Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K1.01
	Importance Rating	3.0

SLC

Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Core spray line break detection: Plant-Specific

Proposed Question: #1

The plant is operating at 100% power when the Liquid Poison isolation valve in the Drywell, BV - LP SYS DISCHARGE TO RX, 42.1-01, develops a significant valve body leak.

Which one of the following describes a potential impact of this leak?

This leak could result in...

A. an incorrect indication of Core Plate D/P.

B. an incorrect indication of Control Rod Drive Cooling D/P.

C. annunciator K3-4-1, Core Spray Sparger Diff Press, alarming.

D. annunciator F2-4-4, Reactor Vessel Seal Leak Press High, alarming.

Proposed Answer: A

Explanation: The LP Sparger (inner pipe of the pipe-within-a-pipe) serves as the high pressure side of the core plate D/P transmitter input. The outer pipe of the pipe-within-a-pipe surrounds the LP Sparger, continues through the core plate and serves as the lower pressure side of the core plate D/P transmitter input. A significant leak in the LP Isolation valve (42.1-01) would lower the sensed pressure on the high pressure side of the Core Plate D/P transmitter and result in an incorrect D/P reading.

KA Match Justification: The question satisfies the randomly selected K/A by requiring knowledge of the cause/effect relationship between a Liquid Poison (SLC) component (leak on 42.1-01) and Core Spray line break detection. The candidate must understand the physical connections between Liquid Poison (SLC), Core Spray line break detection, and the "pipe-within-a-pipe" to both prove choice A and disprove choice C.

B. Plausible – The CRD cooling water D/P transmitter uses a pressure from the outer pipe of the pipe-within-a-pipe. The inner pipe of the pipe-within-a-pipe does not affect the CRD cooling water D/P indication.
C. Plausible – The Core Spray Sparger D/P transmitter uses pressures from the outer pipe of the pipe-within-a-pipe and the CS Sparger. The inner pipe of the pipe-within-a-pipe does not affect the CS Sparger D/P indication.
D. Plausible – This alarm is caused by leakage and pressure buildup between the two Reactor vessel head seals. The pressure tap for this alarm is a stand-alone connection, and does not interface with the LP system.

Technical Reference(s): C-18015-C, C-18019-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO-8

Question Source: Bank – 2010 Audit #5

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K1.04
	Importance Rating	3.2

EDGs

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

Proposed Question: #2

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

- Lines 1 and 4 de-energize.
- Emergency Diesel Generator (EDG) 102 Raw Water pump fails to start.

Which one of the following describes the resulting operation of EDG 102?

EDG 102...

A. does NOT start.

B. starts with NO cooling water flow to the Raw Water side of the heat exchanger.

C. starts with cooling water flow automatically supplied from EDG 103 Raw Water pump.

D. starts with backup cooling water flow automatically supplied from the Diesel Fire pump.

Proposed Answer: B

Explanation: When Lines 1 and 4 de-energize, an undervoltage condition occurs on Powerboard 102. This gives EDG 102 a start signal. During the EDG 102 start sequence, the associated Raw Water pump also gets a start signal. However, there is no interlock preventing EDG 102 from completing its start sequence if the Raw Water pump fails to start. Therefore EDG 102 does start. There is no automatic backup cooling water supply to EDG 102 based on the failure of the Raw Water pump, therefore it is running with no cooling water flow to the Raw Water side of the heat exchanger.

A. Plausible – The EDG 102 start circuit does call for the start of EDG 102 Raw Water pump before EDG 102 start, but there is no feature that blocks start of EDG 102 if the Raw Water pumps fails to run.
C. Plausible – EDG 103 does have an identical Raw Water pump that would start with a loss of Lines 1 and 4, but the two systems are NOT cross connected such that EDG 103 Raw Water pump would deliver flow to the EDG 102 heat exchanger.
D. Plausible – The Diesel Fire pump discharge can be manually aligned to supply cooling water to the Raw Water side of the EDG 102 heat exchanger, but there is no automatic feature for this function.

Technical Reference(s): C-18026-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-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 #	205000 K2.01
	Importance Rating	3.1

Shutdown Cooling

Knowledge of electrical power supplies to the following: Pump motors

Proposed Question: #3

The plant is in cold shutdown with the following:

- Shutdown Cooling (SDC) pumps 11 and 13 in service.
- Then, Powerboard 16B de-energizes due to a sustained electrical fault.

Which one of the following describes the resulting status of the SDC pumps?

SDC pump(s)...

A. 11 and 13 both trip.

B. 11 and 13 both remain running.

C. 11 trips, but SDC pump 13 remains running.

D. 13 trips, but SDC pump 11 remains running.

Proposed Answer: A

Explanation: Both SDC pumps 11 and 13 are powered by PB 16B and trip when this board de-energizes. SDC pump 12 is powered by PB 17B. Power board 16 is equipped with an undervoltage lockout device which will trip the breakers of most loads being supplied by that power board, including SDC pumps.

B. Plausible – PB 17B supplies power for SDC pump 12, but both SDC pumps 11 and 13 are powered by PB 16B.
C. Plausible – PB 17B supplies power for SDC pump 12, but both SDC pumps 11 and 13 are powered by PB 16B.
D. Plausible – PB 17B supplies power for SDC pump 12, but both SDC pumps 11 and 13 are powered by PB 16B.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-4

Question Source: Modified Bank – 2009 Audit #3

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

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

RPS

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

Proposed Question: #4

Which one of the following describes the 600VAC power source that normally supplies power for the Reactor Protection System (RPS) Analog Trip System (ATS) cabinet D?

A. Powerboard 16

B. Powerboard 17

C. Powerboard 131

D. Powerboard 141

Proposed Answer: B

Explanation: 600VAC PB 17 normally supplies power to UPS 172, which in turn supplies power to RPS bus 12 and ATS cabinets B and D.

A. Plausible – PB 16 normally supplies power to UPS 162, which in turn supplies power to RPS bus 11 and ATS cabinets A and C.
C. Plausible – PB 131 normally supplies power to RPS MG Set 131 and RPS trip bus 131.
D. Plausible – PB 141 normally supplies power to RPS MG Set 141 and RPS trip bus 141.

Technical Reference(s): N1-212000-RBO-4, C-19409-C Sh 1B, N1-OP-40, N1-SOP-40.1, C-19957-C Sh 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-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 #	218000 K3.02
	Importance Rating	4.5

ADS

Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Ability to rapidly depressurize the reactor

Proposed Question: #5

The plant was operating at 100% power when the following events occurred:

- An un-isolable steam leak in the Reactor Building has led to the need for an RPV Blowdown.
- Fuse failures caused three (3) ERVs to fail closed.

Which one of the following describes the effect of the ERV loss on the ability to rapidly depressurize the Reactor, in accordance with N1-EOP-8, RPV Blowdown?

The Minimum Number of ERVs Required for Emergency Depressurization ___(1)___ and ___(2)___.

	(1)	(2)
A.	is available	the use of alternate Blowdown Systems to rapidly depressurize the Reactor is allowed, but NOT required
B.	is available	the use of alternate Blowdown Systems to rapidly depressurize the Reactor is NEITHER required NOR allowed
C.	is NOT available	Turbine Bypass Valves may be used to rapidly depressurize the Reactor even if MSIV isolations must be defeated
D.	is NOT available	Turbine Bypass Valves may be used to rapidly depressurize the Reactor but MSIV isolations must NOT be defeated

Proposed Answer: C

Explanation: The Minimum Number of ERVs Required for Emergency Depressurization is 4. With three ERVs unavailable, only three remain available. N1-EOP-8 step 15 allows use of alternate Blowdown systems since less than 4 ERVs can be opened. This step and detail O also both give instructions allowing all isolation signals to be defeated.

A. Plausible – The Minimum Number of ERVs Required for Emergency Depressurization is 4. With 3 ERVs unavailable, only 3 remain available.
B. Plausible – The Minimum Number of ERVs Required for Emergency Depressurization is 4. With 3 ERVs unavailable, only 3 remain available.
D. Plausible – N1-EOP-8 step 15 allows use of alternate Blowdown systems since less than 4 ERVs can be opened. This step and detail O also both give instructions allowing all isolation signals to be defeated.

Technical Reference(s): N1-EOP-8, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-12

Question Source: Bank – 2010 NRC #5

Question History: 2010 NRC #5

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 #	215004 K3.02
	Importance Rating	3.4

Source Range Monitor

Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on the following: Reactor manual control: Plant-Specific

Proposed Question: #6

A Reactor startup is in progress with the following:

- All SRMs are fully inserted and indicate between 200 and 300 cps.
- All IRMs indicate downscale on Range 2.
- REFUEL INST TRIP BYPASS CH 11 and 12 keylock switches are in COINCIDENT.
- Then, the following malfunctions occur:
  - SRM 11 fails downscale.
  - SRM 14 fails upscale.

Which one of the following describes the impact of these malfunctions, if any, on the Reactor Protection System (RPS) and/or Reactor Manual Control System (RMCS)?

	Impact of SRM 11 Failing Downscale	Impact of SRM 14 Failing Upscale
A.	None	Rod block, but NO scram
B.	None	Rod block and a scram
C.	Rod block, but NO half scram	Rod block, but NO scram
D.	Rod block, but NO half scram	Rod block and a scram

Proposed Answer: A

Explanation: With SRM 11 fully inserted, the downscale rod block (<100 cps) is bypassed. Additionally, there is no scram on SRM downscale. Therefore, SRM 11 failing downscale has no effect on either RMCS or RPS. With the keylock switches in COINCIDENT, the SRM upscale scram is bypassed for all SRMs. However, the SRM upscale rod block (>1 x 10^5 cps) is still active. Therefore, SRM 14 failing upscale causes a rod block, but no scram.

B. Plausible – If the keylock switches were in NON-COINCIDENT, SRM 14 failing upscale would cause a scram.
C. Plausible – If SRM 11 were partially withdrawn, its downscale failure would cause a rod block.
D. Plausible – If SRM 11 were partially withdrawn, its downscale failure would cause a rod block. If the keylock switches were in NON-COINCIDENT, SRM 14 failing upscale would cause a scram.

Technical Reference(s): N1-OP-38A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Bank – SYSID 79116

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 K4.01
	Importance Rating	3.4

Component Cooling Water

Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump

Proposed Question: #7

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

- Turbine Building Closed Loop Cooling (TBCLC) pump 11 is in service.
- TBCLC pump 12 is in standby.
- TBCLC discharge header pressure is 110 psig and stable.
- TBCLC discharge header pressure lowers to 75 psig.

Which one of the following describes the resulting status of TBCLC pump 12?

TBCLC pump 12...

A. auto-starts due to a low discharge header pressure signal.

B. remains in standby and will only auto-start if TBCLC pump 11 breaker trips.

C. remains in standby, but will auto-start if discharge header pressure lowers further.

D. remains in standby unless Operators manually start it.

Proposed Answer: A

Explanation: TBCLC pumps have an auto-start feature on a low discharge pressure of 80 psig. Since header pressure is below this value, TBCLC pump 12 auto-starts.

B. Plausible – Some pump auto-start features are based on associated pump breaker position, however the TBCLC auto-start feature is based on discharge header pressure.
C. Plausible – TBCLC header pressure is still fairly high, however it is below the auto-start setpoint of 80 psig.
D. Plausible – The RBCLC system does NOT have an auto-start feature, therefore it's pumps would only start in this situation if an Operator took manual action.

Technical Reference(s): N1-OP-24

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-274000-RBO-5

Question Source: Modified Bank – September 2012 JAF #53

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 K4.02
	Importance Rating	2.6

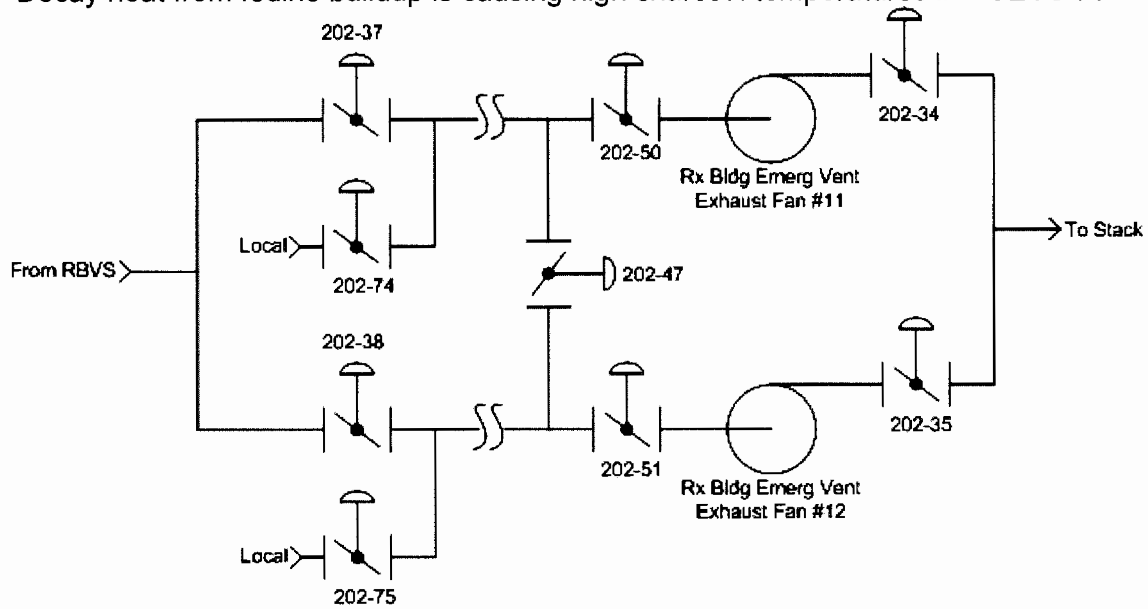
SGTS

Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Charcoal bed decay heat removal

Proposed Question: #8

A steam leak has occurred in the Reactor Building, resulting in the following conditions:

- Reactor Building Emergency Ventilation System (RBEVS) train 12 is in service.
- Reactor Building exhaust radiation levels are above 5 mr/hr.
- Decay heat from Iodine buildup is causing high charcoal temperatures in RBEVS train 12.



Which one of the following actions is required to cool the RBEVS train 12 charcoal bed in accordance with N1-OP-10, Reactor Building Heating, Cooling, and Ventilating System?

A. Maintain RBEVS fan 12 in service and open 202-75, EM VENTILATION LOOP 12 COOLING BV.

B. Maintain RBEVS fan 12 in service, place RBEVS fan 11 in service in parallel, then open 202-75, EM VENTILATION LOOP 12 COOLING BV.

C. Place RBEVS fan 11 in service, maintain RBEVS fan 12 running, close 202-38, EM VENTILATION LOOP 12 INLET BV and open 202-75, EM VENTILATION LOOP 12 COOLING BV.

D. Place RBEVS fan 11 in service, stop RBEVS fan 12 and place its control switch in PTL, then open 202-75, EM VENTILATION LOOP 12 COOLING BV and open 202-47, EM VENTILATION TIE BV.

Proposed Answer: D

Explanation: To place a train of RBEVS in cooling requires shutting down the train, placing its control switch in pull to lock, and then opening the cooling valve and the cross tie valve with the other train running.

A. Plausible – This would pull cooling air through the filter if allowed, however with an initiation signal present (> 5 mr/hr), the cooling BV can only be opened if the associated fan control switch is in PTL.
B. Plausible – This would pull cooling air through the filter if allowed, however with an initiation signal present (> 5 mr/hr), the cooling BV can only be opened if the associated fan control switch is in PTL.
C. Plausible – This would pull cooling air through the filter if allowed, however with an initiation signal present (> 5 mr/hr), the cooling BV can only be opened if the associated fan control switch is in PTL.

Technical Reference(s): N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-10

Question Source: Bank – 2009 Audit #16

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 #	215005 K5.05
	Importance Rating	3.6

APRM / LPRM

Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: Core flow effects on APRM trip setpoints

Proposed Question: #9

The plant is operating at approximately 85% power with the following:

- Recirculation pump 12 has tripped.
- N1-SOP-1.3, Recirc Pump Trip at Power, is being executed.

Which one of the following describes how the APRM scram setpoints will change in response to Operator actions performed in N1-SOP-1.3?

APRM scram setpoints...

A. rise when the Recirc pump 12 discharge valve is closed.

B. lower when the Recirc pump 12 discharge valve is closed.

C. rise when the Recirc pump 12 control switch is green flagged.

D. lower when the Recirc pump 12 control switch is green flagged.

Proposed Answer: B

Explanation: When Recirc pump 12 tripped, reverse flow through the Recirc loop caused total Recirculation flow to indicate erroneously high. This causes high (non-conservative) APRM scram setpoints, since indicated total Recirculation flow is used to establish these setpoints. When the Recirc pump 12 discharge valve is closed, reverse flow through the loop lowers, causing indicated total Recirculation flow to lower, and APRM scram setpoints to lower.

A. Plausible – Reactor power will rise while Recirc pump 12 discharge valve, however APRM scram setpoints lower in response to lowering indicated total Recirculation flow.
C. Plausible – Reactor power will rise while Recirc pump 12 discharge valve, however APRM scram setpoints lower in response to lowering indicated total Recirculation flow. N1-SOP-1.3 does direct green flagging the Recirc pump 12 control switch, however APRM scram setpoints are calculated based on indicated total Recirculation flow, not calculated flow based on Recirc pump switch/breaker positions.
D. Plausible – N1-SOP-1.3 does direct green flagging the Recirc pump 12 control switch, however APRM scram setpoints are calculated based on indicated total Recirculation flow, not calculated flow based on Recirc pump switch/breaker positions.

Technical Reference(s): N1-SOP-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215005-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

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

IRM

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

Proposed Question: #10

A plant startup is in progress with the following:

- The Reactor is critical.
- The Mode Switch is in STARTUP.
- All IRMs are indicating mid-scale on range 2.
- The FULL OUT pushbutton for IRM 16 is inadvertently depressed.

Which one of the following describes the IRM 16 response and the effect on control rod withdrawal?

	IRM 16 Response	Effect On Control Rod Withdrawal
A.	Withdraws	Rod block occurs
B.	Does NOT withdraw	Rod block occurs
C.	Withdraws	Rod block does NOT occur
D.	Does NOT withdraw	Rod block does NOT occur

Proposed Answer: A

Explanation: IRM 16 will withdraw, but a control rod block will be generated because the detector is not fully inserted with the Mode Switch in STARTUP.

B. Plausible – While it is desired for the IRM detector to remain fully inserted until the Mode Switch is taken to RUN, there is no interlock preventing the detector from retracting.
C. Plausible – Even though IRMs are indicating mid-scale on the lowest range, a control rod block is generated.
D. Plausible – While it is desired for the IRM detector to remain fully inserted until the Mode Switch is taken to RUN, there is no interlock preventing the detector from retracting. Even though IRMs are indicating mid-scale on the lowest range, a control rod block is generated.

Technical Reference(s): N1-OP-38B

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Bank – 2009 Audit #7

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 #	262001 K6.02
	Importance Rating	3.6

AC Electrical Distribution

Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION: Off-site power

Proposed Question: #11

A plant startup is in progress with the following:

- The Reactor Mode Switch has just been taken to RUN.
- No subsequent actions in the startup procedure have been completed.
- Then, the 115KV offsite power lines de-energize and remain de-energized.

Which one of the following describes the resulting status of the 4160V powerboards (PBs)?

A. PBs 11 and 12 remain energized without interruption. PB 101 de-energizes and remains de-energized. PBs 102 and 103 de-energize and then re-energize.

B. PBs 11 and 12 remain energized without interruption. PBs 101, 102, and 103 de-energize and then re-energize.

C. PBs 11 and 12 de-energize and remain de-energized. PBs 101, 102, and 103 de-energize and then re-energize.

D. PBs 11, 12, and 101 de-energize and remain de-energized. PBs 102 and 103 de-energize and then re-energize.

Proposed Answer: D

Explanation: Since the Reactor Mode Switch has just been taken to RUN and no subsequent actions have been performed in the startup, the Main Generator is not yet online and therefore all 4160V PBs are supplied by the 115KV offsite power lines. When the 115KV offsite power lines de-energize, all 4160V PBs initially de-energize. EDGs 102 and 103 start and re-energize PBs 102 and 103. PBs 11, 12, and 101 remain de-energized as long as the 115KV offsite power lines remain de-energized.

A. Plausible – This describes the correct response of PBs 101, 102, and 103, as well as the correct response when house loads are supplied by the Main Generator. However, since the Main Generator is not yet online, PBs 11 and 12 are supplied by 115KV offsite power and therefore de-energize.
B. Plausible – This describes the correct response of PBs 102 and 103. However, since the Main Generator is not yet online, PBs 11 and 12 are supplied by 115KV offsite power and therefore de-energize. Additionally, PB 101 does not re-energize.
C. Plausible – This describes the correct response of PBs 11, 12, 102, and 103. However, PB 101 does not re-energize.

Technical Reference(s): N1-OP-43A, N1-OP-33A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K6.04
	Importance Rating	3.1

Reactor Water Level Control

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Reactor feedwater flow input

Proposed Question: #12

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

- Feedwater Level Control is in 3-element.
- Feedwater pump 13 is operating with FCV 13 in automatic.
- Feedwater pumps 11 and 12 are in standby.
- Reactor water level is 72" and stable.

Then, a Feedwater Pump 13 flow element failure occurs resulting in a maximum flow signal input to the Feedwater Level Control circuitry.

Which one of the following identifies the direction and magnitude of the resulting Reactor water level change if NO operator action is taken?

Reactor water level...

A. lowers and causes a Reactor scram.

B. rises and causes a Main Turbine trip.

C. lowers and stabilizes at a lower value without causing a Reactor scram.

D. rises and stabilizes at a higher value without causing a Main Turbine trip.

Proposed Answer: A

Explanation: With the sensed Feedwater flow input to FWLC failed high, FWLC closes FCV 13 to lower flow. This causes Reactor water level to lower. FWLC will sense Reactor water level lowering, but will not sense the change in Feedwater flow due to the failed input. FWLC is level-dominant, however a complete failure of the Feedwater flow input is expected to cause approximately 40 inches of level control error before stabilizing level. Since normal Reactor water level is in the range of 70-72 inches and the Reactor scram setpoint is 53 inches, this 40 inches of level response will cause a Reactor scram.

B. Plausible – Reactor water level lowers, not rises, as would be the case if either sensed Feedwater flow failed low or Feedwater flow demand signal failed high.
C. Plausible – FWLC is level-dominant, however the magnitude of level error required to offset a complete failure of the sensed Feedwater flow is large enough to cause a Reactor scram.
D. Plausible – Reactor water level lowers, not rises, as would be the case if either sensed Feedwater flow failed low or Feedwater flow demand signal failed high. FWLC is level-dominant, however the magnitude of level error required to offset a complete failure of the sensed Feedwater flow is large enough to cause a Reactor scram.

Technical Reference(s): N1-OP-16, N1-259002-RBO-8

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-11

Question Source: Modified Bank – 2009 Audit #24

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 #	262002 A1.02
	Importance Rating	2.5

UPS (AC/DC)

Ability to predict and/or monitor changes in parameters associated with operating the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) controls including: Motor generator outputs

Proposed Question: #13

The plant is shutdown with the following:

- Motor Generator (MG) Set 167 is being returned to service following maintenance.
- Computer Panel Board 167 is currently energized from I&C Bus 130.

Which one of the following describes the mode of operation used to (1) initially start MG Set 167 and (2) synchronize the MG Set 167 output with Computer Panel Board 167 in accordance with N1-OP-48, Motor Generator Sets?

	(1) Mode for Initially Starting MG Set 167	(2) Mode for Synchronizing MG Set 167
A.	DC RUN	DC RUN
B.	DC RUN	AC RUN
C.	AC RUN	DC RUN
D.	AC RUN	AC RUN

Proposed Answer: C

Explanation: N1-OP-48 requires initially starting MG Set 167 using the AC RUN mode. MG Set 167 is synchronized using the DC RUN mode to allow for matching frequency between the MG Set 167 output and the energized board.

A. Plausible – DC RUN is used for synchronizing, but not for starting.
B. Plausible – DC RUN is used for synchronizing, but not for starting. AC RUN is used for starting, but not for synchronizing.
D. Plausible – AC RUN is used for starting, but not for synchronizing.

Technical Reference(s): N1-OP-48

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 A1.04
	Importance Rating	3.7

LPCS

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

Proposed Question: #14

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

- Core Spray testing is in progress in accordance with N1-ST-Q1A, CS 111 Pump Valve and SDC Water Seal Check Valve Operability Test.
- 40-12, CORE SPRAY DISCHARGE IV 11 (OUTSIDE), is closed.
- 40-06, CORE SPRAY TEST VALVE 11, is open.

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

- Reactor water level is -20 inches and slowly lowering.
- Reactor pressure is 400 psig and slowly lowering.
- Drywell pressure is 10 psig and slowly rising.
- All Core Spray pumps are running.
- No operator action has been taken to re-position any Core Spray valves.
- Core Spray valves are aligned as shown on the next page:

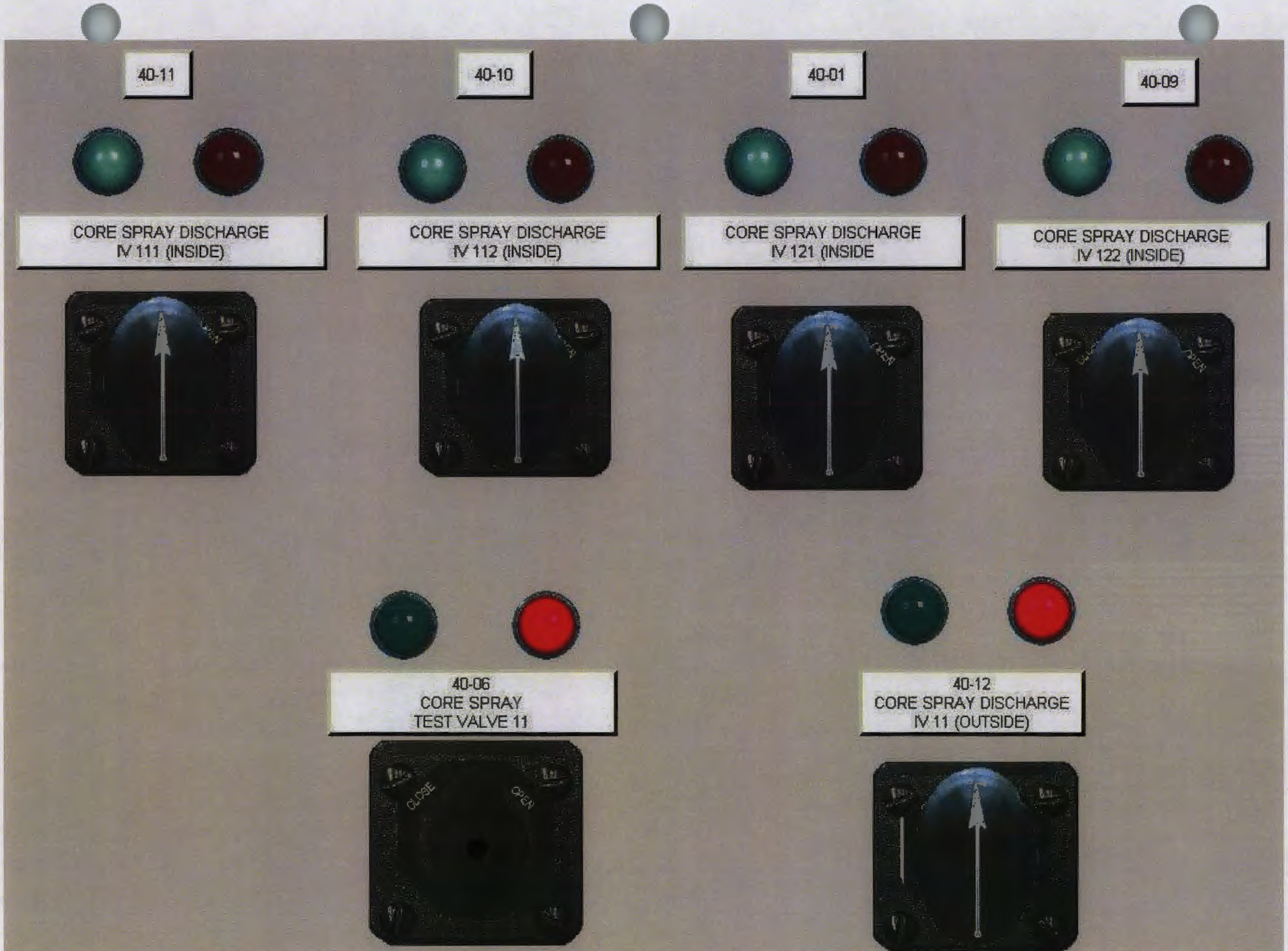
Which one of the following describes the status of the given Core Spray valves?

A. All of the given Core Spray valves have responded correctly.

B. 40-06 has failed to close as designed. All of the other given Core Spray valves have responded correctly.

C. 40-06 has failed to close as designed. 40-12 has opened earlier than designed. All of the other given Core Spray valves have responded correctly.

D. 40-06 has failed to close as designed. The Core Spray inside IVs (40-01, 40-09, 40-10, and 40-11) have failed to open as designed. All of the other given Core Spray valves have responded correctly.



40-11
40-10
40-01
40-09
CORE SPRAY DISCHARGE IV 111 (INSIDE)
CORE SPRAY DISCHARGE IV 112 (INSIDE)
CORE SPRAY DISCHARGE IV 121 (INSIDE
CORE SPRAY DISCHARGE IV 122 (INSIDE)
40-06
CORE SPRAY
TEST VALVE 11
CLOSE
OPEN
40-12
CORE SPRAY DISCHARGE
IV 11 (OUTSIDE)

Proposed Answer: C

Explanation: 40-06 is designed to automatically close on either Reactor water level <5 inches or Drywell pressure >3.5 psig. 40-06 still indicates full open, therefore it failed to close as designed. 40-12, if open, is designed to automatically open with either of these signals present AND when Reactor pressure drops below 365 psig. 40-12 indicates full open with Reactor pressure still above 365 psig, therefore it has opened earlier than designed. The Core Spray inside IVs are designed to also automatically open when 40-12 opens (<365 psig). These valves indicate full closed, as designed.

A. Plausible – 40-06 has failed to close, since it is not designed to wait for Reactor pressure to lower below 365 psig to re-position. 40-12 has opened early, since it is designed to wait for Reactor pressure to lower below 365 psig to re-position.
B. Plausible – 40-06 has failed to operate per design, but so has 40-12, which should not automatically open until Reactor pressure lowers below 365 psig.
D. Plausible – 40-06 has failed to operate per design, but the Core Spray inside IVs are not designed to automatically open until Reactore pressure lowers below 365 psig.

Technical Reference(s): N1-OP-2, C-19859-C sheet 9

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 A2.03
	Importance Rating	4.1

SRVs

Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV

Proposed Question: #15

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

- ERV 111 inadvertently opens due to an erroneous high Reactor pressure signal.
- N1-SOP-1.4, Stuck Open ERV, is being executed.

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

(1) Pulling ERV 111 control power fuses in the Control Room.
(2) Pulling ERV 111 control power fuses in the Reactor Building.

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

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

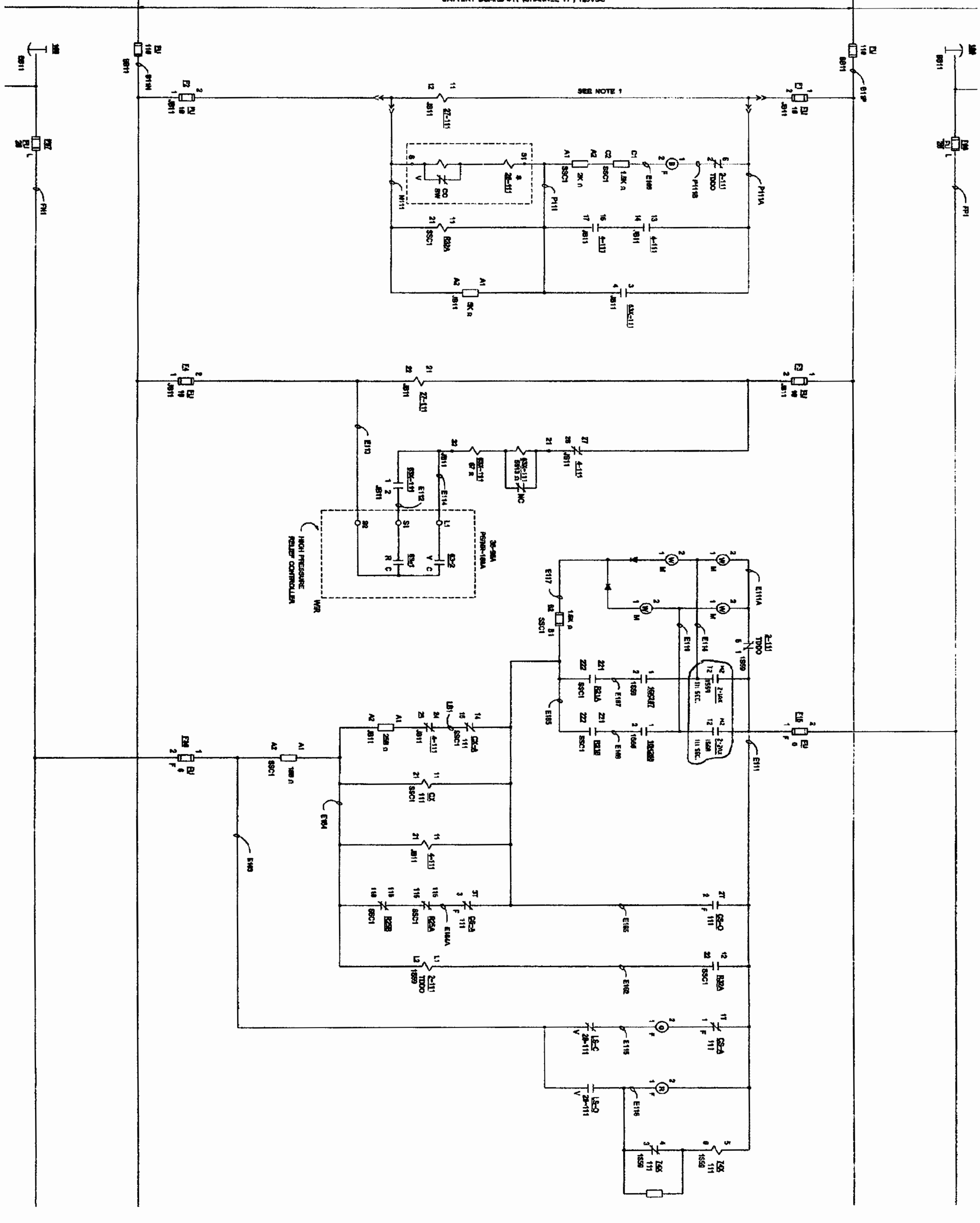
A. Neither of these methods will close ERV 111.

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

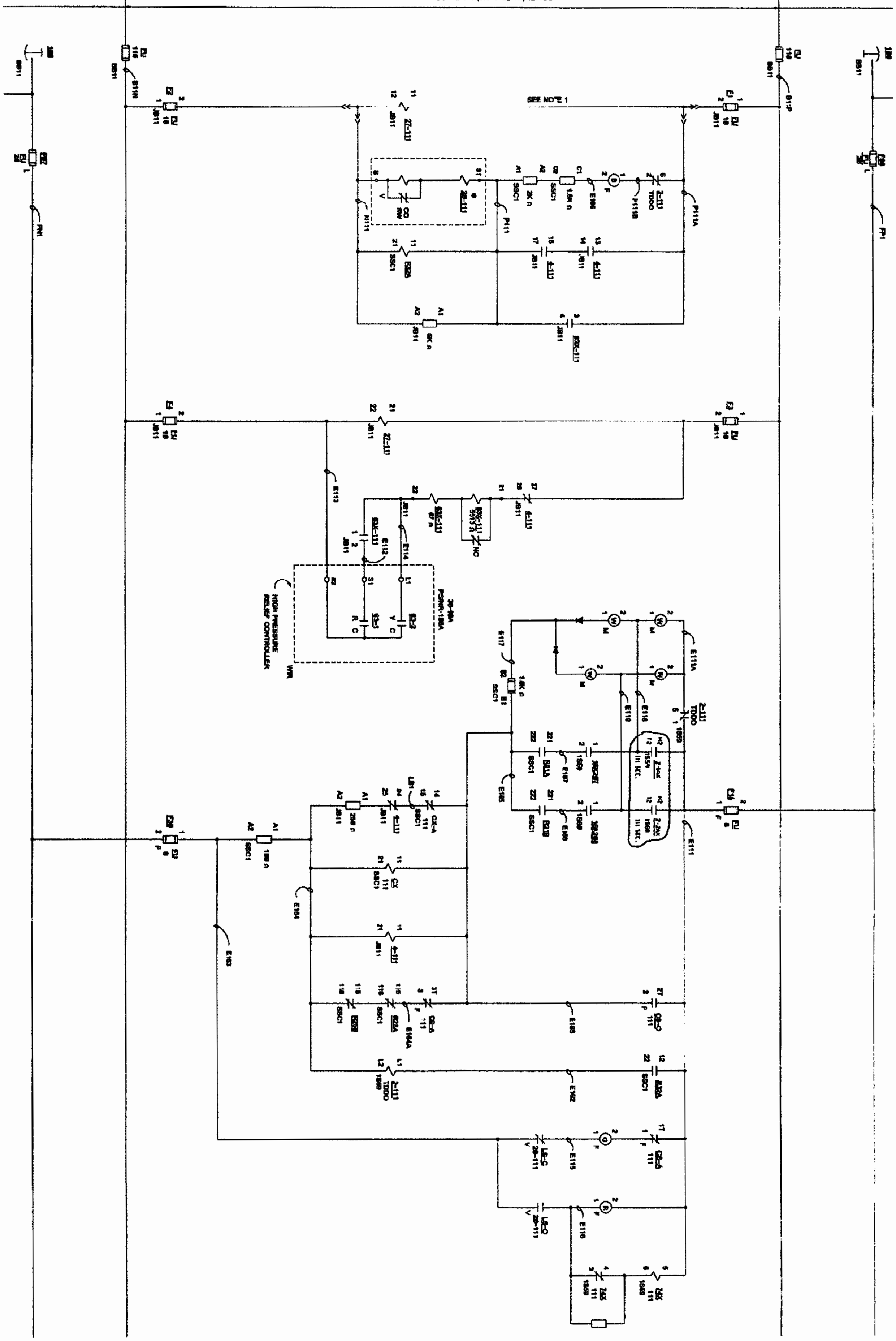
C. Method (2) will close ERV 111, but method (1) will NOT.

D. Both of these methods will close ERV 111.

BATTERY BOARD #11 (CHANNEL 11) 125VDC



BATTERY BOARD #11 (CHANNEL 11) 125VDC



Proposed Answer: C

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

A. Plausible – Method (1) will NOT close ERV 111 with the given failure, however Method (2) will close ERV 111.
B. Plausible – Method (1) will NOT close ERV 111 with the given failure, however Method (2) will close ERV 111.
D. Plausible – Method (1) will NOT close ERV 111 with the given failure.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 A2.01
	Importance Rating	2.9

Instrument Air

Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Air dryer and filter malfunctions

Proposed Question: #16

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

- Instrument Air Dryer (IAD) 94-168 is tagged out of service for extended maintenance.
- Annunciator L1-3-7, Inst Air System, is in alarm.
- IAD 94-169 differential pressure is 12.5 psid and rising.
- Automatic bypass of IAD 94-169 has failed.
- Manual bypass of IAD 94-169 has failed.
- N1-SOP-20.1, Instrument Air Failure, has been entered.

Which one of the following describes the response of the Instrument Air (IA) system as IAD 94-169 differential pressure continues to rise and the required action in accordance with N1-SOP-20.1?

A. Both Scram Air Header pressure AND IA Receiver 11 pressure will lower below normal values. Scram the Reactor if Scram Air Header pressure CANNOT be maintained above the limit of 60 psig. Scram the Reactor if IA Receiver 11 pressure CANNOT be maintained above the limit of 70 psig.

B. Both Scram Air Header pressure AND IA Receiver 11 pressure will lower below normal values. Scram the Reactor if Scram Air Header pressure CANNOT be maintained above the limit of 70 psig. Scram the Reactor if IA Receiver 11 pressure CANNOT be maintained above the limit of 60 psig.

C. Scram Air Header pressure will lower below the normal value, but IA Receiver 11 pressure will remain in the normal range. Scram the Reactor if Scram Air Header pressure CANNOT be maintained above the limit of 60 psig.

D. Scram Air Header pressure will lower below the normal value, but IA Receiver 11 pressure will remain in the normal range. Scram the Reactor if Scram Air Header pressure CANNOT be maintained above the limit of 70 psig.

Proposed Answer: C

Explanation: IADs 94-168 and 94-169 are located in parallel, downstream of Instrument Air Receiver 11, but upstream of the Scram Air Header pressure instrument. With IAD 94-168 tagged out, IAD 94-169 D/P rising, and both automatic and manual bypasses failing, Scram Air Header pressure will lower below the normal value, but IA Compressors will maintain IA Receiver 11 pressure in the normal range. N1-SOP-20.1 requires a scram if Scram Air Header pressure cannot be maintained above the limit of 60 psig.

A. Plausible – The IADs are downstream of IA Receiver 11, therefore pressure in the receiver will not be affected.
B. Plausible – The IADs are downstream of IA Receiver 11, therefore pressure in the receiver will not be affected. The Scram Air Header pressure limit requiring a scram is 60 psig, not 70 psig. The IA Receiver 11 pressure limit requiring a scram is 70 psig, not 60 psig.
D. Plausible – The Scram Air Header pressure limit requiring a scram is 60 psig, not 70 psig.

Technical Reference(s): C-18011-C sheet 2, N1-SOP-20.1, L1-3-7

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A3.03
	Importance Rating	3.9

HPCI

Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: System lineup: BWR-2,3,4

Proposed Question: #17

The plant has scrammed due to a loss of coolant accident with the following:

- All Condensate, Feedwater Booster, and Feedwater pumps are running.
- Then, a grid disturbance results in a loss of 115 KV power.
- Breakers R10 and R40 open.
- Protective relays at Lighthouse Hill clear necessary busses.
- Bennett's Bridge auto-transfers to re-energize Line 4 back to Nine Mile Point.
- Reactor water level is 30" and slowly lowering.

Which one of the following describes the response of the Feedwater system when power is restored?

A. All Condensate, Feedwater Booster, and Feedwater pumps start.

B. Condensate pump 11, Feedwater Booster pump 11, and Feedwater pump 11 start. All other Condensate, Feedwater Booster, and Feedwater pumps remain off.

C. Condensate pump 13, Feedwater Booster pump 13, and Feedwater pump 12 start. All other Condensate, Feedwater Booster, and Feedwater pumps remain off.

D. All Condensate, Feedwater Booster, and Feedwater pumps remain off.

Proposed Answer: C

Explanation: With a major power disturbance resulting in temporary loss of power to Nine Mile Point 115 KV lines, undervoltage relays trip the line breakers, R10 and R40. Protective relay schemes at Lighthouse Hill will automatically clear all necessary busses. One of the two generators at Bennett's Bridge will be started and switched to the line supplying the Lighthouse Hill Station. With the line energized to Lighthouse Hill, the necessary breakers are automatically reclosed to energize Line 4 to Nine Mile Point. The HPCI undervoltage logic locks out start of Condensate pump 11, Feedwater Booster pump 11, and Feedwater pump 11. The undervoltage logic also trips supply breakers to Powerboard 101, which prevents start of Condensate pump 12 and Feedwater Booster pump 12. Condensate pump 13 and Feedwater Booster pump 13 auto-start on low discharge header pressure. With a HPCI initiation signal in (<53"), Feedwater pump 12 then starts and injects in HPCI mode.

KA Match Justification: At NMP Unit 1, HPCI is a function of the Feedwater System. There is no separate HPCI turbine. With reactor water level <53", the feed water system automatically transfers to the HPCI mode of level control.

A. Plausible – All pumps were running prior to the loss of power, however specific HPCI logic prevents the start of non-preferred pumps.
B. Plausible – Condensate pump 11, Feedwater Booster pump 11, and Feedwater pump 11 are the backup pumps that would start if the preferred pump control switches were in pull-to-lock, but without this condition, they remain off.
D. Plausible – All non-preferred pumps remain off, however the preferred pumps automatically start.

Technical Reference(s): N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-5

Question Source: Modified Bank – 2013 Audit #47

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 #	207000 A3.02
	Importance Rating	3.8

Isolation (Emergency) Condenser

Ability to monitor automatic operations of the ISOLATION (EMERGENCY) CONDENSER including: Reactor pressure: BWR-2,3

Proposed Question: #18

The plant was operating at 100% power when a Turbine Trip with partial failure of the Turbine Bypass Valves resulted in the following:

- All control rods inserted.
- Reactor water level lowered to 4 inches and then recovered to 72 inches.
- Reactor water level was below 5 inches for 2 seconds.
- Reactor pressure rose to 1120 psig and then lowered to 918 psig.
- Reactor pressure was above 1080 psig for 14 seconds.
- The CRS has directed a cooldown at a rate of <100°F/hr using Emergency Condensers, in accordance with N1-OP-13, Emergency Cooling System.

Which one of the following is required to perform the directed cooldown, in accordance with OP-13?

A. Manually open Condensate Return Valve(s) 39-05 and/or 39-06. Both Emergency Condenser systems may be placed continually in service.

B. Verify Condensate Return Valves 39-05 and 39-06 automatically opened. Both Emergency Condenser systems may be maintained continually in service.

C. Manually open Condensate Return Valve(s) 39-05 and/or 39-06. At least one Emergency Condenser system must be cycled or secured to maintain the cooldown rate within limits.

D. Verify Condensate Return Valves 39-05 and 39-06 automatically opened. At least one Emergency Condenser system must be cycled or secured to maintain the cooldown rate within limits.

Proposed Answer: D

Explanation: Since Reactor pressure was above 1080 psig for >12 seconds, an automatic EC initiation was received. Therefore, both Condensate Return Valves automatically opened. Analysis shows that two Emergency Condensers continuously in service will result in violation of the 100°F/hr Technical Specification cooldown rate. OP-13 directs securing/cycling of Emergency Condenser systems to prevent violation of cooldown rate.

A. Plausible – Since Reactor water level was above 1080 psig for >12 seconds, both Condensate Return Valves automatically opened. Analysis shows that two Emergency Condenser systems continuously in service will result in violation of the 100°F/hr Technical Specification cooldown rate. N1-OP-13 directs securing/cycling of Emergency Condenser systems to prevent violation of cooldown rate.
B. Plausible – Analysis shows that two Emergency Condenser systems continuously in service will result in violation of the 100°F/hr Technical Specification cooldown rate. N1-OP-13 directs securing/cycling of Emergency Condenser systems to prevent violation of cooldown rate.
C. Plausible – Since Reactor water level was above 1080 psig for >12 seconds, both Condensate Return Valves automatically opened.

Technical Reference(s): N1-OP-13, N1-SOP-33A.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-5

Question Source: Modified Bank – 2010 Audit #9

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 #	263000 A4.04
Importance Rating	3.0

DC Electrical Distribution

Ability to manually operate and/or monitor in the control room: Ground detection circuit: Plant-Specific

Proposed Question: #19

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

- A3-4-3, BATTERY BOARD 11 GROUND, alarms.
- Ground isolation activities are in progress.

Which one of the following describes (1) the indication used in the control room to check the status of the ground and (2) the indication when the ground is cleared?

	Indication	Indication when ground is clear
A.	Ground Indicating Lights	Indicating lights lit
B.	Ground Indicating Lights	Indicating lights extinguished
C.	Ground Indicating Voltmeter	Voltmeter indicates 0V
D.	Ground Indicating Voltmeter	Voltmeter indicates 125V

Proposed Answer: C

Explanation: The ground indication in the control room is a voltmeter on the A3 panel, not lights. When the ground is clear, the voltmeter will indicate 0V

A. Plausible – Ground indicating lights are a common method used to indicate grounds, however that is not what is used in the control room.
B. Plausible – Ground indicating lights are a common method used to indicate grounds, however that is not what is used in the control room.
D. Plausible – 125V is the nominal board voltage. Plausible if the applicant believes the voltmeter should indicate nominal bus voltage if no ground exists

Technical Reference(s): N1-OP-47A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 A4.02
	Importance Rating	3.9

PCIS/Nuclear Steam Supply Shutoff

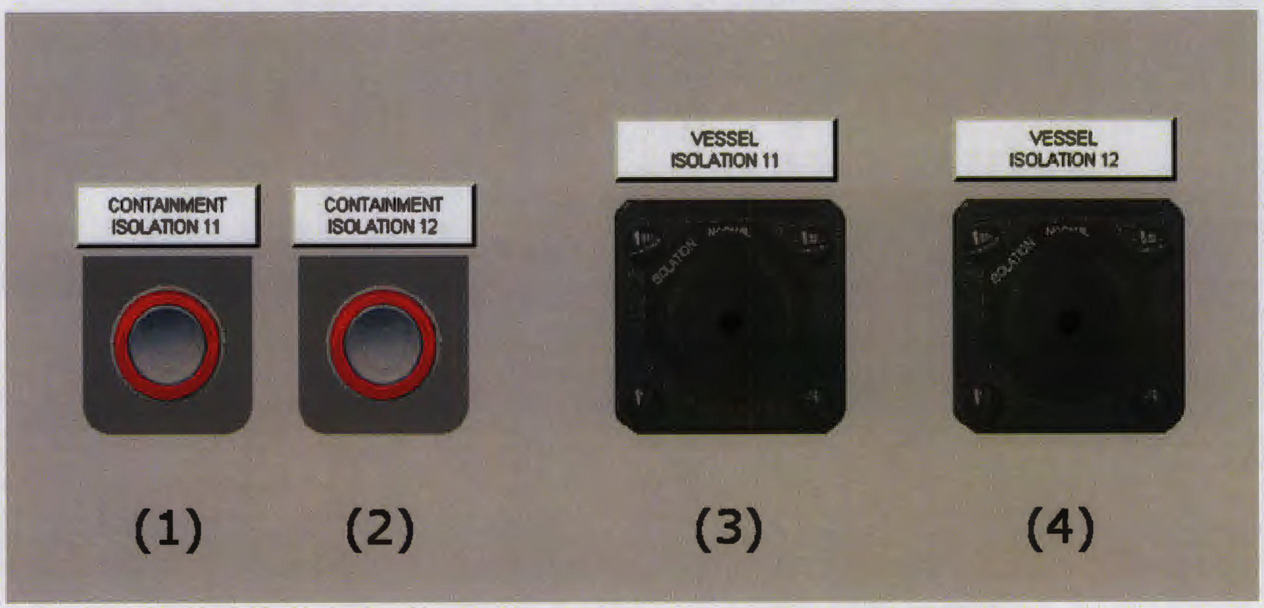
Ability to manually operate and/or monitor in the control room: Manually initiate the system

Proposed Question: #20

The plant has experienced a Main Steam leak into the steam tunnel with the following:

- Steam tunnel temperatures have exceeded the isolation setpoint, but have NOT caused any isolation signals.
- MSIVs 01-02 and 01-04 have been closed using the F-panel controls.
- MSIVs 01-01 and 01-03 have failed to close using the F-panel controls.
- The CRS has directed attempting to manually isolate MSIVs 01-01 and 01-03 using the isolation control(s) on E-console.

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



Which one of the following describes the manipulation(s) required to initiate the manual isolation the CRS has requested?

A. Both pushbutton (1) and (2) must be depressed.

B. Both control switches (3) and (4) must be rotated to ISOLATION.

C. Pushbutton (1) must be depressed. Depressing pushbutton (2) is NOT required.

D. Control switch (3) must be rotated to ISOLATION. Control switch (4) does NOT need to be manipulated.

Proposed Answer: B

Explanation: The MSIVs close as part of a Vessel isolation. A Vessel isolation is initiated by rotating both of the given control switches to ISOLATION. If only one control switch is rotated, one of the two solenoids (20-1 or 20-2) for each MSIV will remain energized and keep the valve open.

A. Plausible – Both pushbuttons must be depressed to initiate a Containment isolation, however MSIVs are part of a Vessel isolation, not a Containment isolation.
C. Plausible – Pushbutton (1) alone does cause a half Containment isolation signal, however no valves actually go closed until a full Containment isolation signal is present.
D. Plausible – Control switch (3) alone does cause a half Vessel isolation signal, however no valves actually go closed until a full Vessel isolation signal is present.

Technical Reference(s): C-19859-C sheets 10, 10a, and 11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 2.1.23
	Importance Rating	4.3

HPCI

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

Proposed Question: #21

A plant startup is in progress with the following:

- The Reactor is at the point of adding heat.
- Reactor pressure is 2 psig and slowly rising.
- Feedwater pumps 11 and 12 are in Pull-to-Lock.
- Feedwater Pump 11 Blocking Valve is closed.

In accordance with N1-OP-43A, Plant Startup, which one of the following identifies:

(1) the threshold Reactor pressure requiring Feedwater pumps 11 and 12 to be taken out of Pull-to-Lock, and

(2) the threshold Reactor power requiring Feedwater Pump 11 Blocking Valve to be opened?

	(1)	(2)
A.	110 psig	5%
B.	110 psig	25%
C.	350 psig	5%
D.	350 psig	25%

Proposed Answer: B

Explanation: N1-OP-43A requires Feedwater pumps 11 and 12 to be taken out of Pull-to-Lock prior to reaching a Reactor pressure of 110 psig. N1-OP-43A requires having both Feedwater Blocking Valves open prior to exceeding a Reactor power of 25%. Both of these requirements are in place to ensure Technical Specification 3.1.8, High Pressure Coolant Injection, is met.

KA Match Justification: At NMP Unit 1, HPCI is a function of the Feedwater System. There is no separate HPCI turbine. With FW Pumps 11 and 12 in PTL, HPCI is inoperable since those pumps are considered the HPCI pumps. The actions listed in the stem are making HPCI operable.

A. Plausible – This is the correct Reactor pressure, but the wrong Reactor power. N1-OP-43A uses 5% Reactor power as part of the criteria for placing the Reactor Mode Switch in RUN.
C. Plausible – This is the correct Reactor power, but the wrong Reactor pressure. N1-OP-43A uses 350 psig as the limit for starting the first Feedwater pump.
D. Plausible – N1-OP-43A uses 350 psig as the limit for starting the first Feedwater pump. N1-OP-43A uses 5% Reactor power as part of the criteria for placing the Reactor Mode Switch in RUN.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-10

Question Source: Modified Bank – 2010 Audit #69

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 #	259002 2.1.7
	Importance Rating	4.4

Reactor Water Level Control

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

Proposed Question: #22

The plant is performing a routine shutdown in preparation for refueling. A normal cooldown is being performed.

Which one of the following describes the desired Reactor water level control strategy and the reason for this strategy, in accordance with N1-OP-43C, Plant Shutdown?

Control Reactor water level...

A. low on GEMAC level instruments to avoid high level trips from Yarway level instruments.

B. high on GEMAC level instruments to avoid low level trips from Yarway level instruments.

C. low on Yarway level instruments to avoid high level alarms from GEMAC level instruments.

D. high on Yarway level instruments to avoid low level alarms from GEMAC level instruments.

Proposed Answer: A

Explanation: The GEMAC level instruments are compensated for various pressures and temperatures, while the Yarway level instruments are not. Therefore the GEMAC level instruments are the preferred instruments for monitoring and control whenever the Reactor is not at normal operating temperature and pressure. As the cooldown progresses, Yarway level instruments will begin to indicate higher than GEMAC level instruments. The Yarway level instruments are also the instruments that cause the high level Turbine trips and Feedwater pump trips. Therefore, to avoid trips from the Yarway level instruments, it is necessary to control Reactor water low on the GEMAC level instruments. N1-OP-43C P&L 27.0 gives an example of 66-68 inches on GEMAC level instruments, which is below a more typical range of 70-72 inches.

B. Plausible – Yarway level instruments do cause low level trips, however they indicate higher than GEMACs when at reduced Reactor pressure, not lower.
C. Plausible – GEMAC level instruments do cause high level alarms, however they indicate lower than Yarway level instruments when at reduced Reactor pressure, not higher.
D. Plausible – Yarways do indicate higher than GEMACs when at reduced Reactor pressure, however GEMACs are the desired level control instrument and the concern is avoiding high level trips from Yarways.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-9

Question Source: Bank – 2009 NRC #44

Question History: 2009 NRC #44

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 2.4.21
	Importance Rating	4.0

EDGs

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: #23

The plant is operating at 100% power with the following Emergency Diesel Generator (EDG) parameters:

Parameter	EDG 102	EDG 103
Room Temperature	45°F	80°F
Jacket Water Temperature	75°F	95°F

Which one of the following describes the operability of EDGs 102 and 103?

A. Both EDGs 102 and 103 are operable.

B. EDG 102 is operable, but EDG 103 is inoperable.

C. EDG 103 is operable, but EDG 102 is inoperable.

D. Both EDGs 102 and 103 are inoperable.

Proposed Answer: C

Explanation: EDG 102 room temperature is below the limit of 50°F and EDG 102 jacket water temperature is below the limit of 85°F. Therefore EDG 102 is inoperable. EDG 103 room and jacket water temperatures are sufficient to support operability.

Note: This question asks knowledge required at the Reactor Operator license level, as the information to answer the question is located in the Precautions and Limitations of the Operating Procedure. Therefore this question would not be considered SRO level.

KA Match Justification: This question meets the K/A by testing knowledge of EDG parameters and their associated limits for operability. Determining EDG operability is required to verify the status of the electrical safety function.

A. Plausible – EDG 103 is operable. EDG 102 room and jacket water temperatures are slightly below the limits for operability.
B. Plausible – EDG 102 room and jacket water temperatures are slightly below the limits for operability. EDG 103 room temperature is higher than the thermostat setpoint of 65°F, but not in violation of any operability requirement.
D. Plausible – EDG 102 is inoperable based on room and jacket water temperatures. EDG 103 room temperature is higher than the thermostat setpoint of 65°F, but not in violation of any operability requirement.

Technical Reference(s): N1-OP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

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

IRM

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

Proposed Question: #24

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- IRM 11 is reading 12 out of 125 on range 2.
- Then, an Operator shifts IRM 11 to range 3.

Which one of the following describes the resulting status of control rod blocks?

A control rod block…

A. does NOT occur because the IRM remains above the downscale setpoint.

B. does NOT occur because the downscale rod block is currently bypassed.

C. occurs. Control rod withdrawal is blocked, but insertion is allowed.

D. occurs. Both control rod withdrawal and insertion are blocked.

Proposed Answer: C

Explanation: Shifting an IRM from range 2 to range 3 will cause the indication to lower by a factor of approximately 3 (square root of 10) on the 0-125 scale. In this case, that will result in IRM 11 indicating approximately 4 out of 125. This is below the downscale rod block setpoint of 7.5 out of 125. The IRM downscale rod block prevents control rod withdrawal, but not insertion.

A. Plausible – IRM 11 is initially above the downscale setpoint, but close enough that changing by one range will cause indication to lower below the setpoint.
B. Plausible – The IRM 11 downscale rod block was initially bypassed when the range switch was on 2. However, the downscale rod block is NOT bypassed once the range switch is taken to 3 or above.
D. Plausible – A rod block does occur, however the IRM downscale rod block only blocks control rod withdrawal. Other rod blocks from the Rod Worth Minimizer do enforce control rod insertion rod blocks.

Technical Reference(s): N1-OP-38B, ARP F2-3-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K6.04
	Importance Rating	3.3

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: Nuclear boiler instrumentation

Proposed Question: #25

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

- Lo-Lo Rosemount Reactor Water Level Transmitters, LT-36-04B and LT-36-04D, fail high.
- Then, a loss of injection causes a Reactor scram.
- The Mode Switch is placed in SHUTDOWN.
- Reactor water level lowers to 0 inches.

Which one of the following describes the effect, if any, of these instrument failures on the required automatic isolations?

A. All required automatic isolations will still occur.

B. A required Vessel isolation will NOT occur, only.

C. A required Containment isolation will NOT occur, only.

D. Both a required Vessel and Containment isolation will NOT occur.

Proposed Answer: D

Explanation: Lo-Lo Reactor water level (<5") requires both a Vessel and Containment isolation. The given instruments both input to RPS 12. With both of these inputs failed high, RPS 12 will NOT process the required Vessel and Containment isolations. Both RPS 11 and RPS 12 must process the isolation for it to occur. Therefore, both the required Vessel and Containment isolations will NOT occur, even though two Lo-Lo Reactor water level instruments remain operable.

A. Plausible – Although the A and C Lo-Lo level instruments will still function, they input to the same RPS channel, and both RPS channels must function to cause either a Vessel or Containment isolation.
B. Plausible – A Containment isolation is also required on Lo-Lo level, and it will NOT occur due to the given instrument failures.
C. Plausible – A Vessel isolation is also required on Lo-Lo level, and it will NOT occur due to the given instrument failures.

Technical Reference(s): C-19859-C sheets 2, 5, 10, and 13

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: Modified Bank – 2010 Audit #15

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 A4.08
	Importance Rating	3.7

ADS

Ability to manually operate and/or monitor in the control room: Suppression pool level

Proposed Question: #26

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

- A large Torus leak develops.
- The Reactor is scrammed.
- Containment Spray Raw Water is aligned to add water to the Torus.
- Torus water level is 7.5 feet and lowering.
- Reactor pressure is 900 psig and slowly lowering on Turbine Bypass Valves.

Which one of the following describes the need for an RPV Blowdown?

An RPV Blowdown is...

A. currently required. Initiate Emergency Condensers and open four ERVs.

B. currently required. Initiate Emergency Condensers, but do NOT open ERVs.

C. NOT currently required, but will be if Torus water level reaches the threshold of 3.5 feet.

D. NOT currently required, but will be if Torus water level reaches the threshold of 4.25 feet.

Proposed Answer: B

Explanation: EOP-4 requires an RPV Blowdown if Torus water level cannot be maintained above 8 feet. With Torus water level at 7.5 feet, an RPV Blowdown is required. EOP-8 requires initiating Emergency Condensers. However, since Torus water level is below 8 feet, EOP-8 does NOT direct opening ERVs to prevent steam from being released directly in the Torus airspace.

A. Plausible – An RPV Blowdown is currently required, however with Torus water level below 7.5 feet, ERVs are not to be opened.
C. Plausible – An RPV Blowdown is currently required. 3.5 feet is the value of minimum downcomer submergence allowed by the Torus water level Technical Specification, but equates to an absolute Torus water level of 10.5 feet.
D. Plausible – An RPV Blowdown is currently required. 4.25 feet is the value of maximum downcomer submergence allowed by the Torus water level Technical Specification, but equates to an absolute Torus water level of 11.25 feet.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-12

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 #	2
	K/A #	202002 K1.04
	Importance Rating	3.1

Recirculation Flow Control

Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION FLOW CONTROL SYSTEM and the following: Reactor/turbine pressure regulating system: Plant-Specific

Proposed Question: #27

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

- All five Recirculation pumps are running and on the Recirc Master controller.
- An Operator adjusts the Recirc Master controller one half turn in the direction indicated by the arrow in the picture below.

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

MHC throttles Turbine Control Valves further...

A. open and stabilizes Reactor pressure at the initial value.

B. closed and stabilizes Reactor pressure at the initial value.

C. open and stabilizes Reactor pressure slightly higher than the initial value.

D. closed and stabilizes Reactor pressure slightly lower than the initial value.

Proposed Answer: D

Explanation: The picture shows the Recirc Master controller adjusted in the direction that causes lowering Recirculation flow. One half turn of the Recirc Master controller will cause a significant change in Recirculation flow. As Recirculation flow lowers, Reactor power lowers, Reactor pressure lowers, and Turbine inlet steam pressure lowers. MHC senses lowering steam pressure at the Turbine inlet and throttle Turbine Control Valves further in the closed direction to limit the pressure drop. The MHC system controls pressure proportional to steam flow, such that as steam flow lowers, Reactor pressure will be controlled at a lower value.

A. Plausible – The given manipulation causes Reactor power and pressure to lower, which causes Turbine Control Valves to further close, not open.
B. Plausible – Turbine Control Valves will further open, however the MHC system is setup to control Reactor pressure proportional to steam flow. Since steam flow will be lower, Reactor pressure will be controlled at a lower value.
C. Plausible – The given manipulation causes Reactor power and pressure to lower, which causes Turbine Control Valves to further close, not open.

Technical Reference(s): N1-OP-31, N1-248000-RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-8

Question Source: Bank – 108915

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 K2.01
	Importance Rating	3.2

Main and Reheat Steam

Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids

Proposed Question: #28

Which one of the following describes the electrical power supply to outboard Main Steam Isolation Valve (MSIV) solenoids and how the solenoids function?

A. 120 VAC; energize to close MSIVs

B. 120 VAC; de-energize to close MSIVs

C. 125 VDC; energize to close MSIVs

D. 125 VDC; de- energize to close MSIVs

Proposed Answer: D

Explanation: The outboard MSIVs (01-03 and 01-04) are air-operated valves. Their solenoids are powered by 125 VDC Battery Boards 11 and 12. The solenoids de-energize to close the valves.

A. Plausible – The solenoids are powered by 125 VDC, not 120 VAC. 120 VAC supplies power to the RPS relays that feed into the MSIV isolation circuitry. The solenoids are energized to open the MSIVs and de-energized to close the MSIVs.
B. Plausible – The solenoids are powered by 125 VDC, not 120 VAC. 120 VAC supplies power to the RPS relays that feed into the MSIV isolation circuitry.
C. Plausible – The solenoids are energized to open the MSIVs and de-energized to close the MSIVs.

Technical Reference(s): C-18002-C sheet 1, C-19859-C sheet 11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-04

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	233000 K3.01
	Importance Rating	3.2

Fuel Pool Cooling/Cleanup

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

Proposed Question: #29

The plant is in an outage with the following:

- The core has been offloaded to the Spent Fuel Pool (SFP).
- The SFP gates are removed.
- SFP temperature is 85°F and stable.

Then, an inadvertent valve closure causes a loss of Instrument Air to the SFP Cooling system.

Which one of the following describes the effect on SFP temperature and the associated SFP design temperature limit that must be avoided?

	SFP Temperature	Maintain SFP temperature...
A.	Rises	below the design limit of 110°F.
B.	Rises	below the design limit of 140°F.
C.	Lowers	above the design limit of 75°F.
D.	Lowers	above the design limit of 68°F.

Proposed Answer: B

Explanation: Loss of Instrument Air to the SFP Cooling system causes the in-service flow control valve to fail closed and the SFP circ water pumps to trip. This stops SFP Cooling flow through the heat exchanger and causes SFP temperature to rise. With the SFP in communication with the Reactor coolant (gates removed) the associated high temperature limit is the SFP design temperature of 140°F.

A. Plausible – 110°F is the limit for SFP temperature to ensure Secondary Containment operability when the SFP is not in communication with the Reactor coolant.
C. Plausible – The SFP Cooling flow control valves fail closed, not open, on loss of Instrument Air. This causes a loss of cooling and SFP temperature to rise, not lower. 75°F is the lower limit of the normal SFP temperature band.
D. Plausible – The SFP Cooling flow control valves fail closed, not open, on loss of Instrument Air. This causes a loss of cooling and SFP temperature to rise, not lower. 68°F is the lower design limit for SFP temperature.

Technical Reference(s): C-18008-C, N1-OP-6, N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-233000-RBO-8

Question Source: Modified Bank – 2006 NRC #43

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201003 K4.05
	Importance Rating	3.2

Control Rod and Drive Mechanism

Knowledge of CONTROL ROD AND DRIVE MECHANISM design feature(s) and/or interlocks which provide for the following: Rod position indication

Proposed Question: #30

A plant startup is in progress. The next control rod will be withdrawn from position 46 to 48, using Notch Override to accomplish a coupling check.

If the control rod is properly coupled, which one of the following describes the Rod Position Indication that is observed during the coupling check?

The Rod Position Indication shows...

A. a blank number display with red backlights illuminated.

B. a blank number display with red backlights extinguished.

C. a "48" in the number display with red backlights illuminated.

D. a "48" in the number display with red backlights extinguished.

Proposed Answer: C

Explanation: If the control rod is properly couple, it will not be able to move past the reed switches that provides a "48" in the number display and illuminate the red backlights for full out indication. Therefore both a "48" and red backlights will be illuminated throughout the coupling check.

A. Plausible – A blank display will occur if the control rod is NOT properly coupled, but will NOT occur if the control rod is properly coupled.
B. Plausible – A blank display will occur if the control rod is NOT properly coupled, but will NOT occur if the control rod is properly coupled. The red backlights are illuminated for a normal full out condition, not just specifically for an overtravel/uncoupled condition.
D. Plausible – The red backlights are illuminated for a normal full out condition, not just specifically for an overtravel/uncoupled condition.

Technical Reference(s): N1-OP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201002-RBO-5

Question Source: Modified Bank – 2008 Audit #34

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 #	201006 K5.06
	Importance Rating	2.8

RWM

Knowledge of the operational implications of the following concepts as they apply to ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC): Rod groups and steps: P-Spec(Not-BWR6)

Proposed Question: #31

A plant startup is in progress with the following:

- RWM Group 01 contains 12 rods.
- RWM Group 01 insert limit is position 12.
- RWM Group 01 withdraw limit is position 48.
- The first ten rods of RWM Group 01 been withdrawn to position 48.
- One of the remaining rods in RWM Group 01 is at position 12.
- The other remaining rod in RWM Group 01 is at position 10.
- A RWM Group 02 rod is then selected.

Which one of the following identifies the currently latched RWM Group and the number of RWM insert errors displayed?

	Currently Latched RWM Group	Number of RWM Insert Errors Displayed
A.	01	0
B.	01	1
C.	02	1
D.	02	2

Proposed Answer: D

Explanation: During reactor power increases, the RWM is said to "latch up" to the next step when all of the rods in the currently latched step and in all lower steps, save two rods (total), are at their respective "Withdraw Limits", AND a Control Rod is selected in the next higher step. The two remaining control rods are both displayed as insert errors because they are inserted beyond the Group 01 withdraw limit. The one control rod being inserted beyond the insert limit is NOT an issue because position 10 is the alternate insert limit.

A. Plausible – Two rods in Group 01 still need to be withdrawn, however the RWM is programmed to latch up to the next Group with two rods in the previous group not withdrawn to the limit. If Group 02 were still latched, 0 insert errors would be displayed because both control rods are at either the insert or alternate insert limit.
B. Plausible – Two rods in Group 01 still need to be withdrawn, however the RWM is programmed to latch up to the next Group with two rods in the previous group not withdrawn to the limit. One Group 01 control rod is inserted beyond the normal insert limit, but since it is at the alternate insert limit, it would not cause an insert error while Group 01 was still latched.
C. Plausible – Group 02 is latched and one Group 01 control rod is inserted beyond the normal insert limit. However, both of the Group 01 control rods that are inserted beyond the withdraw limit are displayed as insert errors.

Technical Reference(s): N1-OP-37

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201003-RBO-5

Question Source: Modified Bank – SYSID 25571

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 #	272000 K6.03
	Importance Rating	2.8

Radiation Monitoring

Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM: A.C. power

Proposed Question: #32

The plant is operating at 100% power when RPS bus 12 de-energizes due to a spurious breaker trip.

Which one of the following describes the response of Reactor Building Ventilation (RBV) radiation monitoring and the associated logic?

RBV radiation monitoring and the associated logic fail such that...

A. both trains of Reactor Building Emergency Ventilation automatically start.

B. Reactor Building Emergency Ventilation system 12 automatically starts, only.

C. both trains of Reactor Building Emergency Ventilation lose automatic start capability.

D. Reactor Building Emergency Ventilation system 12 loses automatic start capability, only.

Proposed Answer: A

Explanation: RBV radiation monitors are arranged such that an upscale condition on either monitor causes an initiation of both trains of RBEVS. Loss of RPS bus 12 causes RBV radiation monitor 12 to fail downscale, and prevents an upscale trip from this radiation monitor. However, loss of RPS bus 12 also causes loss of power to the radiation monitor logic, such that both trains of RBEVS automatically start.

B. Plausible – Some systems are divisionalized in this manner, however both trains of RBEVS automatically start on signals from either radiation monitors/logics 11 or 12.
C. Plausible – RBV radiation monitor 12 does fail downscale, which by itself would prevent that radiation monitor from producing its upscale trip. However, loss of RPS bus 12 also causes loss of power to the radiation monitor logic, such that both trains of RBEVS automatically start.
D. Plausible – RBV radiation monitor 12 does fail downscale, which by itself would prevent that radiation monitor from producing its upscale trip. However, loss of RPS bus 12 also causes loss of power to the radiation monitor logic, such that both trains of RBEVS automatically start.

Technical Reference(s): N1-SOP-40.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-272000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	219000 A1.07
	Importance Rating	3.2

RHR/LPCI: Torus/Pool Cooling Mode

Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE controls including: Emergency generator loading

Proposed Question: #33

A stuck open ERV has resulted in the following:

- A manual Reactor scram was inserted.
- Following the scram, Lines 1 and 4 de-energized.
- Only Emergency Diesel Generator (EDG) 102 started.
- N1-SOP-33A.1, Loss of 115 KV, has been entered.
- Torus Cooling is being placed in service on Containment Spray loop 111.
- An operator starts Containment Spray Raw Water pump 111.
- EDG 102 load stabilizes at 2407 KW.
- Then, the operator starts Containment Spray pump 111.
- EDG 102 load rises to 2689 KW.

Which one of the following describes EDG 102 operation with respect to load limits before and after the operator started Containment Spray pump 111, in accordance with N1-SOP-33A.1, ?

Before starting Containment Spray pump 111, EDG 102 load was __(1)__ the ***continuous*** load rating. After starting Containment Spray pump 111, EDG 102 load is __(2)__ the ***emergency*** load rating.

	(1)	(2)
A.	below	below
B.	below	above
C.	above	below
D.	above	above

Proposed Answer: A

Explanation: EDG 102 has a 2586 KW continuous load rating and a 2845 KW emergency load rating. Before starting Containment Spray pump 111, EDG 102 had a load of 2407 KW, which is below the continuous rating of 2586 KW. After starting Containment Spray pump 111, EDG 102 has a load of 2689 KW, which is above the continuous rating but below the emergency rating of 2845 KW.

B. Plausible – The final load is above the continuous rating, but below the emergency rating.
C. Plausible – The initial load is high, but still below the continuous rating.
D. Plausible – The initial load is high, but still below the continuous rating. The final load is above the continuous rating, but below the emergency rating.

Technical Reference(s): N1-SOP-33A.1, N1-OP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-RBO-9

Question Source: Bank - JAF April 2014 NRC #33

Question History: JAF April 2014 NRC #33

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 A2.01
	Importance Rating	3.7

Main Turbine Generator and Auxiliary Systems

Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine trip

Proposed Question: #34

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

- The Main Generator is synchronized to the grid.
- Then, high Turbine vibrations occur and require a manual Turbine trip.

Which one of the following describes the required action(s) to perform the manual Turbine trip, in accordance with N1-SOP-31.1, Turbine Trip?

A. Depress the UNIT EMERGENCY TRIP pushbutton, only.

B. Place the Reactor Mode Switch in SHUTDOWN, then depress the UNIT EMERGENCY TRIP pushbutton.

C. Depress and hold the FEEDWATER RETURN TO NORMAL AFTER HPCI pushbuttons, then place the Reactor Mode Switch in SHUTDOWN.

D. Depress and hold the FEEDWATER RETURN TO NORMAL AFTER HPCI pushbuttons, then depress the UNIT EMERGENCY TRIP pushbutton.

Proposed Answer: D

Explanation: With Reactor power <35%, N1-SOP-31.1 directs accomplishing the manual Turbine trip without inserting a manual Reactor scram. Therefore, the Reactor Mode Switch is NOT placed in SHUTDOWN. Depressing the UNIT EMERGENCY TRIP pushbutton accomplishes the Turbine trip. However, with no other action, this would cause an unnecessary HPCI actuation. N1-SOP-31.1 directs first depressing and holding the FEEDWATER RETURN TO NORMAL AFTER HPCI pushbuttons, then depressing the UNIT EMERGENCY TRIP pushbutton.

A. Plausible – This would cause a manual Turbine trip, however N1-SOP-31.1 also requires actions to prevent an unnecessary HPCI actuation.
B. Plausible – This would be correct if Reactor power was ≥35%. However, with Reactor power <35%, the Reactor Mode Switch is NOT placed in SHUTDOWN.
C. Plausible – The HPCI pushbuttons must be depressed, however with Reactor power <35%, the Reactor Mode Switch is NOT placed in SHUTDOWN.

Technical Reference(s): N1-SOP-31.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245000-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 #	2
	K/A #	286000 A3.06
	Importance Rating	3.0

Fire Protection

Ability to monitor automatic operations of the FIRE PROTECTION SYSTEM including: Fire dampers

Proposed Question: #35

Given the following fire zones:

(1) H-3031, Auxiliary Control Room
(2) C-3011, Cable Spreading Room
(3) SP-5033, Diesel Fire Pump Room

Which one of the following identifies the fire zones that have automatic fire damper closure upon initiation?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: Both the given Auxiliary Control Room and Cable Spreading Room fire zones cause automatic fire damper closure upon initiation. The given Diesel Fire Pump Room fire zone does NOT cause automatic fire damper closure upon initiation.

B. Plausible – The Auxiliary Control Room does have automatic damper closure, but the Diesel Fire Pump Room does not. The Cable Spreading Room also has automatic damper closure.
C. Plausible – The Cable Spreading Room does have automatic damper closure, but the Diesel Fire Pump Room does not. The Auxiliary Control Room also has automatic damper closure.
D. Plausible – Both the Cable Spreading Room and the Auxiliary Control Room have automatic damper closure, but the Diesel Fire Pump Room does not.

Technical Reference(s): N1-OP-21A, N1-OP-21C, N1-OP-21D, N1-286000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-286000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 A4.12
	Importance Rating	3.8

RHR/LPCI: Containment Spray Mode

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

Proposed Question: #36

The plant has experienced a steam leak in the Drywell with the following:

- Containment parameters are in the OK TO SPRAY region of the Containment Spray Initiation Limit curve.
- Drywell pressure is 11 psig and rising.
- Torus pressure is 10 psig and rising.

Which one of the following describes the requirements for initiating and securing Containment Spray based on Containment pressure, in accordance with N1-EOP-4, Primary Containment Control?

Containment Spray initiation is ___(1)___. Once initiated, Containment Spray must be secured if Drywell pressure drops below the threshold of ___(2)___.

	(1)	(2)
A.	currently required	0 psig
B.	currently required	3.5 psig
C.	NOT currently required	0 psig
D.	NOT currently required	3.5 psig

Proposed Answer: D

Explanation: N1-EOP-4 does not require initiation of Containment Spray based on Containment pressure until Torus pressure exceeds 13 psig. Since Torus pressure is less than 13 psig, Containment Spray is not yet required based on Containment pressure. 3.5 psig is the threshold for securing Containment Sprays in N1-EOP-4.

A. Plausible – Both given Containment pressures are well above the 3.5 psig high Drywell pressure setpoint, however Torus pressure is still below 13 psig. Therefore Containment Spray is not yet required based on Containment pressure. 0 psig is the threshold for securing Containment Sprays in the Severe Accident Procedures for Hydrogen / Oxygen control. In N1-EOP-4, 3.5 psig is the threshold.
B. Plausible – Both given Containment pressures are well above the 3.5 psig high Drywell pressure setpoint, however Torus pressure is still below 13 psig. Therefore Containment Spray is not yet required based on Containment pressure.
C. Plausible – 0 psig is the threshold for securing Containment Sprays in the Severe Accident Procedures for Hydrogen / Oxygen control. In N1-EOP-4, 3.5 psig is the threshold.

Technical Reference(s): N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226001-RBO-12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 2.4.46
	Importance Rating	4.2

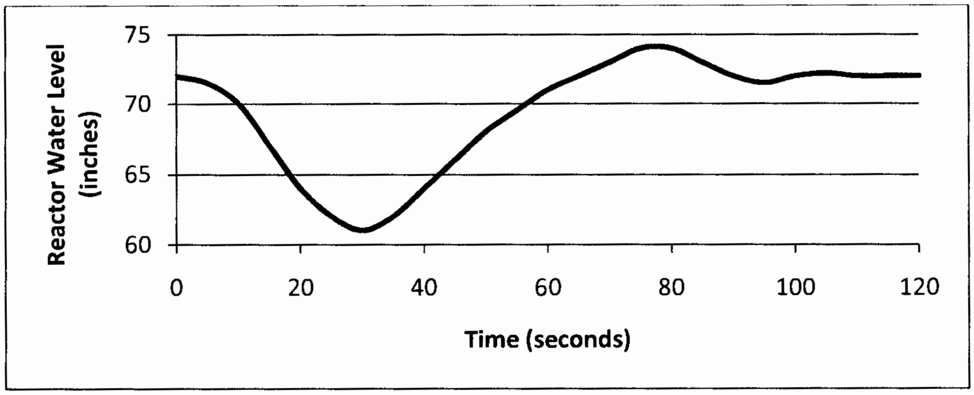
Reactor Feedwater

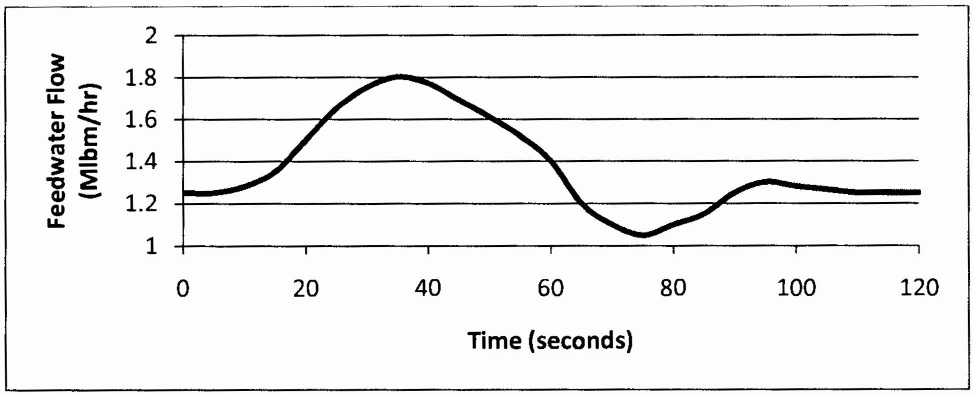
Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: #37

A plant startup is in progress with the following:

- Feedwater pump 11 is operating.
- Feedwater pumps 12 and 13 are secured.
- A transient results in the following Reactor water level and Feedwater flow traces:





The following annunciators remained off throughout the transient:

- F2-3-3, REACT VESSEL LEVEL HIGH-LOW
- F4-4-1, HPCI MODE AUTO INITIATE

Which one of the following describes the status of these annunciators during the transient?

A. Both annunciators were consistent with plant conditions throughout the transient.

B. F2-3-3 should have alarmed during the transient. F4-4-1 was consistent with plant conditions throughout the transient.

C. F4-4-1 should have alarmed during the transient. F2-3-3 was consistent with plant conditions throughout the transient.

D. Both annunciators should have alarmed during the transient.

Proposed Answer: B

Explanation: F2-3-3 should have alarmed since Reactor water level went below the setpoint of 65". F4-4-1 should have stayed off during the entire transient since Reactor water level stayed above 53" and Feedwater flow stayed below 1.9 Mlbm/hr.

A. Plausible – F2-3-3 should have alarmed, even though Reactor water level did not reach the 53" scram setpoint.
C. Plausible – F4-4-1 was consistent with plant conditions since conditions did not degrade enough to warrant HPCI initiation. F2-3-3 should have alarmed, even though Reactor water level did not reach the 53" scram setpoint.
D. Plausible – F4-4-1 was consistent with plant conditions since conditions did not degrade enough to warrant HPCI initiation.

Technical Reference(s): N1-OP-16, ARPs F2-3-3 and F4-4-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-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 #	2
	Group #	2
	K/A #	201001 K6.03
	Importance Rating	3.0

CRD Hydraulic

Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD DRIVE HYDRAULIC System: Plant air systems

Proposed Question: #38

Which one of the following describes a Control Rod Drive (CRD) system valve that would be affected by a loss of Instrument Air and the associated failure position of that valve on loss of Instrument Air?

	Valve	Failure Position
A.	CRD Flow Control Valve	Open
B.	CRD Flow Control Valve	Closed
C.	CRD Drive Water Control Valve	Open
D.	CRD Drive Water Control Valve	Closed

Proposed Answer: B

Explanation: The CRD Flow Control Valve utilizes Instrument Air and fails closed on loss of Instrument Air. The CRD Drive Water Control Valve is motor operated, not air operated.

A. Plausible – The CRD Flow Control Valve fails closed on loss of Instrument Air, not open, such as the CRD scram valves.
C. Plausible – The CRD Drive Water Control Valve is motor operated, not air operated like the Flow Control Valve or scram valves.
D. Plausible – The CRD Drive Water Control Valve is motor operated, not air operated like the Flow Control Valve or scram valves.

Technical Reference(s): C-18016-C sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AK1.03
	Importance Rating	3.5

Main Turbine Generator Trip

Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP: Pressure effects on reactor level

Proposed Question: #39

The plant is operating at 17% power with the following:

- Feedwater pump 13 is the only Feedwater pump operating.
- Feedwater FCV 13 is in automatic.
- Reactor water level is 70" and stable.

Then, a Main Turbine trip occurs. Total Feedwater flow stabilizes at 1.3 Mlbm/hr once steady-state conditions are achieved.

Which one of the following describes the effect on Reactor water level immediately after the trip and once steady-state conditions are reached?

	Immediate Reactor Water Level Response	Reactor Water Level at Steady-State
A.	Lowers	Above 70"
B.	Lowers	Below 70"
C.	Rises	Above 70"
D.	Rises	Below 70"

Proposed Answer: A

Explanation: The sudden pressure rise from the TSV closure collapses voids and causes Reactor water level to shrink, lowering Reactor water level. The Turbine trip also causes HPCI to initiate. This starts the electric Feedwater pumps. Feedwater flow control valve 12 will control Reactor water level at 72" while in the HPCI mode.

B. Plausible – Reactor water level does initially lower, however Feedwater FCV 12 in the HPCI mode will restore and maintain Reactor water level at 72". Feedwater FCV 11 attempts to control Reactor water level at 68" and will fully close.
C. Plausible – Although lowering steam flow would normally cause Reactor water level to rise, the sudden pressure spike following a Turbine trip counteracts this effect and causes Reactor water level to initially lower.
D. Plausible – Although lowering steam flow would normally cause Reactor water level to rise, the sudden pressure spike following a Turbine trip counteracts this effect and causes Reactor water level to initially lower. Reactor water level does initially lower, however Feedwater FCV 12 in the HPCI mode will restore and maintain Reactor water level at 72". Feedwater FCV 11 attempts to control Reactor water level at 68" and will fully close.

Technical Reference(s): N1-SOP-31.1, N1-OP-16, UFSAR XV 3.14.1, N1-239001-RBO-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245000-RBO-11

Question Source: Bank – 2010 NRC #41

Question History: 2010 NRC #41

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AK1.03
	Importance Rating	2.9

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: Electrical bus divisional separation

Proposed Question: #40

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

- A loss of Battery Board 12 occurs.
- Then, an Operator transfers the DC supply switch from "Battery Board 12" to "Battery Board 11" at Powerboard (PB) 103.

Which one of following describes the resulting status of control power for Breaker R-1013, RESERVE SUPPLY TO PB 103, and CORE SPRAY PUMP 112 MOTOR BREAKER?

	Breaker R-1013, RESERVE SUPPLY TO PB 103	CORE SPRAY PUMP 112 MOTOR BREAKER
A.	Has control power	Has control power
B.	Has control power	Does NOT have control power
C.	Does NOT have control power	Has control power
D.	Does NOT have control power	Does NOT have control power

Proposed Answer: C

Explanation: DC control power is divisionalized such that Battery Board 11 normally supplies Powerboard 102 and Battery Board 12 normally supplies Powerboard 103. Some provisions are installed to cross-tie divisions. With a loss of Battery Board 12, the control power for Powerboard 103 can be partially transferred to Battery Board 11. All of the load breakers, such as for Core Spray pump 112, are transferrable, but the supply breakers, such as R-1013, are non-transferrable.

A. Plausible – Control power is restored for all of the load breakers on Powerboard 103, but not for the supply breakers, such as R-1013.
B. Plausible – Control power is restored for all of the load breakers on Powerboard 103, but not for the supply breakers, such as R-1013.
D. Plausible – Both breakers lost control power when Battery Board 12 de-energized, but control power to the Core Spray breaker is restored from Battery Board 11.

Technical Reference(s): N1-SOP-47A.1, N1-OP-47A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-8

Question Source: Bank – 2008 NRC #41

Question History: 2008 NRC #41

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 #	295024 EK1.01
	Importance Rating	4.1

High Drywell Pressure

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

Proposed Question: #41

The plant is operating at 30% power during a startup with the following:

- Pressure in the Drywell rose during the startup requiring opening the following valves to vent the Drywell:
    - 201-18, EM VENTILATION FROM DW & TORUS BV.
    - 201-32, DW N2 VENT & PURGE ISOLATION VALVE 11.
    - 201-31, DW N2 VENT & PURGE ISOLATION VALVE 12.
- Then, a steam leak develops in the Drywell.
- The Reactor is scrammed.
- Drywell pressure is 3.7 psig and slowly rising.

Which one of the following describes the status of the Primary Containment Isolation?

The Primary Containment...

A. fully isolates.

B. does NOT fully isolate until 201-18 is manually closed.

C. does NOT fully isolate until either 201-31 or 201-32 is manually closed.

D. does NOT fully isolate until 201-18 and either 201-31 or 201-32 are manually closed.

Proposed Answer: A

Explanation: High Drywell pressure (>3.5 psig) causes an automatic Primary Containment isolation. 201-18 does NOT automatically close on the Primary Containment isolation signal. However, 201-31 and 201-32 both automatically close. Since 210-18 is downstream of these Primary Containment isolation valves, the Primary Containment is successfully isolated even without closing 201-18.

B. Plausible – 201-18 does remain open, however the Primary Containment is fully isolated since valves upstream have closed.
C. Plausible – 201-31 and 201-32 both automatically close on the high Drywell pressure signal.
D. Plausible – 201-18 does remain open, however the Primary Containment is fully isolated since valves upstream (201-31 and 201-32) have closed. 201-31 and 201-32 both automatically close on the high Drywell pressure signal.

Technical Reference(s): N1-SOP-40.2, N1-OP-9 attachment 5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: Bank – 2008 Audit #20

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EK2.09
	Importance Rating	2.9

High Off-site Release Rate

Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Post accident sample system (PASS): Plant-Specific

Proposed Question: #42

A loss of coolant accident has occurred with the following:

- Reactor water level is -70 inches and slowly rising.
- Drywell pressure is 10 psig and slowly rising.
- The offsite release rate is at the Alert level and rising.
- Chemistry needs to perform post-accident sampling of the Reactor coolant to assist in release calculations.
- Chemistry needs the following valves opened:
    - 63-04, REACTOR SAMPLE RETURN IV
    - 63-05, REACTOR SAMPLE RETURN IV

Which one of the following describes the required action to allow opening these valves?

A. Install the PRIMARY CONT ISOL BYPASS jumpers.

B. Install the RPS SCRAM LOGIC RELAY BYPASS jumpers.

C. Place the AUTO VESSEL ISOL CH 11 and 12 switches in BYPASS.

D. Place the CAD CHANNEL 11 and 12 RPS BYPASS switches in BYPASS.

Proposed Answer: C

Explanation: Placing the AUTO VESSEL ISOL CH 11 and 12 switches in BYPASS allows re-opening of various post-accident sampling valves (including 63-04 and 63-05), even with a sealed-in isolation signal (such as there is based on Reactor water level being well below 5 inches).

A. Plausible – These jumpers are installed per N1-EOP-1 attachment 3 to bypass Primary Containment isolation signals, however they are not needed for opening 63-04 and 63-05.
B. Plausible – These jumpers are installed per N1-EOP-3.1 attachment 3 to bypass RPS signals, however they are not needed for opening 63-04 and 63-05.
D. Plausible – These switches are used, for example in N1-EOP-1 attachment 11, to allow bypassing of some Primary Containment isolation signals.

Technical Reference(s): C-19859-C sheets 10 and 11A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: Modified Bank – SYSID 52162

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 #	295018 AK2.02
Importance Rating	3.4

Partial or Complete Loss of CCW

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: Plant operations

Proposed Question: #43

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

- RBCLC pumps 11 and 12 trip.
- RBCLC pump 13 is started.
- RBCLC supply temperature is 95°F and slowly rising.

Which one of the following actions is to be performed first based on the challenge to RBCLC cooling capability, in accordance with N1-SOP-11.1, RBCLC Failure?

A. Trip the RWCU pumps.

B. Secure Fuel Pool Cooling.

C. Isolate RBCLC to the Drywell.

D. Shutdown two Recirculation pumps.

Proposed Answer: A

Explanation: N1-SOP-11.1 contains the following step:

IF RBCLC cooling capability is challenged THEN trip RWCU pumps.

Since the only available RBCLC pump is running and temperature is rising, cooling capability is being challenged and RWCU pumps must be tripped.

B. Plausible – N1-SOP-11.1 directs monitoring SFPC and also directs securing affected equipment if high temperatures occur. However, nothing in the question indicates specific high temperatures for SFPC and N1-SOP-11.1 specifically directs tripping RWCU pumps due to degraded RBCLC cooling capability.
C. Plausible – N1-SOP-11.1 directs monitoring Drywell cooling and also directs securing affected equipment if high temperatures occur. However, nothing in the question indicates specific high temperatures for Drywell cooling and N1-SOP-11.1 specifically directs tripping RWCU pumps due to degraded RBCLC cooling capability.
D. Plausible – N1-SOP-11.1 directs monitoring Recirculation system temperatures and also directs securing affected equipment if high temperatures occur. However, nothing in the question indicates specific high temperatures for Recirculation components and N1-SOP-11.1 specifically directs tripping RWCU pumps due to degraded RBCLC cooling capability. Plant operation can continue with two Recirculation pumps secured.

Technical Reference(s): N1-SOP-11.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-10

Question Source: Bank – 2010 Audit #8

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 #	295037 EK2.05
	Importance Rating	4.0

SCRAM Conditions 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: CRD hydraulic system

Proposed Question: #44

A failure to scram has occurred with the following:

- Reactor power is 15% and stable.
- The Reactor Mode Switch is in SHUTDOWN.
- The RPS scram pushbuttons have been depressed.
- ARI has been initiated.
- No control rods have inserted.
- All of the white Scram Solenoid Group lights are extinguished.
- All of the blue SCRAM lights on the full core display are extinguished.
- All of the amber accumulator lights on the full core display are extinguished.
- Reactor pressure is 425 psig and stable.
- No CRD pumps are available.

Which one of the following methods in N1-EOP-3.1, Alternate Control Rod Insertion, is available to insert control rods?

A. Attachment 1, Scram Control Rods Electrically

B. Attachment 2, Scram Control Rods by Venting the Scram Air Header

C. Attachment 4, Scram Control Rods by Repeated Manual Scram Signals

D. Attachment 5, Scram Control Rods by Increasing Cooling Water Differential Pressure

Proposed Answer: B

Explanation: The given indications show that the RPS scram groups de-energized (all white lights extinguished), but the scram valves did not open (all blue lights extinguished) and the accumulators did not discharge (all amber lights extinguished). This is indicative of a failure of the scram air header to depressurize. Venting the scram air header per EOP-3.1 attachment 2 is available to insert control rods.

A. Plausible – The given indications show that the RPS scram groups are already de-energized (all white lights extinguished) without causing control rod insertion. Pulling RPS fuses per EOP-3.1 attachment 1 will not result in any change since RPS scram groups are already de-energized.
C. Plausible – The given indications show that the RPS scram groups are already de-energized (all white lights extinguished) without causing control rod insertion. Repeating manual scrams per EOP-3.1 attachment 4 will not do anything to correct the failure that prevented the scram air header from depressurizing on the first scram.
D. Plausible – Raising cooling water D/P per EOP-3.1 attachment 6 would work, however with no CRD pumps available, this method cannot be accomplished.

Technical Reference(s): N1-EOP-3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-10

Question Source: Bank – 2009 NRC #17

Question History: 2009 NRC #17

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AK3.06
	Importance Rating	3.2

SCRAM

Knowledge of the reasons for the following responses as they apply to SCRAM: Recirculation pump speed reduction: Plant-Specific

Proposed Question: #45

N1-SOP-1, Reactor Scram, directs reducing Recirculation flow following a Reactor scram.

Which one of the following describes a value that is within the directed Recirculation flow band and the reason for reducing Recirculation flow, in accordance with N1-SOP-1?

	Acceptable Recirculation Flow	Reason for Reducing Recirculation Flow
A.	28 Mlbm/hr	Prevent damage to Recirculation pumps
B.	28 Mlbm/hr	Prevent damage to Reactor internals
C.	48 Mlbm/hr	Prevent damage to Recirculation pumps
D.	48 Mlbm/hr	Prevent damage to Reactor internals

Proposed Answer: B

Explanation: N1-SOP-1 directs Recirculation flow to be reduced to within a band of 25-43 Mlbm/hr. 48 Mlbm/hr is too high, but 28 Mlbm/hr is within this band. The reason for the reduction is to exit the Reactor Internals Protection Region of the Power-Flow Map. Operation within this region (low Reactor power with high Recirculation flow) may cause damage to the core shroud in the event of a Main Steam Line rupture.

A. Plausible – 28 Mlbm/hr is within the required band, however the reason is to protect Reactor internals, not Recirculation pumps. Individual Recirculation pump flows do rise slightly following a scram due to reduced flow-resistance in the core.
C. Plausible – 48 Mlbm/hr is less than normal full-power Recirculation flow, but still above the required band. The reason is to protect Reactor internals, not Recirculation pumps. Individual Recirculation pump flows do rise slightly following a scram due to reduced flow-resistance in the core.
D. Plausible – 48 Mlbm/hr is less than normal full-power Recirculation flow, but still above the required band.

Technical Reference(s): N1-SOP-1, N1-OP-1 P&L 22, Power-Flow Map

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK3.01
	Importance Rating	3.9

Reactor Low Water Level

Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL: Automatic depressurization system actuation

Proposed Question: #46

Which one of the following describes the Automatic Depressurization System (ADS) logic and the design basis for ADS actuation?

ADS logic requires __(1)__ to actuate. The design basis for ADS actuation is to __(2)__.

	(1)	(2)
A.	both lo-lo-lo Reactor water level and high Drywell pressure	extend the time the core remains adequately cooled during a station blackout
B.	both lo-lo-lo Reactor water level and high Drywell pressure	ensure Core Spray is allowed to inject during a small break loss of coolant accident
C.	either lo-lo-lo Reactor water level or high Drywell pressure	extend the time the core remains adequately cooled during a station blackout
D.	either lo-lo-lo Reactor water level or high Drywell pressure	ensure Core Spray is allowed to inject during a small break loss of coolant accident

Proposed Answer: B

Explanation: The design basis for ADS actuation is to rapidly depressurize the Reactor in the event of a small break LOCA to ensure Core Spray can inject to the Reactor to prevent fuel clad overheating. ADS actuation requires both lo-lo-lo Reactor water level and high Drywell pressure.

A. Plausible – Low Reactor water level and elevated Drywell pressure are expected during a station blackout. Additionally, the station blackout strategy does include reducing Reactor pressure to both keep the fuel clad cool and allow injection with low pressure systems.
C. Plausible – These conditions would each result in Core Spray pumps starting, but both are needed simultaneously for ADS actuation. Low Reactor water level and elevated Drywell pressure are expected during a station blackout. Additionally, the station blackout strategy does include reducing Reactor pressure to both keep the fuel clad cool and allow injection with low pressure systems.
D. Plausible – These conditions would each result in Core Spray pumps starting, but both are needed simultaneously for ADS actuation.

Technical Reference(s): UFSAR Section VII.A.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-1 & 5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EK3.03
	Importance Rating	3.6

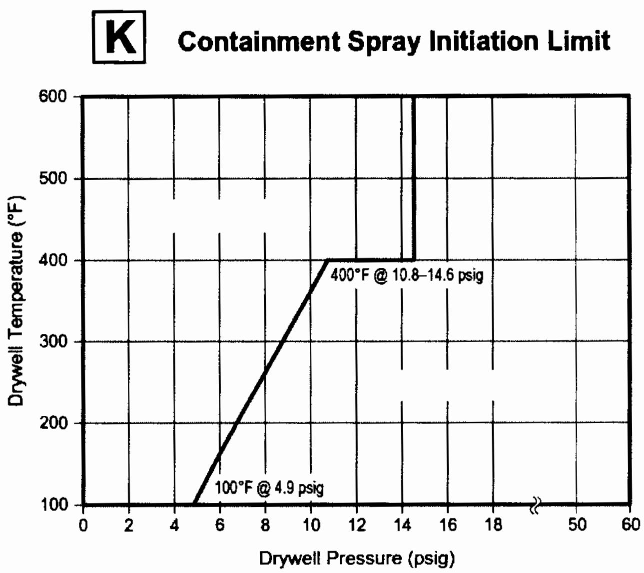
High Drywell Temperature

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: Drywell spray operation: Mark-I&II

Proposed Question: #47

The plant has scrammed with the following:

- Drywell pressure is 10 psig and rising slowly.
- Drywell temperature is 299°F and rising slowly.
- Torus water level is 9 feet and rising slowly.
- The CRS directs Containment Spray initiated.



Which one of the following describes the basis for initiating Containment Spray based on current parameters?

A. Pressure suppression function is about to be lost.

B. Remain within the Containment Spray Initiation Limit.

C. ADS qualification temperature is about to be exceeded.

D. Prevent cyclic stresses on downcomers and vent headers.

Proposed Answer: C

Explanation: N1-EOP-4 gives direction to initiate Containment Spray before Drywell average temperature exceeds 300°F. ADS environmental qualification temperature is 301°F and is the basis for initiating spray before 300°F.

A. Plausible – Drywell pressure is elevated, however it is not yet close to the value at which the pressure suppression function would be lost (~16 psig at this Torus water level).
B. Plausible – Drywell conditions are close to the Containment Spray Initiation Limit, however this is not the reason for initiating Containment Sprays.
D. Plausible – This is the reason for initiating Containment Spray on high Torus pressure, which is not indicated in the given conditions.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2006 NRC #62

Question History: 2006 NRC #62

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 #	295001 AA1.07
	Importance Rating	3.1

Partial or Complete Loss of Forced Core Flow Circulation

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Nuclear boiler instrumentation system

Proposed Question: #48

The plant is operating at power when the following occur:

- Powerboard 11 de-energizes due to an electrical fault.
- Reactor Recirculation pump (RRP) flows stabilize at the following values:
  - RRP 11: 4 Mlbm/hr
  - RRP 12: 4 Mlbm/hr
  - RRP 13: 11 Mlbm/hr
  - RRP 14: 11 Mlbm/hr
  - RRP 15: 11 Mlbm/hr
- Reactor power stabilizes at 52%.
- Main Condenser vacuum is 27 inches Hg and stable.

Note: A portion of the three loop power-to-flow map is provided on the following page.

Which one of the following describes the current rod line and if continuous operation is permitted in the current region of the power-to-flow map, in accordance with N1-SOP-1.5, Unplanned Power Change?

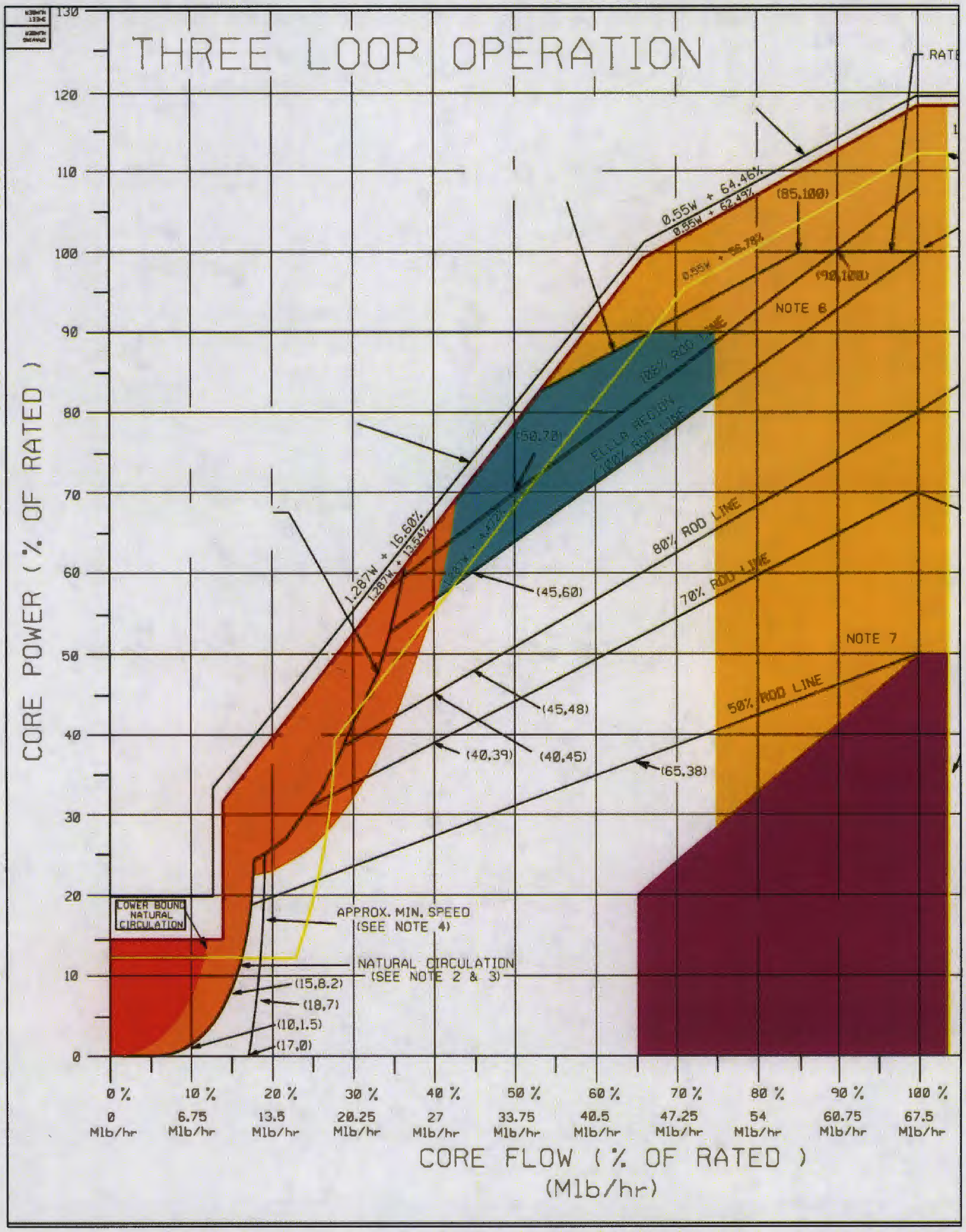
The plant is operating...

A. below the 100% rod line. Action is required to exit the current region.

B. below the 100% rod line. Continuous operation is allowed in the current region.

C. above the 100% rod line. Action is required to exit the current region.

D. above the 100% rod line. Continuous operation is allowed in the current region.



THREE LOOP OPERATION
CORE POWER (% OF RATED)
CORE FLOW (% OF RATED)
(Mlb/hr)
130
120
110
100
90
80
70
60
50
40
30
20
10
0
0 %
0 Mlb/hr
10 %
6.75 Mlb/hr
20 %
13.5 Mlb/hr
30 %
20.25 Mlb/hr
40 %
27 Mlb/hr
50 %
33.75 Mlb/hr
60 %
40.5 Mlb/hr
70 %
47.25 Mlb/hr
80 %
54 Mlb/hr
90 %
60.75 Mlb/hr
100 %
67.5 Mlb/hr
0.55W + 64.46%
0.55W + 62.49%
0.55W + 56.78%
(85,100)
(90,100)
NOTE 6
100% ROD LINE
ELLLA REGION
(100% ROD LINE
(50,70)
1.287W + 16.60%
1.287W + 13.54%
(45,60)
80% ROD LINE
70% ROD LINE
NOTE 7
50% ROD LINE
(45,48)
(40,39)
(40,45)
(65,38)
LOWER BOUND NATURAL CIRCULATION
APPROX. MIN. SPEED
(SEE NOTE 4)
NATURAL CIRCULATION
(SEE NOTE 2 & 3)
(15,8.2)
(18,7)
(10,1.5)
(17,0)
RATE

Proposed Answer: A

Explanation: With Powerboard 11 de-energized, RRPs 11 and 12 have tripped and their indicated flows are actually reverse flow. To calculate total core flow in this condition, the flows from the running RRPs must be added, and then the reverse flows through the tripped pumps must be subtracted from this number (11+11+11-4-4 = 25 Mlbm/hr). With this total core flow at 52% Reactor power, operation is below the 100% rod line and in the restricted zone. N1-SOP-1.5 requires action to exit the restricted zone.

Note: The question meets the K/A by testing a situation with a partial loss of core flow (two Recirc pumps trips on loss of Powerboard 11) and requiring the candidate to interpret nuclear boiler instrumentation (core flow indications) to determine the operating point on the power to flow map.

B. Plausible – If total core flow was calculated as the sum of all indicated RRP flows, it would be 41 Mlbm/hr, which would place the plant below the 100% rod line and in the acceptable portion of the power-to-flow map.
C. Plausible – Actual operation is just below the 100% rod line.
D. Plausible – Actual operation is just below the 100% rod line. If total core flow was calculated as the sum of all indicated RRP flows, it would be 41 Mlbm/hr, which would place the plant in the acceptable portion of the power-to-flow map.

Technical Reference(s): 3 Loop Power-to-Flow Map, N1-SOP-30.1, N1-SOP-1.3, N1-SOP-1.5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA1.03
	Importance Rating	3.0

Partial or Complete Loss of Instrument Air

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air compressor power supplies

Proposed Question: #49

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

- The Reactor scrams.
- Lines 1 and 4 de-energize.
- Emergency Diesel Generator (EDG) 103 fails to start.

Which one of the following air compressors is available to supply Instrument Air?

A. Service Air Compressor

B. Instrument Air Compressor 11

C. Instrument Air Compressor 12

D. Instrument Air Compressor 13

Proposed Answer: B

Explanation: With the Reactor scrammed, Lines 1 and 4 de-energized, and EDG 103 failing to start, Powerboards 11, 12, 101, and 103 are all de-energized. Additionally, downstream Powerboards 14, 15, and 17 are all de-energized. Service Air Compressor is unavailable with Powerboard 15 de-energized. Instrument Air Compressor 12 is unavailable with Powerboard 17 de-energized. Instrument Air Compressor 13 is unavailable with Powerboard 14 de-energized. Only Instrument Air Compressor 11 is available, with Powerboard 16 supplied by EDG 102.

A. Plausible – Service Air Compressor is powered from Powerboard 15, which is unavailable due to the combination of the Reactor scram and loss of Lines 1 and 4.
C. Plausible – Instrument Air Compressor 12 is powered from Powerboard 17, and should available with a loss of offsite power and Reactor scram. However, with EDG 103 failing, Powerboards 103 and 17 are de-energized.
D. Plausible – Instrument Air Compressor 13 is powered from Powerboard 14, which is unavailable due to the combination of the Reactor scram and loss of Lines 1 and 4.

Technical Reference(s): N1-OP-20, C-19409-C sheet 1B

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-4

Question Source: Bank – 2009 NRC #8

Question History: 2009 NRC #8

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 #	295025 EA1.03
	Importance Rating	4.4

High Reactor Pressure

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: Safety/relief valves: Plant-Specific

Proposed Question: #50

The plant is operating at 100% power when MSIVs spuriously close and the Reactor fails to scram.

Which one of the following describes the Reactor pressure at which the first ERV(s) will begin to open and the Reactor pressure at which the last ERV(s) will begin to open?

The first ERV(s) will begin to open at...

A. 1090 psig and the last ERV(s) will begin to open at 1100 psig.

B. 1090 psig and the last ERV(s) will begin to open at 1135 psig.

C. 1218 psig and the last ERV(s) will begin to open at 1228 psig.

D. 1218 psig and the last ERV(s) will begin to open at 1254 psig.

Proposed Answer: A

Explanation: The first two ERVs begin to open at a Reactor pressure of 1090 psig and the last two ERVs begin to open at a Reactor pressure of 1100 psig.

B. Plausible – ERVs do begin to open at 1090 psig, but the last ERVs begin to open at 1100 psig. 1135 psig is the Reactor pressure setpoint for ATWS-ARI/RPT.
C. Plausible – 1218 psig is the Reactor pressure setpoint for the first Reactor head safety valves. 1228 psig is based on the 1218 psig value plus the actual range for ERV pressure setpoints (10 psig).
D. Plausible – 1218 psig is the Reactor pressure setpoint for the first Reactor head safety valves. 1254 psig is the Reactor pressure setpoint for the last Reactor head safety valve.

Technical Reference(s): N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-7

Question Source: Bank – JAF April 2014 NRC #62

Question History: JAF April 2014 NRC #62

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 #	295021 AA2.06
	Importance Rating	3.2

Loss of Shutdown Cooling

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

Proposed Question: #51

The plant is shutdown with the following:

- Shutdown Cooling (SDC) isolated due to an erroneous isolation signal.
- The problem has now been fixed.
- Reactor coolant temperature is 360°F and slowly rising.
- Reactor pressure is 140 psig and slowly rising.

Which one of the following describes the status of Reactor coolant temperature and Reactor pressure for restoring SDC to service?

A. Both the current Reactor coolant temperature and Reactor pressure are acceptable for restoring SDC to service.

B. Reactor coolant temperature is acceptable for restoring SDC to service, but Reactor pressure must be lowered.

C. Reactor pressure is acceptable for restoring SDC to service, but Reactor coolant temperature must be lowered.

D. Both Reactor coolant temperature and Reactor pressure must be lowered to restore SDC to service.

Proposed Answer: D

Explanation: In order to restore SDC after an isolation, the SDC isolation valves must be reopened and the SDC pump(s) must be restarted. SDC isolation valves can only be opened if Reactor pressure is <120 psig, therefore Reactor pressure must be lowered. SDC pumps can only be started if Reactor coolant temperature is <350°F, therefore Reactor coolant temperature must be lowered.

A. Plausible – SDC was originally in service, however Reactor coolant temperature and pressure have risen enough since the isolation that they must be lowered to meet interlocks for re-establishing SDC.
B. Plausible – SDC was originally in service, however Reactor coolant temperature has risen enough since the isolation that it must be lowered to meet interlocks for re-establishing SDC.
C. Plausible – SDC was originally in service, however Reactor pressure has risen enough since the isolation that it must be lowered to meet interlocks for re-establishing SDC.

Technical Reference(s): N1-OP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

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

Partial or Complete Loss of AC Power

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

Proposed Question: #52

A failure to scram has occurred with the following:

- RPS Bus 11 is de-energized.
- Powerboard 17B is de-energized.
- Boron injection is required.

Which one of the following describes the ability to inject boron using the Liquid Poison (LP) system?

A. Both LP pumps 11 and 12 are available for boron injection.

B. LP pump 11 is available for boron injection, but pump 12 is NOT available.

C. LP pump 12 is available for boron injection, but pump 11 is NOT available.

D. Neither LP pump is available for boron injection.

Proposed Answer: B

Explanation: LP pump 12 is powered from PB 17B. Since PB 17B is de-energized, LP pump 12 is unavailable for boron injection. LP pump 11 is powered from PB 16B. Explosive valve 11 is powered from RPS Bus 11. Since RPS Bus 11 is de-energized, explosive valve 11 is unavailable. However, when LP pump 11 is started from the control room, both explosive valve 11 and 12 receive signals to open. Explosive valve is still available with power from RPS Bus 12. Therefore, LP pump 11 is still available for injection even with explosive valve 11 unavailable.

A. Plausible – Loss of just RPS Bus 11 leaves both LP pumps available for injection, but loss of PB 17B makes LP pump 12 unavailable.
C. Plausible – Loss of PB 17B makes LP pump 12 unavailable, not LP pump 11. Also, loss of RPS Bus 11 makes explosive valve 11 unavailable, but not LP pump 11.
D. Plausible – Even though explosive valve 11 and LP pump 12 are unavailable, LP pump 11 can still inject through explosive valve 12.

Technical Reference(s): N1-OP-12, C-18019-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO-4

Question Source: Modified Bank – 2009 Audit #4

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AA2.16
	Importance Rating	3.0

Plant Fire On-site

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Vital equipment and control systems to be maintained and operated during a fire

Proposed Question: #53

The plant has experienced a significant fire with the following:

- The Reactor has been scrammed.
- A Reactor cooldown has been performed.
- Reactor pressure is 50 psig and stable.
- Shutdown Cooling is unavailable due to the fire.
- Continued Reactor cooldown is desired.

Which one of the following describes the Alternate Shutdown Cooling lineup to be used to continue the Reactor cooldown, in accordance with N1-SOP-21.1, Fire in Plant?

A. Initiate both Emergency Condensers while injecting to maintain Reactor water level 53 to 95 inches.

B. Raise Reactor water level to the Main Steam Lines to circulate water to the Torus through ERVs.

C. Maximize Reactor Water Cleanup reject flow while injecting to maintain Reactor water level 53 to 95 inches.

D. Raise Reactor water level to the Main Steam Lines to circulate water to the Main Condenser through Turbine Bypass Valves.

Proposed Answer: ~~B~~ A and C correct, per revised key TFish 4/14/15

Explanation: N1-SOP-21.1 contains specific guidance on how to establish an Alternate Shutdown Cooling if normal Shutdown Cooling is unavailable and Reactor pressure is too low to further cooldown by steaming through ERVs (approximately 50 psig). This Alternate Shutdown Cooling lineup isolates the Main Steam lines and Emergency Condenser steam lines, then raises Reactor water level to the Main Steam lines to inject through the ERVs into the Torus. Torus Cooling is placed in service to remove decay heat.

A. Plausible – With 50 psig of steam pressure left in the Reactor, initiating Emergency Condensers would provide some decay heat removal, however this is not the Alternate Shutdown Cooling lineup called for by N1-SOP-21.1.
C. Plausible – RWCU reject with injection to the Reactor does provide a flow path that would remove some decay heat, however this is not the Alternate Shutdown Cooling lineup called for by N1-SOP-21.1.
D. Plausible – The Alternate Shutdown Cooling method in N1-SOP-21.1 does required raising Reactor water level to the Main Steam lines, but to circulate water to the Torus, not the Main Condenser.

Technical Reference(s): N1-SOP-21.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP211C01 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 #	295026 2.2.37
	Importance Rating	3.6

Suppression Pool High Water Temperature

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

Proposed Question: #54

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

Which one of the following Torus water temperatures is the ***lowest*** applicable limit in Technical Specification 3.3.2, Pressure Suppression System Pressure and Suppression Chamber Water Temperature and Level?

A. 80°F

B. 85°F

C. 95°F

D. 110°F

Proposed Answer: B

Explanation: Technical Specification (TS) 3.3.2 limits Torus water temperature to ≤85°F during normal power operation.

A. Plausible – 80°F is the setpoint for annunciators F1-2-8(F4-2-1), CORE LEVEL TORUS TEMP. MONITOR SYS 11(12) TROUBLE, but the lowest limit in TS 3.3.2 is 85°F.
C. Plausible – 95°F is the limit for Torus water temperature when testing is in progress that adds heat to the Torus per TS 3.3.2.
D. Plausible – 110°F is an applicable limit in TS 3.3.2, but it is not the lowest limit.

Technical Reference(s): Technical Specification 3.3.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 2.2.42
	Importance Rating	3.9

Generator Voltage and Electric Grid Disturbances

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

Proposed Question: #55

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

- Annunciator A8-1-3, 115 KV BUS LOW VOLTAGE, alarms.
- N1-SOP-33A.3, Major 115 KV Grid Disturbances, is entered.
- 115 KV Bus voltage is 108 KV and slowly lowering on all three phases.
- Power Control reports that the Load Flow Computer is NOT available.

Which one of the following identifies the status of HPCI, Line 1, and Line 4 operability, in accordance with N1-SOP-33A.3?

A. HPCI, Line 1, and Line 4 remain operable.

B. HPCI is inoperable. Line 1 and Line 4 are operable.

C. HPCI is operable. Line 1 and Line 4 are inoperable.

D. HPCI, Line 1, and Line 4 are all inoperable.

Proposed Answer: D

Explanation: HPCI requires 115KV line voltage to be at least 109.3KV to be operable. With the Load Flow Computer unavailable and 115KV line voltage less than 114KV, Lines 1 and 4 must be declared inoperable also.

A. Plausible – Voltage is still present on the offsite power lines, which supply the HPCI pumps. However, voltage is too low to consider HPCI, Line 1, or Line 4 operable.
B. Plausible – HPCI is inoperable. Voltage is still present on the offsite power lines, however it is too low to consider them operable.
C. Plausible – Lines 1 and 4 are inoperable. Even though voltage is still available to the HPCI pumps, it is too low to consider HPCI operable.

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

Proposed references to be provided to applicants during examination: N1-SOP-33A.3 pages 5-6

Learning Objective: 1101-SOP33A3C01 EO-2

Question Source: Bank – 2013 Audit #20

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 #	295016 2.1.20
	Importance Rating	4.6

Control Room Abandonment

Ability to interpret and execute procedure steps.

Proposed Question: #56

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

- A fire develops in the Control Room.
- Evacuation of the Control Room is required before a manual Reactor scram can be inserted.

Which one of the following describes the method for accomplishing the Reactor scram and confirming control rod insertion from outside the Control Room, in accordance with N1-SOP-21.2, Control Room Evacuation?

	Method for Reactor Scram	Method for Confirming Control Rod Insertion
A.	Place RPS MG Set 131 and 141 switches in TRIP.	Observe computer point F187, ALL RODS IN, is in alarm from the TSC.
B.	Place RPS MG Set 131 and 141 switches in TRIP.	Observe the white CONTROL RODS IN light is lit on the Remote Shutdown Panel.
C.	Isolate Instrument Air to CRD and vent the scram air header.	Observe computer point F187, ALL RODS IN, is in alarm from the TSC.
D.	Isolate Instrument Air to CRD and vent the scram air header.	Observe the white CONTROL RODS IN light is lit on the Remote Shutdown Panel.

Proposed Answer: B

Explanation: N1-SOP-21.2 directs the Reactor scram be accomplished by tripping the RPS MG Sets from the RSPs. N1-SOP-21.2 directs confirmation of the scram by observing the CONTROL RODS IN light is lit on RSP 11 or 12.

A. Plausible – This computer point is available in the TSC, however N1-SOP-21.2 directs use of the RSP indication.
C. Plausible – This is a valid action to scram the Reactor during an ATWS per N1-EOP-3.1, Alternate Rod Insertion, but not the method directed by N1-SOP-21.2. This computer point is available in the TSC, however N1-SOP-21.2 directs use of the RSP indication.
D. Plausible – This is a valid action to scram the Reactor during an ATWS per N1-EOP-3.1, Alternate Rod Insertion, but not the method directed by N1-SOP-21.2.

Technical Reference(s): N1-SOP-21.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP212C01 EO-2

Question Source: Bank – 2013 Audit #6

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 EA1.01
	Importance Rating	3.6

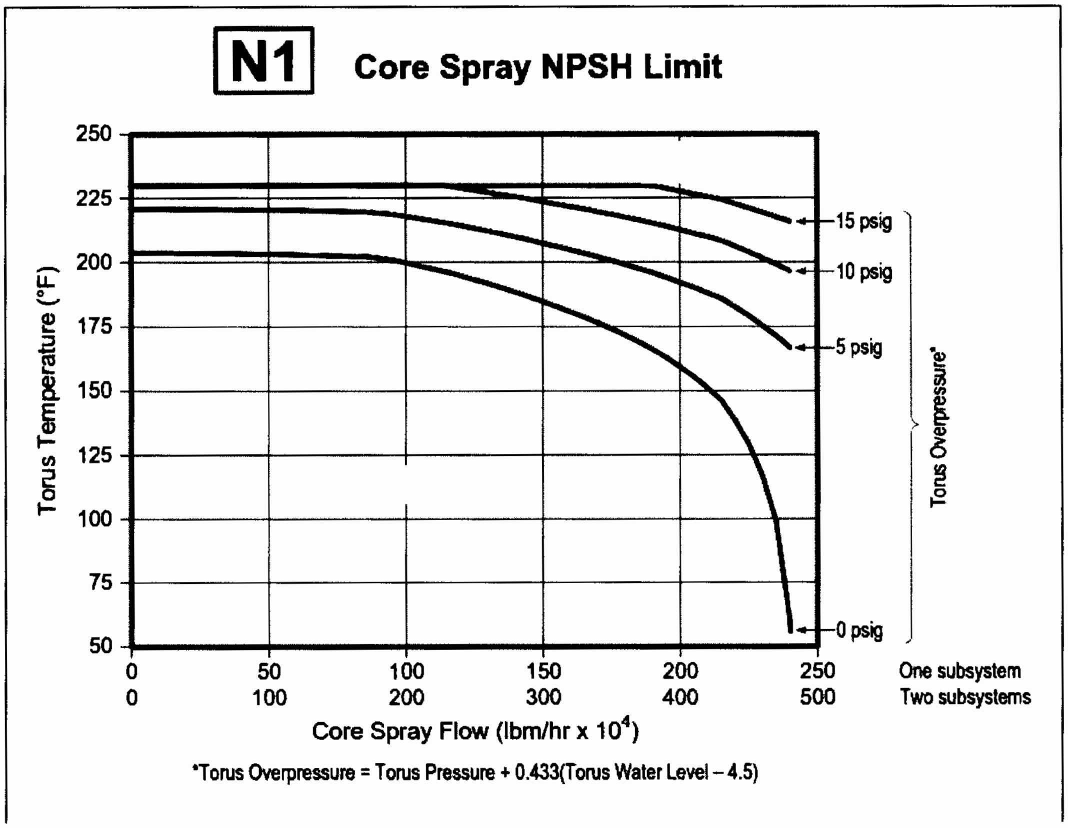
Low Suppression Pool Water Level

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

Proposed Question: #57

A loss of coolant accident has resulted in the following:

- The Reactor has been depressurized using ECs and ERVs.
- Core Spray is injecting and maintaining Reactor water level.
- Containment Sprays have been utilized to lower Containment pressure.
- Torus water temperature is 200°F and stable.
- Torus water level is 8.5 feet and stable.
- Torus pressure is 7 psig and slowly rising.
- Drywell pressure is 9 psig and slowly rising.
- 40-12, Core Spray Discharge IV 11 (Outside), has failed closed.
- Core Spray pump 121 has tripped due to an electrical fault.



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

A. 175

B. 230

C. 350

D. 460

Proposed Answer: A

Explanation: With the given Core Spray failures, only one subsystem (Core Spray pump 122) is available for injection. Torus overpressure is calculated as 7 psig + 0.433 x (8.5 feet – 4.5 feet) = 8.7 psig. Since this is below the 10 psig overpressure curve, the 5 psig overpressure curve is the highest that can be used. At a Torus water temperature of 200°F and using the 5 psig overpressure curve, the highest Core Spray flow with one subsystem is approximately 175 x 10^4 lbm/hr.

B. Plausible – With the given values, the 5 psig overpressure curve must be used. 230 x 10^4 lbm/hr is the flow limit using the 10 psig overpressure curve, which would be used if Torus water level were in the normal band.
C. Plausible – 350 x 10^4 lbm/hr is the flow limit using the 5 psig overpressure curve with two Core Spray subsystems.
D. Plausible – 460 x 10^4 lbm/hr is the flow limit using the 10 psig overpressure curve with two Core Spray subsystems.

Technical Reference(s): N1-EOP-2, C-18007-C

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Modified Bank – 2009 NRC #13

Question History: 2009 NRC #13

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295023 2.4.6
Importance Rating	3.7

Refueling Accidents

Knowledge of EOP mitigation strategies.

Proposed Question: #58

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

- Fuel moves are in progress in the Spent Fuel Pool.
- An irradiated fuel bundle has been dropped in the Spent Fuel Pool.
- Annunciator H1-4-8, Area Radiation Monitors, is in alarm.
- The following Area Radiation Monitors (ARMs) indicate upscale:
  - WEST END OF SHIELD WALL, RB 340 (#18)
  - RX BLDG – EAST WALL, EL. 340 (#25)
- An RP Technician reports the following radiation levels on the Refuel Floor:
  - RB 340' West: 9 R/hr and rising slowly
  - RB 340' East: 11 R/hr and rising slowly.

Note: N1-EOP-5, Secondary Containment Control, Detail R is provided on the following page.

Which one of the following describes the applicability of N1-EOP-5 for this event?

N1-EOP-5 entry is …

A. NOT required because this event does NOT involve a primary system discharge.

B. required. N1-EOP-5 requires a normal Reactor shutdown, but NOT a Reactor scram.

C. required. N1-EOP-5 requires a Reactor scram, but NOT an RPV Blowdown.

D. required. N1-EOP-5 requires a Reactor scram and an RPV Blowdown.

R

Area Radiation Level Alarm Setpoints

(Annunciator H1-4-8)

Area Radiation Level	Setpoint
NEW FUEL ROOM, RB 318 (#2)	10 mr/hr
INNER TIP ROOM RB 249 (#17)	600 mr/hr
WEST END OF SHIELD WALL, RB 340 (#18)	10 mr/hr
RX BLDG – NE CORNER, EL 198 (#19)	10 mr/hr
CLOSED LOOP COOLING AREA, RB 298 (#20)	20 mr/hr
CLEANUP PUMP AREA, RB 261 (#21)	20 mr/hr
RX BLDG – NE, EL. 281 (#22)	5 mr/hr
CRD ACCUMULATOR AREA, RB 237 (#23)	20 mr/hr
RX BLDG – EAST WALL, EL. 340 (#25)	15 mr/hr
RX BLDG – NW, EL. 318 (#27)	20 mr/hr
NORTH INSTR ROOM, RB 237 (#28)	20 mr/hr
REFUEL BRIDGE (HIGH RANGE) (PROCESS MON)	$\leq$1000 mr/hr

Proposed Answer: B

Explanation: N1-EOP-5 entry is required based on area radiation levels above alarm setpoints. The fact that this event is cause by a dropped fuel bundle and not a primary system discharge effects the choice of mitigation strategies within N1-EOP-5, but does not get rid of the need for procedure entry. Since this is NOT a primary system discharge causing the high radiation levels, the leg of N1-EOP-5 that requires a Reactor scram and RPV Blowdown is not applicable. Each side of the Refuel Floor (east and west) counts as a separate General Area. With both sides of the Refuel Floor above the Maximum Safe Radiation value of 8 R/hr, N1-EOP-5 requires a normal Reactor shutdown.

<u>Note:</u> This question asks knowledge required at the Reactor Operator license level, as the information to answer the question is EOP entry conditions and general EOP mitigation strategies, which ROs should be familiar with. Therefore this question would not be considered SRO level.

KA Match Justification: The question meets the K/A by testing a Refueling Accident situation (dropped fuel bundle) and requiring the candidate to determine the appropriate EOP mitigation strategy based on related parameters.

A. Plausible – Because the event is not caused by a primary system discharge, the choice of mitigation strategies within N1-EOP-5 changes, but entry into N1-EOP-5 is still required.
C. Plausible – If this event was caused by a primary system discharge, then a Reactor scram would be required by N1-EOP-5 with even one General Area approaching or exceeding the Maximum Safe Radiation value of 8 R/hr. An RPV Blowdown would be contingent on two General Areas exceeding the Maximum Safe Radiation value.
D. Plausible – If this event was caused by a primary system discharge, then a Reactor scram and RPV Blowdown would be required by N1-EOP-5 because two General Areas have exceeded the Maximum Safe Radiation value of 8 R/hr.

Technical Reference(s): N1-EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295015 AK1.01
	Importance Rating	3.6

Incomplete SCRAM

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

Proposed Question: #59

A failure to scram has occurred with the following:

- N1-EOP-3, Failure to Scram, is being executed.
- Reactor pressure is being controlled 800-1000 psig using Turbine Bypass Valves.
- Reactor water level is being controlled -84 to -41 inches using Feedwater.
- Liquid Poison injection has been initiated.
- Control rod insertion is in progress.
- All control rods have been fully inserted except:
    - Control rod 26-11 is at position 48.
    - Control rod 30-11 is at position 48.
- Then the following sequence of events occurs:

Time (minutes)	Event
00	Liquid Poison has injected 700 gallons of boron solution.
12	Liquid Poison has injected 1050 gallons of boron solution.
24	Control rod 26-11 is fully inserted.
36	Control rod 30-11 is fully inserted.

In accordance with N1-EOP-3, which one of the following is the ***earliest*** time at which sufficient negative reactivity has been inserted to permit exiting N1-EOP-3?

A. 00 minutes

B. 12 minutes

C. 24 minutes

D. 36 minutes.

Proposed Answer: C

Explanation: N1-EOP-3 entry is required in this event because, with at least two control rods withdrawn beyond position 04, neither the Maximum Subcritical Bank Withdrawal Position nor the Technical Specification Shutdown Margin are met. In order to meet the conditional step in N1-EOP-3 that directs exiting N1-EOP-3, either all control rods must be inserted to at least position 04 or the Reactor must be guaranteed to stay shutdown without boron. Neither injecting the Hot Shutdown Boron volume (700 gallons) nor the Cold Shutdown Boron volume (1050 psig) allows exiting N1-EOP-3. Once control rod 26-11 is fully inserted and only control rod 30-11 remains withdrawn, the Reactor will stay shutdown under all conditions without boron based on the Technical Specification Shutdown Margin assumptions. Therefore, at time 24 minutes, N1-EOP-3 allows securing boron injection and exiting N1-EOP-3.

A. Plausible – At time 00 minutes, the Hot Shutdown Boron volume has been injected and the Reactor is still above 212°F. This allows raising Reactor water level back to the normal band if it has been lowered, but does not allow exiting N1-EOP-3.
B. Plausible – At time 12 minutes, the Cold Shutdown Boron volume has been injected. This allows performing a Reactor cooldown, but does not allow exiting N1-EOP-3.
D. Plausible – At time 36 minutes, all control rods are inserted and the Maximum Subcritical Bank Withdrawal Position is satisfied. This does allow N1-EOP-3, however exiting N1-EOP-3 was also allowed at the earlier time 24 minutes based on the Technical Specification Shutdown Margin assumptions.

Technical Reference(s): N1-EOP-3, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO-2

Question Source: Modified Bank – 2008 Audit #59

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 AK2.02
	Importance Rating	3.3

High Drywell Pressure

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Drywell/suppression chamber differential pressure: Mark-I&II

Proposed Question: #60

A loss of coolant accident has resulted in the following:

- Drywell (DW) pressure is 4.0 psig and slowly lowering.
- Torus pressure is 4.8 psig and slowly lowering.
- Reactor Building (RB) pressure is -0.25" H20 and steady.

Which one of the following identifies the status of the Torus to DW Vacuum Breakers and the RB to Torus Vacuum Breakers?

	Torus to DW Vacuum Breakers	RB to Torus Vacuum Breakers
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: B

Explanation: Torus to DW vacuum breakers are open because Torus pressure is greater than DW pressure (by more than 0.5 psid). RB to Torus vacuum breakers are NOT open because Torus pressure is higher than RB pressure.

A. Plausible – A significant differential pressure exists between the Torus and the RB, however it exists in the wrong direction to open the RB to Torus vacuum breakers.
C. Plausible – The Torus to DW differential pressure is only slightly above the value that opens the Torus to DW vacuum breakers. A significant differential pressure exists between the Torus and the RB, however it exists in the wrong direction to open the RB to Torus vacuum breakers.
D. Plausible – The Torus to DW differential pressure is only slightly above the value that opens the Torus to DW vacuum breakers.

Technical Reference(s): N1-OP-9 P&L 10, ARP K1-4-6, TS 3.3.6 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-6

Question Source: Bank – 2010 NRC #35

Question History:

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 #	295008 AK3.04
	Importance Rating	3.3

High Reactor Water Level

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: Reactor feed pump trip: Plant-Specific

Proposed Question: #61

Which one of the following describes the design basis for the Feedwater pump 13 trip on high Reactor water level?

A. Prevent flooding of Main Steam lines.

B. Prevent prolonged Feedwater pump runout.

C. Preserve Emergency Condenser operability.

D. Preserve Hotwell water inventory for HPCI injection.

Proposed Answer: A

Explanation: Feedwater pumps trip on a high Reactor water level of 95". This trip was added as a plant modification to address concerns on flooding of the Main Steam lines.

B. Plausible – High Reactor water level may be caused by Feedwater pump runout. Feedwater pump runout is undesirable, but not the design basis for this Feedwater pump trip.
C. Plausible – The Emergency Condensers are considered inoperable when Reactor water level goes above 95", however this is not the design basis for the Feedwater pump trip at this same Reactor water level.
D. Plausible – High Reactor water level caused by Feedwater pump injection does use water from the hotwell, which reduces available hotwell inventory for future HPCI injection. Tech Spec 3.1.8 requires at least 75,000 gallons of water in the Hotwell for HPCI operability. However, this is not the design basis for the high Reactor water level Feedwater pump trip.

Technical Reference(s): SDBD-402 Section 7.1.11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EA1.01
	Importance Rating	3.4

High Suppression Pool Water Level

Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: HPCI: Plant-Specific

Proposed Question: #62

A loss of coolant accident has resulted in the following:

- Reactor water level is -110 inches and lowering.
- Reactor pressure is 320 psig and lowering.
- Torus water level is 13.4 feet and rising.
- Both Core Spray and HPCI are available for injection.

Which one of the following describes the allowable use of Core Spray and HPCI for injection, in accordance with N1-EOP-2, RPV Control, and N1-EOP-4, Primary Containment Control?

A. Injection with Core Spray is allowed. Injection with HPCI is NOT allowed.

B. Injection with HPCI is allowed. Injection with Core Spray is NOT allowed.

C. Injection with both Core Spray and HPCI is allowed. HPCI is the preferred injection source under these conditions.

D. Injection with both Core Spray and HPCI is allowed. Core Spray is the preferred injection source under these conditions.

Proposed Answer: D

Explanation: With Torus water level approaching 13.5 feet, N1-EOP-4 step TL-3 will direct stopping injection into the Reactor from sources outside the Primary Containment, except if that injection is needed for adequate core cooling or to shutdown the Reactor. Core Spray injects water from inside the Primary Containment (Torus). HPCI injects water from outside the Primary Containment (Hotwell). With Reactor water level below -109 inches, adequate core cooling is not assured. Therefore, both Core Spray and HPCI may be used for injection. However, Core Spray is the preferred injection source and HPCI use should be discontinued if Core Spray is able to restore Reactor water level.

A. Plausible – HPCI injection is undesirable because it will further raise Torus water level, but it is allowed since adequate core cooling is not assured.
B. Plausible – HPCI injection is allowed. Core Spray injection would be unavailable if Reactor pressure were still above 365 psig. Also, Core Spray injection is not allowed under certain conditions (ATWS).
C. Plausible – HPCI is normally the preferred injection source due to the broader range of Reactor pressure at which it can inject and the finer control that it allows. However, with Torus water level so high, HPCI injection should be minimized and secured once Reactor water level is restored.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	2
K/A #	295036 EA2.01
Importance Rating	3.0

Secondary Containment High Sump/Area Water Level

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Operability of components within the affected area

Proposed Question: #63

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

- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, alarms.
- Computer point B129, RBFDT 13 (SW) LVL HIGH, alarms.
- Water level in the area has been confirmed to be right at the alarm setpoint and stable.

Which one of the following describes the equipment located in this area and the effect on the operability of this equipment based on this alarm?

This area contains...

A. Core Spray pumps 111 and 112. These pumps are inoperable because their motors are submerged.

B. Core Spray pumps 111 and 112. These pumps remain operable because their motors are NOT submerged.

C. Containment Spray pumps 111 and 121. These pumps are inoperable because their motors are submerged.

D. Containment Spray pumps 111 and 121. These pumps remain operable because their motors are NOT submerged.

Proposed Answer: B

Explanation: Annunciator H2-2-1 has multiple inputs for various Reactor Building floor drain tank high levels and high area water levels. Computer point B129 indicates that the floor drain tank in the southwest corner room has a high water level condition (approximately 1" below floor level). This room contains Core Spray pumps 111 and 112. Based on the location of the pump motors, these pumps remain operable until area water level reaches 5 feet above floor level, which would be indicated by another computer point, F191 SW RB CORNER RM WTR LEVEL HIGH, which is also an input to annunciator H2-2-1. Therefore, based on the current alarm, these Core Spray pumps remain operable until area water level rises approximately 5 feet.

A. Plausible – Annunciator H2-2-1 combined with computer point F191 would indicate that the Core Spray pumps in this area are inoperable based on water level.
C. Plausible – Containment Spray pumps 111 and 121 are located in the northwest corner room, not the southwest corner room. Annunciator H2-2-1 combined with computer point F191 would indicate that the Core Spray pumps in this area are inoperable based on water level.
D. Plausible – Containment Spray pumps 111 and 121 are located in the northwest corner room, not the southwest corner room.

Technical Reference(s): ARP H2-2-1, NER-1M-095, N1-OP-2, N1-OP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: Modified Bank - 2009 NRC #83

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295032 2.4.1
	Importance Rating	4.6

High Secondary Containment Area Temperature

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: #64

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

- Annunciator K1-1-1, RX BLDG AREA TEMP HIGH, is in alarm.
- Computer point H308, EM COND TK AREA TEMP, is in alarm high at 180°F and slowly rising.
- Annunciator H1-4-8, Area Radiation Monitors, is in alarm.
- Computer point F308, EC SHLD WALL, is in alarm high at 25 mr/hr and slowly rising.
- Annunciator L1-3-4, REACT BLDG/ATM DIFF PRESS, is in alarm.
- Reactor Building differential pressure indicates -0.05" H_2O and stable.
- Annunciator L1-4-3, REACT BLDG VENT RAD MONITOR OFF NORMAL, is in alarm.
- Reactor Building ventilation exhaust radiation monitors indicates 10 mr/hr and slowly rising.

__Note:__ N1-EOP-5, Secondary Containment Control, Details R and T are provided on the following pages.

Which one of the following describes the total number of entry conditions that are met for N1-EOP-5?

A. One

B. Two

C. Three

D. Four

R

Area Radiation Level Alarm Setpoints

(Annunciator H1-4-8)

Area Radiation Level	Setpoint
NEW FUEL ROOM, RB 318 (#2)	10 mr/hr
INNER TIP ROOM RB 249 (#17)	600 mr/hr
WEST END OF SHIELD WALL, RB 340 (#18)	10 mr/hr
RX BLDG – NE CORNER, EL 198 (#19)	10 mr/hr
CLOSED LOOP COOLING AREA, RB 298 (#20)	20 mr/hr
CLEANUP PUMP AREA, RB 261 (#21)	20 mr/hr
RX BLDG – NE, EL. 281 (#22)	5 mr/hr
CRD ACCUMULATOR AREA, RB 237 (#23)	20 mr/hr
RX BLDG – EAST WALL, EL. 340 (#25)	15 mr/hr
RX BLDG – NW, EL. 318 (#27)	20 mr/hr
NORTH INSTR ROOM, RB 237 (#28)	20 mr/hr
REFUEL BRIDGE (HIGH RANGE) (PROCESS MON)	$\leq$1000 mr/hr

T Area Temperature Alarm Setpoints

Area Temperature	Setpoint	Annunciator Location
RWCU Pump Area		
Pumps	≤190°F	K3-3-4
VALVES 33-24 & 33-31	≤190°F	K3-3-4
RWCU Heat Exchanger Room		
ISOL VALVE 33-04	≤190°F	K3-3-4
HEAT EXCH N	≤190°F	K3-3-4
RELIEF VALVE 33-26	≤190°F	K3-3-4
PC VALVES NDLL & 37	≤190°F	K3-3-4
HEAT EXCH S	≤190°F	K3-3-4
RWCU Aux Cleanup Pump Room		
AUX CLEANUP PUMP RM 198	≤190°F	K3-3-4
AUX CLEANUP PUMP RM 199	≤190°F	K3-3-4
RWCU Pump Surge Tank		
VALVES 33-23 & 25 BLOCK VALVES 33-15, 16, & 17	≤190°F	K3-3-4
North Instrument Room		
202-130	110°F	K1-1-1
202-131	110°F	K1-1-1
EC Steam Line Isolation Valve Area		
ISOL VALVE 39-07 & 09	210°F	K1-4-3
ISOL VALVE 39-08 & 10	210°F	K1-4-5
Shutdown Cooling Area		
PUMP 11	≤170°F	K3-3-2
PUMP 12	≤170°F	K3-3-2
PUMP 13	≤170°F	K3-3-2
ISOL VALVE 38-02	≤170°F	K3-3-2
RB 340 Emergency Condenser Area		
EC #11 (202-118)	160°F	K1-1-1
EC #11 (202-119)	160°F	K1-1-1
EC #12 (202-120)	160°F	K1-1-1
EC #12 (202-121)	160°F	K1-1-1
East Instrument Room		
202-126	125°F	K1-1-1
202-127	125°F	K1-1-1
EC Condensate Return Valve Area		
ISOL VALVE 39-05	174°F	K1-4-3
ISOL VALVE 39-06	168°F	K1-4-5
RB 318 North Hall Area		
202-122	140°F	K1-1-1
202-123	140°F	K1-1-1
202-124	140°F	K1-1-1
202-125	140°F	K1-1-1
HCU/Scram Header Area		
SDV (202-132)	118°F	K1-1-1
SDV (202-133)	118°F	K1-1-1
SDV-South (202-134)	118°F	K1-1-1
SDV-North (202-135)	118°F	K1-1-1
West Instrument Room		
202-128	125°F	K1-1-1
202-129	125°F	K1-1-1

Proposed Answer: C

Explanation: N1-EOP-5 contains the following entry conditions:

ENTRY CONDITIONS					
Any of the following conditions exist in the reactor building:					
Area temperature above any alarm setpoint (Detail T)	**Reactor building differential pressure** at or above 0 in.	**Area radiation** above any alarm setpoint (Detail R)	**Ventilation exhaust radiation** above 5 mR/hr (Annunciator (L1-4-3)	**Floor drain sump water level** above alarm setpoint (Annunciator H2-2-1)	**Area water level** above 0 in. (Detail W)

The given indications meet the entry conditions for area temperature, area radiation, and ventilation exhaust radiation. No indications are given for floor drain sump water level or area water level. Reactor Building differential pressure is at the alarm setpoint, but not at or above 0 inches. Therefore, three entry conditions are met for N1-EOP-5.

A. Plausible – While three parameters require N1-EOP-5 entry, they are each just slightly above the threshold requiring entry.
B. Plausible – While three parameters require N1-EOP-5 entry, they are each just slightly above the threshold requiring entry.
D. Plausible – Abnormal values are given for four N1-EOP-5 entry conditions, however the value for Reactor Building differential pressure has not degraded enough to require entry.

Technical Reference(s): N1-EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 AK1.02
	Importance Rating	3.6

Loss of CRD Pumps

Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS: Reactivity control

Proposed Question: #65

A plant startup is in progress with the following:

Time (hh:mm)	Event
09:28	• Preparations are in progress for starting the first Feedwater pump. • CRD pump 11 is in service. • CRD pump 12 is in standby.
09:30	• CRD pump 11 trips. • Annunciator F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, alarms.
09:32	• CRD pump 12 does NOT start when its control switch is placed to START.
09:34	• Annunciator F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, alarms. • Control Rod 10-19 accumulator light illuminates and accumulator pressure is reported as 900 psig and lowering.
09:36	• Control Rod 34-35 accumulator light illuminates and accumulator pressure is reported as 920 psig and lowering.

Which one of the following lists the time at which a manual Reactor scram is first required, in accordance with N1-SOP-5.1, Loss of Control Rod Drive?

A. 09:30

B. 09:34

C. 09:50

D. 09:54

Proposed Answer: B

Explanation: N1-SOP-5.1 contains the following override:

IF	THEN
• NO CRD pump is running, AND • Any accumulator alarm(s) is received, AND • Reactor pressure is **greater than** 900 psig,	Restart at least one CRD pump within 20 minutes AND insert at least one Control Rod at least one notch, OR **SCRAM** the Reactor per N1-SOP-1.
• NO CRD pump is running, AND • Any accumulator alarm(s) is received, AND • Reactor pressure is **less than** 900 psig,	**SCRAM** the Reactor per N1-SOP-1.

Since preparations are underway for starting the first Feedwater pump, Reactor pressure is between 300 and 350 psig. With Reactor pressure less than 900 psig, a Reactor scram is first required when no pumps are running and an accumulator alarm is received (time 09:34).

A. Plausible – At 09:30, two of the three conditional bullets (no CRD pump running, Reactor pressure < 900 psig) are met, but since the third conditional bullet (accumulator alarm) is NOT met, a Reactor scram is NOT yet required.
C. Plausible – This is time when no CRD pump was running with Reactor pressure < 900 psig plus the 20 minute wait time that applies at normal Reactor pressures. However, N1-SOP-5.1 first required a scram at the earlier time of 09:34.
D. Plausible – This is the time a scram would first be required if the Reactor was at normal pressure.

Technical Reference(s): N1-SOP-5.1

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank – 2009 NRC #25

Question History: 2009 NRC #25

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.28
	Importance Rating	4.1

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

Proposed Question: #66

Which one of the following describes an MSIV closure time limit and the associated basis for the stated limit in accordance with the FSAR?

	MSIV Closure Time Limit	Basis for MSIV Closure Time Limit
A.	3 Seconds	In conjunction with Main Steam Line flow limiters, ensures core coverage following a Main Steam Line rupture.
B.	3 Seconds	In conjunction with MSIV position scram, ensures ERV actuation is not required following Main Steam Line isolation at power.
C.	10 Seconds	In conjunction with Main Steam Line flow limiters, ensures core coverage following a Main Steam Line rupture.
D.	10 Seconds	In conjunction with MSIV position scram, ensures ERV actuation is not required following Main Steam Line isolation at power.

Proposed Answer: C

Explanation: Per the UFSAR Section 15 Accident Analysis for the Main Steam Line Break Outside the drywell the assumptions of the accident analysis show that the combination of the flow limiters and isolation valve closure limits when offset by feedwater flow addition limits the loss of water volume to maintain water level above the core. This maintains the core coverage safety limit.

A. Plausible – The 3 second time limit for valve closure is part of the Chapter 15 analysis to limit the pressure transient from a Main Steam Line Isolation Valve closure (with SCRAM) to prevent opening safety valves and maintaining the pressure transient within the capability of the ERV's. The three second isolation time is NOT used as part of the basis for limiting inventory loss and maintaining core coverage during a MSL break accident. This is a plausible distractor because both the isolation time and bases are individually correct but are not associated with one another.
B. Plausible – The 3 second time limit for valve closure is part of the Chapter 15 analysis to limit the pressure transient from a Main Steam Line Isolation Valve closure (with SCRAM) to prevent opening safety valves and maintaining the pressure transient within the capability of the ERV's. This transient is expected to cause ERV actuations and is incorrect because of the reference to no ERV actuations in the bases column. This is a plausible distractor because the isolation time and bases are correct except for the use of safety valve versus ERV.
D. Plausible – The 10 second isolation time is associated with the limitations of coolant loss and not the limitations of the pressure transient due to MSIV closure. This is a plausible distractor because both the isolation time and bases are individually correct but are not associated with one another.

Technical Reference(s): FSAR Section XV 3.5.1 & C.1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-1

Question Source: Bank

Question History: 2009 NRC 60

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

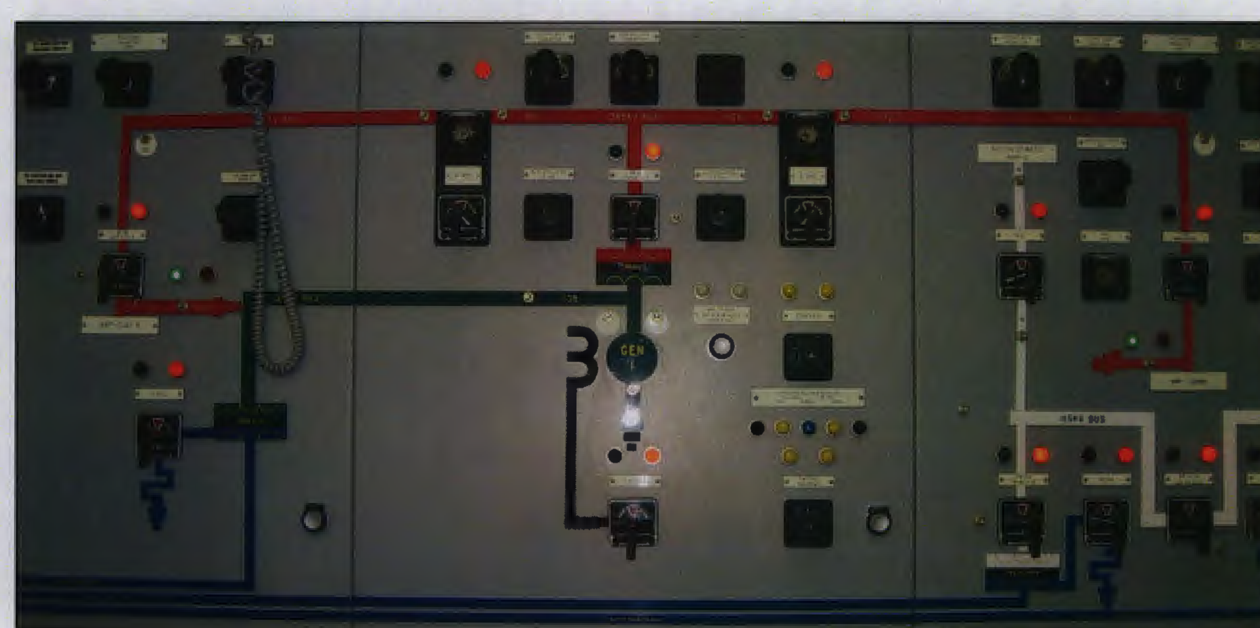
Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.31
	Importance Rating	4.6

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: #67

The plant was operating at 25% power when an automatic Reactor scram occurred. The following pictures show present plant conditions:



Which one of the following indicates the present status of the Main Turbine and Main Generator?

	Main Turbine	Main Generator
A.	Tripped	Connected to the grid
B.	Tripped	NOT connected to the grid
C.	NOT tripped	Connected to the grid
D.	NOT tripped	NOT connected to the grid

Proposed Answer: A

Explanation: The pictures show Turbine Stop Valves closed, indicating the Main Turbine has tripped. The pictures show MOD-18 closed, R-915 closed, and R-925 closed, indicating the Main Generator is still connected to the grid.

B. Plausible – Even though the Main Generator should be disconnected from the grid, the pictures indicate it is still connected.
C. Plausible – The Main Turbine is tripped as indicated by closed Turbine Stop Valves.
D. Plausible – The Main Turbine is tripped as indicated by closed Turbine Stop Valves. Even though the Main Generator should be disconnected from the grid, the pictures indicate it is still connected.

Technical Reference(s): N1-OP-31, N1-OP-32, N1-SOP-31.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245000-RBO-5

Question Source: Bank (105146)

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.35
	Importance Rating	3.6

Ability to determine Technical Specification Mode of Operation.

Proposed Question: #68

The plant is completing a refueling outage with the following:

- Reactor coolant temperature is 180°F.
- The Reactor Mode Switch is in STARTUP.
- The first control rod is being withdrawn to commence the Reactor startup.

Which one of the following is the current Reactor Operating Condition, in accordance with Technical Specifications?

A. Shutdown Condition - Cold

B. Shutdown Condition - Hot

C. Major Maintenance Condition

D. Power Operating Condition

Proposed Answer: D

Explanation: The plant is in the Power Operating Condition because the Reactor Mode Switch is in STARTUP and control rod withdrawal is in progress for a Reactor startup.

A. Plausible – Reactor coolant temperature is below 212°F and the Reactor is still subcritical, but since the Reactor Mode Switch is in STARTUP and control rod withdrawals are in progress, the plant is in the Power Operating Condition, not Cold Shutdown.
B. Plausible – Reactor coolant temperature is close to 212°F and the Reactor is still subcritical, but since the Reactor Mode Switch is in STARTUP and control rod withdrawals are in progress, the plant is in the Power Operating Condition, not Hot Shutdown.
C. Plausible – The plant is just exiting a refueling outage and the Reactor is still subcritical, but since there is fuel in the Reactor vessel and the Reactor Mode Switch is in STARTUP, the plant is in the Power Operating Condition, not the Major Maintenance Condition.

Technical Reference(s): Technical Specification 1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-101001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.12
	Importance Rating	3.7

Knowledge of surveillance procedures.

Proposed Question: #69

N1-ST-Q1C, Core Spray (CS) 112 Pump and Valve Operability Test, is to be performed.

Which one of the following is allowed without causing "pre-conditioning" in accordance with GAP-SAT-01, Surveillance Test Program, and CNG-MN-4.01-1008, Pre/Post-Maintenance Testing?

A. Manually stroking isolation valves to verify they are properly seated.

B. Filling and venting the system prior to starting Core Spray pump 112.

C. Adjusting the packing on isolation valves to ensure no stem binding occurs.

D. Starting and running Core Spray pump 112 to bring it up to operating temperature.

Proposed Answer: B and D, per revised key. TFish 4/10/15

Explanation: Per Attachment 2 of CNG-MN-4.01-1008, Pre/Post Maintenance Testing Requirements, filling and venting a system provided the venting operation has proper controls does not constitute preconditioning. Proper controls are provided directly in N1-ST-Q1C and similar surveillance tests.

A. Plausible – Isolation valve testing is part of this ST, however exercising the valves prior to the ST would constitute pre-conditioning.
C. Plausible – Isolation valve testing is part of this ST, however adjusting packing prior to the ST may affect stroke times and would constitute pre-conditioning.
D. Plausible – Core Spray pump 112 is operated as part of the ST, however it is required to be at normal standby temperature to avoid pre-conditioning.

Technical Reference(s): GAP-SAT-01, CNG-MN-4.01-1008 Attachment 2

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-MN-4.01-1008-CT-01

Question Source: Bank – NMP2 2013 Audit #69

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

A loss of coolant accident has resulted in the following:

- An RPV Blowdown has been performed due to low Reactor water level.
- Reactor pressure is 100 psig and slowly lowering.
- Reactor water level has been restored to a band of 53-95" using Core Spray.
- Liquid Poison is injecting, but is no longer needed for Reactor water level control.

Which one of the following describes the required control of Liquid Poison injection, in accordance with N1-EOP-2, RPV Control, and the EOP Bases?

A. Inject the entire contents of the Liquid Poison tank to minimize radioactive release.

B. Secure Liquid Poison to minimize corrosion of the Reactor vessel and internals.

C. Inject the entire contents of the Liquid Poison tank to provide additional negative reactivity.

D. Secure Liquid Poison to reduce the amount of boron that must be removed from the Reactor coolant.

Proposed Answer: A

Explanation: N1-EOP-2 step L-17 requires injecting the entire contents of the Liquid Poison tank after an RPV Blowdown based on low Reactor water level. This requirement is designed to assist in maintaining radioactive iodine in solution in the Torus water to prevent release of this radioactive iodine in the event of Torus atmosphere venting.

KA Match Justification: From the EOP Bases, "Design basis analyses credit Liquid Poison injection for limiting the radiological dose following loss of coolant accidents ... The injected sodium pentaborate is transported through the primary system break to the torus where it limits the late release of iodine by buffering the suppression pool pH." Question matches KA statement since the action to inject the entire contents of the liquid poison tank is meant to minimize the radiation release.

B. Plausible – Since Liquid Poison is not needed for Reactor water level control, it is reasonable to believe it should be secured. Boric acid is widely known as a corrosive agent.
C. Plausible – Continued Liquid Poison injection is required and it will provide additional negative reactivity, however this is not the basis for the requirement.
D. Plausible – Since Liquid Poison is not needed for Reactor water level control, it is reasonable to believe it should be secured to minimize the amount of boron that needs to be cleaned up from the Reactor coolant system.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: 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 #	2.3.13
	Importance Rating	3.4

Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: #71

A transient is in progress with the following:

- Operators are required to enter a radiologically posted area in order to manually close Primary Containment Isolation Valves.
- Radiation Protection (RP) has determined that the highest dose rate in the area is 750 mRem/hr.

Which one of the following describes the radiological posting requirement for the area and the need for the Operators to be continuously escorted by RP personnel, in accordance with GAP-RPP-08, Control of High, Locked High, and Very High Radiation Areas?

This area is required to be posted as a __(1)__. The Operators __(2)__ required to be continuously escorted by RP personnel while in this area.

	(1)	(2)
A.	Locked High Radiation Area	are
B.	Locked High Radiation Area	are NOT
C.	High Radiation Area	are
D.	High Radiation Area	are NOT

Proposed Answer: D

Explanation: An area that has general dose rates above 100 mRem/hr but less than 1000 mRem/hr is classified as a High Radiation area. An RP escort is required if dose rates have not already been determined. Since RP has already surveyed the area, the Operators do NOT need to have a continuous RP escort while in the area.

A. Plausible – The radiation levels are high in this area, but not above the 1000 mRem/hr threshold that requires posting as a Locked High Radiation Area. Continuous RP escort would be required if surveys had not already been performed.
B. Plausible – The radiation levels are high in this area, but not above the 1000 mRem/hr threshold that requires posting as a Locked High Radiation Area.
C. Plausible – Continuous RP escort would be required if surveys had not already been performed.

Technical Reference(s): GAP-RPP-08

Proposed references to be provided to applicants during examination: None

Learning Objective: GAP-RPP-08-CT-01

Question Source: Modified Bank – 2009 NRC #72

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.45
	Importance Rating	4.1

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

Proposed Question: #72

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

- You are performing a Control Room panel walk-down before taking the shift.
- You notice an annunciator marked with a red dot flagging tool.

Which one of the following describes the significance of this annunciator flagging, in accordance with CNG-OP-1.01-2003, Alarm Response and Control?

The annunciator...

A. is a nuisance alarm.

B. has one or more failed inputs.

C. is out of service due to safety tagging.

D. is expected to alarm due to a maintenance activity.

Proposed Answer: C

Explanation: A red annunciator flagging tool is used to identify an annunciator that it is out of service due to safety tagging.

A. Plausible – Nuisance alarms are flagged, but use a black color, not a red color.
B. Plausible – An alarm with a failed input is flagged, but use a yellow color, not a red color.
D. Plausible – An alarm expected due to maintenance activity is flagged, but use a black color, not a red color.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: S-ODP-OPS-0001-TO01

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 #	2.4.35
	Importance Rating	3.8

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

Proposed Question: #73

N1-SOP-21.2, Control Room Evacuation, is being performed.

Which one of the following describes an action performed per N1-SOP-21.2?

A. Air is isolated to ensure the outboard MSIVs are closed.

B. Breakers are opened to ensure the inboard MSIVs are closed.

C. Jumpers are installed to prevent MSIVs from closing on lo-lo Reactor water level.

D. Mechanical Vacuum pump is started to prevent MSIVs from closing on low vacuum.

Proposed Answer: A

Explanation: Per N1-SOP-21.2 Attachment 1, instrument air is isolated to the outboard MSIVs (01-03 and 01-04). This field action ensures these valves are closed and remain closed.

B. Plausible – Field action is taken to ensure the Main Steam Lines are isolated, however it is by isolating air to the outboard MSIVs, not removing power from the inboard MSIVs.
C. Plausible – Field action is taken to ensure proper control of the MSIVs, however the procedure requires MSIVs closed, not open. Jumpers are installed to defeat MSIV lo-lo Reactor water level closure in other emergency situations.
D. Plausible – Field action is taken to ensure proper control of the MSIVs, however the procedure requires MSIVs closed, not open. The Mechanical Vacuum pump is started to prevent MSIV closure on low vacuum in other emergency situations.

Technical Reference(s): N1-SOP-21.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.2C01

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 #	2.4.13
	Importance Rating	4.0

Knowledge of crew roles and responsibilities during EOP usage.

Proposed Question: #74

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

- The CRS directs a manual Reactor scram due to a pressure regulator malfunction.
- The Operator At-The-Controls (OATC) places the Reactor Mode Switch in SHUTDOWN and depresses the manual scram pushbuttons.
- No control rods insert.
- All RPS white lights remain lit.

Given the following possible actions:

(1) Manually initiating ARI
(2) Tripping Recirculation pumps
(3) Injecting Liquid Poison

Which one of the following identifies which of these actions the OATC may take without further direction from the CRS, in accordance with N1-SOP-1, Reactor Scram, and N1-EOP-3, Failure to Scram?

A. (1) only

B. (1) and (2) only

C. (1) and (3) only

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

Proposed Answer: A

Explanation: When the CRS directs a manual Reactor scram, N1-SOP-1 is entered by the OATC. The override actions of N1-SOP-1 allow the OATC to manually initiate ARI without further direction from the CRS. If the failure to scram persists, N1-EOP-3 contains guidance on tripping Recirculation pumps and injecting Liquid Poison. It is the CRS's role/responsibility to direct these actions.

B. Plausible – N1-EOP-3 does contain guidance for both of these actions. N1-SOP-1 contains specific guidance allowing an RO to initiate ARI as an immediate action without further SRO guidance. Additional SRO direction must be given before tripping Recirculation pumps.
C. Plausible – N1-EOP-3 does contain guidance for both of these actions. N1-SOP-1 contains specific guidance allowing an RO to initiate ARI as an immediate action without further SRO guidance. Additional SRO direction must be given before injecting Liquid Poison.
D. Plausible – N1-EOP-3 does contain guidance for all of these actions. N1-SOP-1 contains specific guidance allowing an RO to initiate ARI as an immediate action without further SRO guidance. Additional SRO direction must be given before tripping Recirculation pumps or injecting Liquid Poison.

Technical Reference(s): N1-SOP-1, N1-EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1C01 EO-2

Question Source: Bank – 2009 Audit #88

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 #	2.3.7
	Importance Rating	3.5

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Proposed Question: #75

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

- TIP runs are in progress
- The detector position indication on TIP machine 1 is not working properly
- TIP machine 1 has been given a withdraw signal
- An operator is being sent into the TIP Room to verify the detector position
- The TIP Room is currently posted as a Very High Radiation Area

Which one of the following describes the type of Radiation Work Permit (RWP) required and the permissions required for TIP room entry in accordance with GAP-RPP-08?

A. A Specific RWP with permission given by the General Supervisor - Rad Protection only.

B. A Specific RWP with permission given by the General Supervisor - Rad Protection, Shift Manager and Plant Manager.

C. An Emergency Response RWP with permission given by the Shift Manager only.

D. An Emergency Response RWP with permission given by the General Supervisor - Rad Protection, Shift Manager and Plant Manager.

Proposed Answer: B

Explanation: Areas with the potential for being Very High Radiation Areas include: TIP Rooms, Upper elevations of the drywell during fuel moves, Spent Fuel Pool during diving operations. A specific RWP is required for entry into Very High Radiation Area.

If the activity is within a Very High Radiation Area, obtain permission before any entry to the area from:

1. General Supervisor - Rad Protection
2. Shift Manager
3. Plant Manager

Note: Reactor Operators need to know this information prior to dispatching plant operators into the field.

A. Plausible – Permission must be given by the Shift Manager, General Supervisor - Rad Protection and Plant General Manager.
C. Plausible – A specific RWP is required for entry into Very High Radiation Area. An Emergency RWP would be used during an emergency; there is nothing in the stem to indicate an emergency exists. General Supervisor - Rad Protection and Plant Manager must also approve of the entry.
D. Plausible – A specific RWP is required for entry into Very High Radiation Area. An Emergency RWP would be used during an emergency; there is nothing in the stem to indicate an emergency exists.

Technical Reference(s): GAP-RPP-02, 3.1.1.b & 3.3.1
GAP-RPP-08, 3.4.3

Proposed references to be provided to applicants during examination: None

Learning Objective: GAP-RPP-02-CT-01

Question Source: Bank (98567)

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

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: #76

The plant is shutdown with the following:

- Core shuffle is in progress.
- A seismic event occurs.
- Significant damage occurs to the Spent Fuel Pool.
- Reactor Building ventilation isolates and RBEVS initiates based on the Refuel Bridge high range radiation monitor.
- Reactor Building differential pressure is maintained negative.
- The Shift Manager (SM) declares an ALERT based on 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 Part 1 Notification Fact Sheet:

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.		

Which one of the following identifies the letter to circle in this section in accordance with EP-Form-ALL31, NMP Station Notification Fact Sheet – Part 1?

A. A – No release

B. B – Release below federal limits (ODCM)

C. C – Release above federal limits (ODCM)

D. D – Unmonitored release requiring evaluation

Proposed Answer: C

Explanation: The Stack release rate that requires declaration of an Alert corresponds to approximately 200x the ODCM limit. Since the SM has made this emergency declaration, a release is in progress, it is monitored by the Stack radiation monitors, and it is above the ODCM limit. Therefore, "C - Release above federal limits (ODCM)" must be circled.

A. Plausible – The "No release" option would be circled if Stack release rates were at normal levels, since the release must be elevated above normal levels based on the emergency event.
B. Plausible – There is no available data confirming federal dose limits are being exceeded at or beyond the site boundary. However, the declaration of an Alert based on Stack radiation level means that ODCM limits are being exceeded.
D. Plausible – An unmonitored release would be in progress if RBEVS was unable to maintain Reactor Building differential pressure negative.

Technical Reference(s): EP-Form-ALL31, EAL Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP6C01 EO-3

Question Source: Bank – SYSID 109122

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 #	295004 AA2.04
	Importance Rating	3.3

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: System lineups

Proposed Question: #77

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

Time (hh:mm)	Event
09:00	Static Battery Charger 161 de-energizes due to a sustained electrical fault.
10:00	Motor Generator (MG) Set 167 is lined up to charge Battery 11.

Which one of the following describes the Technical Specification (TS) impact of these events?

At time 09:00, an LCO is entered which requires returning a battery charger to service within 24 hours or then...

A. initiating a normal orderly shutdown within one hour. At time 10:00, this LCO is exited.

B. reducing Reactor pressure to 110 psig or less within 10 hours. At time 10:00, this LCO is exited.

C. initiating a normal orderly shutdown within one hour. This LCO remains in effect after time 10:00.

D. reducing Reactor pressure to 110 psig or less within 10 hours. This LCO remains in effect after time 10:00.

Proposed Answer: D

Explanation: A loss of a Static Battery Charger is treated as a loss of the battery system. A battery charger must be returned to service within 24 hours in accordance with T.S. 3.6.3.h or take the action required by T.S. 3.1.5 which requires Reactor pressure be reduced to 110 psig or less within 10 hours. Although MG 167 can be used as a battery charger, it is not safety related and cannot be used to exit the LCO.

A. Plausible – Other sections of TS 3.6.3 require initiating a normal orderly shutdown within 1 hour, however failure to meet TS 3.6.3.h specifically requires entering TS 3.1.5.b, which requires Reactor pressure be reduced to 110 psig or less within 10 hours. Although MG 167 can be used as a battery charger, it is not safety related and cannot be used to exit the LCO.
B. Plausible – Although MG 167 can be used as a battery charger, it is not safety related and cannot be used to exit the LCO.
C. Plausible – Other sections of TS 3.6.3 require initiating a normal orderly shutdown within 1 hour, however failure to meet TS 3.6.3.h specifically requires entering TS 3.1.5.b, which requires Reactor pressure be reduced to 110 psig or less within 10 hours.

Technical Reference(s): N1-OP-47A P&L 9.0, Technical Specifications

Proposed references to be provided to applicants during examination: Technical Specifications 3.0.1, 3.1.5, and 3.6.3 (with 3.6.3.f removed)

Learning Objective: N1-263000-RBO-14

Question Source: Modified Bank – 2008 NRC #90

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 #	295005 AA2.06
	Importance Rating	2.7

Main Turbine Generator Trip

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

Proposed Question: #78

The plant is operating at 20% power during a startup with the following:

- Ten minutes ago, the Main Generator was synchronized to the grid and loaded to 60 MWe per N1-OP-32, Generator.
- No subsequent steps in the startup have been performed yet.
- Then, a Main Turbine trip occurs.
- The Reactor does NOT scram.
- Reactor power begins to slowly rise.

Which one of the following describes whether this is the expected Reactor response to the Main Turbine trip and a required action to be directed, in accordance with the SOPs and EOPs?

This is...

A. the expected Reactor response. Enter N1-SOP-31.1, Turbine Trip, and direct a manual Reactor scram.

B. the expected Reactor response. Enter N1-SOP-31.1, Turbine Trip, and direct a Turbine shutdown. A manual Reactor scram is NOT required.

C. NOT the expected Reactor response. Enter N1-SOP-1.5, Unplanned Power Change, to determine the cause of the power rise. A manual Reactor scram is NOT required.

D. NOT the expected Reactor response. Enter N1-EOP-2, RPV Control, and direct a manual Reactor scram. If the Reactor still does NOT scram, transition to N1-EOP-3, Failure to Scram.

Proposed Answer: B

Explanation: This is the proper Reactor response. The Reactor does NOT scram on the Main Turbine trip because Reactor power is below 35%. Reactor power rises due to the loss of Feedwater heating on the Main Turbine trip which causes feed water temperature to lower. N1-SOP-31.1 entry is required. The subsequent actions of N1-SOP-31.1 require directing a Turbine shutdown, but not a Reactor shutdown or Reactor scram.

A. Plausible – The Reactor power rise does diminish the margin to the point where a Turbine trip would cause a Reactor scram. A Reactor scram would only become <u>required</u> if appropriate action was not taken and Reactor power reached 35%.
C. Plausible – This is the expected response. N1-SOP-1.5 entry is required, although the cause of the power rise is already known to be loss of Feedwater heating.
D. Plausible – This is the expected response. N1-EOP-2 entry would only be required if the Reactor was supposed to scram.

Technical Reference(s): N1-SOP-1.5, N1-SOP-31.1, N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP31.1C01

Question Source: Bank – JAF April 2014 NRC #78

Question History: JAF April 2014 NRC #78

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295003 2.2.40
	Importance Rating	4.7

Partial or Complete Loss of AC Power

Ability to apply technical specifications for a system.

Proposed Question: #79

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

Date	Time (hh:mm)	Event
3/21/15	08:00	Core Spray pump 111 is declared inoperable.
3/21/15	12:00	Breaker R10 trips and CANNOT be re-closed.
3/21/15	16:00	Emergency Diesel Generator (EDG) 103 is declared inoperable.

Which one of the following describes the latest time that a plant shutdown can be initiated while still complying with Technical Specifications?

A. 3/21/15 at 13:00

B. 3/21/15 at 17:00

C. 3/22/15 at 17:00

D. 3/28/15 at 09:00

Proposed Answer: B

Explanation: When EDG 103 becomes inoperable at time 1600, Technical Specification 3.6.3.g is no longer met because Core Spray pump 111 (powered by EDG 102) is inoperable. With Technical Specification 3.6.3.g not met, the required action is to initiate a normal orderly shutdown within one hour (by 1700 on 3/21/15).

A. Plausible – This is one hour after Line 1 became inoperable. This would be correct if Technical Specifications included something analogous to TS 3.6.3.g for Lines 1 and 4, and not just EDGs 102 and 103. Additionally, at time 12:00, Core Spray pump 121 must be declared inoperable per TS 3.0.1, since it loses it's normal power source (Line 1) and has a redundant component inoperable (Core Spray pump 111). This requires entry into TS 3.1.4.c, since Core Spray pumps 111 and 121 are in separate Core Spray systems, but still allows 7 days to restore one of these pumps to operable.
C. Plausible – This is 25 hours after EDG 103 became inoperable. When EDG 103 became inoperable, both TS 3.6.3.c and TS 3.6.3.g were required to be entered. TS 3.6.3.c allows 24 hours to restore EDG 103, before then requiring a shutdown to be initiated within 1 hour (25 hours total). However, TS 3.6.3.g has a more restrictive shutdown requirement.
D. Plausible – This is the time when TS 3.1.4.c would require a shutdown to be initiated by for the original Core Spray inoperability, if it were not compounded by the electrical losses.

Technical Reference(s): Technical Specifications 3.1.4 and 3.6.3, N1-OP-2, C-18007-C

Proposed references to be provided to applicants during examination: Technical Specifications 3.0.1, 3.1.4, and 3.6.3 (with 3.6.3.f removed)

Learning Objective: N1-262001-RBO-14

Question Source: Bank – 2006 NRC SRO #17

Question History: 2006 NRC SRO #17

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 #	295006 2.4.31
	Importance Rating	4.1

SCRAM

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #80

The plant is operating at 100% power.

Which one of the following annunciators and indications would require a manual Reactor scram to be directed, in accordance with the associated Alarm Response Procedures?

A. A2-3-5, TURBINE SUPERVISORY SYSTEM, alarms with a Turbine bearing vibration at 7 mils.

B. A1-4-2, GENERATOR HYDROGEN SYSTEM, alarms with Generator temperature at 55°C.

C. A2-1-1, GENERATOR STAT. WATER OUTLET TEMP. HIGH, alarms with Stator Water Cooling outlet temperature at 80°C.

D. A2-2-1, GENERATOR STAT. WATER HIGH CONDUCTIVITY, alarms with Stator Water Cooling conductivity at 10 μmhos/cm.

Proposed Answer: D

Explanation: ARP A2-2-1 requires a Turbine trip if Stator Water Cooling conductivity exceeds 9.9 μmhos/cm. With Reactor power at 100%, a manual Reactor scram is required before the Turbine trip.

A. Plausible – A2-3-5 requires a Turbine trip (and associated Reactor scram) if a Turbine vibration exceeds 10 mils (for 15 minutes) or 12 mils (instantaneous).
B. Plausible – A1-4-2 requires lowering Generator loading, but not a Reactor scram, based on this high temperature condition.
C. Plausible – If Stator Water Cooling temperature degrades further, a Turbine runback could occur. If the Turbine runback were to not clear within the required time, a Turbine trip and Reactor scram would result.

Technical Reference(s): ARPs A1-4-2, A2-1-1, A2-2-1, A2-3-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-253000-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 #	1
	Group #	1
	K/A #	295026 2.1.31
	Importance Rating	4.3

Suppression Pool High Water Temperature

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: #81

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

- A manual Reactor scram was inserted.
- Drywell pressure is 10.0 psig and slowly rising.
- Drywell average temperature is 175°F and slowly rising.
- Torus pressure is 8.5 psig and slowly rising.
- Torus water temperature is 88°F and slowly rising.
- Torus water level is 11.1 feet and slowly rising.
- The current Containment Spray system alignment is shown on the following page.

Which one of the following describes the current alignment of the Containment Spray system, in accordance with N1-EOP-4, Primary Containment Control?

The Containment Spray system...

A. is in the proper alignment.

B. is NOT in the proper alignment. Direct execution of N1-EOP-1 Attachment 16, Torus Cooling.

C. is NOT in the proper alignment. Direct execution of N1-EOP-1 Attachment 15, Torus Water to Waste Collector.

D. is NOT in the proper alignment. Direct execution of N1-EOP-1 Attachment 17, Auto or Manual Initiation of Containment Spray.


11 CS
40-30
40-32
RBEDT
PB 102
PB 103
WASTE COLL

Proposed Answer: A

Explanation: The given Primary Containment conditions require entry into N1-EOP-4 based on high Drywell pressure, high Drywell temperature, and high Torus water temperature. Additionally, Torus pressure and Torus water level are elevated above normal values. Torus pressure is below the value (13 psig) requiring Containment Spray. Torus water level is below the value (11.25 feet) that requires further action. Torus water temperature is above the value (85°F) that requires Torus Cooling. The given Containment Spray system alignment correctly reflects the alignment achieved by N1-EOP-1 Attachment 16, Torus Cooling.

Note: This question meets the K/A and satisfies SRO-only question guidelines by requiring the candidate to analyze various Primary Containment conditions, determine the correct procedure section to be performed based on these conditions, and then further analyze various Containment Spray system indications/controls to determine their status relative to the required procedure section.

B. Plausible – N1-EOP-1 Attachment 16 is required, however Containment Spray is already in the proper alignment for this procedure.
C. Plausible – Torus water level is slightly elevated, but remains within the control band specified in N1-EOP-4. N1-EOP-1 Attachment 15 is allowed to be used in this situation to lower Torus water level within the normal band, but Torus Cooling is required and takes precedent. No procedure exists to simultaneously allow Torus Cooling and Torus letdown.
D. Plausible – Drywell pressure, Torus pressure, and Drywell temperature are all elevated, but remain below the values in N1-EOP-4 that would require initiation of Containment Spray.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295025 EA2.01
	Importance Rating	4.3

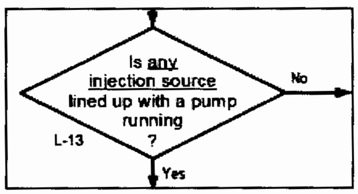
High Reactor Pressure

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

Proposed Question: #82

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

- Reactor water level is -108" and slowly lowering.
- Reactor pressure is 850 psig and slowly lowering with an Emergency Condenser in service.
- No Reactor injection sources are available.
- N1-EOP-2, RPV Control, has been executed to the following step in the alternate level control leg:



Which one of the following describes how Reactor pressure is now required to be controlled in accordance with the Emergency Operating Procedures?

A. Stabilize Reactor pressure around the current value.

B. Lower Reactor pressure. Do NOT exceed a cool-down rate of 100°F/hr.

C. Rapidly lower Reactor pressure. The cool-down rate is allowed to exceed 100°F/hr.

D. Secure the Emergency Condenser and allow Reactor pressure to rise until Emergency Condensers or ERVs automatically actuate.

Proposed Answer: A

Explanation: With no injection sources available and Reactor water level -108" and lowering, N1-EOP-2 must be exited and N1-EOP-9, Steam Cooling, must be entered. N1-EOP-9 requires stabilizing Reactor pressure, even though it is currently above the capacity of many low pressure injection systems. This is done to ensure the Minimum Zero Injection RPV Water Level calculation assumptions remain valid and Reactor inventory loss is minimized.

Note: Although Reactor pressure is not above the normal power operating range, this question meets the K/A because Reactor pressure is much higher than the injection limit of low pressure systems, which are required to inject if they become available based on Reactor water level.

B. Plausible – Lowering Reactor pressure would allow quicker injection with low pressure systems when they become available, however N1-EOP-9 requires stabilizing Reactor pressure.
C. Plausible – Lowering Reactor pressure would allow quicker injection with low pressure systems when they become available, however N1-EOP-9 requires stabilizing Reactor pressure.
D. Plausible – The Emergency Condenser will need to be at least temporarily secured to prevent Reactor pressure from lowering too much more. However N1-EOP-9 directs stabilizing Reactor pressure, not allowing it to rise significantly, such as would be required to reach the Reactor pressure at which Emergency Condensers and ERVs automatically actuate.

Technical Reference(s): N1-EOP-2, N1-EOP-9

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP9C01 EO-2

Question Source: Bank – 2013 Audit #82

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 #	295034 EA2.02
	Importance Rating	4.2

Secondary Containment Ventilation High Radiation

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

Proposed Question: #83

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

- Reactor Building Ventilation exhaust radiation monitors are indicating 6 mR/hr and slowly rising.
- The Reactor Building Continuous Air Monitor (CAM) is indicating normal count rates and stable.
- The Stack radiation monitors are indicating 75 cps and slowly rising.

Which one of the following describes these radiation monitor indications and if an emergency action level (EAL) has been exceeded?

These indications are consistent with contamination coming from the __(1)__.

An EAL __(2)__ been met or exceeded.

	(1)	(2)
A.	Reactor Building Sample Station	has
B.	Reactor Building Sample Station	has NOT
C.	Reactor Water Cleanup Heat Exchanger Room	has
D.	Reactor Water Cleanup Heat Exchanger Room	has NOT

Proposed Answer: B

Explanation: Due to the layout of the Reactor Building Ventilation exhaust piping, the Reactor Building Ventilation exhaust radiation monitors and Stack radiation monitors would both register contamination coming from either the Reactor Building Sample Station or the RWCU Heat Exchanger Room. However, the Reactor Building CAM is upstream of the discharge point for the Reactor Building Sample Station, therefore contamination from this area would not cause a rise in the Reactor Building CAM indication. The Reactor Building Ventilation exhaust radiation monitors are above the alarm/isolation setpoint, but this does not meet or exceed an EAL. Stack radiation monitors are below their EAL thresholds as well. Therefore no indications are given that meet or exceed an EAL.

A. Plausible – While the Reactor Building Ventilation exhaust radiation monitors are above the alarm/isolation setpoint, this does not meet or exceed an EAL. Stack radiation monitors do not meet an EAL until at least 300 cpm.
C. Plausible – Reactor Building Ventilation exhaust radiation monitor indications and Stack radiation monitors would rise due to contamination coming from the RWCU Heat Exchanger Room, but so would the Reactor Building CAM. While the Reactor Building Ventilation exhaust radiation monitors are above the alarm/isolation setpoint, this does not meet or exceed an EAL. Stack radiation monitors do not meet an EAL until at least 300 cpm.
D. Plausible – Reactor Building Ventilation exhaust radiation monitor indications and Stack radiation monitors would rise due to contamination coming from the RWCU Heat Exchanger Room, but so would the Reactor Building CAM.

Technical Reference(s): N1-OP-10, C-18013-C, EAL Chart

Proposed references to be provided to applicants during examination: Hot EAL Chart

Learning Objective: N1-288001-RBO-5

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 #	2
	K/A #	295032 2.4.18
	Importance Rating	4.0

High Secondary Containment Area Temperature

Knowledge of the specific bases for EOPs.

Proposed Question: #84

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

- A steam leak has developed from the Reactor Water Cleanup (RWCU) system.
- Attempts to isolate RWCU have been unsuccessful.
- N1-EOP-2, RPV Control, and N1-EOP-5, Secondary Containment Control, have been entered.
- An Operator in the field reports:
  - Reactor Building 261' West temperature is 136°F and rising slowly.
  - Reactor Building 261' East temperature is 145°F and rising slowly.
  - Reactor Building 281' West temperature is 110°F and rising slowly.
  - Reactor Building 281' East temperature is 119°F and rising slowly.

Which one of the following describes the required action and the basis for the associated temperature limit, in accordance with N1-EOP-2 and/or N1-EOP-5?

	Required Action	Basis for Temperature Limit
A.	Perform an RPV Blowdown	Personnel access
B.	Perform an RPV Blowdown	Reactor Building design limit
C.	Rapidly depressurize using Turbine Bypass Valves	Personnel access
D.	Rapidly depressurize using Turbine Bypass Valves	Reactor Building design limit

Proposed Answer: A and B, per revised key. TFish 4/10/15

Explanation: The given conditions indicate that a primary system is discharging into the Secondary Containment, the system cannot be isolated, and two General Areas (RB 261' West and East) have exceeded the Maximum Safe Temperature of 135°F. With two General Areas above the Maximum Safe Temperature, an RPV Blowdown is required. The basis for the 135°F Maximum Safe Temperature limit is to allow personnel access into the Secondary Containment to perform safe shutdown actions.

B. Plausible – These temperatures are well above normal Reactor Building temperatures, however the basis for the EOP Maximum Safe Temperature is either personnel access (limiting at Nine Mile Point Unit 1) or equipment operability, not Reactor Building design temperature.
C. Plausible – If only one General Area temperature was above Maximum Safe with a second General Area trending towards the limit, then RPV Blowdown would not yet be required and rapid depressurization with Turbine Bypass Valves would be correct.
D. Plausible – If only one General Area temperature was above Maximum Safe with a second General Area trending towards the limit, then RPV Blowdown would not yet be required and rapid depressurization with Turbine Bypass Valves would be correct. These temperatures are well above normal Reactor Building temperatures, however the basis for the EOP Maximum Safe Temperature is either personnel access (limiting at Nine Mile Point Unit 1) or equipment operability, not Reactor Building design temperature.

Technical Reference(s): N1-EOP-5, N1-EOP-2, NER-1M-095, GAI-OPS-20

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295017 2.4.30
	Importance Rating	4.1

High Off-site Release Rate

Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: #85

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

Time (hh:mm)	Condition
12:00	Indications become available in the Control Room that require declaration of Site Area Emergency RS1.2: **RS1.2** **1 | 2 | 3 | 4 | D** **Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the SITE BOUNDARY**
12:15	The Shift Manager declares Site Area Emergency RS1.2.
14:00	Indications become available in the Control Room that require declaration of General Emergency RG1.2: **RG1.2** **1 | 2 | 3 | 4 | D** **Dose assessment using actual meteorology indicates doses > 1,000 mRem TEDE or 5,000 mRem thyroid CDE at or beyond the SITE BOUNDARY**
14:15	The Shift Manager declares General Emergency RG1.2.

Which one of the following identifies the latest acceptable time for initiating notification of initial Protective Action Recommendations (PARs) to the State and County, in accordance with CNG-EP-1.01-1013, Emergency Classification and PAR, and CNG-EP-1.01-1015, Emergency Notifications?

A. 12:15

B. 12:30

C. 14:15

D. 14:30

Proposed Answer: D

Explanation: PARs are only made to the State and County upon declaration of a General Emergency. The PARs are made as part of the Part 1 Notification, which is required to be initiated no later than 15 minutes after the declaration of the General Emergency.

Note: This question is asking knowledge required at the SRO level. Reactor Operators are not expected to know time limits with respect to PARs. Therefore this question is not at the RO license level.

A. Plausible – This is 15 minutes after conditions are available that a significant offsite release is in progress that warrants a Site Area Emergency declaration. However, PARs are NOT made until a General Emergency is declared, even with a release in progress.
B. Plausible – This is 15 minutes after declaration of a Site Area Emergency based on a significant offsite release. However, PARs are NOT made until a General Emergency is declared, even with a release in progress.
C. Plausible – This is 15 minutes after conditions are available that warrant declaration of a General Emergency due to a significant offsite release. However, the PAR notification is NOT required to be initiated until 15 minutes after the declaration of the General Emergency, which can take up to an additional 15 minutes.

Technical Reference(s): CNG-EP-1.01-1013, CNG-EP-1.01-1015

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP6C01 EO-3

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 #	2
	Group #	1
	K/A #	206000 A2.06
	Importance Rating	3.5

HPCI

Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow: BWR-2,3,4

Proposed Question: #86

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

- N1-ST-Q3, High Pressure Coolant Injection Pump and Check Valve Operability Test, has been completed for Feedwater pump 11.
- Condensate storage tank inventory is 125,000 gallons.
- The maximum Feedwater pump 11 flow capability at normal Reactor operating pressure has been determined to be 3180 gpm.

Which one of the following describes the status of HPCI in accordance with Technical Specifications?

A. One train of HPCI is inoperable, only. A 7 day LCO must be entered.

B. One train of HPCI is inoperable, only. A 15 day LCO must be entered.

C. Both trains of HPCI are inoperable. The Reactor must be in cold shutdown within a limit of 10 hours.

D. Both trains of HPCI are inoperable. Reactor pressure must be 110 psig or less within a limit of 24 hours.

Proposed Answer: B

Explanation: Technical Specification 3.1.8 requires each Feedwater pump to be able to provide a flow rate of at least 3420 gpm at normal Reactor operating pressure. Since Feedwater pump 11 is only capable of providing a flow rate of 3180 gpm, it is inoperable. Technical Specification 3.1.8 also requires a minimum Condensate storage tank (CST) inventory of 105,000 gallons, or else both trains of HPCI are inoperable and Reactor pressure must be lowered to 110 psig or less within a limit of 24 hours.

A. Plausible – One train of HPCI is inoperable, which requires a 15 day LCO. 7 days is the LCO length for one inoperable Core Spray pump.
C. Plausible – CST inventory is low, but still adequate to support HPCI operability. Therefore, only one train of HPCI is inoperable, based on inadequate flow from Feedwater pump 11. Cold shutdown within 10 hours is the LCO length for an inoperable Core Spray system.
D. Plausible – CST inventory is low, but still adequate to support HPCI operability. Therefore, only one train of HPCI is inoperable, based on inadequate flow from Feedwater pump 11. Reactor pressure of 110 psig or less within 24 hours is the correct LCO if both HPCI trains were inoperable.

Technical Reference(s): Technical Specification 3.1.8 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-14

Question Source: Modified Bank – SYSID 88406

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 #	207000 A2.01
	Importance Rating	4.5

Isolation (Emergency) Condenser

Ability to (a) predict the impacts of the following on the ISOLATION (EMERGENCY) CONDENSER; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Tube bundle leak: BWR-2,3

Proposed Question: #87

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

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

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

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

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

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

D. These indications are higher than expected for an in service EC, but do NOT validate an EC 11 tube leak. Direct Chemistry to obtain an EC 11 shell side sample.

Proposed Answer: A

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	205000 2.4.8
	Importance Rating	4.5

Shutdown Cooling

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

Proposed Question: #88

A plant shutdown is in progress with the following:

- Reactor coolant temperature is 190°F and slowly lowering.
- The only running Shutdown Cooling pump trips.
- N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal, is entered.
- While placing an alternate Shutdown Cooling pump in service, Reactor water level lowers below 53 inches.

Which one of the following describes the correct procedure implementation?

A. Enter N1-EOP-2, RPV Control. Continue performing N1-SOP-6.1. In the event of a conflict between the procedures, N1-EOP-2 is the overriding document.

B. Enter N1-EOP-2, RPV Control. Continue performing N1-SOP-6.1. In the event of a conflict between the procedures, N1-SOP-6.1 is the overriding document.

C. Exit N1-SOP-6.1 and enter N1-EOP-2, RPV Control. N1-SOP-6.1 is re-entered at the step in progress after exiting N1-EOP-2.

D. Exit N1-SOP-6.1 and enter N1-EOP-2, RPV Control. N1-SOP-6.1 entry conditions are re-evaluated after exiting N1-EOP-2.

Proposed Answer: A

Explanation: N1-EOP-2 is entered in this case based on Reactor water level less than 53". There is no requirement to exit SOPs when EOPs are entered. In fact, both procedures are executed concurrently. The Emergency Operating Procedures are higher-tiered documents than the Special Operating Procedures, therefore in the event of a conflict, the EOP must be followed.

B. Plausible – The SOP does contain specific guidance to assist with this problem, whereas the EOP contains more general guidance. However, EOPs are higher-tiered documents than SOPs. followed.
C. Plausible – The EOP is the higher-tiered document, but the SOP is still used in parallel.
D. Plausible – The EOP is the higher-tiered document, but the SOP is still used in parallel.

Technical Reference(s): NER-1M-095, GAI-OPS-20, N1-SOP-6.1, N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP00C01 TO #1

Question Source: Modified Bank – 2008 NRC #97

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 2.4.47
	Importance Rating	4.2

SLC

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #89

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

- A breaker failure has resulted in a loss of the Liquid Poison (LP) tank heater.
- LP tank volume is 1341 gallons.
- LP tank concentration is 17.9 weight %, with a Boron-10 enrichment of 65.95 atom %.
- LP tank temperature has trended as follows:

Time (hh:mm)	LP Tank Temperature
08:00	96°F
09:00	94°F

Note: Assume the LP tank temperature trend continues at the given rate.

Which one of the following describes the Technical Specification impact of these conditions?

At approximately time __(1)__, a __(2)__ LCO will need to be entered.

	(1)	(2)
A.	16:00	one (1) hour
B.	16:00	seven (7) day
C.	21:00	one (1) hour
D.	21:00	seven (7) day

Proposed Answer: A

Explanation: Technical Specification Figure 3.1.2.b requires a minimum Liquid Poison tank temperature of approximately 80°F, based on a boron solution concentration of 17.9 weight %. With the given temperature trend, this temperature will be reached at approximately 16:00. Technical Specification 3.1.2.d will then NOT be met. Technical Specification 3.1.2.e must be entered and requires initiating a normal orderly shutdown within one hour.

B. Plausible – 16:00 is the correct time, however since there is only one LP tank, a one hour LCO applies, not the 7 day LCO for a redundant component.
C. Plausible – LP tank temperature will be approximately 70°F at time 21:00, which is the alarm setpoint for the low temperature annunciator (K1-3-1). However, TS Figure 3.1.2.b requires a higher temperature for operability.
D. Plausible – LP tank temperature will be approximately 70°F at time 21:00, which is the alarm setpoint for the low temperature annunciator (K1-3-1). However, TS Figure 3.1.2.b requires a higher temperature for operability. Since there is only one LP tank, a one hour LCO applies, not the 7 day LCO for a redundant component.

Technical Reference(s): Technical Specification 3.1.2

Proposed references to be provided to applicants during examination: Technical Specification 3.1.2 and Figure 3.1.2.b

Learning Objective: N1-211000-RBO-14

Question Source: Modified Bank – 2013 NRC #90

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 #	215005 A2.07
	Importance Rating	3.4

APRM / LPRM

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation flow channels flow mismatch

Proposed Question: #90

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

- Recirc pump 14 is shutdown and isolated for maintenance.
- The other four Recirc pumps are in service.

Then, Recirc pump 11 trips due to a sustained electrical fault with the following:

- Initial attempts to close Recirc pump 11 discharge and suction valves have failed.
- The FIN team is investigating.

Which one of the following describes the most restrictive Technical Specification impact?

A. Technical Specifications allow continued power operation as long as Reactor power is maintained below 90%.

B. A limit of 12 hours is available to close Recirc pump 11 discharge or suction valve, or then take action to insert control rods.

C. A limit of 1 hour is available to close Recirc pump 11 discharge or suction valve, or then take action to insert control rods.

D. A manual Reactor scram is required now.

Proposed Answer: C

Explanation: Two Technical Specifications apply to this situation. Technical Specification 3.1.7 requires Reactor power to be limited to less than 90% since only three Recirc loops are in operation. Technical Specification 3.6.2 requires verifying sufficient APRM channels remain operable or tripped to main trip capability within one hour. All 8 APRMs are inoperable because, with Recirc pump 11 tripped and both the discharge and suction valves open, there is a mismatch between actual core flow and what the Recirc flow channels are measuring. This mismatch makes the flow-biased scram and rod block setpoints non-conservative for all 8 APRMs. Therefore, either the Recirc pump 11 suction or discharge valve must be closed within a limit of one hour, or else Technical Specification 3.6.2.a(1) requires control rods to be inserted.

A. Plausible – Technical Specification 3.1.7 does require Reactor power to be limited to 90% in this situation. However there is also a shutdown LCO in effect per Technical Specification 3.6.2 that limits how long continued power operation can continue.
B. Plausible – 12 hours is the limit in Technical Specification 3.6.2 if one channel required by Table 3.6.2.a is inoperable in one or more parameters. Since more than one required channel is inoperable, the more restrictive 1 hour limit applies.
D. Plausible – If one less Recirc pump were operating, a manual Reactor scram would be required.

Technical Reference(s): Technical Specifications 3.1.7 and 3.6.2, N1-SOP-1.3

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.7 and 3.6.2 with setpoint column blocked out

Learning Objective: N1-215005-RBO-14

Question Source: Modified Bank – 2009 NRC #90

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 #	239001 A2.01
	Importance Rating	3.9

Main and Reheat Steam

Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Malfunction of reactor turbine pressure regulating system

Proposed Question: #91

The plant is operating at 50% power with the following:

- The EPR pressure setpoint is 918 psig.
- The MPR pressure setpoint is 926 psig.
- Reactor pressure is 1020 psig and stable.

Then, a malfunction causes the EPR pressure setpoint to drift to 935 psig and then stabilize.

Which one of the following describes the impact of this EPR failure on Thermal Limits, if any, in accordance with N1-OP-31, Tandem Compound Reheat Turbine, and the Core Operating Limits Report?

A. No thermal limit penalties are required.

B. A thermal limit penalty applies to LHGR, only.

C. A thermal limit penalty applies to MCPR, only.

D. A thermal limit penalty applies to both LHGR and MCPR.

Proposed Answer: D

Explanation: With the MPR pressure setpoint less than the EPR pressure setpoint, the MPR is the controlling pressure regulator. The EPR is considered a poor backup regulator, therefore N1-OP-31 requires the thermal limit restrictions for operation with one pressure regulator inoperable to be satisfied. At 50% power, the Core Operating Limits Report (COLR) includes penalties on both the LHGR and MCPR thermal limits without an operable backup pressure regulator.

Note: This question meets the K/A by requiring the candidate to determine the effect of the EPR pressure setpoint failure on system operation (ie. which regulator is now controlling and if there is an actual loss of backup pressure regulator) and the resulting procedural limit to mitigate this effect (ie. N1-OP-31 and COLR requirements for thermal limit penalty). The question meets SRO question guidelines by requiring the candidate to assess conditions and determine the applicable procedural section and limitation that applies in this situation.

A. Plausible – The pressure setpoint difference is within a normal range for when the EPR is in service and the MPR is the backup pressure regulator. However since the MPR is in service, N1-OP-31 requires applying the thermal limit penalties for operation without a backup pressure regulator. With the Reactor at less than 100% power, these penalties do require adjustment to thermal limits.
B. Plausible – A penalty does apply to LHGR, but a penalty also applies to MCPR.
C. Plausible – A penalty does apply to MCPR, but a penalty also applies to LHGR.

Technical Reference(s): N1-OP-31, COLR

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-248000-RBO-14

Question Source: Modified Bank – 2010 Audit #84

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	234000 2.4.9
	Importance Rating	4.2

Fuel Handling Equipment

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

Proposed Question: #92

Refueling is in progress with the following:

- An irradiated fuel bundle is grappled and is being removed from the Reactor core.
- The Refuel Bridge team observes that Spent Fuel Pool and Reactor cavity water level is lowering.
- Spent Fuel Pool and Reactor cavity water level has lowered approximately one foot below the normal level.
- The Refuel Bridge radiation monitors indicate 25 mr/hr and slowly rising.
- The Control Room enters N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal.
- The irradiated fuel bundle is currently half way out of the Reactor core.

Which one of the following describes the required direction to be given in accordance with N1-SOP-6.1?

Direct evacuation of...

A. all personnel from the Refuel Floor, leaving the irradiated fuel bundle in the current position.

B. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to transfer the irradiated fuel bundle to the Spent Fuel Pool.

C. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to lower the irradiated fuel bundle back into the original Reactor core location.

D. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to halt fuel movement and leave the irradiated fuel bundle in the current position.

Proposed Answer: C

Explanation: N1-SOP-6.1 entry is required due to reported or observed loss of SFP/Reactor cavity inventory. A conditional override step requires evacuation of all personnel from the Refuel Floor if either irradiated fuel is uncovered or the Refuel Bridge high range radiation monitor alarms. Neither of these conditions are met, as evidenced by water level being one foot below normal level (therefore fuel bundle in SFP are covered by over 20 feet of water still) and Refuel Bridge radiation monitors indicating 20 mr/hr and rising (well below the 50 mr/hr and 1000 mr/hr nominal high alarm setpoints). A subsequent step in N1-SOP-6.1 requires evacuating non-essential personnel from the Refuel Floor and returning any core component being transferred to the nearest storage location in the SFP or Reactor core. Since the irradiated fuel bundle is only half way out of the Reactor core, the nearest storage location is back in the Reactor core.

A. Plausible – If conditions were worse, such that either irradiated fuel was uncovered or the Refuel Bridge high radiation alarm sounded, immediate evacuation of the Refuel Floor would be the required action in N1-SOP-6.1.
B. Plausible – If the irradiated fuel bundle had already been transferred closer to the SFP, that would be the correct storage location to continue to. Additionally, going to the SFP instead of the Reactor core does eliminate a positive reactivity addition, which is desirable in certain situations.
D. Plausible – Leaving the irradiated fuel bundle in the current position does keep it covered with more water than going to the SFP and eliminates a positive reactivity addition associated with re-inserting it into the Reactor core.

Technical Reference(s): N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP6.1CO1 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	271000 2.4.11
	Importance Rating	4.2

Off-gas

Knowledge of abnormal condition procedures.

Proposed Question: #93

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

- Annunciator H1-2-6, Air Ejector Off Gas Flow – Press Temp High, alarms due to high pressure and temperature.
- Annunciator H1-1-6, Air Ejector Hi – Lo Stm Pr V Closed, alarms.
- N1-SOP-25.3, Offgas Hydrogen Explosion, is entered.
- Main Condenser vacuum is 27.5 inches Hg and slowly lowering.
- N1-SOP-25.1, Unplanned Loss of Condenser Vacuum, is entered.

Which one of the following is required to be directed in accordance with these SOPs?

Direct a(n)...

A. manual Reactor scram per N1-SOP-1, Reactor Scram, and shutdown of Hydrogen Water Chemistry.

B. manual Reactor scram per N1-SOP-1, Reactor Scram, and isolation of steam to the Offgas mixing jet.

C. emergency power reduction per N1-SOP-1.1, Emergency Power Reduction, and shutdown of Hydrogen Water Chemistry.

D. emergency power reduction per N1-SOP-1.1, Emergency Power Reduction, and isolation of steam to the Offgas mixing jet.

Proposed Answer: C

Explanation: The given conditions indicate an explosion has occurred in the Offgas piping and the Steam Jet Air Ejector interstage blocking valves have closed. These events have also led to lowering Main Condenser vacuum. N1-SOP-25.3 requires shutdown of Hydrogen Water Chemistry to prevent adding more hydrogen and/or oxygen to the Offgas system. This procedure requires continuing to add dilution steam through the Offgas mixing jet to reduce concentrations of hydrogen and/or oxygen to avoid another explosion or further fire. N1-SOP-25.1 requires an emergency power reduction under these conditions to attempt to control vacuum.

A. Plausible – N1-SOP-25.1 contains guidance for required manual Reactor scrams if conditions degrade further, but a scram is not currently required.
B. Plausible – N1-SOP-25.1 contains guidance for required manual Reactor scrams if conditions degrade further, but a scram is not currently required. The steam added through the Offgas mixing jet is adding energy (high temperature) to a potentially explosive or fire-affected area, but it is required to be left in service to attempt to dilute hydrogen and/or oxygen.
D. Plausible – An emergency power reduction is required. The steam added through the Offgas mixing jet is adding energy (high temperature) to a potentially explosive or fire-affected area, but it is required to be left in service to attempt to dilute hydrogen and/or oxygen.

Technical Reference(s): N1-SOP-25.3, N1-SOP-25.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-271000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.35
	Importance Rating	3.9

Knowledge of the fuel-handling responsibilities of SROs.

Proposed Question: #94

Given the following evolutions:

(1) The plant is operating at 100% power and an irradiated fuel bundle is being moved within the Spent Fuel Pool in preparation for an outage.

(2) A refueling outage is in progress and an irradiated fuel bundle is being removed from the Reactor core and transferred to the Spent Fuel Pool.

(3) A refueling outage is in progress with fuel in the Reactor and a control rod blade is being removed from the Reactor core for replacement.

Which one of the following identifies the evolution(s) that must be directly supervised by a Senior Reactor Operator (SRO) or SRO Limited to Fuel Handling (LSRO) who has no other concurrent responsibilities, in accordance with S-ODP-NFM-0101, Refueling Operations, Fuel Handling Procedures (N1-FHPs), and the UFSAR?

A. (2) only

B. (1) and (2) only

C. (2) and (3) only

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

Proposed Answer: D

Explanation: UFSAR Table XIII-2, Minimum Shift Crew Composition, requires an SRO or LSRO to directly supervise, with no other concurrent responsibilities, all core alterations. Technical Specifications define a core alteration as "the addition, removal, relocation, or other manual movement of fuel or controls in the reactor core". Therefore, evolutions (2) and (3) are core alterations and require an SRO or LSRO to directly supervise with no other concurrent responsibilities. Evolution (1) is also required to be supervised by an SRO or LSRO per S-ODP-NFM-0101.

A. Plausible – Evolution (2) is correct, but so are evolutions (1) and (3). Evolution (1) is required to be supervised by an SRO or LSRO per S-ODP-NFM-0101. Evolution (3) does not include movement of fuel, but removing a control rod blade from the Reactor core is also a core alteration requiring direct supervision by an SRO.
B. Plausible – Evolutions (1) and (2) are correct, but so is evolution (3). Evolution (3) does not include movement of fuel, but removing a control rod blade from the Reactor core is also a core alteration requiring direct supervision by an SRO.
C. Plausible – Evolutions (2) and (3) are correct, but so is evolution (1). Evolution (1) is required to be supervised by an SRO or LSRO per S-ODP-NFM-0101.

Technical Reference(s): UFSAR Table XIII-2, S-ODP-NFM-0101, N1-FHP-021, N1-FHP-24, N1-FHP-25

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-234000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.1
	Importance Rating	4.4

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

Proposed Question: #95

Preparations for a plant startup are in progress with the following:

- Based on plant conditions, one step in N1-OP-43A, Plant Startup, Section E.1.0, Plant Startup Preparations and Prerequisites, CANNOT be performed and is NOT required to be performed.
- All other steps in N1-OP-43A Section E.1.0 are complete and signed off.
- It is desired to continue on to N1-OP-43A Section E.2.0, Approach to Criticality and Vessel Heatup.

Which one of the following describes the required action to proceed with the startup in accordance with N1-OP-43A?

A. Record the step on a Variance Form. Only one SRO is required to review and approve the variance.

B. Record the step on a Variance Form. Two SROs are required to review and approve the variance.

C. Process a Technical Procedure Step Deletion Screening Form. Only one SRO is required to review and approve the step deletion.

D. Process a Technical Procedure Step Deletion Screening Form. Two SROs are required to review and approve the step deletion.

Proposed Answer: B and D, per revised Key. TFish 4/10/15

Explanation: N1-OP-43A contains specific guidance in section C.1.0 regarding processing of a step that cannot be performed and is not required to be performed. The procedure requires recording the step on a Variance Form located as an attachment in N1-OP-43A. Both the Control Room Supervisor and Shift Manager (2 SROs) must sign the variance form entry.

A. Plausible – Both the Control Room Supervisor and Shift Manager must sign the variance form entry, therefore it is two SROs, not just one, that must review and approve the variance.
C. Plausible – CNG-PR-1.01-1009 Attachment 1, Technical Procedure Step Deletion Screening Form, is the normal method for such a situation with a technical procedure. However, a plant startup is a unique situation where the technical procedure contains its own method for tracking such steps. Both the Control Room Supervisor and Shift Manager must sign the variance form entry, therefore it is two SROs, not just one, that must review and approve the variance.
D. Plausible – CNG-PR-1.01-1009 Attachment 1, Technical Procedure Step Deletion Screening Form, is the normal method for such a situation with a technical procedure. However, a plant startup is a unique situation where the technical procedure contains its own method for tracking such steps.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-OP-43A-CT-01

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 #	3
	Group #	
	K/A #	2.3.11
	Importance Rating	4.3

Ability to control radiation releases.

Proposed Question: #96

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

- Annunciator H1-4-5, Liquid Process Rad Monitor, alarms.
- An Operator reports that the EQUIP FAIL light on the Service Water Rad Monitor is ON.
- Chemistry has been notified and determines that Service Water release rates are normal.

Which one of the following actions is required to allow Service Water operation to continue, in accordance with the Offsite Dose Calculation Manual (ODCM)?

A. Verify the other Service Water Radiation Monitor is operable within 12 hours.

B. Collect and analyze Service Water effluent grab samples at least once per 12 hours.

C. Sample Reactor Building and Turbine Building Service Water return lines alternately every 15 minutes.

D. Collect and analyze two independent Service Water effluent grab samples and have two technically qualified individuals verify calculations and valving.

Proposed Answer: B

Explanation: ODCM Table 3.6.14-1 requires a Service Water effluent line radiation monitor to be operable. With the only installed Service Water effluent line radiation monitor inoperable, Note (d) applies and requires "With the number of channels functional less than required by the minimum channels functional requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed…".

A. Plausible – ODCM Table 3.6.14-1 requires only one Service Water radiation monitor to be operable, so if there was a second one, the table would be met as long as it was operable. However, only one Service Water radiation monitor is installed.
C. Plausible – This is patterned after ODCM Table 3.6.14-1 Note (c), but this note applies to liquid radwaste radiation monitoring, not Service Water.
D. Plausible – ODCM Table 3.6.14-1 Note (i), "Monitoring will be conducted continuously by alternately sampling the reactor building and turbine building service water return lines for approximately 15-minute intervals", applies to the normal sampling during all modes of operation, but not this situation.

Technical Reference(s): ARP H1-4-5, ODCM 3.6.14, N1-OP-50B

Proposed references to be provided to applicants during examination: ODCM 3.6.14

Learning Objective: N1-272000-RBO-14

Question Source: Bank – 2010 NRC #96

Question History: 2010 NRC #96

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.41
	Importance Rating	4.6

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #97

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

Time (hh:mm)	Condition
00:10	Conditions at Unit 1 justify declaration of a Site Area Emergency. Conditions at Unit 2 justify declaration of an Unusual Event.
00:15	Conditions at Unit 1 change and only justify declaration of an Alert. Conditions at Unit 2 continue to justify declaration of an Unusual Event. Emergency declaration has NOT yet been made.
00:20	The Unit 1 and Unit 2 Shift Managers have discussed the situation and are ready to make the proper emergency declaration(s).

Which one of the following describes the proper emergency declaration(s), in accordance with CNG-EP-1.01-1013, Emergency Classification and PAR?

A. Only one declaration must be made for an Alert.

B. Only one declaration must be made for a Site Area Emergency.

C. Unit 1 must declare an Alert and Unit 2 must declare an Unusual Event.

D. Unit 1 must declare a Site Area Emergency and Unit 2 must declare an Unusual Event.

Proposed Answer: A

Explanation: When an emergency declaration is made, it must be for the highest level emergency classification for which an EAL is currently being met or exceeded. Unit 1 has experienced a transitory event, but since the Site Area Emergency is no longer met, the remaining Alert must be declared. The Nine Mile Point site only makes a single emergency declaration, so the Unit 2 Unusual Event is NOT separately declared.

B. Plausible – A Site Area Emergency condition did occur, but since it has cleared before the time of declaration, it is NOT declared.
C. Plausible – Each Unit is individually experiencing conditions that meet these classifications, but the Nine Mile Point site only makes a single emergency declaration.
D. Plausible – Each Unit has individually experienced conditions that meet these classifications, but the Nine Mile Point site only makes a single emergency declaration. A Site Area Emergency condition did occur at Unit 1, but since it has cleared before the time of declaration, it is NOT declared.

Technical Reference(s): CNG-EP-1.01-1013

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPL001-TO-01

Question Source: Modified Bank – 2009 NRC #99

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.14
	Importance Rating	4.3

Knowledge of the process for controlling equipment configuration or status.

Proposed Question: #98

The plant is operating at 100% power and Liquid Poison system ***11*** is to be removed from service for maintenance.

Which one of the following processes is required to be used to identify Liquid Poison system ***12*** while Liquid Poison system ***11*** is out of service?

A. CNG-OP-1.01-1000, Conduct of Operations, Section 5.11, Quarantine.

B. CNG-OP-1.01-1000, Conduct of Operations, Section 5.22, Temporary Protected Equipment.

C. CNG-OP-1.01-2010, Operator Workaround/Challenge Control, Section 5.1, Identification and Origination.

D. CNG-OP-1.01-1007, Clearance and Safety Tagging, Section 5.25, Guarantee Tagouts, Tags, And Protection.

Proposed Answer: B

Explanation: CNG-OP-1.01-1000 Section 5.22, Temporary Protected Equipment, is required during maintenance activities to ensure equipment/systems are not inadvertently removed from service that would require a Technical Specification shutdown within four hours. If both Liquid Poison systems become inoperable at the same time, Technical Specifications require a shutdown be initiated within one hour.

A. Plausible – CNG-OP-1.01-1000 Section 5.11, Quarantine, is used to limit access to equipment, similar to the Protected Equipment process. However, Quarantine is specific to equipment that has failed or malfunctioned, to preserve physical evidence for investigation.
C. Plausible – Removal of one Liquid Poison system could present a challenge to operators during transient response. However, the CNG-OP-1.01-2010, Operator Workaround/Challenge Control, process is specifically used to identify equipment and provide compensatory actions for deficiencies that require extra operator actions to make up for the deficiency. Nothing is deficient with Liquid Poison system 12, it is just the only remaining operable system.
D. Plausible – CNG-OP-1.01-1007, Clearance and Safety Tagging, is required to be used on Liquid Poison system 11 to provide personnel protection during the maintenance. Section 5.25, Guarantee Tagouts, Tags, And Protection, is specifically used to guarantee hazardous energy is removed from a system, not to protect a system from being removed from service.

Technical Reference(s): CNG-OP-1.01-1000, CNG-OP-1.01-GL012

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-OP-1.01-1000-CE-01

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 #	3
	Group #	
	K/A #	2.4.38
	Importance Rating	4.4

Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

Proposed Question: #99

An emergency is in progress and you are the Shift Manager / Emergency Director (SM/ED).

Given the following responsibilities:

(1) Determining the necessity for a local area/building evacuation
(2) Authorizing emergency workers to exceed normal radiation exposure limits
(3) Making the decision to notify off-site emergency management agencies
(4) Making Protective Action Recommendations (PARs) to off-site emergency management agencies

Which one of the following identifies which of these responsibilities shall NOT be delegated, in accordance with the Site Emergency Plan?

A. (4) only

B. (2) and (3) only

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

D. (1), (2), (3), and (4)

Proposed Answer: ~~A~~ D is correct, per revised key. TFish 4/16/15

Explanation: The Site Emergency Plan lists the following responsibilities of the SM/ED that shall NOT be delegated:

- Classification and declaration of the emergency event as an Unusual Event, Alert, Site Area Emergency or General Emergency.
- Determining the necessity for an **exclusion** area evacuation.
- Authorizing emergency workers to exceed normal radiation exposure limits.
- Making the decision to notify off-site emergency management agencies.
- Making Protective Action Recommendations (PARs) as necessary to offsite emergency management agencies.

Items (2), (3), and (4) correspond to three of these bullets. Item (1) is similar to the second bullet, but not the same. The need for local area/building evacuations may be determined by other members of the operating crew, such as the CRS or an RO carrying out an SOP with evacuation requirements.

A. Plausible – Item (4) shall NOT be delegated, and is one of the more severe items given. However, items (2) and (3) also shall NOT be delegated.
B. Plausible – Items (2) and (3) shall NOT be delegated, and are two of the more severe items given. However, item (4) also shall NOT be delegated.
D. Plausible – Items (2), (3), and (4) shall NOT be delegated, however item (1) may be performed by other members of the operating crew.

Technical Reference(s): Site Emergency Plan section 5.2

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPL000-TO

Question Source: Bank – NMP2 2014 Audit #100

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 #	2.1.39
	Importance Rating	4.3

Knowledge of conservative decision making practices.

Proposed Question: #100

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

- The high pressure seal on Recirculation pump 11 has failed.
- N1-SOP-1.2, Reactor Recirculation Pump Seal Failure, has been entered.
- The low pressure seal on Recirculation pump 11 is showing signs of degradation, but the failure is NOT catastrophic.

Which one of the following describes the applicability of CNG-OP-1.01-1001, Operation Decision Making, for this situation?

The CNG-OP-1.01-1001 process...

A. is NOT warranted for this situation.

B. for a Type 1 ODMI is warranted for this situation.

C. for a Type 2 ODMI is warranted for this situation.

D. for a Type 3 ODMI is warranted for this situation.

Proposed Answer: B

Explanation: The CNG-OP-1.01-1001 Type 1 ODMI process is applicable since this off-normal situation is within the bounds of approved procedures (N1-SOP-1.2) and would require the Shift Manager to lead the Control Room to make immediate decisions. Attachment 1, Type 1 Decision-Making Checklist/Event Response Checklist, is warranted since it provides direction on making the appropriate notifications and contains questions designed to ensure all aspects of the situation are addressed.

A. Plausible – N1-SOP-1.2, as well as N1-OP-1, does contain guidance for response to a Recirculation pump seal failure, however the CNG-OP-1.01-1001 Type 1 ODMI process is designed to supplement this procedural guidance. CNG-OP-1.01-1001 Attachment 1, Type 1 Decision-Making Checklist/Event Response Checklist, provides direction on making the appropriate notifications and contains questions designed to ensure all aspects of the situation are addressed.
C. Plausible – CNG-OP-1.01-1001 is warranted, however this is a Type 1 ODMI, not the less severe Type 2 ODMI (for conditions that fall below action thresholds in existing procedures).
D. Plausible – CNG-OP-1.01-1001 is warranted, however this is a Type 1 ODMI, not the less severe Type 3 ODMI.

Technical Reference(s): CNG-OP-1.01-1001, N1-SOP-1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-OP-1.01-CT-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments: