Rating	3.4

SCRAM

Knowledge of the operational implications of the following concepts as they apply to SCRAM: Shutdown margin

Proposed Question: #39

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

- A Turbine trip causes a Reactor scram.
- One control rod fails to insert and is at position 48.
- All other control rods fully insert.

Which one of the following correctly completes the below statements?

The MAXIMUM SUBCRITICAL BANK WITHDRAW POSITION __(1)__ being exceeded.

The Reactor __(2)__ considered shutdown under all conditions

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

Proposed Answer: A

Explanation: The Maximum Subcritical Bank Withdraw position of 00 is being exceeded. However the Reactor is still considered shutdown under all conditions.

B. Plausible – Even with one rod out, the Reactor is considered shutdown under all conditions.
C. Plausible – The Maximum Subcritical Bank Withdraw position of 00 is being exceeded.
D. Plausible – The Maximum Subcritical Bank Withdraw position of 00 is being exceeded.

Technical Reference(s): NER-2M-039 Section 3.2 (EOP-RPV Step 2)

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01 EO-2

Question Source: Bank – 2012 Audit #39

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

02072014: Reworded question stem and changed format to two column format. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK1.04
	Importance Rating	2.5

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Limiting cycle oscillation: Plant-Specific

Proposed Question: #40

Which one of the following identifies:

(1) the threshold value for APRM peak-to-peak oscillations requiring action, and

(2) the required action for exceeding this threshold,

in accordance with N2-SOP-29, Sudden Reduction in Core Flow?

A. (1) 10%
(2) Insert CRAM rods using RMCS.

B. (1) 10%
(2) Scram the Reactor.

C. (1) 25%
(2) Insert CRAM rods using RMCS.

D. (1) 25%
(2) Scram the Reactor.

Proposed Answer: B

Explanation: N2-SOP-29 directs inserting a manual Reactor scram if core oscillations are observed. One of the criteria qualifying as a core oscillation is "any APRM peak-to-peak value exceeds 10%".

A. Plausible – N2-SOP-29 directs inserting CRAM rods using RMCS for other reasons. However, exceeding APRM peak to peak oscillations of 10% requires action to insert a manual Reactor scram.
C. Plausible – 25% is the threshold value for power oscillations requiring boron injection in N2-EOP-C5. N2-SOP-29 directs inserting CRAM rods using RMCS for other reasons. However, exceeding APRM peak to peak oscillations of 10% requires action to insert a manual Reactor scram.
D. Plausible – 25% is the threshold value for power oscillations requiring boron injection in N2-EOP-C5.

Technical Reference(s): N2-SOP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP29C01 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 AK1.01
	Importance Rating	3.5

Partial or Complete Loss of CCW

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations

Proposed Question: #41

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

- Annunciator 601246, REACTOR BLDG CLOSED LOOP COOLING SYS TROUBLE, alarms.
- The alarm is determined to be due to low expansion tank level.
- The expansion tank make-up valve is open.
- The expansion tank level is out of sight low.

Then...

- The running CCP pumps trip.
- The standby CCP pumps fail to auto start and cannot be manually started.

Which one of the following describes the required actions, in accordance with N2-SOP-13, Loss or Degraded CCP System?

A. Scram the Reactor and trip both Recirc pumps and WCS pumps.

B. Scram the Reactor and trip WCS pumps. Trip the Recirc Pumps only when high temperature alarms are received.

C. Reduce power to 85% using Recirc flow or CRAM rods while attempting to locate and isolate the leak. If tank level cannot be readily restored, then scram the Reactor.

D. Reduce power to 85% using Recirc flow or CRAM rods while removing WCS from service and continuing efforts to start a CCP pump. If a pump cannot be started, then scram the Reactor.

Proposed Answer: A

Explanation: Per N2-SOP-13, if all pumps trip and none can be started then the required actions per the 1st override is to scram, trip both Recirc pumps and the WCS pumps. The running CCP pumps tripped on low suction and the standby pumps will not start due to the low suction condition. The conditions provided are symptomatic of a leak in the system.

B. Plausible – Tripping the Recirc pumps is not conditional on receiving the high temperature alarms.
C. Plausible – A scram criteria has been met. Plausible in that the power reduction specified is the manner in which power is reduced if N2-SOP-101D, Rapid Power Reduction, is entered. Also plausible in that the indications provided are symptomatic of a leak in the system.
D. Plausible – A scram criteria has been met. Also plausible in that if the system is degraded, direction is provided to remove the WCS from service in order to remove a major heat load on the system. A standby pump cannot be started due to the low suction condition.

Technical Reference(s): N2-SOP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP13C01 EO-2

Question Source: Bank – 2012 Audit #57

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 EK2.03
	Importance Rating	3.6

High Off-site Release Rate

Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Plant ventilation systems

Proposed Question: #42

The plant is shutdown for a refueling outage. Conditions are as follows:

- An irradiated fuel bundle is dropped in the Spent Fuel Pool.
- Reactor Building ventilation automatically isolates due to high radiation levels.
- Both Standby Gas trains and Emergency Recirc Units fail to start.
- Reactor Building differential pressure is +0.05 inches WG.

Which one of the following identifies the type of offsite radiation release that will occur, if any?

A. No release

B. Treated, elevated release

C. Untreated, elevated release

D. Untreated, ground level release

Proposed Answer: D

Explanation: With high airborne radiation levels, no Reactor Building ventilation running, and positive Reactor Building pressure, a radiation release will occur from the Reactor Building. The release will be untreated since SGTS is not operating. The release will be ground level, NOT elevated, since no fans are operating to direct the release to elevated release points.

A. Plausible – Since an irradiated fuel bundle was dropped in the Spent Fuel Pool, causing high enough airborne contamination levels to isolate Reactor Building ventilation, and Reactor Building D/P is positive, a release will occur. Plausible if a new fuel bundle were dropped or the irradiated fuel bundle did not sustain damage.
B. Plausible – This would be correct if SGTS was operating.
C. Plausible – This would be correct if normal Reactor Building ventilation remained in service.

Technical Reference(s): NER-2M-039 Section for N2-EOP-SC Step SC-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Bank – 2005 NRC #64

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 #	1
	Group #	1
	K/A #	295028 EK2.04
	Importance Rating	3.6

High Drywell Temperature

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell ventilation

Proposed Question: #43

A loss of coolant accident has occurred. Conditions are as follows:

- Drywell pressure is 3.0 psig and slowly rising.
- The hottest Drywell temperature is 275°F and slowly rising.
- All Drywell cooling fans are tripped.

Which one of the following describes the ability to restore Drywell cooling, in accordance with N2-EOP-6.24, Drywell Unit Cooler Operation?

Drywell cooling...

A. can be fully restored without further evaluation and without defeating interlocks.

B. can be fully restored without further evaluation, but only after defeating interlocks.

C. CANNOT be restored without further evaluation because of the potential for CCP piping damage.

D. CANNOT be restored without further evaluation because of the potential for Drywell cooling fan motor damage.

Proposed Answer: C

Explanation: N2-EOP-6.24 does not allow restoration of Drywell cooling until further evaluation has been conducted because the hottest Drywell temperature is ≥ 250°F. Steam voiding in the Drywell CCP piping is of concern, which could result in significant water hammer and piping damage in the event of CCP flow restoration.

A. Plausible – This would be true if Drywell temperature were < 250°F and no LOCA signal was present.
B. Plausible – This would be true if Drywell temperature were < 250°F.
D. Plausible – The specific concern preventing restoration of Drywell cooling is water hammer in CCP piping, not Drywell cooling fan motor damage. Drywell cooling fan motor damage is the concern that leads to tripping Drywell cooling fans before initiating Drywell sprays.

Technical Reference(s): N2-EOP-6.24 Steps 5.8 and 6.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223004-RBO-12

Question Source: Bank – 2005 NRC #22

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 #	1
	Group #	1
	K/A #	295003 AK2.06
	Importance Rating	3.4

Partial or Complete Loss of AC Power

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

Proposed Question: #44

A station blackout has occurred. Given the following actions:

Action #	Action Description
1	De-energize the RCIC Gland Seal Compressor.
2	De-energize non-essential station lighting.
3	Degas the generator, then de-energize the Hydrogen Seal Oil and Emergency Seal Oil Pumps.
4	Break condenser vacuum and when the turbine stops, place the Emergency Bearing Oil pump control switch in Pull-To-Lock.

Which one of the following identifies the actions required within the first two hours of the Station Blackout, in accordance with N2-SOP-01, Station Blackout?

A. Actions 1 & 3.

B. Actions 1 & 4.

C. Actions 2 & 3.

D. Actions 2 & 4.

Proposed Answer: D

Explanation: N2-SOP-01 requires de-energizing non-essential station lighting and securing the Emergency Bearing Oil pump within 2 hours to control battery discharge rate.

A. Plausible – N2-SOP-01 does not specify securing the RCIC Gland Seal Compressor. Plausible in that RCIC will continue to operate with the compressor secured.
B. Plausible – N2-SOP-01 does not specify securing the RCIC Gland Seal Compressor. Plausible in that RCIC will continue to operate with the compressor secured.
C. Plausible – N2-SOP-01 does not specify that the Emergency Seal Oil pump be secured. Plausible in that this DC pump would be operating.

Technical Reference(s): N2-SOP-01

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP01C01 RBO-2

Question Source: Bank – 2010 Audit #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 #	1
	Group #	1
	K/A #	700000 AK3.02
	Importance Rating	3.6

Generator Voltage and Electric Grid Disturbances

Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Actions contained in abnormal operating procedures for voltage and grid disturbances

Proposed Question: #45

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

- Thunderstorms are causing significant grid voltage instability.
- N2-SOP-70, Major Grid Disturbances, has been entered.
- Power Control reports that the Post LOCA Contingency Alarms are in for both Line 5 and 6.

Which one of the following describes the required action based on this report and the reason for this response?

A. Lines 5 and 6 must be declared inoperable because their loss is imminent.

B. The Div 1 and 2 Diesels must be started because loss of Lines 5 and 6 is imminent.

C. Lines 5 and 6 must be declared inoperable because they may not supply adequate voltage under accident conditions.

D. The Div 1 and 2 Diesels must be started because Lines 5 and 6 may not supply adequate voltage under accident conditions.

Proposed Answer: C

Explanation: N2-SOP-70 contains the following conditional step:

IF	THEN
Notification is received that the Post (LOCA) Contingency alarm is in,	Declare Line 5 AND/OR Line 6 inoperable.

The SOP discussion states, "The Load Flow computer at National Grid runs a program that looks at 115 KV voltage for Line 5 and Line 6. This program determines if there is sufficient voltage (depending on grid loading) to supply NMP 2 ECCS loads during a LOCA (plant trip). If the voltage is too low, a contingency low voltage alarm is received at Power Control. If the contingency low voltage alarm is received OR actual voltage is less than 110 KV with the load Flow computer unavailable, then the offsite power source cannot perform its function under accident conditions and is, therefore, inoperable.

A. Plausible – Lines 5 and 6 must be declared inop because voltage may be too low during potential accident conditions, not because they are in immediate danger of tripping on undervoltage.
B. Plausible – N2-SOP-70 contains a step to start Diesels, but it is based on loss of off-site power being imminent, not the Load Flow Computer prediction.
D. Plausible – N2-SOP-70 contains a step to start Diesels, but it is based on loss of off-site power being imminent, not the Load Flow Computer prediction.

Technical Reference(s): N2-SOP-70

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP70C01 EO-2

Question Source: Modified – SYSID 97158

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 #	295021 AK3.02
	Importance Rating	3.3

Loss of Shutdown Cooling

Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Feeding and bleeding reactor vessel

Proposed Question: #46

The plant is cooling down in Mode 3. Conditions are as follows:

- A loss of Shutdown Cooling has occurred.
- RCS temperature is 205°F and rising slowly.
- The CRS has directed that Alternate Shutdown Cooling (ASC) Preferred Lineup be initiated.

Per N2-SOP-31, Loss of SDC, which one of the following describes 1) the RHS mode used to remove heat from the primary containment for ASC Preferred Lineup and 2) the reason why the ASC Preferred Lineup is preferred over the ASC Alternate Lineup?

A. (1) LPCI Injection Mode
(2) A shorter time is required to reach cold shutdown.

B. (1) Suppression Pool Cooling Mode
(2) A shorter time is required to reach cold shutdown.

C. (1) LPCI Injection Mode
(2) There is a greater chance of violating cooldown limits while in ASC Alternate Lineup as opposed to ASC Preferred Lineup.

D. (1) Suppression Pool Cooling Mode
(2) There is a greater chance of violating cooldown limits while in ASC Alternate Lineup as opposed to ASC Preferred Lineup.

Proposed Answer: A

Explanation: N2-SOP-31, Section 5.2 explains that when in the preferred Lineup, it will take less time to reach cold shutdown conditions because heat is being removed from the coolant by the RHS HX prior to injecting water into the reactor (LPCI Injection Mode). The Alternate Lineup does not cool the water prior to going into the RPV but instead relies on a separate loop of RHS being run in Suppression Pool Cooling Mode.

Note: This question meets the K/A because both the ASC Preferred and Alternate flow paths use RPV Feed and Bleed to remove heat from the fuel during a Loss of SDC and the candidate has to know the reason for why it is preferred to use the Preferred Lineup over the Alternate Lineup.

B. Plausible – Suppression Pool Cooling mode is an allowable mode of heat removal from the primary containment, however this mode is only used in the Alternate, not the Preferred Lineup.
C. Plausible – Per N2-SOP-31, Section 5.4, both Preferred and Alternate lineups for ASC have an equal chance of violating cooldown rates because the concern deals with the Suppression Pool temperature, not which lineup you are in.
D. Plausible – Suppression Pool Cooling mode is an allowable mode of heat removal from the primary containment, however this mode is only used in the Alternate, not the Preferred Lineup. Additionally, per N2-SOP-31, Section 5.4, both Preferred and Alternate lineups for ASC have an equal chance of violating cooldown rates because the concern deals with the Suppression Pool temperature, not which lineup you are in.

Technical Reference(s): N2-SOP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP31C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK3.04
	Importance Rating	3.7

High Drywell Pressure

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Emergency depressurization

Proposed Question: #47

Which one of the following identifies:

(1) the required action if Primary Containment parameters CANNOT be restored and maintained inside the Pressure Suppression Pressure (PSP), and

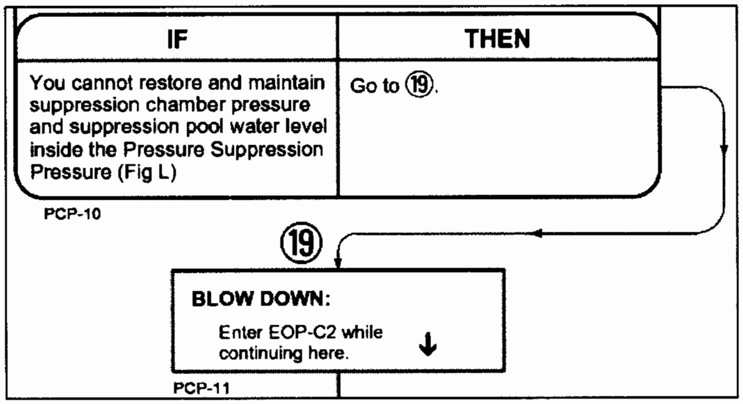
(2) the reason for this action,

in accordance with N2-EOP-PC, Primary Containment Control and the Unit 2 EOP Bases?

	(1)	(2)
A.	Perform an RPV Blowdown	The design pressure of the Primary Containment is being exceeded.
B.	Perform an RPV Blowdown	Ensure Reactor depressurization while the Primary Containment is still capable of sufficiently accommodating a Blowdown.
C.	Vent the Primary Containment	The design pressure of the Primary Containment is being exceeded.
D.	Vent the Primary Containment	Ensure venting is performed while the Primary Containment vent valves are still capable of being opened and closed.

Proposed Answer: B

Explanation: N2-EOP-PC contains the following steps:



The limiting basis for the Pressure Suppression Pressure curve at NMP2 is the highest Suppression Chamber pressure which can occur without steam in the Suppression Chamber airspace. NER-2M-039 states that the RPV Blowdown due to PSP is required "to ensure that pressure suppression capability sufficient to accommodate a Blowdown is maintained while the RPV is at pressure".

A. Plausible – RPV Blowdown is correct, however the basis is NOT that the design pressure of the Primary Containment is being exceeded. This is part of the basis behind the higher Primary Containment Pressure Limit (PCPL).
C. Plausible – This is the action for approaching PCPL, as well as part of the basis for PCPL.
D. Plausible – This is the action for approaching PCPL, as well as part of the basis for PCPL.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

02072014: Deleted the word "capability" from distracters A and C. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AA1.01
	Importance Rating	3.8

Control Room Abandonment

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: RPS

Proposed Question: #48

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

- A Control Room evacuation has been ordered.
- The crew has entered N2-SOP-78, Control Room Evacuation.

Which one of the following identifies actions taken outside the Control Room to ensure a reactor scram and/or MSIV isolation during a Control Room Evacuation in accordance with N2-SOP-78?

(1) Trip the RPM EPA breakers.
(2) Vent the scram air header.
(3) Open breakers on the VBS panels supplied by UPS3A/3B.
(4) Place SRI TEST switches in TEST.

A. (1) and (2)

B. (1) and (3)

C. (2) and (4)

D. (3) and (4)

Proposed Answer: B

Explanation: When a control room evacuation is directed, the Lower Control Building Operator takes action to backup the OATC to ensure the reactor is scrammed and the MSIVs are closed. This is accomplished by tripping the RPS MG Set EPAs and opening the RPS and MSIV logic power breakers on the VBS (UPS) panels supplied by UPS3A/B.

A. Plausible – Both of these actions are possible methods to insert control rods, however SOP-78 does NOT direct venting the scram air header. This action is directed in EOP-6.14.
C. Plausible – Both of these actions are possible methods to insert control rods, however SOP-78 does NOT direct either of them. These actions are directed in EOP-6.14.
D. Plausible – Both of these actions are possible methods to insert control rods, however SOP-78 does NOT direct using SRI TEST switches. This action is directed in EOP-6.14.

Technical Reference(s): N2-SOP-78

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP78C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

02072014: Rearranged stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AA1.08
	Importance Rating	2.6

Plant Fire On-site

Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire fighting equipment used on each class of fire

Proposed Question: #49

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

- Annunciator 849128, FIRE DETECTED PNL129 W STAIR/288, alarms.
- Zone 353SG E288 RR NW Area, is in alarm.
- The zone is in ALARM ONLY at the Main Fire Control Panel.

Which one of the following describes:

(1) the type of fire suppression agent associated with this zone, and

(2) the required action to actuate suppression for this zone,

in accordance with N2-OP-47, Fire Detection?

A. (1) Halon
(2) Place zone control switch to Discharge at 2CEC-PNL849.

B. (1) Halon
(2) Place zone disconnect switch in Disconnect at the Local Fire Control Panel.

C. (1) Water
(2) Place zone control switch to Discharge at 2CEC-PNL849.

D. (1) Water
(2) Place zone disconnect switch in Disconnect at the Local Fire Control Panel.

Proposed Answer: A

Explanation: The zone designator (353SG) provides information about the type of fire suppression agent associated with the zone. The "G" indicates this zone is protected by Halon. With the zone in ALARM ONLY at the Main Fire Control Panel, discharge may still be initiated by placing the zone control switch in DISCHARGE at the Main Fire Control Panel (2CEC-PNL849).

B. Plausible – Placing the zone disconnect switch in Disconnect does not actuate the system, but prevents actuation.
C. Plausible – Numerous other zones on this annunciator are protected by water, however this zone is protected by Halon.
D. Plausible – Numerous other zones on this annunciator are protected by water, however this zone is protected by Halon. Placing the zone disconnect switch in Disconnect does not actuate the system, but prevents actuation.

Teohnical Reference(s): N2-OP-47 Section B

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-286000-RBO-5

Question Source: Modified – 2009 NRC #42

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 #	295005 AA1.07
	Importance Rating	3.3

Main Turbine Generator Trip

Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: A.C. electrical distribution

Proposed Question: #50

The plant is at 100% power when the Main Turbine spuriously trips.

Given the following electrical buses:

(1) 2NPS-SWG001
(2) 2NPS-SWG002
(3) 2NPS-SWG003

Which one of the following identifies electrical buses that automatically transfer to an alternate power supply?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: B

Explanation: At 100% power, 2NPS-SWG001 and 2NPS-SWG003 are powered from the Main Generator output. Upon the Main Turbine trip, the Main Generator will lockout. When this occurs, 2NPS-SWG001 and 2NPS-SWG003 auto transfer to offsite power via fast transfer. 2NPS-SWG002 is fed from Auxiliary Boiler Service Transformer 2ABS-X1. 2ABS-X1 is fed from offsite power. Therefore, 2ABS-X1 and 2NPS-SWG002 do not auto transfer on the Main Generator lockout.

A. Plausible – 2NPS-SWG002 is normally fed from offsite power and therefore does not auto transfer on Main Generator lockout.
C. Plausible – 2NPS-SWG002 is normally fed from offsite power and therefore does not auto transfer on Main Generator lockout.
D. Plausible – 2NPS-SWG002 is normally fed from offsite power and therefore does not auto transfer on Main Generator lockout.

Technical Reference(s): N2-OP-71A Section B and SOP-21

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262000-RBO-5

Question Source: Modified – SYSID 80124

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EA2.02
	Importance Rating	3.8

Suppression Pool High Water Temperature

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

Proposed Question: #51

The plant is shutdown following a Seismic Event and LOCA. Conditions are as follows:

- Suppression Pool level is 196.5 feet and steady.
- RHR Loop A is operating in LPCI mode.
- RHR Loop B is operating in Suppression Pool Cooling mode.

Which one of the following describes the method for determining Suppression Pool temperature, in accordance with NER-2M-039, NMP 2 EOP Basis Document?

Use...

A. Suppression Pool temperature indication on SPDS.

B. RHR heat exchanger inlet temperature on RHR Loop B.

C. RHR heat exchanger outlet temperature on RHR Loop A.

D. post-accident monitor Suppression Pool temperature indicators.

Proposed Answer: B

Explanation: With Suppression Pool level <197 ft, one method given for determining temperature is the HTX inlet temp provided that the pump is aligned with suction from the Suppression Pool, such as RHR Loop B operating in Suppression Pool Cooling mode.

A. Plausible – Normally correct, but all installed Suppression Pool temperature indicators are uncovered.
C. Plausible – Heat exchanger outlet temperature is available, but would not accurately reflect Suppression Pool temperature due to cooling provided by heat exchanger.
D. Plausible – Normally correct, but all installed Suppression Pool temperature indicators are uncovered.

Technical Reference(s): NER-2M-039 Section for SPT-6

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank -2012 Audit #51

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

02072014: Added the words “Seismic Event” to question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EA2.04
	Importance Rating	3.9

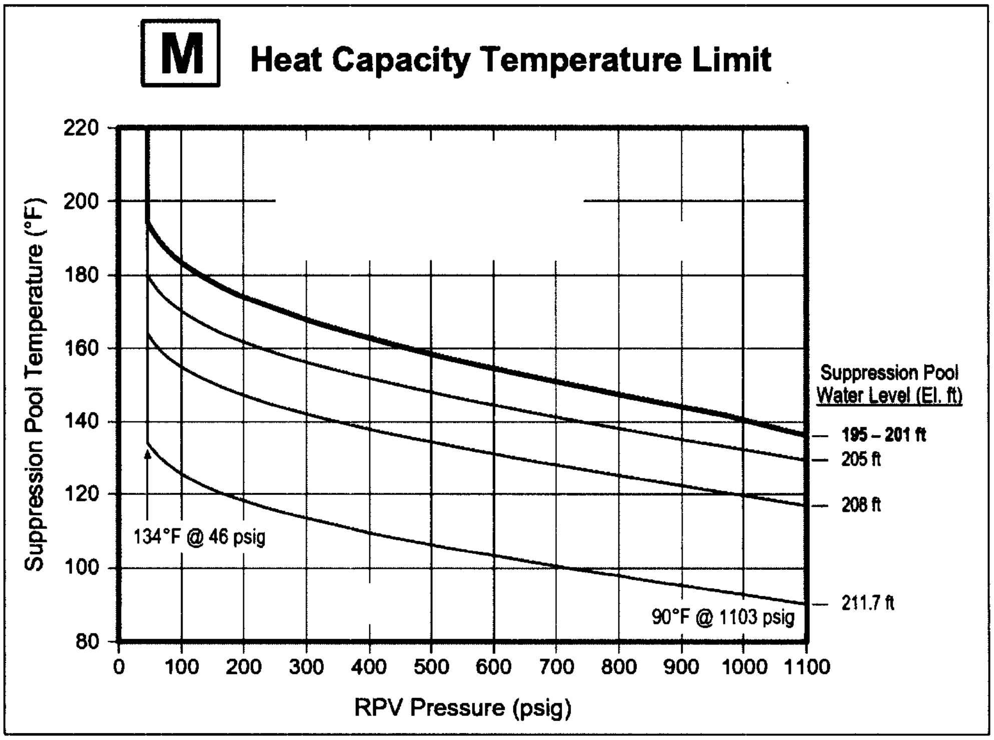
High Reactor Pressure

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression pool level

Proposed Question: #52

The plant has experienced a Failure to Scram. Conditions are as follows:

- Reactor pressure is 850 psig and is being controlled 800-1000 psig with RCIC and SRVs.
- Suppression Pool temperature is 130°F and very slowly rising.
- Suppression Pool water level is 202 feet and very slowly rising.



Which one of the following describes the impact of these parameters on the required Reactor pressure control strategy, in accordance with N2-EOP-C5, Failure to Scram, N2-EOP-PC, Primary Containment Control, and the EOP Bases?

A. An RPV Blowdown is required.

B. Reactor pressure must be maintained at or below approximately 900 psig.

C. Reactor pressure must be maintained at or below approximately 500 psig.

D. Continued use of the entire current Reactor pressure band is acceptable.

Proposed Answer: D

Explanation: With Suppression Pool water level at 202 feet, the operator may interpolate between the 201 and 205 foot curve. With Suppression Pool temperature at 130°F, the maximum Reactor pressure allowed is approximately 1100 psig. Since the entire 800 to 1000 psig pressure range is on the good side of the curve, then the operator can continue to use the current pressure band

A. Plausible – HCTL is not being violated with the given conditions.
B. Plausible – This would be true if suppression pool temperature were higher.
C. Plausible – Reactor pressure above ~1100 psig would be on the BAD side of the HCTL curve. This would be correct if Suppression Pool Temperature were higher.

Technical Reference(s): N2-EOP-PC, EOP Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Modified – NMP1 2009 NRC #15

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 #	295031 EA2.02
	Importance Rating	4.0

Reactor Low Water Level

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Reactor power

Proposed Question: #53

The plant has experienced a Failure to Scram. Conditions are as follows:

- Reactor water level reaches a low of 115 inches and then begins rising.
- Reactor power stabilizes at 1-2% on APRMs.
- Reactor pressure is being controlled on the Main Turbine Bypass Valves.

In accordance with the Emergency Operating Procedures, which one of the following correctly completes the below statements?

Terminating and preventing all RPV injection except boron, CRD, and RCIC __(1)__ required .

The ability to use High Pressure Core Spray (HPCS) for RPV injection once in the proper level band is __(2)__.

	(1)	(2)
A.	IS	RESTRICTED
B.	IS	UNRESTRICTED
C.	IS NOT	RESTRICTED
D.	IS NOT	UNRESTRICTED

Proposed Answer: C

Explanation: Reactor power is below 4%, N2-EOP-C5 directs maintaining Reactor water level between -39 and 202.3 inches and does NOT direct terminating and preventing injection. Even though Reactor water level is NOT required to be intentionally lowered, use of HPCS is not allowed unless Reactor water level cannot be maintained with Preferred ATWS Systems after an RPV Blowdown.

A. Plausible – While conditions require entry into N2-EOP-C5, since Reactor power is <4%, RPV injection is not terminated and prevented.
B. Plausible – While conditions require entry into N2-EOP-C5, since Reactor power is <4%, RPV injection is not terminated and prevented. Even though Reactor power is <4%, use of HPCS for injection is still restricted.
D. Plausible – Even though Reactor power is <4%, use of HPCS for injection is still restricted.

Technical Reference(s): N2-EOP-RPV, N2-EOP-C5

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Modified – SYSID105247

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

02102014: Rearranged question stem. Changed column 1 answers from YES/NO to IS/IS NOT. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 2.1.20
	Importance Rating	4.6

Partial or Complete Loss of Instrument Air

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

Proposed Question: #54

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

- Instrument air pressure on Panel 851 (2IAS-PI101) indicates 84 psig and slowly lowering.

Which one of the following actions is required, in accordance with N2-SOP-19, Loss of Instrument Air?

A. Scram the Reactor.

B. Perform an emergency power reduction.

C. Dispatch an Operator to monitor IAS pressures in the Reactor Building.

D. Verify 2IAS-AOV171, Instrument Air to Service Air Cross-Tie, auto-opens.

Proposed Answer: C

Explanation: If Instrument air pressure, as indicated on Panel 851 2IAS-PI101, lowers to < 85 psig, then the override of N2-SOP-19 requires verifying 2IAS-AOV171 auto-closes and dispatching an Operator to monitor two air pressure indications in the Reactor Building. These air pressure indications must be monitored to assess subsequent conditional scram steps in the procedure.

A. Plausible – A scram is required if 2IAS-PI101 lowers to < 70 psig.
B. Plausible – Since a scram will be required if conditions continue to degrade, it may be prudent to perform an emergency power reduction prior to the scram. However, this is not a requirement of N2-SOP-19.
D. Plausible – 2IAS-AOV171 does automatically re-position, but should be verified closed, not open.

Technical Reference(s): N2-SOP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-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 #	295030 2.1.27
	Importance Rating	3.9

Low Suppression Pool Water Level

Conduct of Operations: Knowledge of system purpose and / or function.

Proposed Question: #55

The plant is at 100% power.

Which one of the following describes:

(1) the minimum Suppression Pool water level required by Technical Specifications, and

(2) the threshold Suppression Pool water level requiring RPV Blowdown, in accordance with N2-EOP-PC, Primary Containment Control?

A. (1) 195 feet
(2) 190 feet

B. (1) 195 feet
(2) 192 feet

C. (1) 199.5 feet
(2) 190 feet

D. (1) 199.5 feet
(2) 192 feet

Proposed Answer: D

Explanation: The Suppression Pool functions to condense steam discharged from Downcomers, SRV discharge lines, and/or the RCIC turbine exhaust line. To ensure these functions are met, Technical Specifications require Suppression Pool water level to be maintained ≥ 199.5 feet while in Modes 1, 2, and 3. The Downcomer openings first become exposed to the Suppression Chamber air space when water level reaches 190 feet, however the N2-EOP-PC threshold value for RPV Blowdown is 192 feet based on the lowest readable Suppression Pool water level.

Note: The question satisfies the K/A by testing knowledge of the minimum Suppression Pool water levels required for the Primary Containment system to meet its design functions.

A. Plausible – Technical Specifications require 199.5 feet. 195 feet corresponds to the N2-EOP-PC warning regarding ECCS and RCIC damage. 190 feet corresponds to the actual Suppression Pool water level at which Downcomer openings would become exposed to the Suppression Chamber air space. However, the N2-EOP-PC threshold value for RPV Blowdown is 192 feet based on the lowest readable Suppression Pool water level.
B. Plausible – Technical Specifications require 199.5 feet. 195 feet corresponds to the N2-EOP-PC warning regarding ECCS and RCIC damage.
C. Plausible – 190 feet corresponds to the actual Suppression Pool water level at which Downcomer openings would become exposed to the Suppression Chamber air space. However, the N2-EOP-PC threshold value for RPV Blowdown is 192 feet based on the lowest readable Suppression Pool water level.

Technical Reference(s): Technical Specification 3.6.2.2, N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223001-RBO-10 and N2-223001-RBO-14

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 #	1
	Group #	1
	K/A #	295023 2.1.32
	Importance Rating	3.8

Refueling Accidents

Conduct of Operations: Ability to explain and apply system limits and precautions.

Proposed Question: #56

Which one of the following describes:

(1) the maximum number of New Fuel Storage Vault covers that may be removed at a time during fuel handling, and

(2) the reason for this limitation,

in accordance with N2-OP-39, Fuel Handling and Reactor Service Equipment, and the UFSAR?

A. (1) One
(2) Criticality control

B. (1) One
(2) Contamination control

C. (1) Two
(2) Criticality control

D. (1) Two
(2) Contamination control

Proposed Answer: A

Explanation: Only one New Fuel Storage Vault cover may be removed at a time during fuel handling. This is part of the New Fuel Storage Vault criticality control strategy.

B. Plausible – New fuel assemblies are radioactive material with the potential for contamination in the event of a manufacturing defect, however this limitation is designed for criticality control, not contamination control.
C. Plausible – Candidate may believe two covers are allowed to transfer fuel between areas of the vault. There are 32 covers total.
D. Plausible – Candidate may believe two covers are allowed to transfer fuel between areas of the vault. There are 32 covers total. New fuel assemblies are radioactive material with the potential for contamination in the event of a manufacturing defect, however this limitation is designed for criticality control, not contamination control.

Technical Reference(s): N2-OP-39 P&L 13.1, UFSAR 9.1.1.3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-234000-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(13)

Comments:

02102014: No change to the question, but Brian Fuller required information regarding N2-OP-39 P&L 13.1 and 13.2. After talking with Mark Rushmore of Operations, he did some research and determined that P&L 13.1 and 13.2 of OP-39 are saying essentially the same. Assuming all of the new fuel is in the New Fuel Vault, if you follow P&L 13.1 which states that you cannot have more than one New Fuel Storage Vault Cover removed at a time, then it is not possible to uncover more than 10 new Fuel Assemblies at a time as stated P&L 13.2. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AK1.05
	Importance Rating	3.3

Partial or Complete Loss of DC Power

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Loss of breaker protection

Proposed Question: #57

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

- A blown fuse results in loss of DC control power to the closed supply breaker for 2NPS-SWG001.

Which one of the following describes the effect on the supply breaker operation?

The supply breaker...

A. trips open and CANNOT be re-closed from the Control Room.

B. remains closed, can still be operated from the Control Room, but loses protective trips.

C. remains closed, CANNOT be operated from the Control Room, and loses protective trips.

D. remains closed, CANNOT be operated from the Control Room, but retains all protective trips.

Proposed Answer: C

Explanation: 2BYS-SWG001A supplies DC control power for 2NPS-SWG001 supply breakers. Upon loss of the DC control power, the closed 2NPS-SWG001 supply breaker remains closed, but loses remote operation capability and protective trips.

A. Plausible – Candidate may believe that control power is required to maintain the breaker closed, or the breaker has a protective feature to automatically open upon loss of control power.
B. Plausible – Candidate may believe there is separate control power for remote operation versus protective features.
D. Plausible – Candidate may believe there is separate control power for remote operation versus protective features.

Technical Reference(s): N2-SOP-04 Section 2.4 and Attachment 5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-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 #	1
	Group #	1
	K/A #	295037 EK2.11
	Importance Rating	3.8

SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: RMCS: Plant-Specific

Proposed Question: #58

The plant has experienced a Failure to Scram. Conditions are as follows:

- The RO is manually inserting control rods using the Reactor Manual Control System (RMCS) per N2-EOP-6.14, Alternate Control Rod Insertion.

Per N2-EOP-6.14, which one of the following correctly completes the following statement?

When using RMCS to manually drive rods, the Rod Worth Minimizer is __(1)__. Control Rods are inserted __(2)__.

A. (1) bypassed.
   (2) starting in the center and spiraling to the outside.

B. (1) bypassed.
   (2) starting on the outside and spiraling to the center.

C. (1) not bypassed.
   (2) starting in the center and spiraling to the outside.

D. (1) not bypassed.
   (2) starting on the outside and spiraling to the center.

Proposed Answer: A

Explanation: When inserting control rods using RMCS during a Failure to Scram condition, N2-EOP-6.14 requires the RWM to be in Bypass. This is because the order in which the rods are inserted is different from the programmed rod sequence. Not having the RWM in bypass will cause a rod block which will stop RMCS from inserting the control rods. Per N2-EOP-6.14, the rod insertion sequence starts with rods in the center and spirals out to the outside.

B. Plausible – Figure 2 and 3 of N2-EOP-6.14 have the control rods starting in the center and spiraling out not outside spiraling in.
C. Plausible – The RWM is normally not bypassed when performing a rod insertion sequence for a normal shutdown.
D. Plausible – The RWM is normally not bypassed when performing a rod insertion sequence for a normal shutdown.

Technical Reference(s): N2-EOP-6.14

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

02102014: Rearranged question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 AK1.05
	Importance Rating	3.3

Inadvertent Containment Isolation

Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION: Loss of drywell/containment cooling

Proposed Question: #59

The plant is at 100% power when a malfunction within the PCIS system causes an inadvertent isolation of the Reactor Building Closed Loop Cooling Water to the Drywell Coolers.

Per N2-OP-60, Drywell Cooling System, which one of the following is the operational implication of this isolation if NO operator actions are taken?

The Drywell Cooling fans will...

A. trip and the plant will eventually scram on a subsequent high Drywell pressure.

B. trip and Drywell temperature will slowly rise, however no scram will occur.

C. continue to operate, but the plant will scram on a subsequent high Drywell pressure.

D. continue to operate and Drywell temperature will slowly rise, however no scram will occur.

Proposed Answer: A

Explanation: An isolation of the CCP cooling water valves will cause an automatic trip of the Drywell Cooling Fans. If all drywell cooling is lost during power operation, the drywell temperature/pressure rise will be rapid and immediate action should be taken to prevent a high drywell pressure signal (1.68 psig) from occurring.

B. Plausible – Drywell cooling must be restored or a Reactor scram on high Drywell pressure will occur.
C. Plausible – Drywell Cooling fans automatically trip on isolation of the RBCLC cooling water valves.
D. Plausible – Drywell Cooling fans automatically trip on isolation of the RBCLC cooling water valves. Drywell cooling must be restored or a reactor scram on high drywell pressure will occur.

Technical Reference(s): N2-OP-60 Section B.3 and P&L 3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223004-RBO-8

Question Source: Bank – 2009 NRC #59

Question History: 2009 NRC #59

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EK2.08
	Importance Rating	2.6

High Suppression Pool Water Level

Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: Drywell/suppression chamber ventilation

Proposed Question: #60

A loss of coolant accident with fuel damage has occurred. Conditions are as follows:

- Primary Containment venting has been directed using N2-EOP-6.21, Containment Venting, due to high Drywell pressure.
- Drywell pressure is 45 psig and stable.
- Suppression Chamber pressure is 35 psig and slowly rising
- Primary Containment water level is 220 feet and slowly rising.

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

Vent from the...

A. Drywell, because this is the preferred venting lineup.

B. Suppression Chamber, because this is the preferred venting lineup.

C. Drywell, because the preferred venting lineup from the Suppression Chamber is NOT available.

D. Suppression Chamber, because the preferred venting lineup from the Drywell is NOT available.

Proposed Answer: C

Explanation: The preferred venting lineup is from the Suppression Chamber, however Primary Containment water level is too high (> 217 feet) to vent from the Suppression Chamber. Therefore, the Drywell should be vented.

A. Plausible – While Drywell pressure is higher, the Suppression Chamber is the preferred vent source because it allows the Suppression Pool to scrub the vented gases.
B. Plausible – The preferred venting lineup is from the Suppression Chamber, however Primary Containment water level is too high (> 217 feet) to vent from the Suppression Chamber.
D. Plausible – Primary Containment water level is too high (> 217 feet) to vent from the Suppression Chamber. The Drywell is NOT the preferred vent source and is available for venting, even though Drywell pressure is higher.

Technical Reference(s): N2-EOP-PC Step PCP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

02102014: Changed Drywell pressure form 40 psig to 45 psig based on NRC direction. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 AK3.01
	Importance Rating	3.7

Loss of CRD Pumps

Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: Reactor SCRAM

Proposed Question: #61

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

- The running CRD pump has tripped.
- The standby CRD pump has failed to start.
- Two accumulator trouble alarms have been received for withdrawn control rods.
- Both control rods have accumulator pressures of 920 psig.
- The CRD pumps cannot be started.

Per N2-OP-30, Control Rod Drive Failures, which one of the following correctly completes the below statement?

A Reactor scram is required ___(1)___. This scram is required due to degraded ___(2)___.

A. (1) immediately
   (2) control rod scram capability

B. (1) immediately
   (2) cooling to the CRD mechanisms

C. (1) 20 minutes after the loss of CRD Charging Header pressure
   (2) control rod scram capability

D. (1) 20 minutes after the loss of CRD Charging Header pressure
   (2) cooling to the CRD mechanisms

Proposed Answer: C

Explanation: With Reactor power at 100%, Reactor pressure is above 900 psig. With Reactor pressure above 900 psig, N2-SOP-30 requires a Reactor scram if two or more accumulators for withdrawn control rods are inoperable AND CRD charging water header pressure is <940 psig for 20 minutes. The reason for the scram is degraded control rod scram capability.

A. Plausible – Only if Reactor pressure were < 900 psig would an immediate Reactor scram be required.
B. Plausible – Only if Reactor pressure were < 900 psig would an immediate Reactor scram be required. Cooling is lost to the CRD mechanisms with the loss of CRD pressure, however this is NOT the reason for the scram.
D. Plausible – Cooling is lost to the CRD mechanisms with the loss of CRD pressure, however this is NOT the reason for the scram.

Technical Reference(s): N2-SOP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP30C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

02102014: Added bullet to stem stating that CRD pumps cannot be started. Rearranged question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	2
K/A #	295009 AA1.03
Importance Rating	3.0

Low Reactor Water Level

Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Recirculation system: Plant-Specific

Proposed Question: #62

The plant is at 100% power when a LOCA occurs. Conditions are as follows:

[illegible]	[illegible]
0	180
30	155
60	115
90	100

Reactor water level has been continuously lowering throughout this transient.

Which one of the following identifies the status of the Recirc pumps at Time = 30 seconds and at Time = 90 seconds?

	Time = 30 Seconds	**Time = 90 Seconds**
A.	Running at High Speed	Running at Low Speed
B.	Running at High Speed	Tripped
C.	Running at Low Speed	Running at Low Speed
D.	Running at Low Speed	Tripped

Proposed Answer: D

Explanation: The Recirc pumps are initially in high speed, based on the 100% initial power level. The Recirc pumps downshift from high to low speed with no time delay when Reactor water level reaches 159.3" (Level 3). Therefore, at Time = 30 seconds (155"), the pumps are running in low speed. The Recirc pumps trip with no time delay when Reactor water level reaches 108.8" (Level 2). Therefore, at Time = 90 seconds (100"), the pumps have tripped.

A. Plausible – The Recirc pumps are initially running at high speed and downshift to low speed before tripping.
B. Plausible – The Recirc pumps are initially running at high speed.
C. Plausible – The Recirc pumps downshift to low speed before tripping.

Technical Reference(s): N2-OP-29 P&Ls 8.0 and 9.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-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 #	2
	K/A #	295032 EA2.01
	Importance Rating	3.8

High Secondary Containment Area Temperature

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

Proposed Question: #63

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

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

ATTACHMENT 28
DETERMINING REACTOR BUILDING TEMPERATURES
Table 28-3 Reactor Bldg Areas
(Partial Table Only)

RX BLDG RCIC PMP ROOM RX BLDG RADIOACTIVE PIPE CHASE EL 206	
E31-N602A *(P632)*	198°F
E31-N612A *(P632)*	185°F
E31-N602B *(P642)*	150°F
E31-N612B *(P642)*	162°F

RX BLDG GENL AREA EL 206	
E31-N641A (P632)	131°F

RX BLDG GENL AREA EL 206	
E31-N641B (P642)	124°F

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

A. Zero

B. One

C. Two

D. Three

Proposed Answer: B

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Modified – 2012 Audit #85

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 #	295010 2.4.11
	Importance Rating	4.0

High Drywell Pressure

Emergency Procedures / Plan: Knowledge of abnormal condition procedures.

Proposed Question: #64

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

- Annunciator 602115, RECIRC PUMP 1A SEAL STAGING FLOW HIGH/LOW is in alarm.
- Upper seal pressure (B35-R602A) indicates 980 psig.
- Lower seal pressure (B35-R603A) indicates 1000 psig.
- Drywell pressure is 0.41 psig and slowly rising.

Which one of the following actions is required in accordance with N2-SOP-29.1, Recirc Pump Failures?

A. Trip Recirc pump A and then close the suction and discharge isolation valves.

B. Trip Recirc pump A, but do NOT isolate the pump unless Drywell pressure rises further.

C. Perform a normal shutdown of Recirc pump A per N2-OP-29, Reactor Recirculation System.

D. Continue to monitor Recirc pump A. Do NOT trip the pump unless seal pressures degrade further.

Proposed Answer: A

Explanation: With upper seal cavity pressure >920 psig and Drywell pressure rising, N2-SOP-29.1 requires tripping the affected pump and then closing the suction and discharge isolation valves.

B. Plausible – Once the pump is required to be tripped, no further evaluation of parameters is conducted before also isolating the pump.
C. Plausible – The conditional step in N2-SOP-29.1 directs tripping the pump. If the conditional step were not met, then a subsequent step would give the option of performing this normal shutdown.
D. Plausible – Upper seal cavity pressure is above the 920 psig threshold in N2-SOP-29.1 which, when combined with rising Drywell pressure, requires pump trip. Lower seal cavity pressure is still below its 1200 psig threshold.

Technical Reference(s): N2-SOP-29.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-RBO-10

Question Source: Modified – SYSID 94255

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 #	500000 EK3.03
	Importance Rating	3.0

High Containment Hydrogen Concentrations

Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Operation of hydrogen and oxygen recombiners

Proposed Question: #65

Which one of the following describes (1) the operation of the Hydrogen (H_2) Recombiners and (2) an associated reason for this requirement, in accordance with N2-EOP-PCH, Hydrogen Control, and the EOP Bases?

H_2 Recombiners are ___(1)___ when H_2 concentration is above 5% to ___(2)___.

A. (1) only allowed to be operated
(2) prevent damage to the heater elements

B. (1) only allowed to be operated
(2) ensure an adequate reaction can be maintained

C. (1) secured
(2) prevent gas ignition or excessive reaction temperatures

D. (1) secured
(2) prevent excessive moisture buildup in the reaction chamber

Proposed Answer: C

Explanation: N2-EOP-PCH requires securing the H_2 Recombiners when H_2 concentration exceeds 5%. There are two reasons associated with this requirement: (1) eliminate a potential ignition source for a deflagration in the containment and (2) prevent excessive reaction temperatures from damaging the H_2 Recombiner.

A. Plausible – H_2 Recombiners are secured, not started, when H_2 concentration exceeds 5%. Excessive temperatures damaging heater elements is a concern, but at high H_2 concentration, not low.
B. Plausible – H_2 Recombiners are secured, not started, when H_2 concentration exceeds 5%. Some H_2 is required to make the reaction self-sustaining.
D. Plausible – Excessive moisture buildup in the reaction chamber is the reason for continuous operation of the H_2 Recombiner trickle heaters.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCHC01 EO-2

Question Source: Modified – 2009 NRC #67

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 #	2.1.44
	Importance Rating	3.9

Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.

Proposed Question: #66

The plant is in Mode 5 with a full core offload in progress. Conditions are as follows:

- You are the Operator At-The-Controls.
- Manipulation of RHR Shutdown Cooling flow rate is required.

Per N2-OP-31, Residual Heat Removal, and N2-FHP-003, Refueling Manual, which one of the following correctly completes the below statement?

Total drive flow through the jet pumps must be maintained...

A. greater than 5700 gpm to maintain stable jet pump operation.

B. greater than 5700 gpm to prevent thermal stratification in the core.

C. less than 5700 gpm to maintain water visual clarity for the Refuel Bridge team.

D. less than 5700 gpm to prevent damage to instrument tubes or lifting blade guides.

Proposed Answer: D

Explanation: With core off load in progress, in-core instrumentation may NOT be fully surrounded by fuel assemblies and/or blade guides. In this condition, it is required to maintain total drive flow through the jet pumps to less than 5700 gpm to prevent damage to the instrument tubes from flow induced vibration and to prevent lifting the blade guides.

A. Plausible – 5700 gpm is the limitation, but operation must be below, not above this limit.
B. Plausible – 5700 gpm is the limitation, but operation must be below, not above this limit.
C. Plausible – Water clarity is a concern to allow Refuel Bridge operators to see during fuel movements, but it is not the basis for this limitation. Higher SDC flow rates could potentially cause more particulates to be swept into the coolant and reduce visibility.

Technical Reference(s): N2-OP-31 Section D.6.0, N2-FHP-003 Section 4.1.4

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(2)

Comments:

02102014: Rearranged question stem per NRC direction. DH.

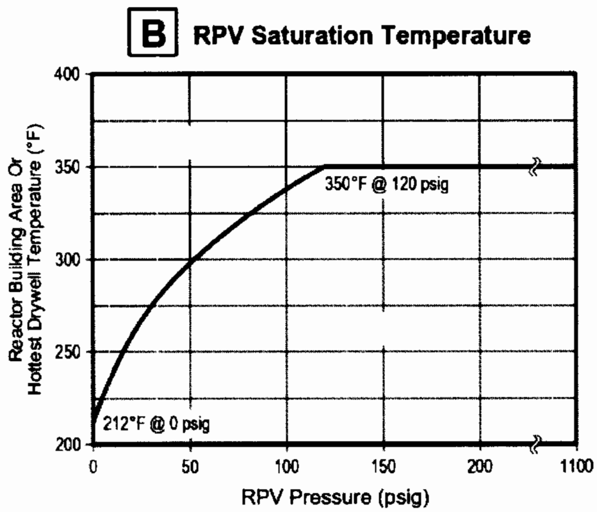
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.45
	Importance Rating	4.3

Ability to identify and interpret diverse indications to validate the response of another indicator.

Proposed Question: #67

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

- The hottest Reactor Building temperature is 110°F and slowly rising.
- The hottest Drywell temperature is 310°F and slowly rising.
- Reactor pressure is 50 psig and stable.
- There is no indication of instrument leg flashing for any RPV Level Instrument
- The SRO has directed Reactor water level be controlled in the preferred control band given in N2-EOP-RPV, RPV Control.



Which one of the following identifies the Reactor water level instrument(s) (if any) that may be used to trend Reactor water level across the entire preferred level band, in accordance with N2-EOP-RPV?

A. Wide Range, only

B. Wide Range and Narrow Range, only

C. Wide Range, Narrow Range, and Fuel Zone, only

D. No Reactor water level indications are valid across this entire level band

Proposed Answer: B

Explanation: The given Drywell / Reactor Building temperatures and Reactor pressure place operation in the BAD region of the RPV Saturation Temperature curve. Even though operation is in the BAD region of the RPV Saturation Temperature curve, instruments may still be used as long as indications of flashing are not present and as long as they are above the minimum indicated level. The preferred RPV level band per N2-EOP-RPV is 159.3 inches to 202.3 inches (160-200 inches). Both Wide Range and Narrow Range instruments are capable of trending over the entire 160 to 200 inch band, however, Fuel Zone would be off scale high for this band. Upset and Shutdown Range cannot be used because 160 inches is below their minimum indicated levels per N2-EOP-RPV.

A. Plausible – Although Wide Range is able to be used in this level band, Narrow Range is also on scale.
C. Plausible – Although Fuel Zone is still available to determine if water level is above the TAF, it cannot trend level between 160 to 200 inches as it would be off scale high.
D. Plausible – If there were evidence of reference leg flashing, then no level instruments could be used.

Technical Reference(s): N2-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.0

Knowledge of the process used to track inoperable alarms.

Proposed Question: #68

Which one of the following identifies the flagging tool color used to indicate that a single input to a multiple input Control Room annunciator has been disabled, in accordance with CNG-OP-1.01-2003, Alarm Response and Control?

A. Red

B. Blue

C. Black

D. Yellow

Proposed Answer: D

Explanation: A yellow flagging tool is used to identify a multiple input annunciator which has had one or more inputs removed from service.

A. Plausible – A red flagging tool is used for an annunciator that is completely tagged out.
B. Plausible – A blue flagging tool is used for an annunciator that is locked in or removed from service.
C. Plausible – A black flagging tool is used for an annunciator that is in for maintenance or nuisance.

Technical Reference(s): CNG-OP-1.01-2003 Section 5.2.D.1

Proposed references to be provided to applicants during examination: None

Learning Objective: S-ODP-OPS-0001, Conduct Of Operations TO-01

Question Source: Bank – 2010 Audit #68

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 #	2.2.7
	Importance Rating	2.9

Knowledge of the process for conducting special or infrequent tests.

Proposed Question: #69

Preparations are under way for an Infrequently Performed Test or Evolution (IPTE). Conditions are as follows:

- The IPTE Integrated Pre-Job Brief is about to begin.
- One Reactor Operator (RO) who will be involved in performing the IPTE is currently unavailable to attend the brief.

Which one of the following describes the requirement for briefing this RO, in accordance with CNG-HU-1.01-1002?

A. The RO may be briefed separately, but this brief must occur before the RO performs in the IPTE.

B. The Integrated Pre-Job Brief must be re-scheduled for when all individuals involved in the IPTE can attend at the same time.

C. The RO can perform in the IPTE without receiving the Integrated Pre-Job Brief as long as they are NOT performing any critical steps.

D. The RO can perform in the IPTE without receiving the Integrated Pre-Job Brief as long as a peer-check is received from someone who was briefed.

Proposed Answer: A

Explanation: IPTEs requires an Integrated Pre-Job Brief. All workers involved in a job requiring a brief must be briefed before participation in the job. A worker may be granted permission to be absent from the initial brief, as long as they are briefed separately before taking part in the job.

B. Plausible – The purpose of the Integrated Pre-Job Brief is to allow all team members to participate in a discussion of the job, so it is preferred, although not required, that all members attend the same brief.
C. Plausible – Critical steps are those identified as having significant negative consequences if performed incorrectly, and are thus the most important to be briefed. However, the RO must receive the brief even before participating at all in the IPTE.
D. Plausible – A peer check from a briefed individual would theoretically provide protection from an error, however the RO is required to actually be briefed personally before participating in the IPTE.

Technical Reference(s): CNG-OP-4.01-1000 Section 5.7.D, CNG-HU-1.01-1000 Section 5.1.D

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-OP-4.01-1000-CT-01

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 #	2.3.11
	Importance Rating	3.8

Ability to control radiation releases.

Proposed Question: #70

The plant has experienced an un-isolable Main Steam Line break inside the Turbine Building.

Which one of the following describes how to control Turbine Building ventilation (HVT), in accordance with N2-EOP-RR, Radioactivity Release Control?

A. Isolate HVT to minimize the release from the Turbine Building.

B. Verify HVT is operating to maintain an elevated, monitored release.

C. Isolate HVT to prevent transferring air between the Turbine Building and Reactor Building.

D. Verify HVT is operating to maintain Turbine Building pressure lower than Reactor Building pressure.

Proposed Answer: B

Explanation: HVT is verified operating to maintain a monitored, elevated release pathway. If HVT trips or is shutdown, Turbine Building pressure will rise and radioactivity may be released through unmonitored, ground level pathways, such as building doors and cracks in walls.

A. Plausible – If the steam leak were in the Reactor Building HVR would be isolated to minimize release.
C. Plausible – If the steam leak were in the Reactor Building HVR would be isolated. It is desirable to minimize how much contamination enters the RB, but HVT is not isolated for this purpose. Some HVT exhaust may be drawn into HVR inlet as a result of continued HVT operation.
D. Plausible – HVT is verified operating, but to maintain monitored, elevated release. While in this situation is desirable to prevent leakage from TB to RB, TB pressure will not be less than RB pressure under even normal circumstances.

Technical Reference(s): N2-EOP-RR Step RR-1 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRRC01 EO-2

Question Source: Bank – NMP1 2010 NRC #56

Question History: NMP1 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 #	2.3.4
	Importance Rating	3.2

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: #71

The plant is shutdown. Conditions are as follows:

- A 31 year old Operator is entering the Drywell for a job.
- General area dose rate is 4 Rem/hr.
- The Operator's TEDE for the year is 1210 mRem.
- The Operator's lifetime exposure is 25 Rem.
- No dose extensions have been obtained.
- It will take 45 minutes to complete the job.

Which one of the following describes the dose gained in order to complete the job, in accordance with GAP-RPP-07, Internal and External Dosimetry Program?

The operator's dose...

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

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

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

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

Proposed Answer: C

Explanation: The normal administrative dose control limit without any extensions is 2000 mRem TEDE, the CENG annual administrative limit is 4000 mRem, and the Federal annual limit is 5000 mRem. The total dose the worker will have received after completing the job will be 1210 + 4000 (45 / 60) = 4210 mRem. This will require an extension beyond the 2000 mRem dose control limit and will also exceed the CENG annual administrative limit of 4000 mRem, but will NOT exceed the Federal annual limit of 5000 mRem.

A. Plausible – The dose will remain within the 5000 mRem federal limit, but exceeds the normal 2000 mRem dose control level.
B. Plausible – The expected dose does exceed the normal dose control level, but also exceeds the CENG annual administrative limit.
D. Plausible – The expected dose does exceed the CENG annual administrative limit of 4000 mRem, but does not exceed the Federal annual limit of 5000 mRem.

Technical Reference(s): GAP-RPP-07

Proposed references to be provided to applicants during examination: None

Learning Objective: GAP-RPP-07-TO01

Question Source: Modified – 2010 NMP1 NRC #75

Question History: 2010 NMP1 NRC #75

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(12)

Comments:

TRH 02072014 – Revised answer choices to raise plausibility of one distractor, based on NRC comment.

02102014: Reviewed question. No comments. DH.

TRH 02202014 – Revised answer choices based on validator comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.31
	Importance Rating	4.2

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #72

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

- An evolution is in progress.
- The operator controlling the evolution identifies an alarm that will be received after the next manipulation.

Given the following statements:

(1) Alarm must be discussed with Control Room SRO before receipt of alarm.
(2) Alarm response procedure must be reviewed before receipt of alarm.
(3) Alarm must be specifically identified as expected in the procedure that controls the evolution.

Which one of the following identifies which of these statements are requirements to classify the alarm as an "Expected Alarm", in accordance with CNG-OP-1.01-2003, Alarm Response and Control?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: The alarm must be discussed with the Control Room SRO and the ARP must be reviewed before receipt of the alarm to classify it as an "Expected Alarm" per CNG-OP-1.01-2003.

B. Plausible – The ARP also must be reviewed before receipt of the alarm. Many alarms are listed as part of a procedure and this may help in identifying an upcoming alarm, but it is not required to classify an alarm as an "Expected Alarm".
C. Plausible – The alarm also must be discussed with the Control Room SRO before receipt of the alarm. Many alarms are listed as part of a procedure and this may help in identifying an upcoming alarm, but it is not required to classify an alarm as an "Expected Alarm".
D. Plausible – Many alarms are listed as part of a procedure and this may help in identifying an upcoming alarm, but it is not required to classify an alarm as an "Expected Alarm".

Technical Reference(s): CNG-OP-1.01-2003 Section 3.4

Proposed references to be provided to applicants during examination: None

Learning Objective: S-ODP-OPS-0001, Conduct Of Operations TO-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

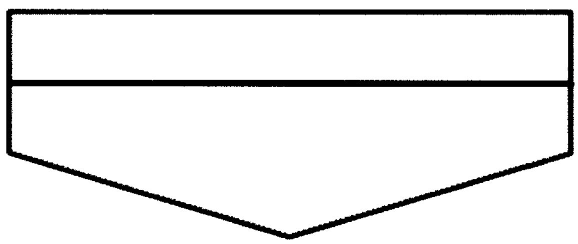
Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.19
	Importance Rating	3.4

Knowledge of EOP layout, symbols, and icons.

Proposed Question: #73

Which one of the following is indicated by an arrow-shaped pentagon (see symbol below) in the Emergency Operating Procedures, in accordance with NER-2M-039, NMP2 Emergency Operating Procedures and Severe Accident Procedures Basis Document?



A. Hold Point

B. Before Step

C. Override Step

D. Decision Point

Proposed Answer: B

Explanation: Arrow-shaped pentagons symbolize conditions coordinating the timing of subsequent steps. The associated actions should be performed, if possible, before the specified conditions occur.

A. Plausible – Hold points are shown as octagons.
C. Plausible – Override steps are shown as rectangles with rounded corners.
D. Plausible – Decision points are shown as diamonds.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-TMG-INTRO-EO-01

Question Source: Bank – NMP1 2013 Audit #74

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 02072014 – Resampled K/A based on NRC comment.

02102014: Reviewed question. No comments. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.37
	Importance Rating	3.6

Ability to determine operability and / or availability of safety related equipment.

Proposed Question: #74

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

- A piece of safety-related equipment required by Technical Specifications has been out-of-service for two days for maintenance overhaul.
- All maintenance is now complete.
- All safety tags have been removed.
- The equipment has been restored to a normal standby lineup per the associated Operating Procedure.
- Post-Maintenance Testing of the equipment is about to start.

Which one of the following describes the current status of the equipment per CNG-OP-1.01-2002, Operations Shift Turnover and Relief?

The equipment is...

A. inoperable, but available.

B. non-functional, but available.

C. both inoperable and unavailable.

D. both non-functional and unavailable.

Proposed Answer: A

Explanation: CNG-OP-1.01-2002 contains definitions for various system status designations, such as operable, available, and functional. Since the equipment has been physically restored to a normal standby lineup, it is available for use. However, operability is not restored until the equipment is tested (PMT) to ensure it meets all required design functions. Additionally, the term functional does not apply per CNG-OP-1.01-2002 because this is equipment required by Technical Specifications, and functional is used to describe non-Technical Specification required equipment.

B. Plausible – If this equipment was not required by Technical Specifications, the term "functional" would apply instead of "operable", and the equipment is currently inoperable.
C. Plausible – The equipment is available and should be operable, however operability must be proven through testing before being formally declared.
D. Plausible – The equipment was inoperable and unavailable during the maintenance, however availability is restored when the equipment is returned to the normal standby lineup. If this equipment was not required by Technical Specifications, the term "functional" would apply instead of "operable".

Technical Reference(s): CNG-OP-1.01-2002, Sections 3.1, 3.4, and 5.3.B6

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-COO-OBJ-02

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

TRH 02072014 – Added information to the question stem and revised distracters due to one implausible distracter. Completed per NRC direction.

02102014: Reviewed question. No comments. DH.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.15
	Importance Rating	2.9

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #75

Given the following radiation monitors:

(1) Offgas radiation monitors
(2) Drywell area radiation monitors
(3) Main Steam Line radiation monitors
(4) Reactor Building area radiation monitors

Which one of the following identifies which radiation monitors are directly used to determine entry into N2-SOP-17, Fuel Failure?

A. (1) or (2) only

B. (1) or (3) only

C. (2) or (4) only

D. (3) or (4) only

Proposed Answer: B

Explanation: N2-SOP-17 is entered due to either high Offgas or Main Steam Line radiation.

A. Plausible – Drywell area rad monitors are used in determining extent of fuel damage in the E-Plan.
C. Plausible – Drywell area rad monitors are used in determining extent of fuel damage in the E-Plan. RB area rad monitors may experience elevated readings due to fuel damage and are used to determine entry into N2-EOP-SC.
D. Plausible – RB area rad monitors may experience elevated readings due to fuel damage and are used to determine entry into N2-EOP-SC.

Technical Reference(s): N2-SOP-17

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP17C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

02102014: In the answer and each distracter, changed “and” to “or” and added the word “only” to the end. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.03
	Importance Rating	2.9

Partial or Complete Loss of DC Power

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Battery voltage

Proposed Question: #76

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

- Power has been lost to 2CES*IPNL414.
- N2-SOP-04, Loss of DC Power is being executed.
- The cause of the loss of DC power has been determined and corrected.
- The crew is attempting to restore DC power to 2CEC*IPNL414
- Electrical Maintenance has informed the control room that the Division 3 battery voltage is at 103 VDC.

What actions shall the Control Room Supervisor direct to restore the Division 3 DC system and why in accordance with N2-SOP-04?

The CRS shall direct the crew to restore the Division 3...

A. Battery Charger first based on the concern that the Battery may be unable to re-energize de-energized devices.

B. Battery Charger first based on the concern that the Battery may be unable to maintain DC loads for at least 2 hours.

C. Battery first based on the concern that the Battery Charger will not have sufficient load to operate correctly.

D. Battery first based on the concern that the Battery Charger may go into current limiting mode and trip.

Proposed Answer: A

Explanation: Per N2-SOP-04, attempting to reenergize the Division 3 DC bus with a battery <105 VDC may cause deenergized devices not to re-energize. In cases where battery voltage is <105 VDC, N2-SOP-04 directs restoring the Battery Charger first.

B. Plausible – The ability to maintain loads for 2 hours is part of the basis for battery operability in Technical Specifications, however it is not a concern when attempting to restore DC power to the bus.
C. Plausible – The battery would be restored to service first if voltage was greater than 105 VDC.
D. Plausible – The battery would be restored to service first if voltage was greater than 105 VDC.

Technical Reference(s): N2-SOP-04 and Section 5.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

02102014: Changed “should” to “shall” in question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295023 AA2.03
	Importance Rating	3.8

Refueling Accidents

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Airborne contamination levels

Proposed Question: #77

The plant is in Mode 5 with fuel movement in progress. Conditions are as follows:

- A seismic event occurs
- Significant damage has occurred to the Spent Fuel Pool
- The Reactor Building has isolated based on high airborne radiation levels
- GTS Trains A and B are operating and maintaining the RB DP negative
- The Shift Manager has declared an ALERT based on Main Stack Effluent radiation levels.
- Radiation Protection Personnel have not yet begun on and offsite surveys in support of dose assessment.
- The SM is filling out the following section on the NMP 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.		

For the above Part 1 Fact Sheet, which one of the following describes which letter to circle on Block 5?

A. A – No release

B. B – Release below federal limits (ODCM)

C. C – Release above federal limits (ODCM)

D. D – Unmonitored release requiring evaluation

Proposed Answer: C

Explanation: Per EPMP-EPP-0102, the DRMS RED alarm setpoint is conservatively set to ensure the ODCM radioactivity release limits are not exceeded. With an ALERT being declared due to main stack effluent radiation levels, it means that release rates are 200 times the high RED alarm setpoint. 200 times the setpoint is indicative of release rates above the ODCM limits.

A. Plausible – Since both GTS Trains A and B are both running and designed to filter out airborne contamination, the candidate may determined that filtered release could be considered no release.
B. Plausible – The candidate may assume that a higher level of classification is needed before release limits exceed the ODCM limits.
D. Plausible – The candidate might assume that since RP personnel are not yet performing surveys, that this release could be considered an unmonitored release.

Technical Reference(s): EPMP-EPP-0102

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRRC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.01
	Importance Rating	3.6

Loss of Shutdown Cooling

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

Proposed Question: #78

The plant is in Mode 4 during a shutdown. Conditions are as follows:

- Shutdown Cooling has been interrupted due to an equipment malfunction.
- The following Reactor coolant temperature readings have been taken:

Time (hhmm)	Reactor Coolant Temperature (°F)
0000	140
0005	152

Assuming the Reactor coolant temperature trend remains constant, which one of the following describes the resulting heat-up rate and when the plant Mode will change, in accordance with Technical Specifications (TS)?

	Heat-Up Rate	Time of Mode Change (hhmm)
A.	Below TS limit	0025
B.	Below TS limit	0030
C.	Above TS limit	0025
D.	Above TS limit	0030

Proposed Answer: C

Explanation: The current heat-up rate is 144°F/hr [(152°F – 140°F)*(60 min/hr)/(5 min)]. This is above the 100°F/hr limit of Technical Specification Surveillance Requirement 3.4.11.1. The plant transitions from Mode 4 to Mode 3 at 200°F. Based on the current heat-up rate, this temperature will be reached at time 0025.

Note: This question meets SRO level guidelines because it requires knowledge of a Technical Specification Surveillance Requirement and cannot be answered solely on "above-the-line" or definition information.

A. Plausible – Candidate may either miscalculate heat-up rate or not know limit.
B. Plausible – Candidate may either miscalculate heat-up rate or not know limit. 0030 is the time at which Reactor coolant temperature will reach 212°F and begin to boil. However, the Mode change is at 200°F, not 212°F.
D. Plausible – 0030 is the time at which Reactor coolant temperature will reach 212°F and begin to boil. However, the Mode change is at 200°F, not 212°F.

Technical Reference(s): Technical Specification Table 1.1-1 and Surveillance Requirement 3.4.11.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-101001-RBO-14

Question Source: Modified – JAF 9/2012 NRC #96

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295018 2.4.46
	Importance Rating	4.2

Partial or Complete Loss of CCW

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

Proposed Question: #79

The plant is in Mode 4 with preparations in progress for a Reactor Startup. Conditions are as follows:

- Reactor Coolant Temperature is 105°F.
- Recirculation Pumps A and B are running in low speed.
- RHS B is operating in Shutdown Cooling Mode.

Then...

- An event occurs in the Reactor Building Closed Loop Cooling System (CCP).
- Indications for the CCP System after the event are found on the following page.
- The crew has entered N2-SOP-13, Loss or Degraded CCP System.
- A Plant Operator reports that CCP Surge Tank Level is in the normal band.
- Other than to silence and acknowledge/reset applicable panel alarms, no operator actions have been taken.

Per N2-SOP-13, which one of the following correctly completes the below statement?

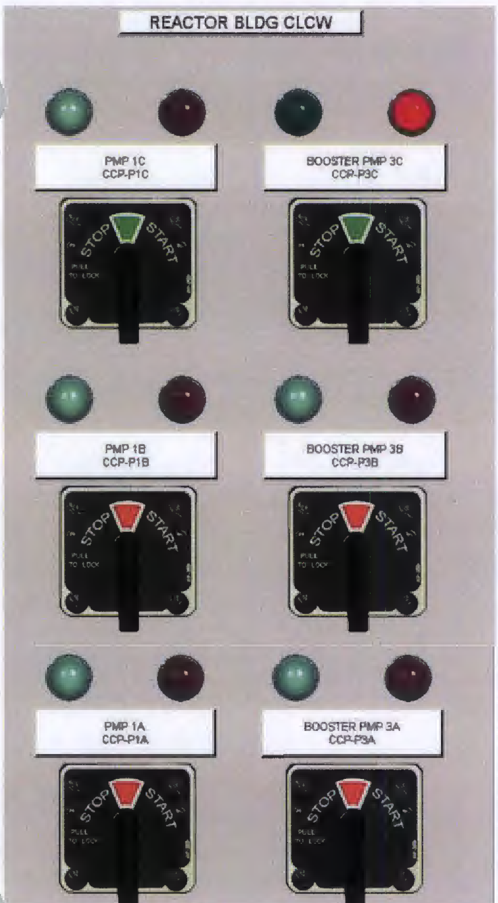
CCP annunciators ___(1)___ consistent with the current system configuration. The CRS must direct ___(2)___ to mitigate this event.

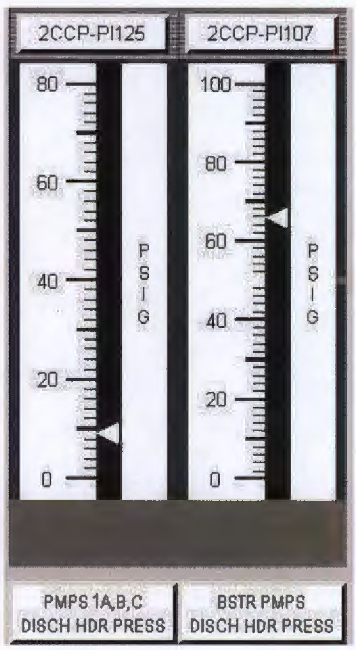
	(1)	(2)
A.	ARE	shifting RHS Seal Cooling to Service Water per N2-OP-13
B.	ARE	exiting N2-SOP-13 and recovering the CCP system per N2-OP-13
C.	ARE NOT	shifting RHS Seal Cooling to Service Water per N2-OP-13
D.	ARE NOT	exiting N2-SOP-13 and recovering the CCP system per N2-OP-13

Note: All lit annunciators are solid (i.e. not flashing)

DIVISION II SERVICE WATER SYSTEM INOPERABLE 201	SERVICE WATER PUMP 1B/1D/1F PMP/MOTOR BRG TEMP HIGH 202	SER WTR VALVE FV54B/FV47B HYDR UNIT TROUBLE 203		DIVISION I RBCLCW ISOL VALVES INOPERABLE 205	DIVISION II RBCLCW ISOL VALVES INOPERABLE 206				
DIVISION II SERVICE WATER VALVES MOT OVERLOAD 211	SERVICE WATER SUPPLY HEADER PRESS B LOW 212		SERVICE WATER PUMP 1B/1D/1F MOTOR OVERLOAD 214	SERVICE WATER PUMP 1B/1D/1F AUTO START 215	SERVICE WATER PUMP 1B/1D/1F AUTO TRIP / FAIL TO START 216	SERVICE WATER PUMP 1B/1D/1F MOTOR/FEEDER ELEC FAULT 217	SERVICE WATER PUMP 1B/1D/1F SUCTION PRESS LOW 218	SERVICE WATER PUMP 1B/1D/1F DISCH FLOW LOW 219	INTAKE SHAFT GATE 30B CLOSED 220
DIVISION III DSL GEN SER WTR VLVS MOT OVERLOAD 221	SERVICE WATER STRAINER 4B/4D/4F MOT OVERLOAD 222	SER WTR STR 4B/4D/4F DIFF PRESSURE HI-HI 223					CHILLER 1B CONDENSING WTR PUMP 2B SUCT PRESS LO 228		
RBCLCW PUMPS 3A/3B/3C AUTO START 231	RBCLCW PUMPS 3A/3B/3C AUTO TRIP / FAIL TO START 232	RBCLCW PUMPS 3A/3B/3C MOTOR OVERLOAD 233	RBCLCW PUMPS 3A/3B/3C DISCH PRESS LOW 234						
			TURBINE BLDG CLOSED LOOP COOLING SYS TROUBLE 244		REACTOR BLDG CLOSED LOOP COOLING SYS TROUBLE 246				
RBCLC PUMP 1A/1B/1C AUTO START 251	RBCLC PUMP 1A/1B/1C AUTO TRIP / FAIL TO START 252	RBCLC PUMP 1A/1B/1C MOTOR OVERLOAD 253	RBCLC PUMP 1A/1B/1C DISCH PRESS LOW 254	RBCLCW TO RHR PUMP 1A/1B/1C PRESSURE LOW 255	RBCLCW TO RWCU FLOW HIGH 256	RBCLCW TO REACTOR RECIRC PUMP A CLRS PRES LOW 257	RBCLCW TO REACTOR RECIRC PUMP B CLRS PRES LOW 258	RBCLC FROM DRYWELL UNIT COOLERS TEMP HIGH 259	DRYWELL UNIT COOLERS LEAKAGE HIGH 260

ANNUNCIATOR 601200





Proposed Answer: C

Explanation: The given indications show that Booster Pump 3C is green flagged but its associated red indicating light is lit. Based on this information and the indication of pressure on the discharge header, the candidate can assume that Pump 3C auto started as expected in this situation. Because the pump control switch is still green flagged and its associated breaker is shut, Annunciator 601231 should be in alarm and it is not. Because Booster Pump 3C is running and Annunciator 601231 is not lit, plant conditions are not consistent with the alarms. Additionally, P1C red light is not lit which indicates that it is not running. However Annunciator 601251 is lit which indicates that the pump has auto started (green flagged but breaker shut). Annunciator 601251 should not be lit in this situation and is another plant condition that is not consistent with the plant indications.

An override in N2-SOP-13, states that if an RHS pump is operating, then shift seal cooling to Service Water per N2-OP-13, H.7.0. Because RHS B is operating in shutdown cooling mode, N2-SOP-13 requires that seal cooling water be shifted to RHS B.

A. Plausible – With the exception of Annunciator 601231, all other indications and alarms are consistent with plant conditions.
B. Plausible – With the exception of Annunciator 601231, all other indications and alarms are consistent with plant conditions. Additionally, you would only exit N2-SOP-13 and recover per N2-OP-13 if the CCP Expansion tank were found to be empty.
D. Plausible – You would only exit N2-SOP-13 and recover per N2-OP-13 if the CCP Expansion tank were found to be empty.

Technical Reference(s): N2-SOP-13, ARP 601231

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-276000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

02102014: Rearranged question stem per NRC direction. Changed first column answer choices from YES/NO to ARE/ARE NOT. Completed per NRC direction. DH.

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

High Reactor Pressure

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

Proposed Question: #80

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

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

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

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

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

Proposed Answer: A

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

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

Technical Reference(s): Technical Specification 3.4.12 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-101001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

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

Partial or Complete Loss of AC Power

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #81

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

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

The following sequence then occurs:

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

Which one of the following describes the emergency action levels resulting from these events, in accordance with EPIP-EPP-02, Classification of Emergency Conditions at Unit 2?

Note: Apply all required time limits.

The first EAL exceeded was an ___(1)___.
The highest EAL exceeded was a(n) ___(2)___.

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

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

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

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

Proposed Answer: B

Explanation: At time 00:00, Unusual Event SU1.1 was exceeded due to loss of all offsite power to Division 1 and 2 switchgear. At time 00:15, the loss of the Division 2 EDG, combined with the inoperability of the Division 3 EDG, started a clock for an Alert to be met at time 00:25. At time 00:45, the highest EAL exceeded is Site Area Emergency SS1.1 due to loss of all offsite and onsite power to Division 1 and 2 switchgear.

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

Technical Reference(s): EPIP-EPP-02 Attachment 1

Proposed references to be provided to applicants during examination: EPIP-EPP-02 Attachment 1

Learning Objective: N2-EAL-UP-CE-1.09

Question Source: Bank – 2012 NRC #79

Question History: 2012 NRC #79

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

02102014: Deleted 00:45 in question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	700000 2.4.8
	Importance Rating	4.5

Generator Voltage and Electric Grid Disturbances

Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question: #82

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

- N2-SOP-70, Major Grid Disturbances, is being executed.
- The Reactor scrams.
- Reactor water level is 150 inches and slowly rising.

Which one of the following sets of actions is correct?

A. Enter N2-EOP-RPV, RPV Control. Continue performing N2-SOP-70. In the event of a conflict between the procedures, N2-SOP-70 is the overriding document.

B. Enter N2-EOP-RPV, RPV Control. Continue performing N2-SOP-70. In the event of a conflict between the procedures, N2-EOP-RPV is the overriding document.

C. Exit N2-SOP-70 and enter N2-EOP-RPV, RPV Control. N2-SOP-70 is re-entered at the step in-progress after exiting N2-EOP-RPV.

D. Exit N2-SOP-70 and enter N2-EOP-RPV, RPV Control. N2-SOP-70 entry conditions are re-evaluated after exiting N2-EOP-RPV.

Proposed Answer: B

Explanation: N2-EOP-RPV is entered due to Reactor water level less than 159.3 inches. There is no requirement to exit SOPs when EOPs are entered. In fact, both procedures are executed concurrently. The EOPs are higher-tiered documents than the SOPs, therefore in the event of a conflict, the EOP must be followed.

A. Plausible – N2-SOP-70 was entered first and is event-specific, however the EOP is a higher-tiered document.
C. Plausible – N2-EOP-RPV is a higher-tiered document, however there is no requirement to exit the SOP.
D. Plausible – N2-EOP-RPV is a higher-tiered document, however there is no requirement to exit the SOP.

Technical Reference(s): GAI-OPS-20 Section 3.3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOP00C01 TO #1

Question Source: Bank – SYSID 105324

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	500000 EA2.04
	Importance Rating	3.3

High Containment Hydrogen Concentrations

Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for wetwell

Proposed Question: #83

A LOCA has occurred. Conditions are as follows:

- Suppression Pool water level is 200 ft and stable.
- Drywell H_2 concentration is 6%.
- Suppression Chamber H_2 concentration is 9%.
- Drywell O_2 concentration is 4%.
- Suppression Chamber O_2 concentration is 3%.
- The offsite release rate from any Containment venting/purging is expected to reach the Alert level.

Which one of the following describes the status of the Primary Containment H_2/O_2 deflagration limit and the need to direct Primary Containment purging, in accordance with N2-EOP-PCH, Hydrogen Control, and the EOP Bases?

	H_2/O_2 Deflagration Limit	**Direct Primary Containment Purge?**
A.	Below	No
B.	Below	Yes
C.	Above	No
D.	Above	Yes

Proposed Answer: A

Explanation: Both the Drywell and Suppression Chamber are below the deflagration limit (both 6% H_2 and 5% O_2 needed together). N2-EOP-PCH Circles 31 and 34 are applicable. These steps direct purging only if the release rate will stay below ODCM limits. Since release rate is expected to reach the Alert level (which is greater than the ODCM limits) if any venting/purging is performed, purge should NOT be directed.

B. Plausible – Purging would be directed if the projected release rate were lower.
C. Plausible – The H_2 concentrations would support deflagration is more O_2 were present.
D. Plausible – The H_2 concentrations would support deflagration is more O_2 were present. Purging would be directed if the projected release rate were lower.

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

Proposed references to be provided to applicants during examination: N2-EOP-PCH with all notes regarding SP level restrictions and all directions on Recombiner use deleted

Learning Objective: 2101-EOPPCHC01 EO-2

Question Source: Modified – 2005 Audit SRO #10

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 #	295029 2.4.47
	Importance Rating	4.2

High Suppression Pool Water Level

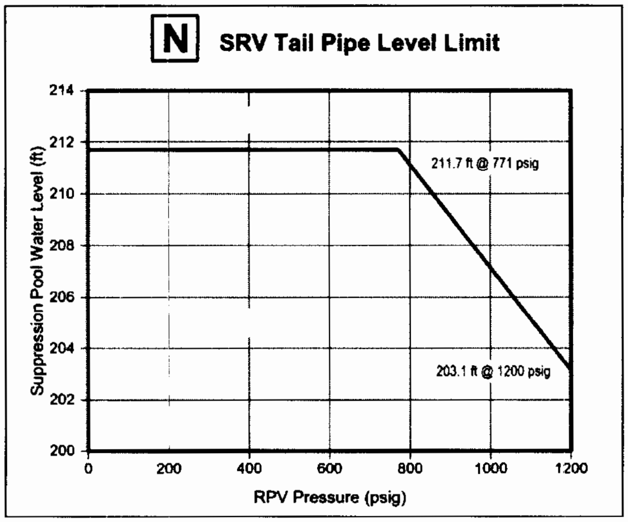
Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #84

A LOCA has occurred. Conditions are as follows:

- Reactor water level is -10 inches and stable with only Condensate injecting.
- No other injection systems are available.
- Reactor pressure is 230 psig and stable.
- Suppression Pool water temperature is 100°F and stable.
- Suppression Pool water level is 209 feet and rising 0.1 feet per minute.
- Attempts to lower Suppression Pool water level have been unsuccessful.

Assume all parameter trends remain constant.



Which one of the following describes:

(1) When Suppression Pool water level will initially reach the SRV Tail Pipe Level Limit, and

(2) the action required to be directed if Suppression Pool water level CANNOT be restored and maintained below this level?

A. (1) ≤ 30 minutes
(2) Perform an RPV Blowdown.

B. (1) ≤ 30 minutes
(2) Stop injection into the Reactor from sources outside Primary Containment.

C. (1) > 30 minutes
(2) Perform an RPV Blowdown.

D. (1) > 30 minutes
(2) Stop injection into the Reactor from sources outside Primary Containment.

Proposed Answer: A

Explanation: For a Reactor pressure of 230 psig, the SRV Tail Pipe Level Limit is 211.7 feet. With the current Suppression Pool level and trend, this level will be reached in 27 minutes [(211.7 feet – 209 feet)/0.1 feet per minute]. N2-EOP-PC requires an RPV Blowdown if Suppression Pool water level cannot be restored and maintained below the SRV Tail Pipe Level Limit.

B. Plausible – Stopping injection from external sources is directed before RPV Blowdown, but only if not needed for adequate core cooling. In this case, Reactor water level is just above TAF with no other injection systems available, so Condensate injection must not be stopped.
C. Plausible – 30 minutes is only slightly greater than the actual time of 27 minutes. 80 minutes is the time to reach 217 feet in the Suppression Pool, which is another action level in this EOP leg to secure Drywell sprays.
D. Plausible – 30 minutes is only slightly greater than the actual time of 27 minutes. 80 minutes is the time to reach 217 feet in the Suppression Pool, which is another action level in this EOP leg to secure Drywell sprays. Stopping injection from external sources is directed before RPV Blowdown, but only if not needed for adequate core cooling. In this case, Reactor water level is just above TAF with no other injection systems available, so Condensate injection must not be stopped.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

02102014: Rearranged question stem. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295022 AA2.01
	Importance Rating	3.6

Loss of CRD Pumps

Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: Accumulator pressure

Proposed Question: #85

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

- The running CRD pump trips.
- Annunciator 603441, Rod Drive Accumulator Trouble, alarms.
- The standby CRD pump is started and charging water header pressure is restored.
- Four HCUs are still alarming for withdrawn control rods.
- An Operator reports the following accumulator pressures for the alarming HCUs:

HCU	Accumulator Pressure (psig)
02-19	910
10-19	1015
10-11	960
22-11	850

Which one of the following describes the current impact of these conditions on Technical Specifications?

__(1)__ control rod scram accumulator(s) is(are) inoperable. Unless accumulator pressure(s) is(are) restored, declare the associated control rod(s) scram time "slow" or declare the associated control rod(s) inoperable within __(2)__ hour(s).

A. (1) One
(2) one

B. (1) One
(2) eight

C. (1) More than one
(2) one

D. (1) More than one
(2) eight

Proposed Answer: C

Explanation: Technical Specification Surveillance Requirement 3.1.5.1 requires accumulator pressure ≥ 940 psig. Therefore, HCUs 02-19 and 22-11 are inoperable. Technical Specification 3.1.5 Condition B then applies, which requires either declaring the associated control rod scram times "slow" or declaring the associated control rods inoperable within 1 hour.

A. Plausible – Only HCU 22-11 has pressure < 900 psig, which is the Reactor pressure referenced in TS 3.1.5.
B. Plausible – Only HCU 22-11 has pressure < 900 psig, which is the Reactor pressure referenced in TS 3.1.5. Eight hours is the correct completion time for a single inoperable HCU.
D. Plausible – Eight hours is the correct completion time for a single inoperable HCU.

Technical Reference(s): Technical Specification 3.1.5

Proposed references to be provided to applicants during examination: Technical Specification 3.1.5 without bases or surveillance requirements

Learning Objective: N1-201001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	209001 A2.03
Importance Rating	3.6

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: #86

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

- CSL Pump Room Unit Cooler, 2HVR*UC402A, has been removed from service for maintenance.
- The AC breaker for CSL Pump Room Unit Cooler, 2HVR*UC402B fails open and cannot be shut
- Concurrently, 2CSH*P2, WTR LEG PMP 2 trips on motor electric fault.

Which one of the following describes the impact of these conditions on continued plant operation, in accordance with Technical Specifications?

A. Place the plant in Mode 2 within 7 hours.

B. Plant operation may continue for up to 72 hours before entering a shutdown LCO.

C. Plant operation may continue for up to 7 days before entering a shutdown LCO.

D. Plant operation may continue for up to 14 days before entering a shutdown LCO.

Proposed Answer: A

Explanation: With both CSL Pump Room Unit Coolers out of service, CSL must be declared inoperable. When 2CSH*P2 trips concurrently with the power loss, HPCS is also considered inoperable., Technical Specification 3.5.1 Condition H requires entering LCO 3.0.3. LCO 3.0.3 requires Mode 2 within 7 hours.

Note: The question meets the K/A by presenting an AC electrical loss which must be analyzed to predict/determine the effect on Core Spray based on Unit Cooler loss. Based on this prediction/determination, Technical Specifications must be used to determine the allowable operating time, which is required to ensure correct implementation of the plant shutdown procedure.

B. Plausible – TS 3.5.1 Condition C would allow 72 hours before entering a shutdown LCO if the conditions resulted in either 2 ECCS injection subsystems or one ECCS injection and one ECCS spray subsystem inoperable.
C. Plausible – TS 3.5.1 Condition A is also entered, which by itself would not result in a shutdown LCO for 7 days.
D. Plausible – TS 3.5.1 Condition B is also entered, which by itself would not result in a shutdown LCO for 14 days.

Technical Reference(s): N2-OP-52 Attachment 3, Technical Specifications 3.0.3 and 3.5.1

Proposed references to be provided to applicants during examination: Technical Specification 3.5.1 without bases

Learning Objective: N1-209001-RBO-14

Question Source: Modified – 2012 Audit #86

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 #	212000 A2.03
	Importance Rating	3.5

RPS

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Surveillance testing

Proposed Question: #87

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

- Surveillance testing is in progress on the RPS System
- At 04:00 today, I&C reported that RPS Reactor Pressure High trip unit B22-N678A failed downscale and cannot generate an RPS trip on high RPV pressure.
- At 06:00 today, RPS Reactor Pressure High trip unit B22-N678B also failed downscale and cannot generate an RPS trip on high RPV pressure.
- I&C has determined that the remaining RPS Reactor Pressure high trip units are all functioning correctly.

As of 06:00 today, which one of the following meets the requirements of Technical Specifications?

A. Place RPS A trip system in trip by 16:00 today. There are no additional actions required per TS 3.3.1.1.

B. Place RPS A in trip by 12:00 today and be in Mode 3 by 06:00 tomorrow.

C. Place RPS B in trip by 12:00 today and be in Mode 3 by 06:00 tomorrow.

D. Restore RPS trip capability for the High Reactor Pressure function per TS 3.3.1.1 Condition C by 07:00 today or be in Mode 3 by 19:00 today.

Proposed Answer: B

At 04:00, one of two high pressure RPS trip units for RPS A fails. Condition A of TS 3.3.1.1 is entered and the RPS A trip system is required to be tripped by 16:00 (12 hours later). At 06:00, one of the two high pressure RPS trip units for RPS B also fails. Condition A and B are then entered at 06:00. Condition B requires that one of the two trips systems be placed in trip by 12:00 (6 hours later). Because of the Separate Condition entry allowed note at the beginning of TS 3.3.1.1, Condition A requires RPS B trip system be tripped by 18:00 as well. By tripping RPS A at 12:00, the actions of Condition A and B are satisfied for RPS A, however Condition A is still in effect for RPS B being tripped at 18:00. At 18:00, Condition D is entered which directs you to Condition H for being in Mode 3 within 12 hours. 12 hours from 18:00 is 06:00 the next day.

A. Plausible – Tripping RPS A by 16:00 today is an acceptable action per Condition A, however Condition B is also in effect and would require that RPS B be tripped as well by 12:00.
C. Plausible – Tripping RPS B would be an acceptable action per Condition B, however if the applicant chose to trip RPS B, then the plant would have to be in Mode 3 by 04:00 tomorrow, not 06:00.
D. Plausible – This would be true if both of the failed trip units came from the same trip system, however the trip units are from separate trip systems.

Technical Reference(s): TS 3.3.1.1 and associated bases.

Proposed references to be provided to applicants during examination: TS 3.3.1.1, No Bases.

Learning Objective: N2-278001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

02132014: Resampled K/A. Wrote new question. Completed per NRC direction. DH.

02202014: Added "As of 06:00 today" to question stem. Revised distracter A. Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262002 2.1.23
	Importance Rating	4.4

UPS (AC/DC)

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

Proposed Question: #88

A plant shutdown is in progress. Conditions are as follows:

- Containment purging is in progress.
- Then, 2VBB-UPS1A is lost due to a sustained electrical fault.
- 2VBB-UPS1A loads CANNOT yet be re-energized.

Which one of the following describes how to control the Containment purge, in accordance with N2-SOP-71, Loss of UPS?

A. Station an Operator with no concurrent duties to perform a Containment purge manual isolation if it becomes necessary.

B. Verify Div 1 Containment purge manual isolation tripped within one hour. Div 2 Containment purge manual isolation does NOT need to be tripped.

C. Verify Div 2 Containment purge manual isolation tripped within one hour. Div 1 Containment purge manual isolation does NOT need to be tripped.

D. Verify Div 1 and 2 Containment purge manual isolations tripped within one hour or isolate the Containment purge penetrations within two hours.

Proposed Answer: D

Explanation: N2-SOP-71 has a specific step for loss of 2VBB-UPS1A requiring either tripping Div 1 and Div 2 Containment purge manual isolations within 1 hour or isolating the Containment purge isolations within 2 hours. The reason for this requirement is to satisfy TS 3.3.6.1 Condition B or F due to inoperability of the Group 9 isolation on SGTS exhaust radiation from the loss of 2VBB-UPS1A.

A. Plausible – The concern is loss of automatic isolation capability, so a dedicated Operator for manual isolation is a reasonable compensatory measure, just not the procedurally required measure.
B. Plausible – Div 1 must be tripped, but Div 2 is also affected and must also be tripped.
C. Plausible – Div 2 must be tripped, but Div 1 is also affected and must also be tripped.

Technical Reference(s): N2-SOP-71

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262002-RBO-10

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 #	2
	Group #	1
	K/A #	218000 2.4.6
	Importance Rating	4.7

ADS

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: #89

The reactor is shutdown after a LOCA and the CRS has determined an RPV Blowdown is required. Conditions are as follows:

1. An RO has been directed to open all ADS valves.
2. The RO reports five ADS valves have opened.
3. The RO is then directed to open two additional SRVs.
4. The RO reports only one additional SRV could be opened.
5. Reactor pressure is 750 psig and lowering.

Which one of the following describes the alternate mitigation strategy, if any, the CRS must now use for these conditions, in accordance with N2-EOP-C2, RPV Blowdown?

A. Direct the RO to depressurize the RPV using other Blowdown systems. Isolations and interlocks are NOT allowed to be defeated.

B. Direct the RO to depressurize the RPV using other Blowdown systems. Isolations and interlocks are allowed to be defeated, but only if release rate limits are NOT exceeded.

C. Direct the RO to depressurize the RPV using other Blowdown systems. Isolations and interlocks are allowed to be defeated, even if release rate limits are exceeded.

D. No additional Blowdown systems are required to be used. Wait until the Shutdown Cooling interlock clears and then direct Shutdown Cooling placed in service.

Proposed Answer: D

Explanation: N2-EOP-C2 directs opening all 7 ADS valve. If any ADS valves fail to open, additional SRVs are opened until a total of 7 SRVs are open. After these actions are carried out, the CRS must make a determination as to the adequacy of the number of open SRVs. If less than 6 SRVs are open, additional Blowdown systems must be used. Additionally, isolations and interlocks are allowed to be defeated and release rate limits are allowed to be exceeded if necessary. However, since 6 SRVs are open, no additional Blowdown systems are required to be used. N2-EOP-C2 requires waiting until the SDC interlock clears, then placing SDC in service.

A. Plausible – If less than 6 SRVs were open, N2-EOP-C2 would require using additional Blowdown systems. Isolations/interlocks are allowed to be defeated under these circumstances and release rates are allowed to be exceeded.
B. Plausible – If less than 6 SRVs were open, N2-EOP-C2 would require using additional Blowdown systems. Isolations/interlocks are allowed to be defeated under these circumstances and release rates are allowed to be exceeded.
C. Plausible – If less than 6 SRVs were open, N2-EOP-C2 would require using additional Blowdown systems. Isolations/interlocks are allowed to be defeated under these circumstances and release rates are allowed to be exceeded.

Technical Reference(s): N2-EOP-C2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC2C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

TRH 02072014 – Revised question to have higher difficulty level, based on NRC comment. The revision puts the question farther down the N2-EOP-C2 decision tree and does not allow using RO knowledge to help arrive at correct answer.

02112014: Reviewed question. Changed bullets to numbers for sequencing and added that the reactor was shutdown in the question stem. DH.

02202014: Minor revision to question stem based on NRC direction. DH.

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	261000 2.2.12
	Importance Rating	4.1

SGTS

Equipment Control: Knowledge of surveillance procedures.

Proposed Question: #90

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

- The crew has just completed monthly surveillance testing on GTS Train A.
- 2GTS*PV5A FILTER TRAIN A RECIRCULATION VALVE did not function correctly and has been declared inoperable.
- 2GTS*PV5A is currently closed.

Which of the following action(s) is(are) required to be directed by the CRS to maintain GTS Train A operable with 2GTS*PV5A inoperable in accordance with N2-OP-61B, Standby Gas Treatment System?

(1) Shut 2GTS*V52, FAN 1A RECIRC LINE ISOL.
(2) Shut 2GTS*V103, GTS*PV5A AIR ISOLATION.
(3) Install a locking device on 2GTS*PV5A.

A. (3)

B. (1) OR (2)

C. (1) AND (3)

D. (2) AND (3)

Proposed Answer: B

Explanation: Per N2-OSP-GTS-M001, Note in Section 8.1, GTS Train A may be considered operable with an inoperable 2GTS*PV5A if EITHER 2GTS*V52 is shut, OR 2GTS*PV5A is closed and the air isolated.

A. Plausible – N2-OSP-GTS-M001 only requires one of the actions, not both.
C. Plausible Although isolating air to PV5A would be an acceptable action to take to maintain GTS operability, no procedural guidance exists to install a locking device on PV5A.
D. Plausible – Although isolating air to PV5A or shutting V52 would be an acceptable action to take to maintain GTS operability, no procedural guidance exists to install a locking device on PV5A.

Technical Reference(s): N2-OSP-GTS-M001, Section 8.1 and N2-OP-61B.

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-261000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

02112014: Rearranged question stem and answer/distracters based on NRC direction. DH.

02202014: Revised distracters C and D to "and" vs. "or". Completed per NRC direction. DH.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	215002 A2.04
	Importance Rating	2.8

RBM

Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply losses: BWR-3,4,5

Proposed Question: #91

The plant is at 80% power. Conditions are as follows:

- A control rod pattern adjustment is in progress.
- An electrical fault causes a complete loss of power to Rod Block Monitor (RBM) A.

Which one of the following describes the impact of this power loss and the appropriate direction to be given in response to this event?

RBM A...

A. enforces a rod block. Enter N2-SOP-97, Reactor Protection System Failures, and direct further control rod movements to be suspended.

B. enforces a rod block. Direct bypassing RBM A per N2-OP-92, Neutron Monitoring, if desired to continue with control rod movements.

C. is prevented from enforcing a rod block. Enter N2-SOP-97, Reactor Protection System Failures, and direct placing the channel in trip.

D. is prevented from enforcing a rod block. Direct continuation of control rod movements for a period of time as allowed by Technical Specifications.

Proposed Answer: B

Explanation: Upon loss of power, RBM A enforces a rod block. TS 3.3.2.1 Conditions A and B allow one inoperable RBM for up to 25 hours before requiring the channel be placed in trip. Therefore, it is allowable to bypass the RBM for up to 25 hours to clear the rod block and enable continued control rod movements. This is accomplished using N2-OP-92

A. Plausible – RBM A does enforce a rod block and both RBMs are required channels per Tech Specs, however Tech Specs do allow for continued control rod movement with one RBM inoperable (ie bypassed) for up to 25 hours. N2-SOP-97 may be entered for a half scram, but no half scram results from this event.
C. Plausible – RBM A does require electrical power to operate properly, but is designed to cause a rod block upon loss of power. Both RBMs are required channels per Tech Specs, however Tech Specs do allow for continued control rod movement with one RBM inoperable (ie bypassed) for up to 25 hours. N2-SOP-97 may be entered for a half scram, but no half scram results from this event.
D. Plausible – RBM A does require electrical power to operate properly, but is designed to cause a rod block upon loss of power. Continued control rod movement is allowed.

Technical Reference(s): ARP 603204, Technical Specification 3.3.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

02112014: Reworded distracter D per NRC direction. DH.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	272000 2.2.22
	Importance Rating	4.7

Radiation Monitoring

Conduct of Operations: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #92

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

- Annunciator 851257, Stack Effluent Monitor Trouble, alarms.
- Chemistry reports the noble gas monitoring function of WRGMS has failed.

Which one of the following is the required action in response to this condition, in accordance with the Offsite Dose Calculation Manual?

A. Place the non-functional channel in the tripped condition within 12 hours.

B. Use auxiliary sampling equipment to start continuous collection of samples within 8 hours.

C. Take grab samples of the effluent stream within 12 hours and once per 12 hours thereafter.

D. Immediately suspend the gaseous effluent release monitored by the inoperable equipment.

Proposed Answer: C

Explanation: With a gaseous effluent monitoring channel non-functional, ODCM D 3.3.2 Condition B applies. Table D 3.3.2-1 3.a references Condition F for the noble gas activity monitor. Condition F requires taking grab samples within 12 hours and once per 12 hours thereafter.

A. Plausible – This is ODCM D 3.3.2 Required Action C.1, which applies to the Offgas noble gas activity monitor, but not the Stack noble gas activity monitor.
B. Plausible – This is ODCM D 3.3.2 Required Action E.1, which applies to the Stack iodine and particulate samplers, but not the Stack noble gas activity monitor.
D. Plausible – This is ODCM D 3.3.2 Required Action A.1, which does not apply because the fault was a failure of the instrument, not of the setpoint of the instrument.

Technical Reference(s): ODCM D 3.3.2

Proposed references to be provided to applicants during examination: ODCM D 3.3.2 and Table D 3.3.2-1 without bases

Learning Objective: N2-272000-RBO-14

Question Source: Bank – 2005 Audit SRO #8

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 #	241000 2.1.31
	Importance Rating	4.3

Reactor/Turbine Pressure Regulating System

Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: #93

The plant is at 80% power. The following indication is present on 2CEC*PNL851:

- The MAXIMUM COMBINED FLOW LIMIT potentiometer is set to achieve 110% of maximum combined flow.

Which one of the following describes:

- the indicated status of the MAXIMUM COMBINED FLOW LIMIT potentiometer, in accordance with N2-OP-23, Main Turbine EHC, and
- a thermal limit that takes a penalty when the backup pressure regulator is out of service, in accordance with the Core Operating Limits Report?

	MAXIMUM COMBINED FLOW LIMIT POTENTIOMETER	**Thermal Limit Penalized When Backup Pressure Regulator Out Of Service**
A.	Properly set	MCPR
B.	Properly set	APLHGR
C.	NOT properly set	MCPR
D.	NOT properly set	APLHGR

Proposed Answer: C

Explanation: The MAXIMUM COMBINED FLOW LIMIT potentiometer is properly set at 115% of rated flow. The COLR places a penalty on the MCPR thermal limit when the backup pressure regulator is out of service.

A. Plausible – The potentiometer is properly set when maximum flow is 115%, not 110%. 110% is the maximum SRV steam flow used to calculate the Minimum Steam Cooling Pressure.
B. Plausible – 110% is the maximum SRV steam flow used to calculate the Minimum Steam Cooling Pressure. APLHGR is a thermal limit that incurs a penalty for single loop operation, but not for operation without a backup pressure regulator.
D. Plausible – APLHGR is a thermal limit that incurs a penalty for single loop operation, but not for operation without a backup pressure regulator.

Technical Reference(s): N2-OP-23 Section E.1.0 Note, COLR

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-248000-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 #	3
	Group #	
	K/A #	2.1.20
	Importance Rating	4.6

Ability to interpret and execute procedure steps.

Proposed Question: #94

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

- A Reactor Operator is performing an Operating Procedure section to restore a piece of equipment following a maintenance activity.
- Due to an abnormal lineup from the maintenance activity, one step in the procedure does not apply and CANNOT be performed.
- The procedure step does not contain a specific allowance to mark the step "N/A".
- The Reactor Operator requests to move on without performing the step.

Which one of the following describes the required direction to the Reactor Operator, in accordance with CNG-PR-1.01-1009, Procedure Use and Adherence Requirements?

A. Process an Editorial Change.

B. Process a Technical Procedure Step Deletion Screening Form.

C. Obtain technical concurrence from a second Reactor Operator, then "N/A" the step and move on in the procedure.

D. Obtain technical concurrence from a Senior Reactor Operator, then "N/A" the step and move on in the procedure.

Proposed Answer: B

Explanation: CNG-PR-1.01-1009 provides the guidance for this situation in section 5.5.C (equipment out of service, equipment is not required to be operated, and the procedure does NOT contain a statement authorizing the partial performance). The procedure directs completion of Attachment 1, Technical Procedure Step Deletion Screening Form. This form requires two SROs to approve of the step deletion (First Line Supervisor, which would be an SRO for an Ops procedure, and another SRO).

A. Plausible – CNG-PR-1.01-1011 does provide for an Editorial Change process, but this situation does not meet the requirements.
C. Plausible – This is similar to the normal peer check process expected for most procedure execution, but does not meet the requirements for this situation.
D. Plausible – This is slightly more restrictive than the normal peer check process expected for most procedure execution, but does not meet the requirements for this situation.

Technical Reference(s): CNG-PR-1.01-1009

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-PR-1.01-1009-CT-01

Question Source: Bank – SYSID 105336

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.6
	Importance Rating	3.6

Knowledge of the process for making changes to procedures.

Proposed Question: #95

An Immediate Change was generated for a Special Operating Procedure (SOP). The Plant Management Staff approval has been completed.

Which one of the following identifies:

(1) the individual(s) required to approve the Immediate Change BEFORE it can be implemented, and

(2) when final approval of the change is required,

in accordance with CNG-PR-1.01-1011, Station-Specific Procedure Process?

	(1)	(2)
A.	Any active Senior Reactor Operator	Within 5 days of initial approval
B.	Any active Senior Reactor Operator	Within 14 days of initial approval
C.	General Supervisor – Shift Operations or Operations Manager	Within 5 days of initial approval
D.	General Supervisor – Shift Operations or Operations Manager	Within 14 days of initial approval

Proposed Answer: B

Explanation: CNG-PR-1.01-1011 allows any active SRO to approve an Immediate Change for initial implementation. CNG-PR-1.01-1011 requires final approval by the Approval Authority within 14 days of initial implementation.

A. Plausible – Within 5 days would be acceptable, however final approval is not required until 14 days.
C. Plausible – GSO / Operations Manager would perform the final approval, but are not required for initial approval. Within 5 days would be acceptable, however final approval is not required until 14 days.
D. Plausible – GSO / Operations Manager would perform the final approval, but are not required for initial approval.

Technical Reference(s): CNG-PR-1.01-1011

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-CGPR01-CE-01

Question Source: Bank – NMP1 2013 Audit #97

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.12
	Importance Rating	3.7

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: #96

The plant is in Mode 5. Conditions are as follows:

- An LPRM string is being removed from the core.
- Communications between the Control Room and the Refuel Bridge have been lost.

Which one of the following describes the required actions, if any per N2-FHP-011, Instrument Tube Removal and Installation?

A. No actions are required. LPRM string removal may continue.

B. Place the LPRM string in a safe condition and then suspend Core Alterations.

C. Immediately stop movement of the LPRM string and suspend Core Alterations.

D. Continue with LPRM string removal, but restore communications with one hour.

Proposed Answer: A

Explanation: Communications are required during LPRM string removal between the Controller, Under vessel, and the Refuel Floor, however communication with the Control Room is not required. Therefore LPRM string removal may continue.

Note: The question meets the K/A by presenting a situation where core maintenance activities would result in adverse radiological consequences and testing the required operational response to ensure radiological safety.

B. Plausible – Core Alterations must be suspended, however LPRM string removal does not qualify as a Core Alteration.
C. Plausible – Core Alterations must be suspended, however LPRM string removal does not qualify as a Core Alteration.
D. Plausible – Communications with the Control Room are not required for this evolution.

Technical Reference(s): N2-FHP-003 Section 4.1.1, N2-FHP-011 Section 5.16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-234000-RBO-10

Question Source: Bank – 2009 NRC #99

Question History: 2009 NRC #99

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4) and (7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.26
	Importance Rating	3.6

Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.

Proposed Question: #97

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

- Maintenance needs to block open a USAR-required fire door in the Reactor Building.
- The fire door will be blocked open for equipment removal for 4 hours.
- No preplanned fire provisions or fire compensatory actions have been designed or taken for this evolution.

Which one of the following identifies the requirements for a breach permit to be issued and for a fire watch/patrol to be established, in accordance with GAP-FPP-03, Breach Permit?

	Issuance of a Breach Permit	Establishing of a Fire Watch/Patrol
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: A

Explanation: GAP-FPP-03 allows a fire door breach without issuing a Breach Permit if the breach does not exceed one hour and does not involve an air lock or work on the door/door hardware. Since this activity goes over one hour, a breach permit is required. The USAR requires a fire watch/patrol for this breach (unless preplanned provision(s) were designated by a qualified Fire Protection Engineer).

B. Plausible – A fire watch/patrol is required because this is a USAR required fire door and no preplanned provisions have been designated by Fire Protection Engineering.
C. Plausible – A Breach Permit is required because the activity goes over one hour.
D. Plausible – A Breach Permit is required because the activity goes over one hour. A fire watch/patrol is required because this is a USAR required fire door and no preplanned provisions have been designated by Fire Protection Engineering.

Technical Reference(s): GAP-FPP-03 Section 3.3.4 and Attachment 1, USAR Section 9A.3.5.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: GAP-FPP-03-CT-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.42
	Importance Rating	3.4

Knowledge of new and spent fuel movement procedures.

Proposed Question: #98

The plant is in Mode 5. Conditions are as follows:

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

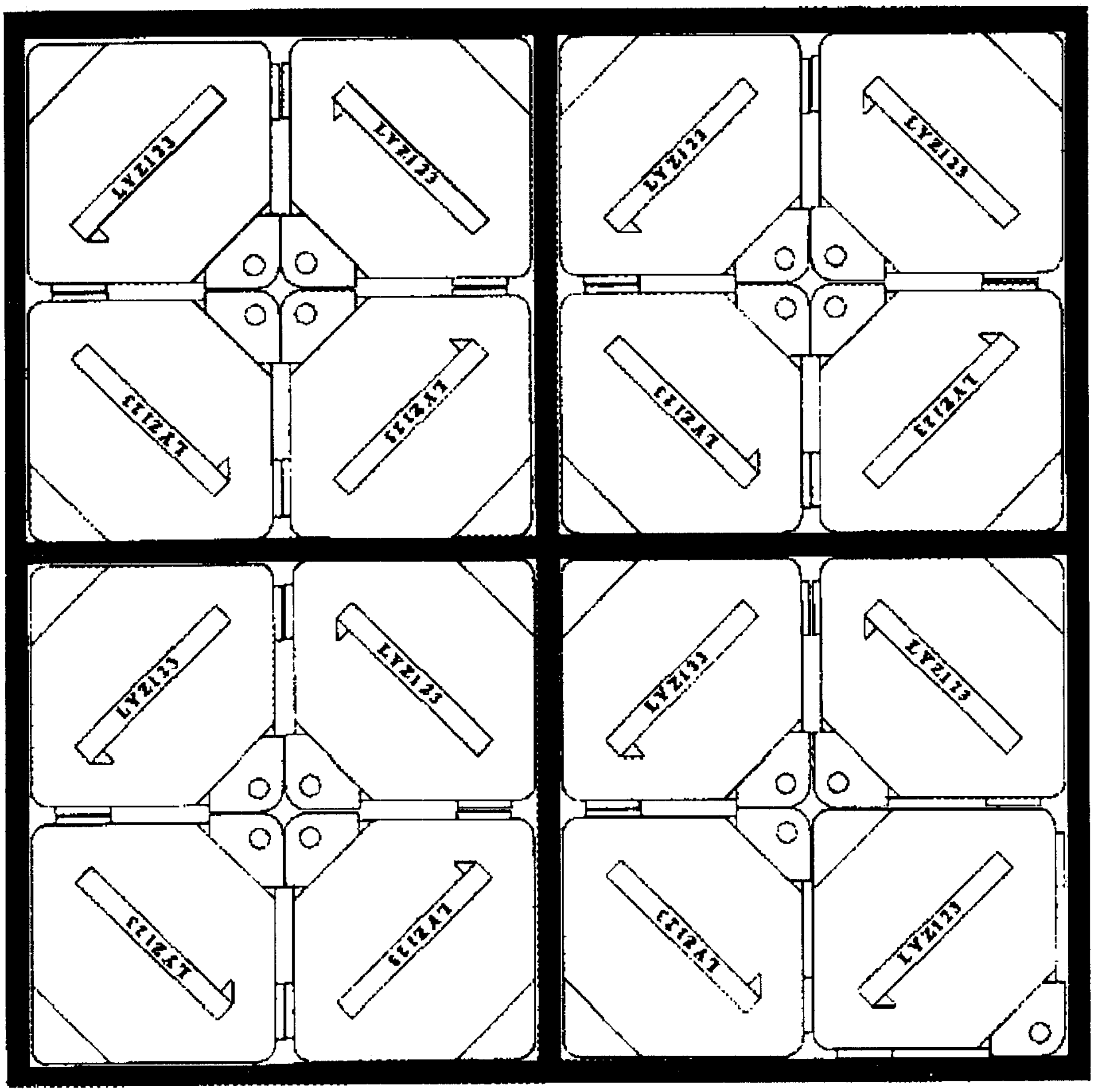
Which one of the following describes the status of this portion of the core and the required action(s), if any, in accordance with N2-FHP-13.3, Core Shuffle?

A. The fuel is loaded correctly. Core shuffle may continue with no additional required actions.

B. A discrepancy exists in the fuel loading. Core shuffle may continue without interruption as long as the discrepancy is fixed prior to startup.

C. A discrepancy exists in the fuel loading. The discrepancy must be immediately fixed and then fuel movements may continue with permission from the Refuel Floor SRO.

D. A discrepancy exists in the fuel loading. Fuel movements must be immediately stopped. Completion of a formal evaluation is required before fuel movement may resume.



Proposed Answer: D

Explanation: The fuel assembly in the lower right corner is oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell. N2-FHP-13.3 section 4.1.4 defines this as a Fuel Movement Discrepancy (FMD). N2-FHP-13.3 section 4.3.2 requires fuel movement to be stopped and requires completion of a formal evaluation, via Attachments 3 and 4, prior to resuming fuel movements.

A. Plausible – Three of the four fuel cells are oriented correctly.
B. Plausible – No significant adverse consequence would be expected from this fuel movement discrepancy until Reactor operation recommenced.
C. Plausible – The orientation issue could be resolved using the original fuel movement instructions, however N2-FHP-13.3 requires a formal evaluation before proceeding. This formal evaluation requires more than just Refuel Floor SRO permission to continue (Reactor Engineering, Shift Manager, and Designated Lead Point of Contact signatures required).

Technical Reference(s): N2-FHP-13.3

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-101002-RBO-9

Question Source: Bank – SYSID 105340

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.21
	Importance Rating	4.6

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: #99

A LOCA has occurred. Conditions are as follows:

- No Reactor injection sources are currently available.
- N2-EOP-C3, Steam Cooling, is being executed.
- Reactor water level is -35 inches (actual) and lowering slowly.

Then five minutes later...

- A Control Rod Drive pump is restored and is now injecting to the Reactor.
- Reactor water level is -42 inches (actual) and continuing to lower.

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

	Status of Adequate Core Cooling	Required Action
A.	Assured	Return to N2-EOP-RPV, RPV Control.
B.	Assured	Remain in N2-EOP-C3, Steam Cooling.
C.	NOT assured	Enter N2-EOP-C2, RPV Blowdown.
D.	NOT assured	Enter the Severe Accident Procedures.

Proposed Answer: C

Explanation: Initially, adequate core cooling (ACC) was assured because there was no RPV injection and Reactor water level was above -58 inches. However, once CRD began injecting, adequate core cooling is not assured between -39 and -58 inches. With an RPV injection source injecting and Reactor water not able to be restored/maintained above -39 inches, N2-EOP-C3 requires entering N2-EOP-C2, RPV Blowdown.

A. Plausible – ACC was assured until the CRD pump was injecting with Reactor water level < -39 inches.
B. Plausible – ACC was assured until the CRD pump was injecting with Reactor water level < -39 inches.
D. Plausible – If core cooling cannot be restored after the Blowdown, the Severe Accident Procedures would then be entered from N2-EOP-RPV.

Technical Reference(s): N2-EOP-C3 Step 1, NER-2M-039 Section 2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC3C01 EO-2

Question Source: Modified – 2012 NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.15
	Importance Rating	3.1

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #100

Which one of the following radiation monitors may be used to provide an initial estimate of core damage during an accident, in accordance with EPIP-EPP-09, Determination of Core Damage Under Accident Conditions?

A. Off-gas radiation monitor

B. Drywell radiation monitor

C. Main Stack radiation monitor

D. Main Steam Line radiation monitor

Proposed Answer: B

Explanation: EPIP-EPP-09 uses the Drywell radiation level to provide an initial estimate of core damage under accident conditions.

Note: This question tests appropriate SRO-level knowledge under 10CFR55.43(b)(4) by presenting a situation with extreme radiation hazards (a LOCA with fuel damage) and testing knowledge of how to analyze conditions to assess the magnitude of core damage (which radiation monitors are used for assessment). This assessment is required within the Emergency Plan and would be required to select the appropriate operational response during a beyond design-basis event. This is similar in nature to the example item in the Guidance for SRO-only Questions, Section II.D bullet #3. Drywell radiation level can be used to estimate core damage more quickly than coolant activity level.

A. Plausible – The Off-gas radiation monitors are used to determine entry into N2-SOP-17, Fuel Failure, but not for core damage estimation in EPIP-EPP-09.
C. Plausible – The Main Stack radiation monitor are used during an accident to estimate off-site release rate, but not for core damage estimation in EPIP-EPP-09.
D. Plausible – The Main Steam Line radiation monitors are used to determine entry into N2-SOP-17, Fuel Failure, but not for core damage estimation in EPIP-EPP-09.

Technical Reference(s): EPIP-EPP-09 Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: EPIP-EPP-09-TO-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments: