Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K1.09
	Importance Rating	3.2

LPCS

Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Nuclear boiler instrumentation

Proposed Question: #1

The plant was starting up following a refueling outage with the following:

- An electrical fault occurred on RPS Bus 12.
- Reactor water level was inadvertently lowered to 0".
- Reactor pressure is 310 psig and steady.

Which one of the following describes the resulting status of the Core Spray system?

A. All Core Spray pumps inject.

B. 111 and 112 Core Spray pumps inject, only.

C. 121 and 122 Core Spray pumps inject, only.

D. NO Core Spray pumps inject.

Proposed Answer: A

Explanation: Core Spray logic is de-energize to function. The loss of RPS bus 12 produces a half initiation signal. The other half initiation occurs when Reactor water level lowers to <+5". The Core Spray injection valves open when Reactor pressure is <365 psig. Therefore, the injection valves would be open. Additionally, Reactor pressure is sufficiently lower than 365 psig to allow flow to the Reactor (below Core Spray discharge pressure capability).

B. Incorrect – All Core Spray pumps inject. Plausible because one half of RPS is de-energized.
C. Incorrect – All Core Spray pumps inject. Plausible because one half of RPS is de-energized.
D. Incorrect – All Core Spray pumps inject. Plausible because this would be correct if the loss of RPS 12 prevented Core Spray logic from starting pumps (energize to function), if Reactor water level stayed higher (>+5"), or if Reactor pressure was higher.

Technical Reference(s): C-19859-C Sheet 9

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-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 #	400000 K1.02
	Importance Rating	3.2

Component Cooling Water

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

Proposed Question: #2

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

- H1-1-4, TB COOLING WTR PUMP 11 TRIP/TROUBLE
- H1-2-4, TB COOLING WTR PUMP 12 TRIP/TROUBLE

N1-SOP-24.1, TBCLC Failure, is entered.

Which one of the following identifies a component that shall be monitored for high temperatures in accordance with N1-SOP-24.1, TBCLC Failure?

A. Instrument Air Compressor 11

B. Recirculation MG Set coolers

C. Condensate Pump 13 coolers

D. Feedwater Pump 11 coolers

Proposed Answer: B

Explanation: The provided alarms show a loss of TBCLC. In accordance with N1-SOP-24.1, systems cooled by TBCLC should be monitored. Of the systems listed, only Recirc MG set coolers are cooled by TBCLC.

A. Incorrect – Instrument Air Compressor (IAC) 11 is cooled by RBCLC. Plausible because IAC 13 is cooled by TBCLC.
C. Incorrect – Condensate pump 13 cooler is cooled by RBCLC. Plausible because Feedwater pump 13 jacket coolers are cooled by TBCLC.
D. Incorrect – Feedwater Pump 11 coolers are cooled by RBCLC. Plausible because Feedwater pump 13 jacket coolers are cooled by TBCLC and Feedwater Pump 11 is in the Turbine Building.

Technical Reference(s): N1-SOP-24.1, N1-SOP-11.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-274000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 11/9/18 – Changed choice D to raise plausibility, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K2.01
	Importance Rating	3.2

HPCI

Knowledge of electrical power supplies to the following: System valves: BWR-2,3,4

Proposed Question: #3

The plant was operating at 100% power when a failure to scram resulted in the following:

- The US ordered the RO to terminate and prevent Feedwater injection.
- The RO closed the following valves:
  - Feedwater Isolation Valve 11
  - Feedwater Isolation Valve 12
  - Feedwater FCV 11
  - Feedwater FCV 12

Then, an electrical fault caused a loss of Powerboard 171B.

Which one of the following valves CANNOT be manually re-opened from the Control Room due to this power loss?

A. Feedwater FCV 11

B. Feedwater FCV 12

C. Feedwater Isolation Valve 11

D. Feedwater Isolation Valve 12

Proposed Answer: D

Explanation: Feedwater Isolation Valve 12 is a motor operated valve powered from PB 171B. With the loss of PB 171B, Feedwater Isolation Valve 12 fails as is.

Note: This question meets the K/A because the HPCI system at NMP1 is a mode of feedwater operation. From N1-OP-16, the "HPCI mode of Feedwater operation consists of the condensate surge and storage tanks, main condenser hotwell, condensate pumps (#11 and #13), condensate prefilters, condensate demineralizers, feedwater booster pumps (#11 and #13), feedwater heaters, motor-driven feedwater pumps (#11 and #12), an integrated control system, and all associated piping and valves."

A. Incorrect – Feedwater FCV 11 is an air-operated valve, with control power coming from RPS bus 11. The loss of PB 171B does not affect RPS Bus 11. Plausible because Powerboards 161B and 171B provide power to the related Feedwater isolation valves.
B. Incorrect – Feedwater FCV 12 is an air-operated valve, with control power coming from RPS bus 12. The loss of PB 171B does not affect RPS Bus 12. Plausible because Powerboards 161B and 171B provide power to the related Feedwater isolation valves.
C. Incorrect – Feedwater Isolation Valve 11 is powered from PB 161B, not PB 171B. Plausible because these are related valves with related power sources.

Technical Reference(s): N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-4

Question Source: Bank - 2010 NRC #4

Question History: 2010 NRC #4

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K2.02
	Importance Rating	3.0

Instrument Air

Knowledge of electrical power supplies to the following: Emergency air compressor

Proposed Question: #4

The plant is operating at 100% power with the following Instrument Air Compressor (IAC) lineup:

- IAC 13 is in service.
- IAC 11 is red flagged as the backup compressor.
- IAC 12 is green flagged as the standby compressor.

Then, Powerboard 16A is de-energized due to a sustained electrical fault.

Which one of the following lists the availability of IACs 11 and 12 to supply header pressure?

	IAC 11	IAC 12
A.	NOT available	NOT available
B.	Available	NOT available
C.	NOT available	Available
D.	Available	Available

Proposed Answer: C

Explanation: IAC 12 is powered from PB 17A and remains available. IAC 11 is powered from PB 16A, and is therefore unavailable. Note that IAC 13 is powered from PB 14B and is also unaffected by the loss of PB 16A.

Note: The question meets the K/A by testing knowledge of power supplies to IACs 11 and 12. While there are no air compressors at NMP1 that are specifically referred to as "emergency air compressors", the air compressors are divided into safety-related (IACs 11 and 12) and non-safety-related (IAC 13, Service Air Compressor). The safety-related IACs were utilized for this question to meet the intent of "emergency air compressors", as they are powered from the emergency buses and able to be supplied by the Emergency Diesel Generators.

A. Incorrect – IAC 12 is powered from PB 17A and remains available. Plausible both IAC 11 and 12 are powered by related powerboards.
B. Incorrect – IAC 11 is unavailable. Plausible both IAC 11 and 12 are powered by related powerboards and this would be correct for a loss of PB 17A.
D. Incorrect – IAC 11 is unavailable. Plausible because this would be correct if IAC 11 was supplied by PB 16B instead of PB 16A.

Technical Reference(s): N1-OP-20

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278000-RBO-4

Question Source: Bank - 2010 NRC #3

Question History: 2010 NRC #3

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 11/9/18 – Revised question and answers to avoid calling out IACs 11 and 12 "safety-related", based on NRC comment.
TRH 11/19/18 – Added K/A match statement based on NRC comment.
TRH 11/28/18 – Deleted "if any" based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K3.10
	Importance Rating	2.7

UPS (AC/DC)

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

Proposed Question: #5

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

- UPS 162A was in service when it experienced an internal system fault, tripping the AC and DC input breakers.
- Then, a loss of injection causes a Reactor scram.
- The Reactor Mode Switch is placed in SHUTDOWN.
- Reactor water level lowers to 0".

Which one of the following describes the effect, if any, of these 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. Neither a required Vessel or Containment isolation will occur.

Proposed Answer: A

Explanation: With Reactor water level <+5 inches, both a Vessel and Containment isolation are required. An internal fault on UPS 162A causes RPS 11 to de-energize. This causes two Lo-Lo Rosemount Reactor Water Level Transmitters, LT-36-04A and LT-36-04C to fail low. Vessel and Containment isolations require two instruments to process the isolation. However, it must be an instrument from each RPS channel. LT-36-04A and LT-36-04C are both RPS 11 instruments. Therefore, when UPS 162A is de-energized, a half isolation signal is present. Either Rosemount Reactor Water Level Transmitter from RPS 12 indicating < +5 inches will cause the isolations to occur. Reactor water level lowering to 0 inches satisfies this requirement.

B. Incorrect – The required Vessel isolation will occur as required. Plausible because the loss of UPS results in a partial loss of power to isolation logic.
C. Incorrect – The required Containment isolation will occur as required. Plausible because the loss of UPS results in a partial loss of power to isolation logic.
D. Incorrect – The required Vessel and Containment isolations will occur as required. Plausible because the loss of UPS results in a partial loss of power to isolation logic.

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 - 2015 NRC #25

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K3.02
	Importance Rating	3.2

Shutdown Cooling

Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor water level: Plant-Specific

Proposed Question: #6

The plant is shutdown with the following:

- Shutdown Cooling (SDC) Pump 11 is in service.
- Reactor coolant temperature is 120°F and lowering slowly.

Then, a significant leak from SDC pump 11 suction flange results in the following:

- Reactor water level is 72" and lowering slowly.

If Reactor water level continues to lower, which one of the following describes the Reactor water level at which an isolation signal will ***first*** be generated that will isolate the leak?

A. 65"

B. 53"

C. 5"

D. -10"

Proposed Answer: C

Explanation: 5" Reactor water level will cause a SDC system isolation.

Note: The question meets the K/A because it presents a malfunction of the SDC system (system leak) and requires the applicant to determine the effect on Reactor water level (magnitude of Reactor water level drop before an automatic isolation signal is generated).

A. Incorrect – The low reactor water level alarm will initiate at 65", but no isolation occurs.
B. Incorrect – A Reactor scram and HPCI initiation signal are generated at 53", but SDC isolation does not occur.
D. Incorrect – Inputs to ADS logic are impacted at -10". SDC should have already isolated by this point.

Technical Reference(s): N1-OP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: Bank – SSES LOC28 NRC #6

Question History: SSES LOC28 NRC #6

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 11/9/18 – Revised question slightly ("to" to "that will"), based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 K4.02
	Importance Rating	3.4

Source Range Monitor

Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor SCRAM signals

Proposed Question: #7

A Full Core Reload is in progress with the following:

- All SRMs are fully inserted and indicate on scale <100 cps.
- All IRMs indicate downscale on Range 2.
- REFUEL INST TRIP BYPASS CH 11 and 12 keylock switches are in NONCOINCIDENT.
- 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: B

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 NON-COINCIDENT, the SRM upscale scram is NOT bypassed for all SRMs. The SRM upscale rod block (>1×10^5 cps) is also active. Therefore, SRM 14 failing upscale causes a rod block, and a scram.

A. Incorrect – A scram occurs. Plausible because if the keylock switches were in COINCIDENT, SRM 14 failing upscale would NOT cause a scram.
C. Incorrect – SRM 11 does not cause a rod block. Plausible because if SRM 11 were partially withdrawn, its downscale failure would cause a rod block. A scram occurs. Plausible because if the keylock switches were in COINCIDENT, SRM 14 failing upscale would NOT cause a scram.
D. Incorrect – SRM 11 does not cause a rod block. Plausible because if SRM 11 were partially withdrawn, its downscale failure would cause a rod block.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Modified Bank – 2015 NRC #6

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 K4.03
	Importance Rating	3.1

AC Electrical Distribution

Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Interlocks between automatic bus transfer and breakers

Proposed Question: #8

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

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

Which one of the following describes the electrical distribution system response?

Powerboards 11 and 12 __(1)__ transfer to Reserve power based on a(n) __(2)__ signal.

	(1)	(2)
A.	slow	undervoltage
B.	slow	Generator trip
C.	fast	undervoltage
D.	fast	Generator trip

Proposed Answer: D

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

A. Incorrect – Slow transfer will NOT occur. Plausible because this would occur if degraded voltage conditions were present. Undervoltage will NOT occur. Plausible because this would be correct if the Generator trip did directly cause fast transfer, such that undervoltage occurred before transfer.
B. Incorrect – Slow transfer will NOT occur. Plausible because this would occur if degraded voltage conditions were present.
C. Incorrect – Undervoltage will NOT occur. Plausible because this would be correct if the Generator trip did directly cause fast transfer, such that undervoltage occurred before transfer.

Technical Reference(s): N1-OP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO 5

Question Source: Bank - 2013 NRC #4

Question History: 2013 NRC #4

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

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

RPS

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

Proposed Question: #9

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All IRMs are mid-scale on Range 8.
- All MSIVs are open.
- Reactor pressure is 550 psig.

Then, an RO moves IRM 11 to Range 10 and IRM 18 to Range 10.

Which one of the following describes the plant response and the reason for this response?

All MSIVs __(1)__ and the Reactor __(2)__ scram.

A. (1) close
(2) will

B. (1) close
(2) will NOT

C. (1) remain open
(2) will

D. (1) remain open
(2) will NOT

Proposed Answer: A

Explanation: IRM 11 is in RPS channel 11 and IRM 18 is in RPS channel 12. The bypass of the <850 psig MSIV closure is interrupted in each RPS channel by taking an IRM to range 10. Therefore the MSIVs will close. The <600 psig bypass of the MSIV closure scram is also interrupted by the IRMs in range 10. Therefore a scram will occur.

B. Incorrect – The Reactor will scram. Plausible because initially the MSIV closure scram is bypassed due to the combination of IRM switch positions, Reactor Mode Switch position, and Reactor pressure.
C. Incorrect – MSIVs close. Plausible because initially the low Reactor pressure MSIV closure is bypassed due to the combination of IRM switch positions and Reactor Mode Switch position.
D. Incorrect – MSIVs close. Plausible because initially the low Reactor pressure MSIV closure is bypassed due to the combination of IRM switch positions and Reactor Mode Switch position. The Reactor will scram. Plausible because initially the MSIV closure scram is bypassed due to the combination of IRM switch positions, Reactor Mode Switch position, and Reactor pressure.

Technical Reference(s): N1-OP-38B, N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: Bank - 2009 NRC #35

Question History: 2009 NRC #35

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:
TRH 11/9/18 – Lowered Reactor pressure, reworded IRM descriptions, and reworded all answer choices based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K5.01
	Importance Rating	2.6

IRM

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

Proposed Question: #10

A Reactor startup is in progress with all IRMs on Range 6.

Then, an RO inadvertently ranges IRM 14 (currently at 50% of full scale) down to range 5.

Which one of the following describes the plant response to this action, if any?

A. A rod block occurs, only.

B. A half scram occurs, only.

C. A half scram and a rod block occur.

D. NEITHER a half scram NOR a rod block occur.

Proposed Answer: C

Explanation: Shifting an IRM from range 6 to range 5 will cause the indication to raise by a factor of approximately 3 (square root of 10) on the 0-125 scale. Since the IRM is already 50% of scale, the IRM Hi setpoint of 107.5/125 (88% full scale) would be exceeded causing a rod block and the Hi Hi setpoint of 117.5/125 (96% full scale) would be exceeding causing a scram signal.

Note: The question meets the K/A by testing the operational implication (system response to an operator action) of an aspect of IRM detector operation (how detector signal is processed to allow use across wide range of power levels).

A. Incorrect – A rod block does occur due to exceeding the IRM Hi setpoint. However, a half scram signal is also processed.
B. Incorrect – A half scram signal is generated due to exceeding the IRM Hi Hi setpoint. However a rod block is also generated.
D. Incorrect – IRM 14 is initially operating within its limits to prevent causing a rod block or half scram. However, ranging down on the IRM with cause both thresholds to be exceeded.

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

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:
TRH 11/9/18 – Lowered IRM indication and added K/A match statement, based on NRC comment.
TRH 11/19/18 – Lowered IRM indication further based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	207000 K6.04
	Importance Rating	3.2

Isolation (Emergency) Condenser

Knowledge of the effect that a loss or malfunction of the following will have on the ISOLATION (EMERGENCY) CONDENSER: Plant air systems: BWR-2,3

Proposed Question: #11

The plant was manually scrammed following a complete loss of Instrument Air pressure.

Which one of the following actions is required, in accordance with N1-SOP-20.1, Instrument Air Failure?

A. Return Reactor Building Ventilation to service and secure RBEVS.

B. Control Reactor water level by throttling the Feedwater Isolation Valves.

C. Manually operate the EC Steam Supply Valves to control cooldown rate.

D. Start the Mechanical Vacuum Pump to maintain vacuum in the Main Condenser.

Proposed Answer: C

Explanation: N1-SOP-20.1 directs manually closing the EC Steam Supply Valves since both EC Condensate Return Valves 39-05 and 39-06 fail OPEN on the loss of air. Failing to manually control the EC steam supply valves would result in violating reactor cooldown rate limits.

A. Incorrect – The Reactor Building Ventilation isolation valves fail closed on loss of air. They are not available to restore normal Reactor Building Ventilation. N1-SOP-20.1 requires the use of RBEVS.
B. Incorrect – Although the FCVs have failed as is, RPV level is controlled by starting and stopping the feedwater pumps as necessary or pinning the feedwater flow control valves and operating them locally.
D. Incorrect – Main condenser vacuum will be degrading, however there is no requirement to start the Mechanical Vacuum Pump at this time.

Technical Reference(s): N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-8

Question Source: Bank - 2008 NRC #16

Question History: 2008 NRC #16

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:
TRH 11/9/18 – Revised wording of choice D based on NRC comment.

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	261000 K6.03
Importance Rating	3.0

SGTS

Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: Emergency diesel generator system

Proposed Question: #12

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

- RBEVS fan 11 is tagged out for maintenance.
- An un-isolable steam leak develops from Reactor Water Cleanup into the Reactor Building.
- Reactor Building Ventilation exhaust radiation monitors are reading 30 mR/hr and slowly rising.
- Reactor Building differential pressure is -0.05 psig and stable.

Then, a loss of offsite power occurs with the following:

- EDG 102 re-energizes PB 102.
- EDG 103 fails to start.

Which one of the following describes the effect, if any, of the loss of offsite power and EDG response on Reactor Building differential pressure (D/P) and off-site release rate?

	Reactor Building D/P	Off-site Release Rate
A.	Maintained negative	Remains unchanged
B.	Maintained negative	Rises
C.	NOT maintained negative	Remains unchanged
D.	NOT maintained negative	Rises

Proposed Answer: D

Explanation: Since Reactor Building Ventilation exhaust radiation monitors are greater than 5 mR/hr, normal Reactor Building Ventilation has tripped and RBEVS has auto-started. Since RBEVS fan 11 is tagged out, only RBEVS fan 12 is initially maintaining RB D/P. With the loss of offsite power and failure of EDG 103 to start, RBEVS fan 12 is de-energized. Nothing remains available to maintain RB D/P negative. The un-isolable steam leak into the building will therefore make D/P go positive. With positive RB D/P and elevated building airborne rad levels from the steam leak, this leads to a ground level release and a rise in off-site release rate.

A. Incorrect – RB D/P goes positive. Plausible because this would be correct if RB vent exhaust was <5 mR/hr. Off-site release rate rises due to an uncontrolled ground level release from the Reactor Building because the Reactor Building is NOT designed to be a zero-leakage containment without ventilation. Plausible because this would be correct if Secondary Containment were designed to be zero-leakage, such as the Primary Containment.
B. Incorrect – RB D/P goes positive. Plausible because this would be correct if RB vent exhaust was <5 mR/hr.
C. Incorrect – Off-site release rate rises due to an uncontrolled ground level release from the Reactor Building because the Reactor Building is NOT designed to be a zero-leakage containment without ventilation. Plausible because this would be correct if Secondary Containment were designed to be zero-leakage, such as the Primary Containment.

Technical Reference(s): N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-5

Question Source: Modified Bank – 2013 NRC #44

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:
TRH 11/9/18 – Revised question slightly and changed 1st half choices slightly, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A1.03
	Importance Rating	3.6

SLC

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

Proposed Question: #13

A failure to scram has occurred with the following:

- A Reactor Operator has placed the Liquid Poison keylock switch to the SYS 11 position.
- Reactor pressure is 1020 psig and stable.
- Liquid Poison header pressure is 1115 psig.
- The LIQUID POISON EXPL VALVE 11 CONTINUITY light is illuminated.
- The LIQUID POISON EXPL VALVE 12 CONTINUITY light is extinguished.

Which one of the following describes the status of the Liquid Poison system?

Liquid Poison is...

A. injecting at full design flow.

B. injecting at approximately half of full design flow.

C. NOT injecting. Initiating SYS 12 will result in injection.

D. NOT injecting. Initiating SYS 12 will NOT result in injection.

Proposed Answer: A

Explanation: Reactor pressure and Liquid Poison discharge pressure are normal for injection. Each explosive valve is rated for 100% flow. Even though the explosive valve for system 11 did not fire, full flow will be achieved through explosive valve 12 via a cross-connect line.

B. Incorrect – Full design flow is present. Plausible because only one explosive valve fired.
C. Incorrect – Liquid Poison is injecting. Plausible because only one explosive valve fired. Also plausible because this would be correct if a lower discharge pressure were given, indicating pump 11 was malfunctioning.
D. Incorrect – Liquid Poison is injecting. Plausible because only one explosive valve fired. Also plausible if indication was given of a flow blockage (higher discharge pressure at relief valve setpoint).

Technical Reference(s): N1-OP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO 3

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 #	215005 A1.07
	Importance Rating	3.0

APRM / LPRM

Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: APRM (gain adjustment factor)

Proposed Question: #14

The plant is operating at 99% power, with the following conditions:

- FPAPDR has been determined to be 1.01
- The Shift Manager has directed an APRM Gain Adjustment in accordance with N1-REP-12
- The nominal APRM setting required due to the high FPAPDR has been calculated as 100%

When adjusting the APRM amplifier gain, which one of the following APRM meter readings is an acceptable value for a successful gain adjustment?

A. 97%

B. 99%

C. 101%

D. 103%

Proposed Answer: C

Explanation: In accordance with N1-REP-12, the acceptable reading must be +2% to -0% of the nominal value given. Therefore the desired value is between 100 and 102. Therefore 101% is a successful gain adjustment.

A. Incorrect – This is 3% off nominal. 3% is the allowable APRM margin where a gain adjustment is required for rod blocks in accordance with N1-REP-12.
B. Incorrect – The would be accurate for +0 to -2%.
D. Incorrect – This is 3% off nominal. 3% is the allowable APRM margin where a gain adjustment is required for rod blocks in accordance with N1-REP-12.

Technical Reference(s): N1-REP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-10

Question Source: Bank - 2008 NRC #20

Question History: 2008 NRC #20

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 #	239002 A2.01
	Importance Rating	3.0

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 vacuum breakers

Proposed Question: #15

A plant transient results in the following:

- ERV 111 opens due to high Reactor pressure and then closes.
- An ERV 111 vacuum breaker opens and sticks in the open position.

Which one of the following describes the consequence of the stuck open ERV vacuum breaker?

A. Steam will continue to flow from the Reactor to the Torus through the open ERV vacuum breaker.

B. If a LOCA occurs, Torus and Drywell pressure will equalize, degrading containment pressure suppression capability.

C. If ERV 111 opens again, some of the steam passing through the ERV will be released directly into the Torus airspace.

D. If ERV 111 opens again, some of the steam passing through the ERV will be released directly into the Drywell airspace.

Proposed Answer: D

Explanation: After ERV operation, the vacuum breakers open to equalize pressure between the Drywell and tailpipes. Without vacuum breaker operation, condensation of steam in the tailpipe draws water from the Torus up into the tailpipe. Upon subsequent re-opening of the ERV, high forces would be experienced due to the clearing of the extra water from the tailpipe. With a stuck open vacuum breaker, subsequent ERV opening would admit steam directly to the Drywell airspace, resulting in rising Drywell temperature and pressure.

Note: The question addresses only the first part of the K/A due to limited ability to correct, control, or mitigate this malfunction (there are no blocking valves or controls for the ERV vacuum breakers) and testing higher levels of mitigation would result in crossing into SRO level knowledge.

A. Incorrect – The closed ERV isolates the vacuum breaker from the Reactor. The vacuum breaker is connected between the ERV discharge piping and the Drywell air space, not the Reactor. Plausible if candidate believes ERV vacuum breakers connect ERV discharge piping to the Reactor.
B. Incorrect – The ERV tailpipe vacuum breakers connect to the Drywell airspace, but do not provide a direct connection between the Torus and Drywell airspaces. Plausible because this would be correct for a stuck open Torus-to-Drywell vacuum breaker.
C. Incorrect – The SRV tailpipe vacuum breakers connect to the Drywell airspace, not Torus. Plausible if candidate believes the ERV vacuum breakers connect ERV discharge piping to the Torus air space.

Technical Reference(s): C-18002-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-2

Question Source: Bank - 2017 Cert #16

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 11/9/18 – Replaced choice B to raise plausibility, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 A2.01
	Importance Rating	4.1

ADS

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Small steam line break LOCA

Proposed Question: #16

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

- An un-isolable steam leak in the Turbine Building has led to the need for an RPV Blowdown due to off-site release rate.
- Battery Board 12 is de-energized due to a sustained electrical fault from the effects of the steam in the Turbine Building.

Which one of the following describes the effect of the battery board 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 is __(1)__ and the use of alternate Blowdown Systems to rapidly depressurize the Reactor is __(2)__.

	(1)	(2)
A.	available	allowed, but NOT required
B.	available	NEITHER required NOR allowed
C.	NOT available	allowed, even if isolations must be defeated.
D.	NOT available	allowed, but only if isolations must NOT be defeated.

Proposed Answer: C

Explanation: The Minimum Number of ERVs Required for Emergency Depressurization is 4. With battery board 12 de-energized, three of the six ERVs are 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.

Note: The question meets the K/A by presenting a steam line break and requiring the candidate to determine the impact on the ADS system (availability of ERVs due to electrical fault from steam leak) and, based on that determination, determine the required action in a procedure (N1-EOP-8).

A. Incorrect – The Minimum Number of ERVs Required for Emergency Depressurization is 4. With 3 ERVs unavailable, only 3 remain available. Plausible because the Minimum Number of ERVs Required for Emergency Depressurization used to be 3.
B. Incorrect – The Minimum Number of ERVs Required for Emergency Depressurization is 4. With 3 ERVs unavailable, only 3 remain available. Plausible because the Minimum Number of ERVs Required for Emergency Depressurization used to be 3.
D. Incorrect – 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. Plausible that isolations would not be allowed to be defeated since 3 ERVs are available and will lower Reactor pressure.

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: Modified Bank – 2017 Cert #6

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/9/18 – Revised 2nd half of C and D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 A3.01
	Importance Rating	3.4

PCIS/Nuclear Steam Supply Shutoff

Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: System indicating lights and alarms

Proposed Question: #17

The plant was operating at 100% power when the following annunciators alarmed:

- A3-4-1, BAT. BD. 11 BATTERY BREAKER TRIP
- A3-3-1, SBC 161 A-B DC BKR TRIP

An operator reports the following indications from the A3 panel:

- Battery 11 DC ammeter indicates 0 amps.
- Battery 11 voltmeter indicates 130 volts.

Which one of the following states the impact on the MSIVs?

A. All MSIVs close.

B. Only the outboard MSIVs close.

C. All MSIVS remain open, but auto isolation capability is lost.

D. All MSIVs remain open, but some position indication is lost.

Proposed Answer: D

Explanation: The indications given indicate a loss of battery board 11. Battery board 11 supplies power to the indicating lights for MSIV 112 (01-03) and MSIV 122 (01-04). At rated conditions, both red indicating lights should be lit. When battery board 11 is de-energized, both red indicating lights extinguish.

Note: The K/A defines "system indicating lights and alarms" as an automatic operation of the PCIS/NSSS system. The question meets the K/A by testing the candidates ability to monitor these automatic operations (electrical source alarms related to PCIS/NSSS, knowledge of corresponding MSIV indications vs. actual position).

A. Incorrect – The station batteries do supply power to the MSIV automatic isolation logic, however there are several solenoids and pilot valves for each MSIV. For the MSIVs to auto close, multiple solenoids must be de-energized. Therefore all MSIVs remain open.
B. Incorrect – The station batteries do supply power to the MSIV automatic isolation logic, however there are several solenoids and pilot valves for each MSIV. For the MSIVs to auto close, multiple solenoids must be de-energized. Therefore all MSIVs remain open.
C. Incorrect – All MSIVs do remain open. A loss of battery board 11 does impact the auto isolation function. However, rather than prevent, it provides half of the signal necessary to cause an isolation.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

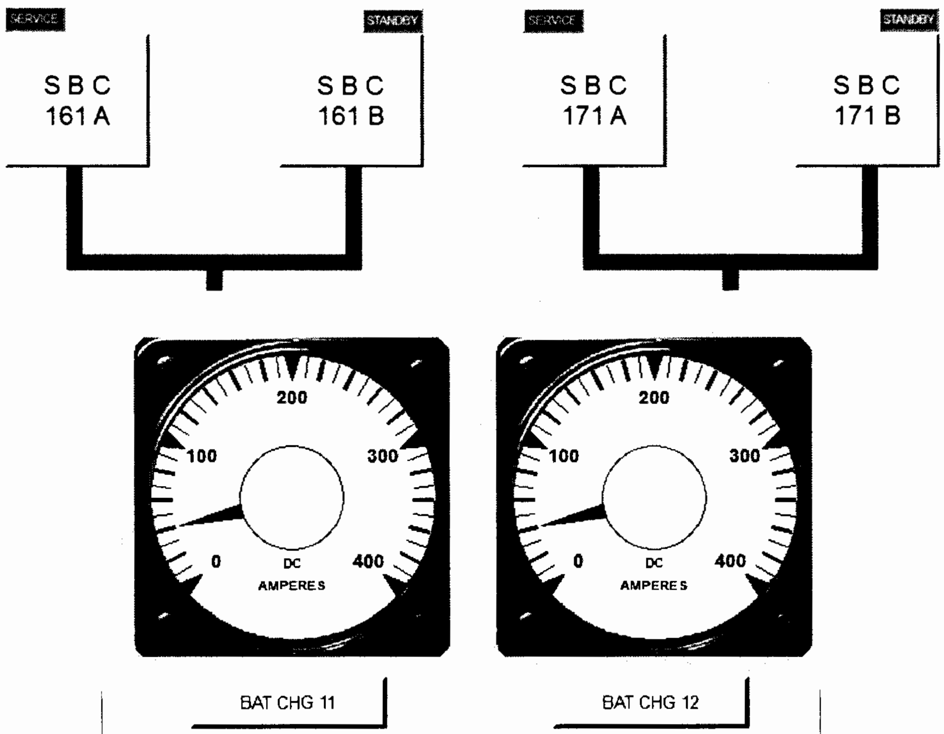
Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 A3.01
	Importance Rating	3.2

DC Electrical Distribution

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

Proposed Question: #18

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



Then, Powerboard 16B is inadvertently de-energized and then re-energized.

Which one of the following describes the status of Batteries 11 and 12 two (2) minutes after Powerboard 16B is re-energized?

	Battery 11	Battery 12
A.	Discharging	Discharging
B.	Discharging	Charging
C.	Charging	Discharging
D.	Charging	Charging

Proposed Answer: D

Explanation: The indications show batteries 11 and 12 being charged by SBCs 161A and 171A respectively. Approximately 90 seconds after AC power is restored to SBC161A (from PB 16B) the SBC will align itself to the Battery 11 and restore the normal float charge. Battery 12 is unaffected by the transient since its SBC 171A receives power from PB 17B, which is unaffected by the loss of PB 16B.

A. Incorrect – Battery 11 will be charging. Plausible because this would be correct at an earlier time during the transient. Battery 12 will be charging. Plausible because this would be correct for a loss of PB 17B before 100 seconds elapsed.
B. Incorrect – Battery 11 will be charging. Plausible because this would be correct at an earlier time during the transient. Battery 12 will be charging.
C. Incorrect – Battery 12 will be charging. Plausible because this would be correct for a loss of PB 17B before 100 seconds elapsed.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-3

Question Source: Bank - 2010 NRC #13

Question History: 2010 NRC #13

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 #	264000 A4.04
	Importance Rating	3.7

EDGs

Ability to manually operate and/or monitor in the control room: Manual start, loading, and stopping of emergency generator: Plant-Specific

Proposed Question: #19

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

- N1-ST-M4A, Emergency Diesel Generator 102 and PB 102 Operability Test, is in progress.
- EDG 102 has been operating at the correct test load for 30 minutes.

Which one of the following describes the response of EDG 102 parameters if the EDG 102 governor control switch is momentarily placed in the LOWER position?

A. Frequency lowers

B. Kilowatts lower

C. Voltage lowers

D. KVARs lower

Proposed Answer: B

Explanation: N1-ST-M4A starts EDG 102, parallels it to PB 102, and then operates it loaded at 2650-2750 KW and 300-800 KVARs for at least 60 minutes. PB 102 remains connected to offsite power during the test, which places the EDG in parallel with another power source. Placing the governor control switch to lower will lower EDG kilowatts since there is another electric power source to transfer load to. EDG frequency will remain constant because the offsite power grid keeps bus frequency constant.

A. Incorrect – Frequency remains unchanged. Plausible because this would be correct when not operating in parallel with another source.
C. Incorrect – Voltage remains unchanged. Plausible because this would be correct for lowering on the voltage regulator when not paralleled with another power source.
D. Incorrect – KVARs remain unchanged. Plausible because this would be correct for lowering on the voltage regulator.

Technical Reference(s): N1-ST-M4A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264000-RBO 5

Question Source: Modified – 2010 NRC #20

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 11/9/18 – Modified to have N1-ST-M4A in progress to test more than GFES knowledge, based on NRC comment.
TRH 11/19/18 – Rearranged order of answer choices based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A4.01
	Importance Rating	3.8

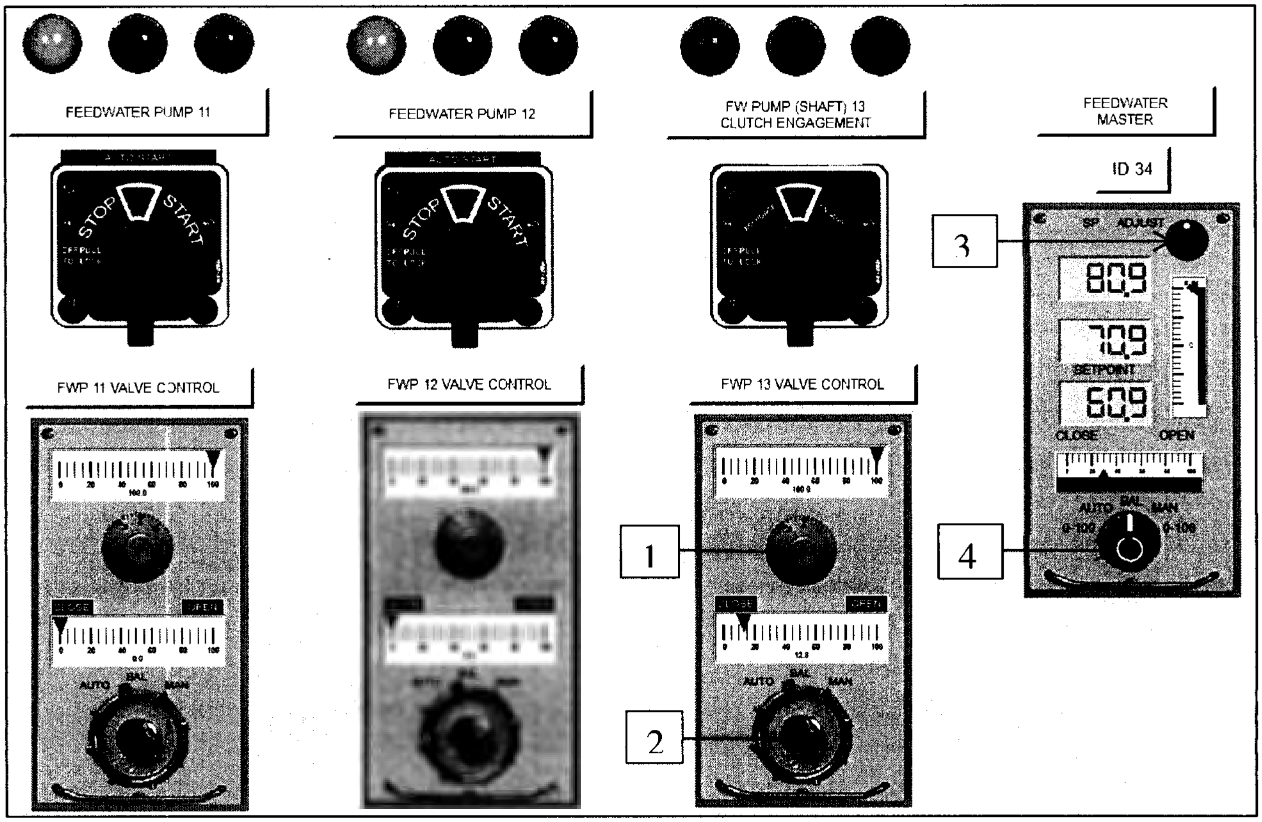
Reactor Water Level Control

Ability to manually operate and/or monitor in the control room: All individual component controllers in the manual mode

Proposed Question: #20

The plant is operating at 20% power.

The status of select portions of the Feedwater system is shown in the following pictures, with potentiometers marked (1), (2), (3), and (4):



Which one of the following identifies the control used to change Reactor water level with the current system lineup?

A. (1)

B. (2)

C. (3)

D. (4)

Proposed Answer: B

Explanation: The given pictures show Feedwater pump 13 operating with its flow controller in manual and the Feedwater Master flow controller in balance. Since the Feedwater pump 13 flow controller is in manual, the Feedwater Master is not used to change Reactor water level (labels (3) and (4)). The Feedwater pump 13 flow control valve adjustment knob (label (2)) is the correct control to change Reactor water level in this alignment.

A. Incorrect – The Feedwater pump 13 flow control valve adjustment knob (label (2)) is the correct control to change Reactor water level in this alignment, not upper knob on the same controller (label (1)). Label (1) adjusts controller bias, but is not used to control level.
C. Incorrect – The Feedwater pump 13 flow control valve adjustment knob (label (2)) is the correct control to change Reactor water level in this alignment, not label (3). Label (3) would be correct in a different alignment with Feedwater pump 13 flow control in AUTO or BAL.
D. Incorrect – The Feedwater pump 13 flow control valve adjustment knob (label (2)) is the correct control to change Reactor water level in this alignment, not label (4). Label (4) would be correct in a different alignment with Feedwater pump 13 flow control in AUTO or BAL and the Feedwater Master in MAN.

Technical Reference(s): N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-5

Question Source: Bank - 2017 Cert #19

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:
TRH 11/9/18 – Revised question to raise plausibility of 2 distractor, based on NRC comment.
TRH 11/19/18 – Revised picture to include FWPs 11 and 12 again based on NRC comment.

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	300000 2.1.30
Importance Rating	4.4

Instrument Air

Ability to locate and operate components, including local controls.

Proposed Question: #21

An outage is in progress with the following:

- Instrument Air Compressors (IAC) 11 and 13 are in service.
- IAC 12 control switch is tagged in pull-to-lock for maintenance.
- Instrument Air Dryer (IAD) 94-168 is in service.
- Breathing Air is being used to support maintenance activities.
- Subsequently, IAC 11 trips and its control switch is placed in pull-to-lock.
- The US directs bypass of IAD 94-168 and IAD 94-169.

Which one of the following identifies the location for bypassing these IADs and the required compensatory action while they are out of service, in accordance with N1-OP-20, Service, Instrument and Breathing Air Systems?

	Location for Bypassing IAD 94-168 and 94-169	Compensatory Action
A.	TB el. 261'	Secure use of Breathing Air to reduce system load
B.	TB el. 261'	Blowdown air manifolds daily
C.	TB el. 291'	Secure use of Breathing Air to reduce system load
D.	TB el. 291'	Blowdown air manifolds daily

Proposed Answer: D

Explanation: The actions to bypass these IADs are performed on TB el. 291'. With these IADs bypassed, N1-OP-20 H.3.0 requires blowing down selected air manifolds once per day per Attachment 4.

A. Incorrect – The location is TB el. 291'. Plausible because many other IA components are located on TB el. 261'. The required compensatory action is to blow down air manifolds once per day. Plausible because securing use of the breathing air header is an action in N1-SOP-20.1 and would also assist in minimizing Instrument Air usage while drying capacity is limited.
B. Incorrect – The location is TB el. 291'. Plausible because many other IA components are located on TB el. 261'.
C. Incorrect – The required compensatory action is to blow down air manifolds once per day. Plausible because securing use of the breathing air header is an action in N1-SOP-20.1 and would also assist in minimizing Instrument Air usage while drying capacity is limited.

Technical Reference(s): N1-OP-20

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-10

Question Source: Modified Bank – 2009 NRC #52

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 11/9/18 – Added to justifications based on NRC comment.
TRH 11/19/18 – Added more to justification based on NRC comment.
TRH 11/28/18 – Revised 2nd half of A and C based on NRC comment.
TRH 11/28/18 – Changed to outage with Breathing Air in use and added reason to A and C, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 2.1.32
	Importance Rating	3.8

APRM / LPRM

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

Proposed Question: #22

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

- Two LPRM D level detectors for APRM 16 are failed and they are bypassed.
- Then, a third LPRM D level detector for APRM 16 fails downscale.

Which one of the following describes the consequence of placing the third LPRM detector switch to BYPASS?

A. APRM 16 generates a trip because of insufficient number of D level detectors.

B. APRM 16 generates a trip because of insufficient number of total detector inputs.

C. APRM 16 is inoperable by Technical Specifications because of insufficient number of D level detectors. APRM 16 does NOT generate a trip.

D. APRM 16 is inoperable by Technical Specifications because of insufficient number of total detector inputs. APRM 16 does NOT generate a trip.

Proposed Answer: C

Explanation: Per N1-OP-38C Precaution and Limitation D.1.1 No more than two C or D level inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed for the APRM to be considered operable. No more than one of the four APRM inputs to each trip system shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. Per T.S. Table 3.6.2.a note e, No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed in order for the APRM to be considered operable.

A. Incorrect – There is an insufficient number of D level detectors, but it does not generate an automatic trip signal.
B. Incorrect – While there are insufficient D level detectors, there is no indication that the total number of LPRM inputs are not met. With only three LPRMs inoperative five inputs remain operable.
D. Incorrect – There is no indication that the total number of LPRM inputs is not met. With only three LPRMs inoperative five inputs remain operable. Only four LPRM inputs are required to maintain APRM operability based on total number of inputs.

Technical Reference(s): N1-OP-38C P&L 1.1, ARP F2-1-6, T.S. Table 3.6.2.a note e

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-9

Question Source: Bank - 2010 Cert #25

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:
TRH 11/9/18 – Made slight revisions to answer choices to avoid possible subset issue, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 K1.07
	Importance Rating	3.6

SRVs

Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: Suppression Pool

Proposed Question: #23

Which one of the following describes the reason for performing an RPV Blowdown due to low Torus water level, in accordance with NER-1M-095, NMP1 Emergency Operation Procedures and Severe Accident Procedures Basis Document?

A. To prevent steam from discharging directly into the Torus airspace if ERVs are opened and challenging the Primary Containment Pressure Limit.

B. To ensure ERV operability is maintained during the blowdown and that the Heat Capacity Temperature Limit is not exceeded.

C. To ensure Core Spray operability is maintained during the blowdown to provide low pressure injection to the Reactor vessel.

D. To prevent loss of adequate Containment Spray pump net positive suction head while the Reactor remains at pressure.

Proposed Answer: A

Explanation: Per NER-1M-095: “Torus water level must be maintained above 8 ft. to ensure that all openings in the ERV discharge devices remain submerged. If torus water level is below the elevation of the discharge holes, opening ERVs would discharge steam directly into the torus airspace. The resulting pressure increase could exceed the maximum pressure capability of the primary containment. Since the RPV may not be kept at pressure under these conditions, a blowdown is required.”

B. Incorrect – Damage to the Primary Containment if ERVs open is the concern. Plausible because when level goes below 8’, the ERV discharges in the Torus become uncovered, which does affect the ability to use the ERVs.
C. Incorrect – Damage to the Primary Containment if ERVs open is the concern. Plausible because low Torus water level does make Core Spray operability a concern (NPSH).
D. Incorrect – Damage to the Primary Containment if ERVs open is the concern. Plausible because low Torus water level does make Containment Spray operability a concern (NPSH).

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 - 2017 Cert #57

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/9/18 - Changed “reason” to “basis for this level” in question based on NRC comment.
TRH 11/14/18 – Replaced question due to overlap issue with operating exam.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K4.07
	Importance Rating	4.3

HPCI

Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Automatic system initiation: BWR-2,3,4

Proposed Question: #24

The plant was manually scrammed due to a steam leak in the primary containment. Current plant conditions are as follows:

- Drywell pressure is 5 psig and rising.
- Feedwater pumps 11 and 12 have tripped on high Reactor water level.
- Reactor water level is 96" and lowering slowly.
- Reactor pressure is 920 psig and being controlled automatically by the EPR.

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

A. Feedwater pumps 11 and 12 will automatically start and inject when Reactor water level reaches 53".

B. Feedwater pumps 11 and 12 will automatically start and inject when Reactor water level reaches 90".

C. Feedwater pumps 11 and 12 will not restart automatically until the operator manually resets the high level trip.

D. Feedwater pump 12 will automatically start and inject when Reactor Level reaches 53". Feedwater pump 11 will NOT automatically start.

Proposed Answer: A

Explanation: Both MDFWPs will receive an auto start signal when reactor water level reaches 53", regardless of the status of the reactor water high level trips being reset.

B. Incorrect – Both MDFWPs will receive an auto start signal, but the level is 53", not 90".
C. Incorrect – Neither pump would be able to be started manually without resetting the high level trips. However the high level trip reset would not prevent HPCI starts. Both MDFWPs would start automatically.
D. Incorrect – FWP 12 is the preferred HPCI pump and would start automatically. However FWP 11 would also start.

Technical Reference(s): N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-206000-RBO-5

Question Source: Bank - 2009 NRC #30

Question History: 2009 NRC #30

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 #	261000 A2.07
	Importance Rating	2.7

SGTS

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

Proposed Question: #25

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

- Emergency Diesel Generator (EDG) 102 is inoperable due to a generator ground.
- Reactor Building Emergency Ventilation (RBEVS) automatically starts due to a valid signal.
- It is desired to secure one train of RBEVS.
- The automatic start signal for RBEVS is still present.

Which one of the following identifies the train of RBEVS that should be secured and the requirements for securing the fan, in accordance with N1-OP-10, Reactor Building Heating, Cooling, and Ventilating System?

Secure RBEVS train...

A. 11. Place the fan control switch in Pull-To-Lock.

B. 11. Place the fan control switch in normal-after-stop.

C. 12. Place the fan control switch in Pull-To-Lock.

D. 12. Place the fan control switch in normal-after-stop.

Proposed Answer: A

Explanation: N1-OP-10 section H.2.0 provides the direction for response to automatic start of RBEVS, including the necessary steps to secure one train of RBEVS. With the RBEVS automatic initiation signal still present, control switch for the RBEVS fan to be secured will have to be placed in pull-to-lock. This both secures the fan and makes it inoperable. With an inoperable EDG, it is preferred to secure the RBEVS fan that is associated with the inoperable EDG. Therefore, since EDG 102 is inoperable, RBEVS fan 11 should be secured (powered from PB/EDG 102 through PB 16B to PB 161B).

B. Incorrect – The control switch must be placed in PTL. Plausible because this would be correct if the automatic start signal were not still present or if the logic were different.
C. Incorrect – Train 11 is the preferred train to secure. Plausible this would be correct if EDG 103 were inoperable instead of EDG 102.
D. Incorrect – Train 11 is the preferred train to secure. Plausible this would be correct if EDG 103 were inoperable instead of EDG 102. The control switch must be placed in PTL. Plausible because this would be correct if the automatic start signal were not still present or if the logic were different.

Technical Reference(s): N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N-261000-RBO-10

Question Source: Modified Bank – 2017 Cert #87

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:
TRH 11/19/18 – Deleted 2 bullets to avoid cueing question #12, based on NRC comment.
TRH 11/28/18 – Revised 2nd bullet to tie responses to a valid signal, based on NRC comment.
TRH 11/28/18 – Deleted 1st half of 2nd bullet based on NRC comment.

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

DC Electrical Distribution

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

Proposed Question: #26

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

- Lines 1 and 4 de-energize.
- Emergency Diesel Generators (EDGs) 102 and 103 start and re-energize Powerboards 102 and 103, respectively.

Then, Battery Board 12 de-energizes due to an electrical fault.

Which one of the following describes the impact of this DC power loss of the EDGs?

A. EDG 102 trips.

B. EDG 103 trips.

C. EDG 102 remains running, but remote tripping capability is lost.

D. EDG 103 remains running, but remote tripping capability is lost.

Proposed Answer: D

Explanation: When Battery Board 12 de-energizes, EDG 103 loses power to the shutdown solenoid. This is an energize-to-function solenoid required to trip the EDG from the control room.

A. Incorrect – EDG 102 is unaffected. Plausible because EDG 102 would be affected by the loss of the related Battery Board 11.
B. Incorrect – EDG 103 does not trip. Plausible because this does result in a loss of control power to EDG 103 and would be correct if the trip solenoid was de-energize-to-function, as many solenoids in the plant are.
C. Incorrect – EDG 102 is unaffected. Plausible because EDG 102 would be affected by the loss of the related Battery Board 11.

Technical Reference(s): N1-OP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264000-RBO-8

Question Source: Bank – 2013 Cert #49

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 #	295001 AA2.02
	Importance Rating	3.1

Partial or Complete Loss of Forced Core Flow Circulation

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

Proposed Question: #27

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: 14 Mlbm/hr
  - RRP 14: 14 Mlbm/hr
  - RRP 15: 14 Mlbm/hr
- APRMs stabilize at 66%.
- Main Condenser vacuum is 27" Hg and stable.

Note: Power-to-flow maps are provided on the following pages.

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?

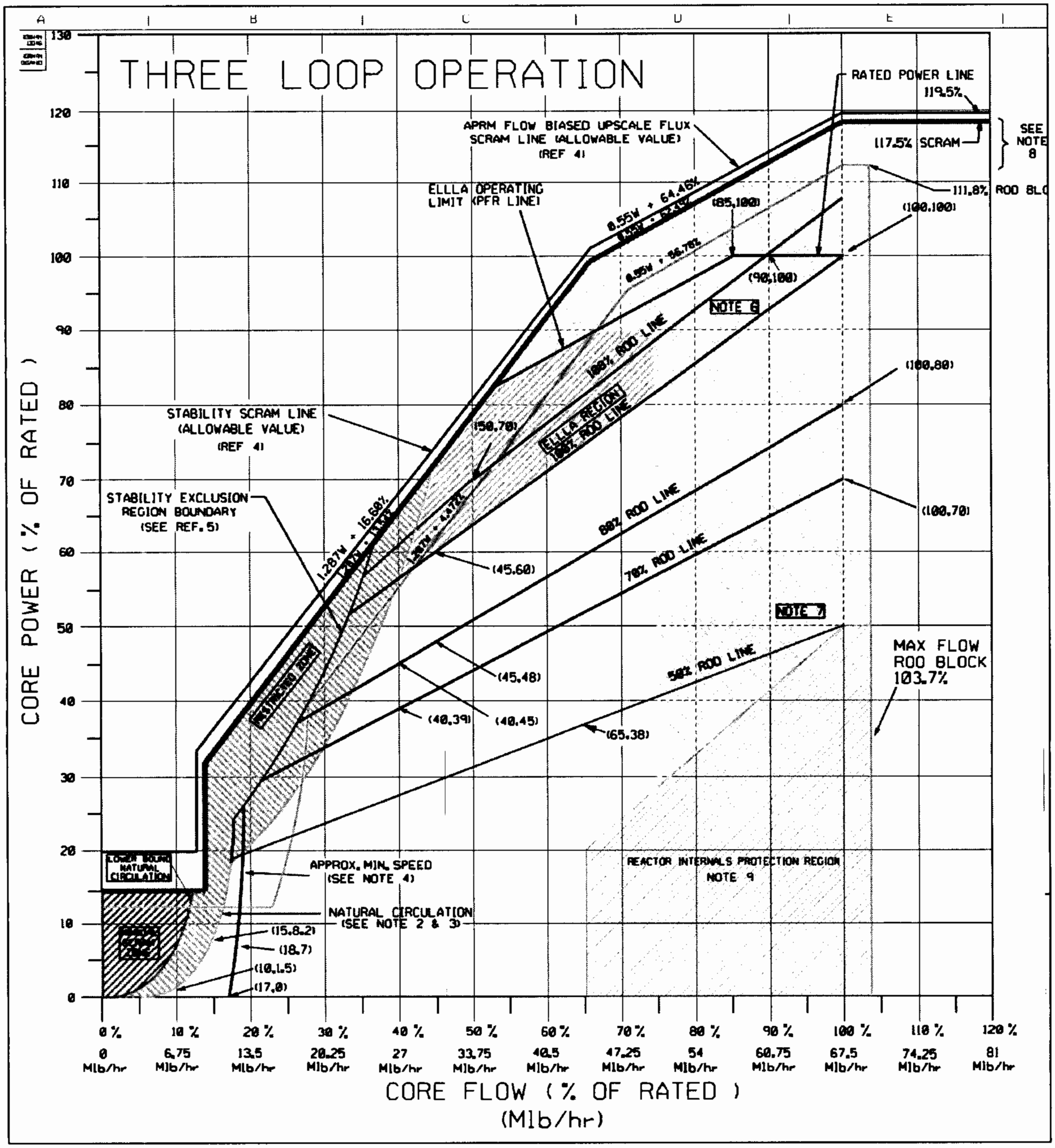
The plant is operating...

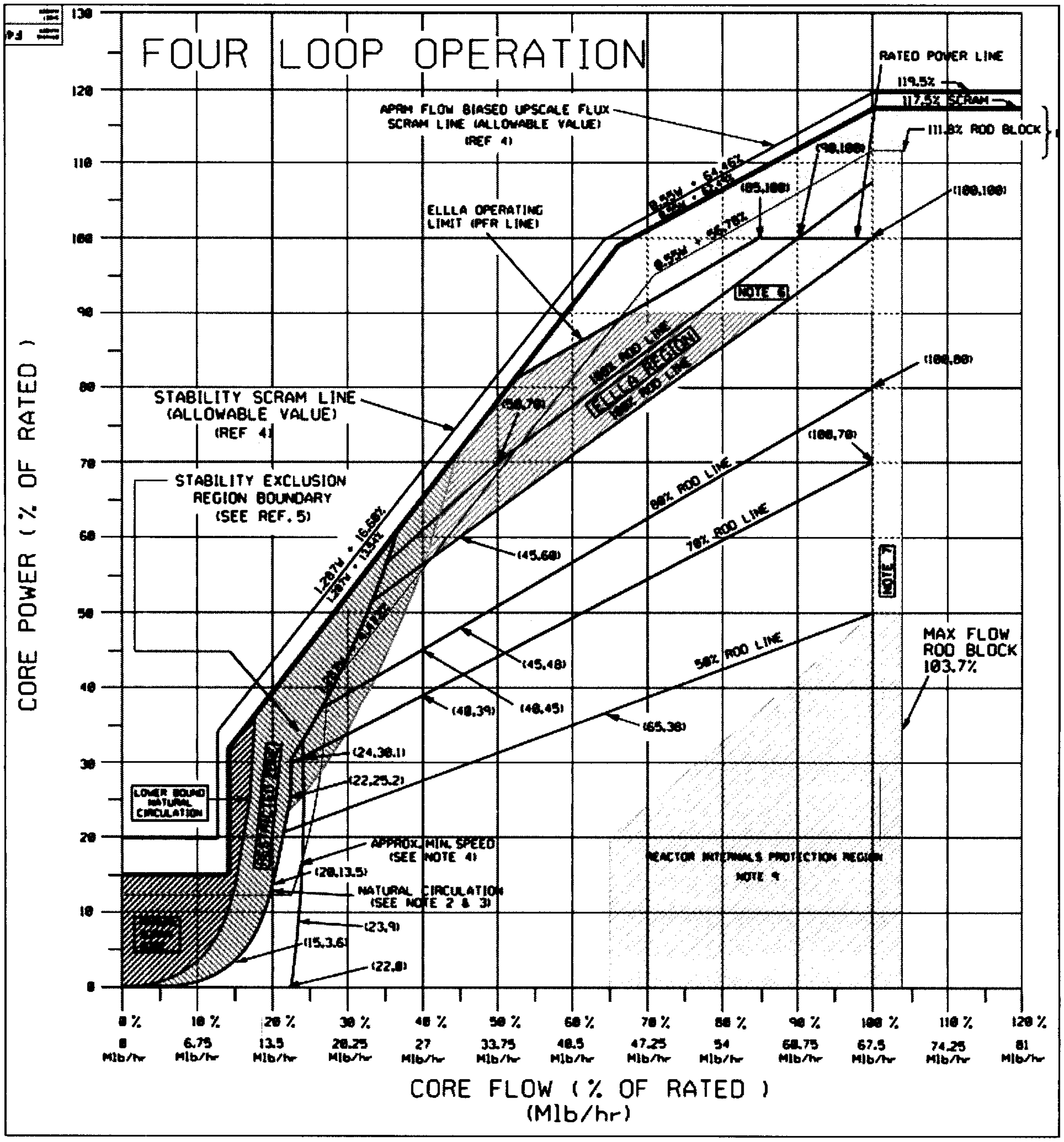
A. below the 100% rod line. Action is required to exit the current region.

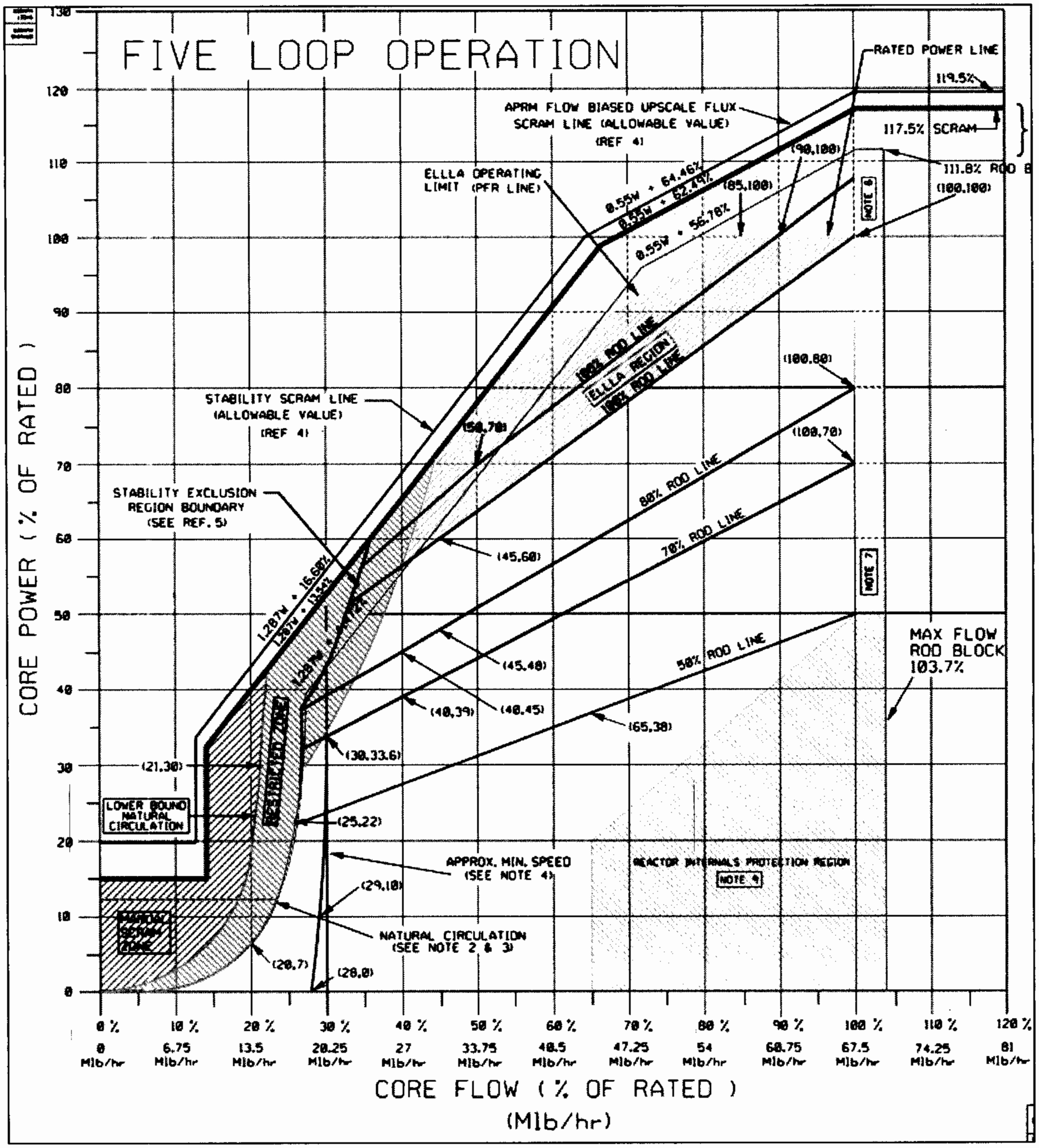
B. below the 100% rod line. Continuous operation is allowed in the current region.

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)
RATED POWER LINE
119.5%
117.5% SCRAM
SEE NOTE 8
111.8% ROD BLO
APRM FLOW BIASED UPSCALE FLUX SCRAM LINE (ALLOWABLE VALUE) (REF 4)
ELLLA OPERATING LIMIT (PFR LINE)
0.55W + 64.46%
0.55W + 56.76%
(85,100)
(100,100)
(90,100)
NOTE 6
100% ROD LINE
ELLLA REGION
100% ROD LINE
(100,80)
STABILITY SCRAM LINE (ALLOWABLE VALUE) (REF 4)
(50,70)
STABILITY EXCLUSION REGION BOUNDARY (SEE REF. 5)
1.287W + 16.60%
(100,70)
80% ROD LINE
(45,60)
70% ROD LINE
NOTE 7
MAX FLOW ROD BLOCK 103.7%
(45,48)
50% ROD LINE
(40,39)
(40,45)
(65,38)
LOWER BOUND NATURAL CIRCULATION
APPROX. MIN. SPEED (SEE NOTE 4)
REACTOR INTERNALS PROTECTION REGION NOTE 9
NATURAL CIRCULATION (SEE NOTE 2 & 3)
(15.8,2)
(18,7)
(10,1.5)
(17,0)
0 % 10 % 20 % 30 % 40 % 50 % 60 % 70 % 80 % 90 % 100 % 110 % 120 %
0 Mlb/hr
6.75 Mlb/hr
13.5 Mlb/hr
20.25 Mlb/hr
27 Mlb/hr
33.75 Mlb/hr
40.5 Mlb/hr
47.25 Mlb/hr
54 Mlb/hr
60.75 Mlb/hr
67.5 Mlb/hr
74.25 Mlb/hr
81 Mlb/hr
0 10 20 30 40 50 60 70 80 90 100 110 120 130


FOUR LOOP OPERATION
RATED POWER LINE
119.5%
117.5% SCRAM
111.8% ROD BLOCK
APRM FLOW BIASED UPSCALE FLUX SCRAM LINE (ALLOWABLE VALUE) (REF 4)
ELLLA OPERATING LIMIT (PFR LINE)
(85,100)
(90,100)
(100,100)
NOTE 6
100% ROD LINE
ELLLA REGION
90% ROD LINE
(100,80)
STABILITY SCRAM LINE (ALLOWABLE VALUE) (REF 4)
(50,70)
(100,70)
STABILITY EXCLUSION REGION BOUNDARY (SEE REF. 5)
80% ROD LINE
70% ROD LINE
(45,60)
NOTE 7
50% ROD LINE
MAX FLOW ROD BLOCK 103.7%
(45,48)
(40,45)
(40,39)
(65,30)
(24,30.1)
(22,25.2)
LOWER BOUND NATURAL CIRCULATION
APPROX. MIN. SPEED (SEE NOTE 4)
(20,13.5)
NATURAL CIRCULATION (SEE NOTE 2 & 3)
REACTOR INTERNALS PROTECTION REGION NOTE 9
(23,9)
(15,3.6)
(22,0)
CORE POWER (% OF RATED)
0 %
10 %
20 %
30 %
40 %
50 %
60 %
70 %
80 %
90 %
100 %
110 %
120 %
0 Mlb/hr
6.75 Mlb/hr
13.5 Mlb/hr
20.25 Mlb/hr
27 Mlb/hr
33.75 Mlb/hr
40.5 Mlb/hr
47.25 Mlb/hr
54 Mlb/hr
60.75 Mlb/hr
67.5 Mlb/hr
74.25 Mlb/hr
81 Mlb/hr
CORE FLOW (% OF RATED)
(Mlb/hr)



FIVE LOOP OPERATION
RATED POWER LINE
119.5%
117.5% SCRAM
111.8% ROD B
APRM FLOW BIASED UPSCALE FLUX SCRAM LINE (ALLOWABLE VALUE) (REF 4)
ELLLA OPERATING LIMIT (PFR LINE)
0.55W + 64.46%
0.55W + 62.44%
0.55W + 56.76%
(90,100)
(85,100)
(100,100)
NOTE 6
100% ROD LINE
ELLLA REGION
108% ROD LINE
(100,80)
(100,70)
STABILITY SCRAM LINE (ALLOWABLE VALUE) (REF 4)
(50,70)
STABILITY EXCLUSION REGION BOUNDARY (SEE REF. 5)
80% ROD LINE
70% ROD LINE
(45,60)
NOTE 7
1.287W + 16.68%
(45,40)
MAX FLOW ROD BLOCK 103.7%
50% ROD LINE
(40,39)
(40,45)
(65,38)
(30,33.6)
(21,30)
LOWER BOUND NATURAL CIRCULATION
RESTRICTED ZONE
(25,22)
APPROX. MIN. SPEED (SEE NOTE 4)
REACTOR INTERNALS PROTECTION REGION
NOTE 4
(29,10)
MANUAL SCRAM ZONE
NATURAL CIRCULATION (SEE NOTE 2 & 3)
(20,7)
(28,0)
CORE POWER (% OF RATED)
CORE FLOW (% OF RATED)
(Mlb/hr)
0% 10% 20% 30% 40% 50% 60% 70% 80% 90% 100% 110% 120%
0 Mlb/hr 6.75 Mlb/hr 13.5 Mlb/hr 20.25 Mlb/hr 27 Mlb/hr 33.75 Mlb/hr 40.5 Mlb/hr 47.25 Mlb/hr 54 Mlb/hr 60.75 Mlb/hr 67.5 Mlb/hr 74.25 Mlb/hr 81 Mlb/hr

Proposed Answer: D

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 (14+14+14-4-4 = 34 Mlbm/hr). With this total core flow at 66% Reactor power, operation is above the 100% rod line and in the blue ELLLA region. N1-SOP-1.5 allows continuous operation in this region.

A. Incorrect – Operation is above the 100% rod line. Plausible because this would be correct if APRMs indicated slightly lower or if core flow were slightly higher. Continuous operation is allowed in this region. Plausible because this would be correct if operation were in the nearby orange Restricted zone.
B. Incorrect – Operation is above the 100% rod line. Plausible because this would be correct if APRMs indicated slightly lower or if core flow were slightly higher.
C. Incorrect – Continuous operation is allowed in this region. Plausible because this would be correct if operation were in the nearby orange Restricted zone.

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: Modified Bank – 2015 NRC #48

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:
TRH 11/9/18 – Fixed a few typos and embedded the 4- and 5-loop maps, based on NRC comment.
TRH 11/19/18 – Fixed note to match new maps.
TRH 11/28/18 – Deleted "three (3)" from note based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EA1.03
	Importance Rating	3.9

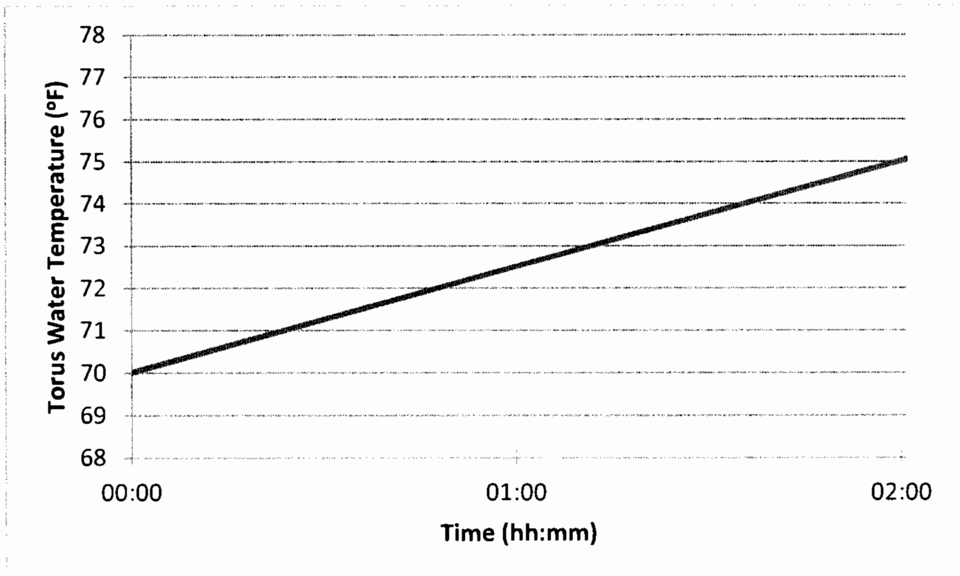
Suppression Pool High Water Temperature

Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring

Proposed Question: #28

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

- An ERV has been leaking for the last two (2) hours.
- Reactor pressure is 1020 psig and stable.
- Torus water level is 10.7 feet and approximately stable.
- Torus water temperature has risen as follows:



Note: Assume the trend remains constant and Torus Cooling is NOT placed in service.

Which one of the following identifies the approximate time N1-EOP-4, Primary Containment Control, entry will first be required?

Enter EOP-4 at time…

A. 04:00

B. 06:00

C. 08:00

D. 10:00

Proposed Answer: B

Explanation: The given Torus water temperature data shows a rate of rise of 5°F every 2 hours. Since Torus water temperature is 75°F at time 02:00, it will take approximately 4 more hours to reach 85°F, which is the N1-EOP-4 entry condition. Therefore, Torus water temperature will reach the N1-EOP-4 entry condition at approximately 06:00.

A. Incorrect – N1-EOP-4 entry will first be required at approximately 06:00. Plausible because this is the time that F1-2-8 and F4-2-1 will alarm due to Torus water temperature at 80°F.
C. Incorrect – N1-EOP-4 entry will first be required at approximately 06:00. Plausible because Torus water temperature will be 90°F at this time, and 90°F is one of the values used in F1-2-8 and F4-2-1.
D. Incorrect – N1-EOP-4 entry will first be required at approximately 06:00. Plausible because Torus water temperature will be 95°F at this time, which is a value used in TS 3.3.2.

Technical Reference(s): N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Modified Bank – 2015 Cert #49

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

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

Partial or Complete Loss of AC Power

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

Proposed Question: #29

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

- An electrical fault that will not clear develops between breaker R10 and disconnect SW 168.
- Breakers R10 and R40 open.
- R10 and R40 Auto-Reclosure switches are in ON.

Which one of the following describes the electrical distribution system lineup five (5) minutes later?

	Breaker R40	MOD 8106
A.	Closed	Closed
B.	Closed	Open
C.	Open	Closed
D.	Open	Open

Proposed Answer: B

Explanation: If a 115 kV bus fault occurs, breakers R10 and R40 open. If 115 kV lines are energized (with 115 kV bus de-energized), breakers R10 and R40 attempt sequential re-closure. Re-closure will fail due to the location of the fault downstream of R10. Since re-closure fails, bus sectionalizing disconnect switch MOD 8106 opens and then R10 and R40 attempt another re-closure to re-energize the un-faulted section of the 115 kV bus. R10 will fail to close due to the location of the fault, but with MOD 8106 open, R40 will be isolated from the fault and stay closed. This entire sequence completes within five minutes.

A. Incorrect – MOD 8106 will open to isolate the fault. Plausible because this MOD is normally closed, does not initially open, and would not open if not for the given fault.
C. Incorrect – R40 will close and remain closed once MOD 8106 opens. Plausible because R40 will initially fail to re-close. MOD 8106 will open to isolate the fault. Plausible because this MOD is normally closed, does not initially open, and would not open if not for the given fault.
D. Incorrect – R40 will close and remain closed once MOD 8106 opens. R40 will close and remain closed once MOD 8106 opens. Plausible because R40 will initially fail to re-close.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262000-RBO-05

Question Source: Bank - 2013 NRC #2

Question History: 2013 NRC #2

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 #	295004 AK1.04
	Importance Rating	2.8

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: Effect of battery discharge rate on capacity

Proposed Question: #30

A Station Blackout is in progress.

Which one of the following identifies the time requirement for completing N1-SOP-33A.2 Attachment 4, Safety Related Battery Load Reduction, and a load (or loads) that is (are) secured as part of this attachment, in accordance with N1-SOP-33A.2, Station Blackout / ELAP?

N1-SOP-33A.2 Attachment 4, Safety Related Battery Load Reduction, must be completed within a maximum of ___(1)___ from the start of the Station Blackout.

___(2)___ is (are) secured per this attachment.

A. (1) 30 minutes
   (2) UPS 175

B. (1) 30 minutes
   (2) EDG oil pumps

C. (1) 1 hour
   (2) UPS 175

D. (1) 1 hour
   (2) EDG oil pumps

Proposed Answer: B

Explanation: N1-SOP-33A.2 Attachment 4, Safety Related Battery Load Reductions, must be completed within a maximum of 30 minutes from the start of the Station Blackout. EDG oil pumps are secured as a part of this attachment.

Note: The question meets the K/A because, it tests an operational implication (load shedding requirements) of the effect of battery discharge rate on capacity (failure to load shed lowers battery capacity and threatens meeting coping time requirements) related to a partial loss of DC power (Station Blackout causes loss of battery chargers).

A. Incorrect – UPS 175 is not secured per N1-SOP-33A.2 Attachment 4. Plausible because it is secured per N1-SOP-33A.2 Attachment 5 to reduce Control Room heat load.
C. Incorrect – UPS 175 is not secured per N1-SOP-33A.2 Attachment 4. Plausible because it is secured per N1-SOP-33A.2 Attachment 5 to reduce Control Room heat load. The time requirement for completing N1-SOP-33A.2 Attachment 4 is 30 minutes. Plausible because there are many 1 hour time requirements in plant procedures and Technical Specifications.
D. Incorrect – The time requirement for completing N1-SOP-33A.2 Attachment 4 is 30 minutes. Plausible because there are many 1 hour time requirements in plant procedures and Technical Specifications.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33A.2C01 EO-2

Question Source: Modified Bank - SSES LOC28 NRC #48

Question History: SSES LOC28 NRC #48

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/19/18 – Changed distractors to 1 hour based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 2.4.31
	Importance Rating	4.2

Main Turbine Generator Trip

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #31

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

- Leakage has developed in the Stator Water Cooling system
- A2-1-1, GENERATOR STAT. WATER OUTLET TEMP. HIGH, is in alarm
- A2-1-2, GENERATOR STAT. WATER INLET PRESS. LOW, is in alarm
- A2-2-1, GENERATOR STAT. WATER HIGH CONDUCTIVITY, is in alarm

Which one of the following requires an immediate trip of the Turbine, in accordance with the associated Annunciator Response Procedure?

Stator water...

A. inlet flow at 400 gpm

B. inlet pressure at 22 psig

C. outlet temperature at 87°C

D. conductivity at 10 µmho/cm

Proposed Answer: D

Explanation: ARP A2-2-1 requires a Turbine trip if Stator Water conductivity exceeds 9.9 μmho/cm.

A. Incorrect – This flow does not require a Turbine trip. Plausible because this flow would drive annunciator A2-1-2 to alarm and is below the runback setpoint.
B. Incorrect – This pressure does not require a Turbine trip. Plausible because this pressure would drive annunciator A2-1-2 to alarm and is below the runback setpoint.
C. Incorrect – This temperature does not require a Turbine trip. Plausible because this temperature is above the runback setpoint.

Technical Reference(s): ARP A2-2-1, ARP A2-2-2, N1-OP-44, N1-SOP-32

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-253000-RBO-12

Question Source: Bank – 2010 NRC #66

Question History: 2010 NRC #66

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AK2.06
	Importance Rating	4.2

SCRAM

Knowledge of the interrelations between SCRAM and the following: Reactor power

Proposed Question: #32

The plant has been manually scrammed with the following:

- All control rod position indication is lost.
- N1-EOP-3, Failure to Scram, has been entered.
- All APRMs indicate downscale.
- All IRMs are fully inserted.
- All IRMs indicate downscale on Range 6 and on-scale on Range 5.

Which one of the following describes the condition of the Reactor, in accordance with N1-EOP-3?

The Reactor is...

A. NOT currently shutdown.

B. currently shutdown based on APRM indication, but NOT guaranteed to be shutdown under all conditions without boron.

C. currently shutdown based on IRM indication, but NOT guaranteed to be shutdown under all conditions without boron.

D. guaranteed to be shutdown under all conditions without boron.

Proposed Answer: C

Explanation: OP-NM-101-111-1001 states that the EOP definition of shutdown equates to IRM range 6 or below with the IRMs fully inserted. Since Reactor power is downscale on IRM range 6 with the IRMs fully inserted, the Reactor is currently shutdown. The Reactor can only be determined to be shutdown under all conditions without boron based on assessment of control rod positions. With a loss of all control rod position indication, the Reactor can NOT be guaranteed to remain shutdown under all conditions without boron.

A. Incorrect – Since IRMs are fully inserted and downscale on range 6, the EOP definition of shutdown is met. Plausible because control rod positions are unknown and IRMs are still indicating higher than normal for being fully inserted after a scram.
B. Incorrect – APRMs being downscale is not enough to determine that the Reactor is shutdown. Plausible because APRMs being downscale is an indication used immediately after a scram to determine the success of the scram and need for N1-EOP-2 entry.
D. Incorrect – Without control rod position indication, it cannot be guaranteed that the Reactor will remain shutdown under all conditions without boron. Plausible because the related shutdown determination can be made based on Reactor power indication alone.

Technical Reference(s): N1-EOP-3, NER-1M-095, OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO-2

Question Source: Bank - 2015 Cert #42

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 2.4.35
	Importance Rating	3.8

Control Room Abandonment

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

Proposed Question: #33

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 2, 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. Incorrect – 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. Incorrect – 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. Incorrect – 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: Bank – 2015 NRC #73

Question History: 2015 NRC #73

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK1.01
	Importance Rating	3.5

Partial or Complete Loss of CCW

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

Proposed Question: #34

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. Incorrect – RWCU pumps must be tripped first. Plausible because N1-SOP-11.1 directs monitoring SFPC and also directs securing affected equipment if high temperatures occur.
C. Incorrect – RWCU pumps must be tripped first. Plausible because N1-SOP-11.1 directs monitoring Drywell cooling and also directs securing affected equipment if high temperatures occur.
D. Incorrect – RWCU pumps must be tripped first. Plausible because N1-SOP-11.1 directs monitoring Recirculation system temperatures and also directs securing affected equipment if high temperatures occur. Also plausible because 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 – 2015 NRC #43

Question History: 2015 NRC #43

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:
TRH 11/12/18 – Bold and italicized "first" based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AK3.03
	Importance Rating	3.2

Partial or Complete Loss of Instrument Air

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service air isolations: Plant-Specific

Proposed Question: #35

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

- A leak develops in the Instrument Air (IA) system.
- N1-SOP-20.1, Instrument Air Failure, is entered.
- IA pressure lowers to 80 psig and is then recovered to 104 psig after isolating a leak in the plant.
- All required operator actions are taken during the transient, in accordance with plant procedures.

Which one of the following describes when 94-19, HSA RECEIVER TO IA SEPARATOR, ABSORBER AFTERFILTER, is ***closed***?

94-19 is closed…

A. manually as IA pressure lowers.

B. automatically as IA pressure lowers.

C. manually when IA pressure is recovered.

D. automatically when IA pressure is recovered.

Proposed Answer: C

Explanation: 94-19 is normally closed and isolates IA from Service Air. 94-19 automatically opens in response to lowering IA pressure to attempt to allow Service Air to recover IA. Once open, 94-19 remains open until manual action is taken. 94-19 is manually closed once IA pressure is recovered to normal to once again isolate IA and Service Air.

Note: The question meets the K/A by requiring the candidate to know why 94-19 is closed to isolate Service Air from Instrument Air (in response to rising pressure vs. lowering pressure).

A. Incorrect – 94-19 is closed manually, not automatically. Plausible this valve automatically opens and similar valve 94-91 has an auto closure. 94-19 is closed as IA pressure is recovered, not as it lowers. Plausible because similar valve 94-91 closes on lowering IA pressure. Also plausible that this would be done to prevent a Service Air leak from dragging down IA pressure.
B. Incorrect – 94-19 is closed as IA pressure is recovered, not as it lowers. Plausible because similar valve 94-91 closes on lowering IA pressure. Also plausible that this would be done to prevent a Service Air leak from dragging down IA pressure.
D. Incorrect – 94-19 is closed manually, not automatically. Plausible this valve automatically opens and similar valve 94-91 has an auto closure.

Technical Reference(s): ARP L1-4-7, N1-OP-20

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:
TRH 11/12/18 – Revised 1st bullet to be a leak vs. a loss, based on NRC comment.
TRH 11/19/18 – Revised noun name of 94-19 to match SOP, based on NRC comment.
TRH 11/28/18 – Revised noun name based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 2.2.37
	Importance Rating	3.6

Loss of Shutdown Cooling

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

Proposed Question: #36

A plant shutdown is in progress due to a steam leak in the Drywell with the following:

- Preparations are underway for placing Shutdown Cooling (SDC) in service.
- The breaker for valve 38-01, SDC SYSTEM IN IV 11 (INSIDE), will NOT close.
- Drywell pressure is 2.2 psig and slowly rising.

Which one of the following describes the availability of SDC pumps to be placed in service?

A. All SDC pumps are available to be placed in service.

B. SDC pump 12 is available to be placed in service, only.

C. SDC pumps 12 and 13 are available to be placed in service, only.

D. NO SDC pumps are available to be placed in service.

Proposed Answer: D

Explanation: During normal power operation, 38-01 is closed with its breaker open. This valve is on a common suction line to all three SDC pumps and is located inside the Drywell. With the breaker failing to close, it cannot be remotely opened. With the Drywell at 2.2 psig due to a steam leak, it is unable to be entered, so the valve cannot be manually opened. Since the valve is common to all three pumps, no SDC pumps are available to be placed in service.

Note: The question meets the K/A by testing knowledge of the operability/availability of SDC isolation valves as they relate to placing SDC in service. The SDC isolation valves are the safety-related portion of the SDC system. The candidate must understand the operability/availability of SDC isolation valve 38-01 (normally closed, inside Drywell, Drywell can't be accessed, unable to be opened due to tripped breaker) and how this impacts the ability to place SDC in service (single SDC suction line with valves in series, not parallel).

A. Incorrect – No SDC pumps are available to be placed in service. Plausible because this would be correct if 38-01 and 38-02 were in parallel, if 38-01 were normally open, or if 38-01 could be accessed to manually open.
B. Incorrect – No SDC pumps are available to be placed in service. Plausible because this would be correct if the SDC suction was designed so that the Powerboard 16B pumps (11 and 13) shared a common suction line that was separate from the Powerboard 17B pump (12).
C. Incorrect – No SDC pumps are available to be placed in service. Plausible because this would be correct if this valve was only in the suction line to SDC pump 11 (as 38-03 is).

Technical Reference(s): C-18018-C, N1-OP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 11/19/18 – Added high Drywell pressure due to steam leak and K/A match statement, based on NRC comment.
TRH 11/28/18 – Deleted 3rd bullet based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AK1.01
	Importance Rating	3.6

Refueling Accidents

Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: Radiation exposure hazards

Proposed Question: #37

A refueling outage is in progress with the following:

- An LPRM string is inadvertently raised partially out of the water and CANNOT be lowered.
- The Fuel Pool high range radiation monitor indicates 1500 mr/hr.
- The following Area Radiation Monitors (ARMs) indicate upscale:
    - #18, West End of Shield Wall, RB 340
    - #25, Rx Bldg – East Wall, El. 340
- The Shift Manager has declared an Unusual Event due to the high ARM readings.
- Radiation Protection reports a General Area radiation level of 2 R/hr on the Refuel Floor.

Which one of the following describes the requirements to enter N1-EOP-5, Secondary Containment Control, and N1-EOP-6, Radioactivity Release Control?

	N1-EOP-5 Entry	N1-EOP-6 Entry
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: B

Explanation: With ARMs #18 and #25 upscale, they are above the alarm setpoints of Detail R, and therefore entry into N1-EOP-5 is required. Since the Shift Manager has only declared an Unusual Event, and no indications are given of an offsite release rate above the Alert level, entry into N1-EOP-6 is NOT required.

A. Incorrect – Entry into N1-EOP-6 is NOT required. Plausible because radiation levels are significantly above normal and the Shift Manager has made a related E-plan declaration.
C. Incorrect – Entry into N1-EOP-5 is required. Plausible because the given General Area temperature is below the Max Safe level and the high radiation is not caused by a primary system, which affects the flow path once in N1-EOP-5. Entry into N1-EOP-6 is NOT required. Plausible because radiation levels are significantly above normal and the Shift Manager has made a related E-plan declaration.
D. Incorrect – Entry into N1-EOP-5 is required. Plausible because the given General Area temperature is below the Max Safe level and the high radiation is not caused by a primary system, which affects the flow path once in N1-EOP-5.

Technical Reference(s): N1-EOP-5, N1-EOP-6

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-1, 1101-EOP6C01 EO-1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/12/18 – Fixed error in explanation based on NRC comment.

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK2.15
	Importance Rating	3.8

High Drywell Pressure

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Containment spray logic: Plant-Specific

Proposed Question: #38

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

- Containment Spray loop 111 is operating in Torus Cooling.
- Containment Spray loops 112, 121, and 122 are in a normal standby lineup.

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

- The Reactor scrams on high Drywell pressure.
- Reactor water level reaches a low of 0" on the scram transient before recovering.

Which one of the following identifies the status of Containment Spray two (2) minutes later?

	Containment Spray Loop 111	Containment Spray Loops 112, 121, and 122
A.	Remains in Torus Cooling	Remain in normal standby lineup
B.	Remains in Torus Cooling	Spraying the Containment
C.	Does NOT remain in Torus Cooling	Remain in normal standby lineup
D.	Does NOT remain in Torus Cooling	Spraying the Containment

Proposed Answer: B

Explanation: With Drywell pressure above 3.5 psig (as evidenced by Reactor scram on high Drywell pressure) and Reactor water level lowering below +5", an automatic Containment Spray signal is received. Containment Spray pumps 112, 121, and 122 automatically start and spray the Containment. With Containment Spray loop 111 initially in Torus Cooling, 80-45 is closed, 80-16 is closed, and 80-118 is open. None of these valves automatically re-position on either the high Drywell or low-low Reactor water level signals. Therefore, Containment Spray loop 111 remains in the Torus Cooling lineup.

A. Incorrect – Containment Spray pumps 112, 121, and 122 automatically start and spray the Containment. Plausible because this would be correct if Reactor water level stayed above +5" and/or Drywell pressure stayed below 3.5 psig.
C. Incorrect – Containment Spray loop 111 remains in the Torus Cooling lineup. Plausible because an automatic Containment Spray signal is received and similar systems would realign to their intended automatic function (eg. Core Spray comes out of test lineup if it gets an auto initiation signal). Containment Spray pumps 112, 121, and 122 automatically start and spray the Containment. Plausible because this would be correct if Reactor water level stayed above +5" and/or Drywell pressure stayed below 3.5 psig.
D. Incorrect – Containment Spray loop 111 remains in the Torus Cooling lineup. Plausible because an automatic Containment Spray signal is received and similar systems would realign to their intended automatic function (eg. Core Spray comes out of test lineup if it gets an auto initiation signal).

Technical Reference(s): N1-OP-14, 1101-226001C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226001-RBO-5

Question Source: Modified Bank – SYSID 50290

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 #	295025 2.4.8
	Importance Rating	3.8

High Reactor Pressure

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

Proposed Question: #39

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

- A loss of all 115 KV power occurs.
- N1-SOP-33A.1, Loss of 115 KV, is being performed.
- A spurious Reactor scram occurs.
- Following the scram, a pressure control malfunction results in Reactor pressure rising to 1100 psig.
- The Unit Supervisor enters N1-EOP-2, RPV Control, due to high Reactor pressure.

Which one of the following describes the correct procedure implementation for N1-SOP-33A.1?

A. Continue performing N1-SOP-33A.1. In the event of a conflict between N1-SOP-33A.1 and N1-EOP-2, N1-EOP-2 is the overriding procedure.

B. Continue performing N1-SOP-33A.1. In the event of a conflict between N1-SOP-33A.1 and N1-EOP-2, N1-SOP-33A.1 is the overriding procedure.

C. Exit N1-SOP-33A.1. N1-SOP-33A.1 is re-entered at the step in progress after exiting N1-EOP-2.

D. Exit N1-SOP-33A.1. N1-SOP-33A.1 entry conditions are re-evaluated after exiting N1-EOP-2.

Proposed Answer: A

Explanation: There is no requirement to exit SOPs when an EOP is entered. Both procedures are executed concurrently. The EOPs are higher-tiered procedures than SOPs, therefore in the event of a conflict, the EOP must be followed.

B. Incorrect – The EOPs are higher-tiered procedures than SOPs, therefore in the event of a conflict, the EOP must be followed. Plausible because the SOP is event-specific while the EOP is not.

C. Incorrect – There is no requirement to exit SOPs when EOPs are entered. Plausible because the EOP is the higher-tiered document. Also plausible because N1-SOP-33A.1 does have an override step that requires moving back to the beginning of the procedure due to the Main Generator trip.

D. Incorrect – There is no requirement to exit SOPs when EOPs are entered. Plausible because the EOP is the higher-tiered document. Also plausible because N1-SOP-33A.1 does have an override step that requires moving back to the beginning of the procedure due to the Main Generator trip.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-EOP00C01 TO #1

Question Source: Modified Bank – SSES LOC27 NRC #21

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 #	295028 EK3.06
	Importance Rating	3.4

High Drywell Temperature

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: ADS

Proposed Question: #40

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

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

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

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

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

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

Proposed Answer: A

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2017 NRC #41

Question History: 2017 NRC #41

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 EA2.04
	Importance Rating	3.5

Low Suppression Pool Water Level

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Drywell/ suppression chamber differential pressure: Mark I & II

Proposed Question: #41

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

- Drywell pressure is 1.1 psig.
- Torus pressure is 1.0 psig.

Then, a leak develops in the Torus.

- Torus water level lowers from 11.1' to 10.6' and stabilizes.

Which one of the following describes:

(1) the effect of this transient on Drywell to Torus differential pressure (D/P),

and

(2) the need to entry N1-EOP-4, Primary Containment Control?

	Effect on Drywell to Torus D/P	N1-EOP-4 Entry
A.	Lowers	Required
B.	Lowers	NOT required
C.	Rises	Required
D.	Rises	NOT required

Proposed Answer: D

Explanation: The lowering Torus water level causes Torus pressure to lower due to expansion of the gases in the Torus air space. Drywell pressure remains constant. Torus pressure is normally maintained slightly lower than Drywell pressure. Therefore, Drywell to Torus D/P rises. Since Torus water level is above 10.5', N1-EOP-4 entry is not yet required.

A. Incorrect – Drywell to Torus D/P rises. Plausible because this would be correct if Torus pressure were normally higher than Drywell pressure, or if the effect of the lowering Torus water level were reversed. N1-EOP-4 entry is not yet required. Plausible because Torus water level has lowered significantly and is close to the EOP entry condition.
B. Incorrect – Drywell to Torus D/P rises. Plausible because this would be correct if Torus pressure were normally higher than Drywell pressure, or if the effect of the lowering Torus water level were reversed.
C. Incorrect – N1-EOP-4 entry is not yet required. Plausible because Torus water level has lowered significantly and is close to the EOP entry condition.

Technical Reference(s): ARP K2-4-4, N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EA1.13
	Importance Rating	4.3

Reactor Low Water Level

Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Reactor water level control

Proposed Question: #42

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

- Feedwater pump 13 is running.
- Feedwater pumps 11 and 12 are in standby.

Then, a leak results in the following:

- Reactor water level lowers.
- A Reactor scram occurs.

Current conditions are:

- Reactor water level is 30" and slowly rising.
- Feedwater pump 11 is running, but NOT injecting.
- Feedwater pump 12 is injecting 1.9 Mlbm/hr.
- Feedwater pump 13 is injecting 2.8 Mlbm/hr.

Which one of the following describes reasons for the status of injection from Feedwater pumps 11 and 12?

	Feedwater pump 11 is NOT injecting because...	Feedwater pump 12 flow is limited to 1.9 Mlbm/hr to prevent...
A.	total Feedwater flow is too high.	RPV overfill.
B.	total Feedwater flow is too high.	pump runout.
C.	Reactor water level is above its setpoint.	RPV overfill.
D.	Reactor water level is above its setpoint.	pump runout.

Proposed Answer: B

Explanation: Total Feedwater flow is >4.5 Mlbm/hr, therefore Feedwater pump 11 FCV is given a close signal. Feedwater pump 12 is limited to 1.9 Mlbm/hr injection to prevent pump runout.

A. Incorrect – FWP 12 flow is limited to prevent pump runout, not RPV overfill. Plausible because other FW limits (such as setpoint setdown) are designed to prevent RPV overfill.
C. Incorrect – FWP 11 is not injecting due to total FW flow being high, not Reactor water level being too high. Plausible because above 65" FWP 11 injection would also be blocked. FWP 12 flow is limited to prevent pump runout, not RPV overfill. Plausible because other FW limits (such as setpoint setdown) are designed to prevent RPV overfill.
D. Incorrect – FWP 11 is not injecting due to total FW flow being high, not Reactor water level being too high. Plausible because above 65" FWP 11 injection would also be blocked.

Technical Reference(s): N1-OP-16, SDBD-402

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259000-RBO-6

Question Source: Bank - 2017 Cert #65

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 #	295037 EA1.01
	Importance Rating	4.6

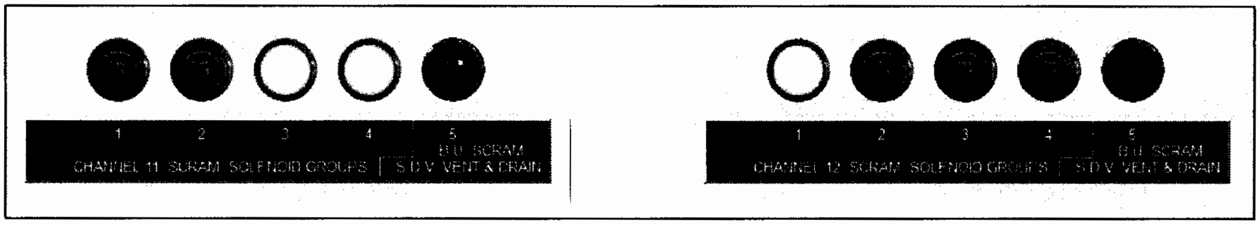
Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor Protection System

Proposed Question: #43

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

- A Main Turbine trip occurs, but RPS fails to insert control rods.
- An Operator places the Reactor Mode Switch in SHUTDOWN and depresses both manual scram pushbuttons.
- The following RPS indications are available in the Main Control Room:



Note: Assume all control rods respond properly for the status of RPS indications.

Which one of the following describes the status of control rods?

A. All control rods have inserted.

B. NO control rods have inserted.

C. Only one group of control rods have inserted.

D. Only two groups of control rods have inserted.

Proposed Answer: C

Explanation: The given indications show that RPS channel 11 has failed to de-energize the Group 3 and 4 scram solenoids, and RPS channel 12 has failed to de-energize the Group 1 scram solenoids and the channel 12 backup scram valve solenoids. This results in only the Group 2 control rods inserting because (1) all other Groups have one side of RPS still energizing their scram solenoids and (2) one backup scram solenoid remains energized.

A. Incorrect – Only Group 2 control rods have inserted. Plausible because many lights are extinguished, including one of the backup scram valve solenoids. However, both backup scram vale solenoids must be de-energized to insert all control rods.
B. Incorrect – Group 2 control rods have inserted. Plausible because many lights are still lit when they should be extinguished, however the combination is sufficient to at least insert Group 2 control rods.
D. Incorrect – Only Group 2 control rods have inserted. Plausible because many lights are extinguished, however the given combination gives only one Group of control rods with both sides of RPS de-energizing their scram solenoids.

Technical Reference(s): N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: Bank – 2017 Cert #21

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 EK2.08
	Importance Rating	2.6

High Offsite Radioactivity Release Rate

Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: SPDS/ERIS/CRIDS/GDS: Plant-Specific

Proposed Question: #44

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

- Multiple annunciators have alarmed, as shown in a picture on the following page.
- NO other annunciators are in alarm.
- Various radiation monitor indications are shown in a picture on the following page.

Which one of the following describes the need for entry into N1-SOP-25.2, Fuel Failure or High Activity in Rx Coolant or Off Gas, and the required control of the Reactor?

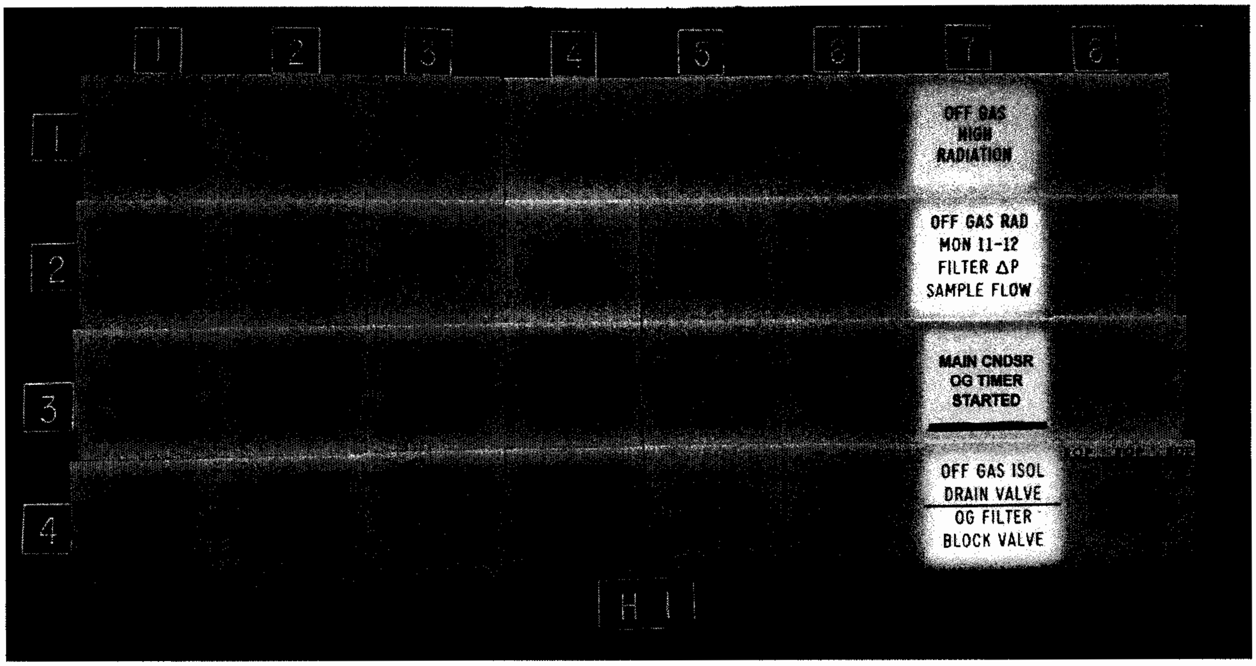
N1-SOP-25.2 entry is...

A. NOT required. The Reactor may remain at the current power level.

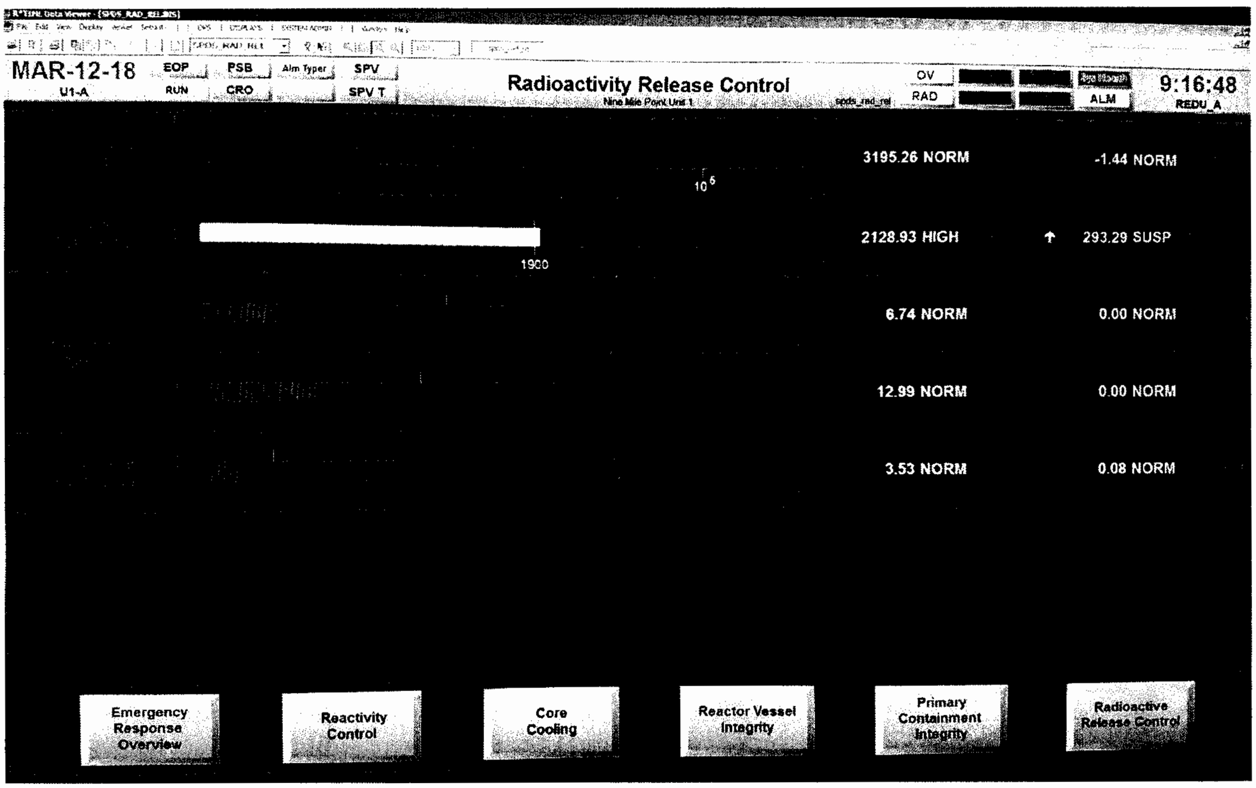
B. required. The Reactor may remain at the current power level.

C. required. A Reactor power reduction is required, but a scram is NOT required.

D. required. A Reactor scram is required.



1
2
3
4
5
6
7
8
OFF GAS HIGH RADIATION
OFF GAS RAD MON 11-12 FILTER ΔP SAMPLE FLOW
MAIN CNDSR OG TIMER STARTED
OFF GAS ISOL DRAIN VALVE
OG FILTER BLOCK VALVE
H 1



MAR-12-18
U1-A
EOP
RUN
PSB
CRO
Alm Typer
SPV
SPV T
Radioactivity Release Control
Nine Mile Point Unit 1
OV
RAD
ALM
9:16:48
REDU_A
3195.26 NORM
-1.44 NORM
10^6
2128.93 HIGH
293.29 SUSP
1900
6.74 NORM
0.00 NORM
12.99 NORM
0.00 NORM
3.53 NORM
0.08 NORM
Emergency Response Overview
Reactivity Control
Core Cooling
Reactor Vessel Integrity
Primary Containment Integrity
Radioactive Release Control

Proposed Answer: C

Explanation: N1-SOP-25.2 entry is required based on H1-1-7, Off Gas High Radiation, being in alarm and valid, as evidenced by elevated Off Gas radiation levels on the SPDS display. Additionally, this confirmed high Off Gas radiation requires reducing Reactor power per N1-SOP-1.1, Emergency Power Reduction. A Reactor scram is not yet required because Main Steam Line radiation monitors are not in alarm.

A. Incorrect – N1-SOP-25.2 entry is required based on H1-1-7, Off Gas High Radiation, being in alarm and valid, as evidenced by elevated Off Gas radiation levels on the SPDS display. Plausible because Main Steam line radiation monitors are not yet in alarm and the Off Gas radiation data on SPDS has not yet turned red.

B. Incorrect – A power reduction is required. Plausible because Main Steam line radiation monitors are not yet in alarm and the Off Gas radiation data on SPDS has not yet turned red.

D. Incorrect – A scram is not yet required. Plausible because Offgas radiation levels have risen high enough to cause isolation of the SJAE interstage block valves, which will cause Main Condenser vacuum degradation.

Technical Reference(s): N1-SOP-25.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP25.2C01 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 #	1
	K/A #	600000 AA2.05
	Importance Rating	2.9

Plant Fire On Site

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Ventilation alignment necessary to secure affected area

Proposed Question: #45

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

- Smoke Purge is running and aligned to the Auxiliary Control Room for fire surveillance testing.
- Then, a fire occurs in the Auxiliary Control Room.
- Fire detection for the Auxiliary Control Room actuates.
- The Auxiliary Control Room high exhaust duct temperature alarm is received at Local Fire Panel 1.
- A small amount of smoke has also entered the Main Control Room.

Which one of the following describes the response of the Smoke Purge system and the ability to operate Smoke Purge for the Main and Auxiliary Control Rooms?

Smoke Purge system...

A. flow continues throughout the fire and will clear smoke from the Auxiliary Control Room, only, with no further operator action.

B. flow continues throughout the fire and will clear smoke from the Main and Auxiliary Control Room with no further operator action.

C. isolates. Following reset, the Smoke Purge system can be aligned to both the Main and Auxiliary Control Room simultaneously.

D. isolates. Following reset, the Smoke Purge system can be aligned to only the Main Control Room or the Auxiliary Control Room at one time.

Proposed Answer: D

Explanation: Upon receipt of the high duct temperature, the smoke purge system will isolate. An interlock prevents simultaneous operation of aux and main control room smoke purge to prevent smoke from traversing from one area to the other.

A. Incorrect – The high duct temperature condition isolates the smoke purge system. Plausible if the candidate does not know that smoke purge automatically isolates on high duct temperature.
B. Incorrect – The high duct temperature condition isolates the smoke purge system. Plausible if the candidate does not know that smoke purge automatically isolates on high duct temperature. The aux and main control rooms must be purged separately. Plausible if the candidate does not understand the interlock preventing simultaneous operation.
C. Incorrect – The aux and main control rooms must be purged separately. Plausible if the candidate does not understand the interlock preventing simultaneous operation.

Technical Reference(s): N1-OP-21F

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-2880003-RBO-5

Question Source: Bank - 2009 NRC #19

Question History: 2009 NRC #19

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 #	700000 AK2.07
	Importance Rating	3.6

Generator Voltage and Electric Grid Disturbances

Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Turbine/generator control

Proposed Question: #46

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

- A grid disturbance has occurred.
- Power Control has requested adjusting Generator reactive power to the grid to the maximum allowed by the Generator capability curve.
- The following Generator conditions are present and stable:
  - Generator power 630 MWe
  - Generator reactive power 430 MVAR
  - Generator hydrogen pressure 45 psig

Note: The Generator capability curve is provided on the following page.

Which one of the following describes the required manipulation to carry out the request from Power Control?

Rotate...

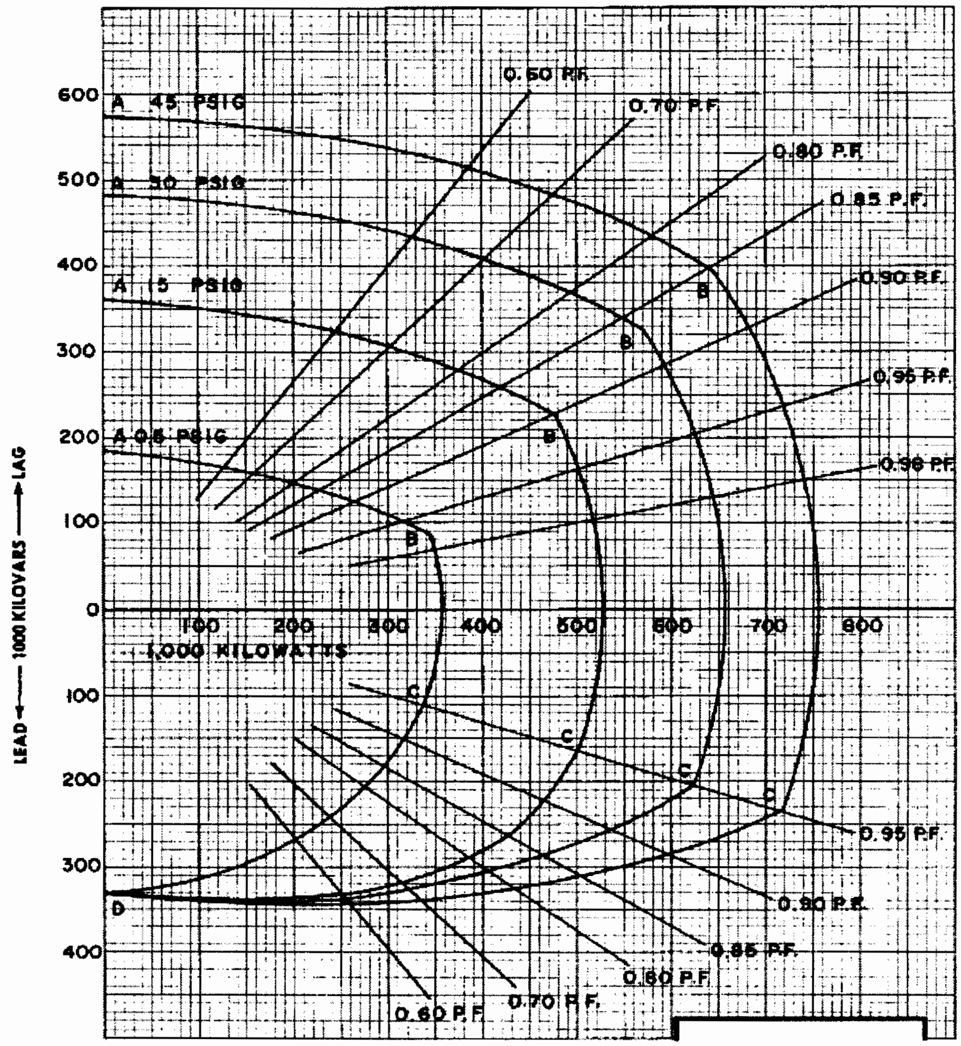
A. GOVERNOR control switch clockwise to RAISE.

B. GOVERNOR control switch counterclockwise to LOWER.

C. EXCITATION control switch clockwise to RAISE.

D. EXCITATION control switch counterclockwise to LOWER.

ATB 4 POLE, 755,000 KVA, 1800 RPM, 24,000 VOLTS
0.85 P.F., 0.58 SCR, 45 PSIG HYDROGEN PRESSURE, 500 VOLTS EXCITATION



CURVE AB LIMITED BY FIELD HEATING
CURVE BC LIMITED BY ARMATURE HEATING
CURVE CD LIMITED BY ARMATURE CORE END HEATING

Proposed Answer: D

Explanation: The given Generator parameters are exceeding the capability curve. MVARs must be lowered. This is accomplished by rotating the EXCITATION control switch counterclockwise to LOWER.

A. Incorrect – The correct control is the EXCITATION control switch. Plausible because this control switch is used to adjust Generator power (MWe) during a startup.
B. Incorrect – The correct control is the EXCITATION control switch. Plausible because this control switch is used to adjust Generator power (MWe) during a startup.
C. Incorrect – MVARs must be lowered, not raised. This is accomplished by rotating the EXCITATION control switch counterclockwise to LOWER, not clockwise to RAISE. Plausible because this would be correct if MVARs were lowered initially, as they normally would be without a grid disturbance altering them.

Technical Reference(s): N1-OP-32

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245001-RBO-5

Question Source: Bank - SSES LOC28 NRC #39

Question History: SSES LOC28 NRC #39

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295002 AK1.03
	Importance Rating	3.6

Loss of Main Condenser Vacuum

Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM: Loss of heat sink

Proposed Question: #47

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

- A loss of Main Condenser vacuum occurs.
- The Reactor is manually scrammed.
- Multiple control rods fail to fully insert.
- N1-EOP-3, Failure to Scram, has been entered.
- All required actions of N1-EOP-3 have been performed.
- Reactor water level is -50" and stable with Feedwater injecting.
- Main Condenser vacuum is 14" Hgv and slowly lowering.

Which one of the following describes the availability of Turbine Bypass Valves for Reactor pressure control?

Turbine Bypass Valves are currently...

A. available for Reactor pressure control. They will ***first*** become unavailable when Main Condenser vacuum lowers to approximately 10" Hgv.

B. available for Reactor pressure control. They will ***first*** become unavailable when Main Condenser vacuum lowers to approximately 7" Hgv.

C. NOT available for Reactor pressure control due to Main Condenser vacuum.

D. NOT available for Reactor pressure control due to Reactor water level.

Proposed Answer: A

Explanation: Reactor water level has been lowered below +5" due to the failure to scram. Since all required N1-EOP-3 actions have been completed, the MSIV low-low Reactor water level isolation has been bypassed. With MSIVs still open and Main Condenser vacuum >10" Hgv, Turbine Bypass Valves are currently available for Reactor pressure control. They will first become unavailable when Main Condenser vacuum lowers to approximately 10" Hgv because this is when Vacuum Trip #2 actuates to trip Turbine Bypass Valves closed.

B. Incorrect – They will first become unavailable when Main Condenser vacuum lowers to approximately 10" Hgv because this is when Vacuum Trip #2 actuates to trip Turbine Bypass Valves closed. Plausible because 7" Hgv is when MSIVs will close, which also makes Turbine Bypass Valves unavailable.

C. Incorrect – Turbine Bypass Valves are currently available for Reactor pressure control. Plausible because Main Condenser vacuum is significantly degraded and below the 22.1" Hgv setpoint for Vacuum Trip #1, which trips the Main Turbine.

D. Incorrect – Turbine Bypass Valves are currently available for Reactor pressure control. Plausible because Reactor water level is below the MSIV isolation setpoint, so this would be correct if not for actions taken in N1-EOP-3.

Technical Reference(s): N1-EOP-3, N1-SOP-25.1, ARP A1-3-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:
TRH 11/12/18 – Bolded and italicized "first" in A and B for consistency, based on other NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 AA1.04
	Importance Rating	2.7

Low Reactor Water Level

Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor water cleanup

Proposed Question: #48

A plant startup is in progress with the following:

- Reactor water level is 71" and stable.
- Feedwater low flow control valve (FCV) 11 is in MAN.
- Reactor water cleanup (RWCU) is in service with the Auxiliary pump running.
- RWCU low-pressure pressure control valve (PCV) is operating in AUTO.
- RWCU reject flow rate is $7x10^4$ lbm/hr.

Then, RWCU system sensed pressure drifts low.

Which one of the following describes the impact on Reactor water level and the required manipulation of the RWCU reject FCV to restore Reactor water level?

Reactor water level...

A. rises. Throttle the RWCU reject FCV further open.

B. rises. Throttle the RWCU reject FCV further closed.

C. lowers. Throttle the RWCU reject FCV further open.

D. lowers. Throttle the RWCU reject FCV further closed.

Proposed Answer: D

Explanation: When RWCU sensed pressure drifts low, the low-pressure PCV throttles further open, raising actual RWCU pressure and reject flow rate. This causes Reactor water level to lower. To restore (raise) Reactor water level, the RWCU reject FCV must be throttled further closed in order to reject less flow.

A. Incorrect – Reactor water level lowers. Plausible because this would be the correct answer for rising sensed pressure or for loss of instrument air pressure.
B. Incorrect – Reactor water level lowers. Plausible because this would be the correct answer for rising sensed pressure or for loss of instrument air pressure.
C. Incorrect – The RWCU reject FCV must be throttled further closed. Plausible because this would be correct for rising Reactor water level or if the effect of reject flow rate on Reactor water level is confused.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-204000-RBO-11

Question Source: Modified Bank - 2009 Cert #29

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295012 AK3.01
	Importance Rating	3.5

High Drywell Temperature

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: Increased drywell cooling

Proposed Question: #49

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

- Drywell cooling fan 11 trips.
- NO additional Drywell cooling fans are available to start.
- Drywell average temperature is 135°F and slowly rising.
- The US has directed additional Drywell cooling per N1-OP-8 Section H, Maintaining Drywell Average Temperature During Steady State Power Operations.

Which one of the following identifies the RBCLC heat exchanger to be verified in service and the reason for using this RBCLC heat exchanger for additional Drywell cooling, in accordance with N1-OP-8, Primary Containment Area Cooling System, and N1-OP-11, Reactor Building Closed Loop Cooling System?

	RBCLC Heat Exchanger Used for Additional Drywell Cooling	Reason
A.	11	Variation in RBCLC heat exchanger design heat removal capacity
B.	11	Variation in piping configuration between RBCLC heat exchangers and Drywell cooling
C.	12	Variation in RBCLC heat exchanger design heat removal capacity
D.	12	Variation in piping configuration between RBCLC heat exchangers and Drywell cooling

Proposed Answer: B

Explanation: For additional Drywell cooling, RBCLC heat exchanger 11 is the preferred heat exchanger. The reason for this is due to differences in piping configuration between the RBCLC heat exchangers and Drywell cooling.

A. Incorrect – The reason is due to differences in piping configuration between the RBCLC heat exchangers and Drywell cooling. Plausible because differences in heat exchanger design heat removal capacity could also cause differences in Drywell cooling (for example, if one heat exchanger had been replaced, modified, etc.).
C. Incorrect – RBCLC heat exchanger 11, not 12, is preferred for additional Drywell cooling. Plausible because there is no inherent design difference between these heat exchangers themselves. The reason is due to differences in piping configuration between the RBCLC heat exchangers and Drywell cooling. Plausible because differences in heat exchanger design heat removal capacity could also cause differences in Drywell cooling (for example, if one heat exchanger had been replaced, modified, etc.).
D. Incorrect – RBCLC heat exchanger 11, not 12, is preferred for additional Drywell cooling. Plausible because there is no inherent design difference between these heat exchangers themselves.

Technical Reference(s): N1-OP-8, N1-OP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-8

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 #	295013 2.4.20
	Importance Rating	3.8

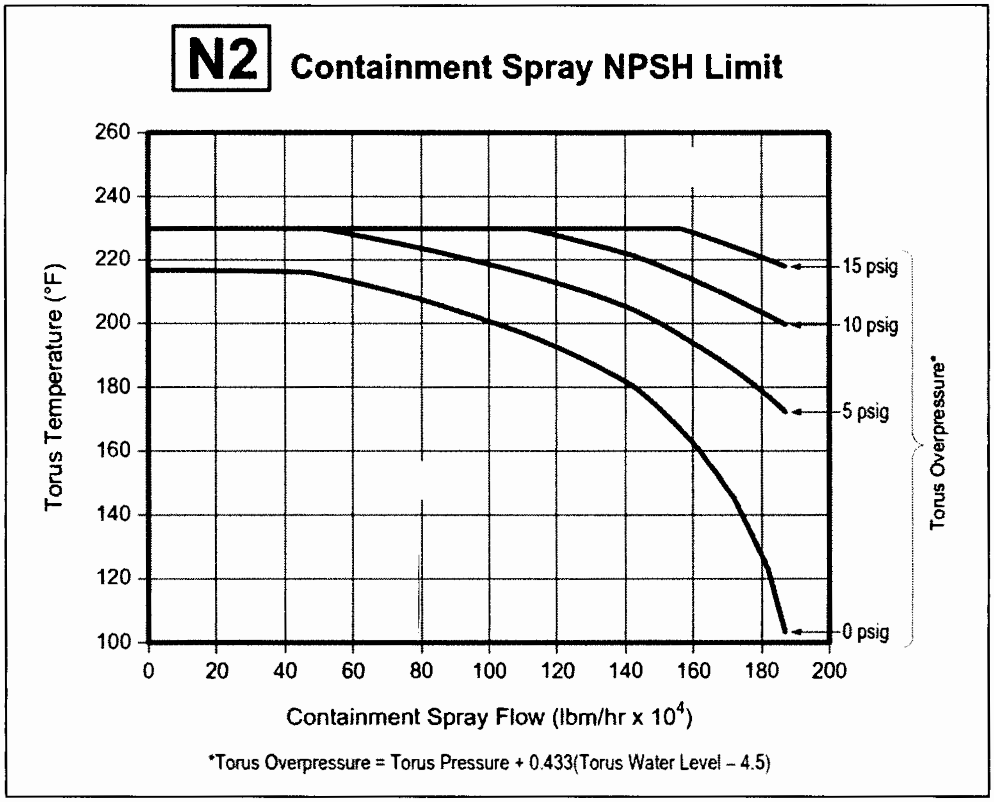
High Suppression Pool Temperature

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

Proposed Question: #50

A loss of coolant accident has resulted in the following:

- Containment Spray is in service.
- Torus pressure is 6 psig.
- Torus water level is 12.0 feet.
- Torus temperature is 180°F.
- Containment Spray flow is 160 x 10^4 lbm/hr.



Which one of the following describes (1) the status of the Containment Spray NPSH Limit and (2) the ability to use Containment Spray if the Containment Spray NPSH Limit is exceeded?

Containment Spray operation is currently in the __(1)__ region of the Containment Spray NPSH Limit. If the Containment Spray NPSH Limit is exceeded, Containment Spray operation __(2)__.

	(1)	(2)
A.	BAD	must be immediately stopped
B.	BAD	may continue, but operation in the BAD region should be minimized
C.	GOOD	must be immediately stopped
D.	GOOD	may continue, but operation in the BAD region should be minimized

Proposed Answer: D

Explanation: The given plant parameters result in a Torus overpressure calculation of 9.25 psig. This allows the 5 psig curve to be used, which results in the given Torus temperature and Containment Spray flow being in the GOOD region of the NPSH curve. An N1-EOP-4 caution states:

⚠ Operating Containment Spray with suction from the torus while above the NPSH Limit (Fig N2) or with torus water level below 6.3 ft may cause system damage.

The EOP bases state "Containment Spray operation with torus temperature above the NPSH limit or torus water level below 6.3 ft. may result in system damage and should be avoided... The operating limits are specified in a caution to provide event-specific flexibility and because it is difficult to define in advance exactly when the limits should be observed. Although the caution does not expressly prohibit operation beyond the limits, challenges to the limits should be considered only if the risk of equipment damage is warranted by the nature of the event... Operation beyond the NPSH limit is not expected to result in immediate or catastrophic pump failure."

A. Incorrect – The given parameters are in the GOOD region of the NPSH limit. Plausible because slight changes in parameters would result in operation in the BAD region. If the NPSH limit is exceeded, operation may continue. Plausible because other limits do not allow such latitude (eg. Containment Spray Initiation Limit).
B. Incorrect – The given parameters are in the GOOD region of the NPSH limit. Plausible because slight changes in parameters would result in operation in the BAD region.
C. Incorrect – If the NPSH limit is exceeded, operation may continue. Plausible because other limits do not allow such latitude (eg. Containment Spray Initiation Limit).

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 - 2013 NRC #60

Question History: 2013 NRC #60

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 AA1.05
	Importance Rating	2.5

Incomplete Scram

Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: Rod worth minimizer: Plant-Specific

Proposed Question: #51

The plant has experienced a Reactor scram. The following conditions now exist:

- Multiple control rods remain fully withdrawn.
- The Reactor Mode Switch is in SHUTDOWN.
- All APRMs indicate 5%.
- All IRMs indicate upscale on range 9.
- N1-EOP-3.1, Alternate Control Rod Insertion, Attachment 3, Driving Control Rods Using Reactor Manual Control, is in progress.
- The Reactor scram has been reset.

Which one of the following identifies whether or NOT the Rod Worth Minimizer (RWM) must be bypassed and the associated reason?

The RWM...

A. MUST be bypassed because IRM upscale rod blocks are being enforced.

B. does NOT have to be bypassed because Reactor power is in the transition zone.

C. MUST be bypassed because the RWM is invoking an insert error control rod block.

D. does NOT have to be bypassed because the Emergency Rod In switch will insert control rods regardless of RWM operation.

Proposed Answer: C

Explanation: The RWM ensures the Operator adheres to a predetermined sequence of control rod withdrawals or insertions when the Reactor is operating at low power levels. Given multiple control rods remain full out, insert blocks will be active and prevent the Operator from inserting control rods. Therefore, the RWM must be bypassed to allow rod insertion. Transition zone is between 10% and 25% core thermal power. While in the transition zone, all normal RWM rod block actions are removed. Displays are still active, with exception of withdraw error windows which will not be displayed in the transition zones. Low Power Setpoint: If steam flow goes < 10%, the RWM is returned to operation.

A. Incorrect – The RWM must be bypassed, but not because of IRM rod blocks. Plausible because IRMs are causing withdrawal blocks.
B. Incorrect – The RWM must be bypassed. Plausible because APRMs are indicating fairly high (above downscales), but below the transition zone (10%).
D. Incorrect – The RWM must be bypassed. Plausible because the Emergency Rod In switch may be used and does bypass some features (RMCS timer).

Technical Reference(s): N1-OP-37, N1-EOP-3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201003-RBO-10

Question Source: Bank – JAF 9/12 NRC #23

Question History: JAF 9/12 NRC #23

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 11/14/18 – Rephrased question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 AK2.07
	Importance Rating	3.4

Loss of Control Rod Drive Pumps

Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Reactor pressure (SCRAM assist): Plant-Specific

Proposed Question: #52

Which one of the following identifies:

(1) the Reactor pressure threshold used in determining the need for a Reactor scram in N1-SOP-5.1, Loss of Control Rod Drive, and

(2) the concern related to this pressure?

A. (1) 800 psig
(2) Control rod scram capability

B. (1) 800 psig
(2) Cooling to CRD mechanisms

C. (1) 900 psig
(2) Control rod scram capability

D. (1) 900 psig
(2) Cooling to CRD mechanisms

Proposed Answer: C

Explanation: The scram overrides in N1-SOP-5.1 use a Reactor pressure value of 900 psig in determining the need for a scram. The concern related to this pressure is control rod scram times and insertion capability, based on lack of Reactor pressure assist during the scram.

A. Incorrect – The pressure value is 900 psig, not 800 psig. Plausible because 800 psig is a Reactor pressure value discussed in both the CRD lesson plan and Technical Specification bases related to scram insertion times.

B. Incorrect – The pressure value is 900 psig, not 800 psig. Plausible because 800 psig is a Reactor pressure value discussed in both the CRD lesson plan and Technical Specification bases related to scram insertion times. The concern is related to scram capability, not CRD mechanism cooling. Plausible because CRD mechanism cooling is lost while in N1-SOP-5.1, would be another concern while in the procedure, and would be affected by Reactor temperature/pressure.

D. Incorrect – The concern is related to scram capability, not CRD mechanism cooling. Plausible because CRD mechanism cooling is lost while in N1-SOP-5.1, would be another concern while in the procedure, and would be affected by Reactor temperature/pressure.

Technical Reference(s): N1-SOP-5.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-10

Question Source: Modified Bank – NMP2 2014 NRC #61

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EA2.03
	Importance Rating	3.4

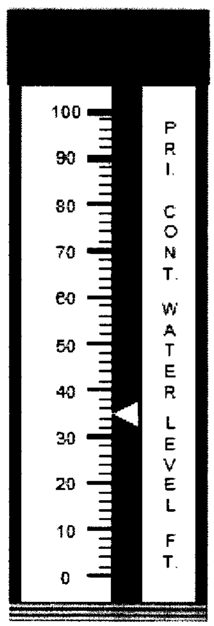
High Suppression Pool Water Level

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level

Proposed Question: #53

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

- All Torus water level indications have malfunctioned.
- The Primary Containment water level indicators are functioning properly and indicate as shown below:



Which one of the following describes the status of Primary Containment water level?

Water level in the Primary Containment is...

A. Below the ring header

B. Above the ring header and below the top of the Torus

C. Above the top of the Torus and below the bottom of the RPV

D. Above the bottom of the RPV

Proposed Answer: C

Explanation: The indicator shows Containment water level at 35'. This is above the top of the Torus (27'), but below the bottom of the RPV (61').

A. Incorrect – Containment water level is above the ring header (13.5'). Plausible because this would be correct if level were elevated but <13.5'.
B. Incorrect – Containment water level is above the top of the Torus (27'). Plausible because this would be correct if level were elevated but 13.5'-27'.
D. Incorrect – Containment water level is below the bottom of the RPV. Plausible because level is elevated and it would be correct if level were >61'.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201006 A2.01
	Importance Rating	2.5

Rod Worth Minimizer

Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply loss: P-Spec(Not-BWR6)

Proposed Question: #54

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

- Control rod movements are in progress.
- The Rod Worth Minimizer (RWM) BYPASS switch is in OFF.
- Power is lost to the RWM.

Which one of the following identifies the ability to continue control rod movements using RMCS per N1-OP-5, Control Rod Drive System?

A. Both control rod withdrawal and insertion are available without bypassing the RWM.

B. Control rod insertion is available using EMERGENCY ROD IN. Control rod movement with the CONTROL ROD MOVEMENT switch is NOT available without bypassing the RWM.

C. Control rod insertion is available using EMERGENCY ROD IN and the CONTROL ROD MOVEMENT switch. Control rod withdrawal is NOT available without bypassing the RWM.

D. NO control rod movements can be made with RMCS without bypassing the RWM.

Proposed Answer: D

Explanation: At 2% power with the RWM BYPASS in OFF, the RWM is in service. When power is lost to the RWM, a program abort occurs, which prevents all control rod movements using RMCS. The RWM must be bypassed per N1-OP-37 in order to perform control rod movements with RMCS using N1-OP-5.

A. Incorrect – An RWM program abort occurs which prevents all control rod movements using RMCS. Plausible if the RWM required power to enforce blocks and/or was automatically bypassed on loss of power.
B. Incorrect – An RWM program abort occurs which prevents all control rod movements using RMCS. Plausible because the EMERGENCY ROD IN bypasses some portions of RMCS, and is used in some situations to insert control rods when the CONTROL ROD MOVEMENT switch will not work. Also plausible that some method of control rod insertion would be maintained for safety reasons.
C. Incorrect – An RWM program abort occurs which prevents all control rod movements using RMCS. Plausible because many control rod blocks only block withdrawal and not insertion. Also plausible that some method of control rod insertion would be maintained for safety reasons.

Technical Reference(s): 1101-201003C01, N1-OP-37

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201003-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:
TRH 11/14/18 – Fixed typo in first bullet and lowered Reactor power slightly, based on NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202002 A3.03
	Importance Rating	3.1

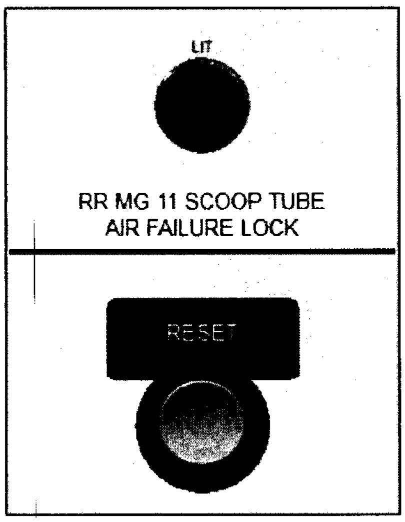
Recirculation Flow Control

Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Scoop tube operation: BWR-2,3,4

Proposed Question: #55

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

- Annunciator F2-2-1, REACT RECIRC M-G SET 11, alarms.
- The following indication is present for Recirculation pump 11 on F panel:



Which one of the following describes a possible cause of this panel light indication and the ability to take local manual control of the associated Recirc MG set Bailey scoop tube positioner given this issue?

	Possible Cause of Panel Light Indication	Local Manual Control of Bailey Scoop Tube Positioner
A.	Loss of speed control signal	Available
B.	Loss of speed control signal	NOT available
C.	Loss of scoop tube oil	Available
D.	Loss of scoop tube oil	NOT available

Proposed Answer: A

Explanation: The F panel indications include the RR MG 11 SCOOP TUBE AIR FAILURE LOCK red light ON. This indicates either loss of speed control signal or loss of instrument air. With this light lit, the scoop tube has automatically locked up, and control of Recirculation pump 11 speed from the Control Room is NOT available. Local manual control of Recirculation pump 11 speed is still available using the Bailey scoop tube positioner.

B. Incorrect – Local manual control of Recirculation pump 11 speed is still available using the Bailey scoop tube positioner. Plausible because the scoop tube has automatically locked up and control of Recirculation pump 11 speed from the Control Room is NOT available. The Bailey scoop tube positioner normally operates by converting the electrical signal from the Control Room controls into a pneumatic signal that repositions the scoop tube.
C. Incorrect – Loss of scoop tube oil would cause the given alarm, but not the given light indications. Plausible because annunciator F2-2-1 is caused by multiple conditions, one of which is low oil level.
D. Incorrect – Loss of scoop tube oil would cause the given alarm, but not the given light indications. Plausible because annunciator F2-2-1 is caused by multiple conditions, one of which is low oil level. Local manual control of Recirculation pump 11 speed is still available using the Bailey scoop tube positioner. Plausible because the scoop tube has automatically locked up and control of Recirculation pump 11 speed from the Control Room is NOT available. The Bailey scoop tube positioner normally operates by converting the electrical signal from the Control Room controls into a pneumatic signal that repositions the scoop tube.

Technical Reference(s): ARP F2-2-1, 1101-202001C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202001-RBO-5

Question Source: Modified Bank - 2009 Cert #35

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:
TRH 11/14/18 – Changed 2nd half of question to raise plausibility of distractors, based on NRC comment.
TRH 11/28/18 – Revised question, 2nd column heading, and plausibility statements to highlight 11/14/18 change, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215001 K4.01
	Importance Rating	3.4

Traversing In Core Probe

Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark I & II (Not-BWR1)

Proposed Question: #56

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

- Traversing In-core Probe (TIP) traces are in progress.
- TIP 1 detector is stuck in the indexer.
- Then, a Main Turbine trip occurs.
- Reactor water level reaches a low of -2" before recovering.

Which one of the following describes the status of the TIP 1 Ball and Shear valves five (5) minutes later?

	Ball Valve	Shear Valve
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: A

Explanation: With Reactor water level <+5", a Containment isolation is required, which should result in the Ball valve closing. However, with the TIP detector stuck in the indexer, the Ball valve will not close. The Shear valve is designed for just this situation, but it only closes if given a manual signal, which is not indicated in the stem conditions. Therefore, both the Ball and Shear valves remain open.

B. Incorrect – The Shear valve remains open. Plausible because it is designed to be closed in this situation, and could easily be designed to have an automatic closure on a time delay.

C. Incorrect – The Ball valve remains open. Plausible because it would be closed if not for the failure of the TIP detector to fully retract.

D. Incorrect – The Ball valve remains open. Plausible because it would be closed if not for the failure of the TIP detector to fully retract. The Shear valve remains open. Plausible because it is designed to be closed in this situation, and could easily be designed to have an automatic closure on a time delay.

Technical Reference(s): N1-OP-39, N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215001-RBO-5

Question Source: Bank - SSES LOC27 NRC #30

Question History: SSES LOC27 NRC #30

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	223001 2.4.3
	Importance Rating	3.7

Primary Containment and Auxiliaries

Ability to identify post-accident instrumentation.

Proposed Question: #57

The plant is shutdown with the following:

- An accident has occurred.
- Multiple Primary Containment instruments have been damaged in the Control Room.

Given the following Primary Containment parameters:

(1) Drywell pressure
(2) Torus water temperature
(3) Containment high range radiation

Which one of the following identifies which of these parameters can be determined using installed indicators at Remote Shutdown Panel 11?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: Remote Shutdown Panel 11 has an installed indicator for Drywell pressure and Torus water temperature, but not Containment high range radiation.

Note: The question meets the K/A because it requires the candidate to identify post-accident instrumentation for the Primary Containment parameters. The RO job function related to identifying post-accident instrumentation includes knowledge of where alternate post-accident instrumentation can be located. All three of the given Primary Containment parameters are post-accident instruments required by Technical Specifications.

B. Incorrect – Remote Shutdown Panel 11 does have an installed indicator for Torus water temperature. Plausible because it has limited instrumentation and does not have instruments for all N1-EOP-4 parameters (eg. Drywell temperature and H2 monitors). Remote Shutdown Panel 11 does not have an installed indicator for Containment high range radiation. Plausible because this is a required instrument in Technical Specification 3.6.11 and is a Reg Guide 1.97 instrument.

C. Incorrect – Remote Shutdown Panel 11 does have an installed indicator for Drywell pressure. Plausible because it has limited instrumentation and does not have instruments for all N1-EOP-4 parameters (eg. Drywell temperature and H2 monitors). Remote Shutdown Panel 11 does not have an installed indicator for Containment high range radiation. Plausible because this is a required instrument in Technical Specification 3.6.11 and is a Reg Guide 1.97 instrument.

D. Incorrect – Remote Shutdown Panel 11 does not have an installed indicator for Containment high range radiation. Plausible because this is a required instrument in Technical Specification 3.6.11 and is a Reg Guide 1.97 instrument.

Technical Reference(s): Technical Specification Table 3.6.13-1, 1101-296000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-296000-RBO-5

Question Source: Bank - SSES LOC27 NRC #73

Question History: SSES LOC27 NRC #73

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 11/14/18 – Revised stem conditions to better clarify K/A match, based on NRC comment.
TRH 11/19/18 – Added K/A match statement based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 K6.11
	Importance Rating	2.8

RHR/LPCI: Containment Spray Mode

Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE: Component cooling water systems

Proposed Question: #58

Which one of the following alternate lineups is NOT available if Containment Spray Raw Water pump 121 trips?

Containment Spray Raw Water injection to...

A. Core Spray loop 11

B. Core Spray loop 12

C. Containment Spray loop 11

D. Containment Spray loop 12

Proposed Answer: C

Explanation: Containment Spray Raw Water pump 121 supplies alternate injection to Containment Spray loop 11.

A. Incorrect – Containment Spray Raw Water pump 121 supplies alternate injection to Containment Spray loop 11, not Core Spray loop 11. Plausible because Containment Spray Raw Water pump 111 supplies alternate injection to Core Spray loop 11.
B. Incorrect – Containment Spray Raw Water pump 121 supplies alternate injection to Containment Spray loop 11, not Core Spray loop 12. Plausible because Containment Spray Raw Water pump 122 supplies alternate injection to Core Spray loop 12.
D. Incorrect – Containment Spray Raw Water pump 121 supplies alternate injection to Containment Spray loop 11, not Containment Spray loop 12. Plausible because Containment Spray Raw Water pump 112 supplies alternate injection to Containment Spray loop 12.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226001-RBO-3

Question Source: Bank – 2017 Cert #31

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 A4.09
	Importance Rating	2.6

Main Turbine Generator/Auxiliary

Ability to manually operate and/or monitor in the control room: Hydrogen seal oil pressure

Proposed Question: #59

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

- A malfunction occurs with the Generator H_2 Seal Oil pressure regulator.
- Annunciator A1-4-1, Generator H_2 Seal Oil Pressure Low, alarms.
- Generator H_2 Seal Oil pressure is 35 psig and steady.
- Main Seal Oil pump discharge pressure is 110 psig and steady.
- Operator control of the Generator H_2 Seal Oil pressure regulator bypass is required to raise seal oil pressure.

Which one of the following identifies the status of the Emergency Seal Oil pump and the location for taking control of the Generator H_2 Seal Oil pressure regulator bypass?

	Emergency Seal Oil Pump	Location for Control of Generator H_2 Seal Oil Pressure Regulator Bypass
A.	Has automatically started	Control Room Panel A1
B.	Has automatically started	Turbine Building El. 277'
C.	Remains in standby	Control Room Panel A1
D.	Remains in standby	Turbine Building El. 277'

Proposed Answer: D

Explanation: The Emergency Seal Oil pump remains in standby because Main Seal Oil pump discharge pressure remains above 90 psig. The pressure regulator is bypassed by opening HS-5, which is controlled on Turbine Building el. 277'.

A. Incorrect – The Emergency Seal Oil pump remains in standby because Main Seal Oil pump discharge pressure remains above 90 psig. Plausible because this would be correct if the malfunction also caused Main Seal Oil pump discharge pressure to lower. Also plausible that the Emergency Seal Oil pump would auto-start based on pressure at the Generator, not at the discharge of the Main Seal Oil pump. The pressure regulator is bypassed by opening HS-5, which is controlled on Turbine Building el. 277'. Plausible because there are multiple Seal Oil indications at Control Room Panel A1 and the Steam Seal regulator bypass is controlled at Control Room Panel A.

B. Incorrect – The Emergency Seal Oil pump remains in standby because Main Seal Oil pump discharge pressure remains above 90 psig. Plausible because this would be correct if the malfunction also caused Main Seal Oil pump discharge pressure to lower. Also plausible that the Emergency Seal Oil pump would auto-start based on pressure at the Generator, not at the discharge of the Main Seal Oil pump.

C. Incorrect – The pressure regulator is bypassed by opening HS-5, which is controlled on Turbine Building el. 277'. Plausible because there are multiple Seal Oil indications at Control Room Panel A1 and the Steam Seal regulator bypass is controlled at Control Room Panel A.

Technical Reference(s): N1-SOP-32, N1-OP-7, 1101-252000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-252000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:
TRH 11/14/18 – Revised last bullet and wording of 1st column A and B, based on NRC comment.
TRH 11/28/18 – Revised to eliminate "manual" based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 K6.06
	Importance Rating	2.7

Feedwater

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: Plant service water

Proposed Question: #60

The plant has experienced a loss of all normal Service Water pumps with the following:

- The Reactor has been manually scrammed.
- Emergency Service Water (ESW) pump 11 has been started.
- ESW pump 12 has failed to start.

Which one of the following identifies the Feedwater pump(s), if any, that is(are) being cooled by ESW (directly or indirectly)?

A. None

B. Feedwater pump 11, only

C. Feedwater pumps 11 and 12, only

D. Feedwater pumps 11, 12, and 13

Proposed Answer: C

Explanation: ESW pump 11 is supplying cooling water to all RBCLC heat exchangers, but none of the TBCLC heat exchangers. Feedwater pumps 11 and 12 are cooled by RBCLC and Feedwater pump 13 is cooled by TBCLC. Therefore, Feedwater pumps 11 and 12 are being indirectly cooled by ESW through the RBCLC system. Feedwater pump 13 is NOT being cooled by ESW because ESW is not cooling TBCLC.

A. Incorrect – Feedwater pumps 11 and 12 are being indirectly cooled by ESW through the RBCLC system. Plausible because the Feedwater pumps are in the Turbine Building, most plants have Feedwater pumps cooled by TBCLC, and TBCLC is not being cooled by ESW.
B. Incorrect – Feedwater pump 12 is also being indirectly cooled by ESW. Plausible because check valves on the ESW pump discharge piping does limit where ESW flow can go and ESW pump 12 is not running even though N1-SOP-18.1 would require it to be started.
D. Incorrect – Feedwater pump 13 is NOT being cooled by ESW because ESW is not cooling TBCLC. Plausible because Feedwater pumps 11 and 12 are indirectly cooled by ESW. Also plausible because all three Feedwater Booster pumps are indirectly cooled by ESW.

Technical Reference(s): N1-SOP-18.1, C-18022-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-8

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 #	268000 K1.05
	Importance Rating	2.9

Radwaste

Knowledge of the physical connections and/or cause-effect relationships between RADWASTE and the following: Drywell equipment drains

Proposed Question: #61

Which one of the following conditions would result in the Drywell Equipment Drain Isolation Valves receiving an automatic closure signal?

A. Reactor water level at 25"

B. Drywell pressure at 4.0 psig

C. Drywell temperature at 160°F

D. Drywell equipment drain leakage at 30 gpm

Proposed Answer: B

Explanation: Drywell equipment drain tank isolation valves (83.1-09 and 83.1-10) automatically close on Drywell pressure >3.5 psig.

A. Incorrect – Reactor water level at 25" does not close Drywell equipment drain tank isolation valves. Plausible because Reactor water level <+5" does close Drywell equipment drain tank isolation valves.
C. Incorrect – Drywell temperature at 160°F does not close Drywell equipment drain tank isolation valves. Plausible because this is an N1-EOP-4 entry condition and an indication of a potential RCS leak into the Drywell which could require Containment isolation.
D. Incorrect – Drywell equipment drain leakage at 30 gpm does not close Drywell equipment drain tank isolation valves. Plausible because Drywell Equipment Drain IV closure is designed to prevent excess leakage from exiting Primary Containment. Also plausible because this is above the 25 gpm leakage value used in Technical Specifications.

Technical Reference(s): N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-291000-RBO-5

Question Source: Modified Bank – 2013 NRC #41

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 11/14/18 – Revised wording of question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 K2.05
	Importance Rating	2.6

Radiation Monitoring

Knowledge of electrical power supplies to the following: Reactor building ventilation monitors: Plant-Specific

Proposed Question: #62

Which one of the following identifies the power supply to Reactor Building Ventilation radiation monitor 11?

A. RPS Bus 11

B. I&C Bus 130

C. I&C Bus 130X

D. Distribution Panel 167A

Proposed Answer: A

Explanation: Reactor Building Ventilation radiation monitor 11 is powered from RPS Bus 11.

B. Incorrect – Reactor Building Ventilation radiation monitor 11 is powered from RPS Bus 11. Plausible because I&C Bus 130 supplies power to area radiation monitors.
C. Incorrect – Reactor Building Ventilation radiation monitor 11 is powered from RPS Bus 11. Plausible because I&C Bus 130X supplies power to a Service Water radiation monitor.
D. Incorrect – Reactor Building Ventilation radiation monitor 11 is powered from RPS Bus 11. Plausible because Distribution Panel 167A supplies power to a Stack radiation monitoring.

Technical Reference(s): N1-SOP-40.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-272000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	288000 K5.02
	Importance Rating	3.2

Plant Ventilation

Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS: Differential pressure control

Proposed Question: #63

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

- High winds are occurring.
- Annunciator L1-3-4, REACT BLDG/ATM DIFF PRESS, is in alarm
- Reactor Building differential pressure is -0.05" H_2O.
- An Operator has taken manual control of Reactor Building differential pressure per N1-OP-10, Reactor Building Heating, Cooling, and Ventilating System.

Which one of the following describes the required action to restore Reactor Building differential pressure, in accordance with N1-OP-10?

Adjust the Reactor Building differential pressure controller to further...

A. open the Reactor Building supply inlet damper.

B. close the Reactor Building supply inlet damper.

C. open the Reactor Building exhaust outlet damper.

D. close the Reactor Building exhaust outlet damper.

Proposed Answer: B

Explanation: The Reactor Building differential pressure controller is used to modulate the Reactor Building supply inlet damper. The given differential pressure is too low (not negative enough). To make differential pressure more negative, the supply inlet damper must be throttle further closed, restricting the flow of air into the building.

A. Incorrect – The supply inlet damper must be throttled further closed, not open. Plausible because this would be correct if D/P was too negative or if the candidate reverses the cause-effect relationship of this system.
C. Incorrect – This controller controls the supply inlet damper, not the exhaust outlet damper. Plausible because this is another possible way to design building D/P control and the exhaust outlet damper does have automatic function.
D. Incorrect – This controller controls the supply inlet damper, not the exhaust outlet damper. Plausible because this is another possible way to design building D/P control and the exhaust outlet damper does have automatic function.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-3

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 #	290001 A3.01
	Importance Rating	3.9

Secondary Containment

Ability to monitor automatic operations of the SECONDARY CONTAINMENT including: Secondary containment isolation

Proposed Question: #64

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

Time (mm:ss)	Condition
00:00	• Annunciator L1-4-3, REACT BLDG VENT RAD MONITOR OFF NORMAL, alarms. • Reactor Building Exhaust radiation monitor 12 indicates downscale.
05:00	• Reactor Building Exhaust radiation monitor 11 fails upscale.

Which one of the following describes the response of Reactor Building Ventilation (RBVS) to these conditions?

RBVS...

A. automatically isolated at time 00:00.

B. automatically isolated at time 05:00.

C. did NOT automatically isolate and has lost automatic isolation capability.

D. did NOT automatically isolate and still has automatic isolation capability.

Proposed Answer: B

Explanation: RBVS isolation logic requires one out of two upscale signals to function. Upon upscale trip of either of two radiation monitors, the system isolates. Downscale conditions do not cause isolation for this system. When Reactor Building Exhaust Radiation Monitor 12 fails downscale at time 00:00, RBVS remains un-isolated. When Reactor Building Exhaust Radiation Monitor 11 fails upscale at time 05:00, both RBVS automatically isolates.

A. Incorrect – There are only two radiation monitors and one of them is inoperable at time 00:00. However, the logic does not have a feature to isolate RBVS on such a downscale failure. Plausible because other rad monitor logics do have a downscale feature that inputs to an actuation.
C. Incorrect – RBVS isolates at time 05:00. Plausible because Reactor Building Exhaust Radiation Monitor 12 has failed downscale, which prevents it from causing an isolation.
D. Incorrect – RBVS isolates at time 05:00. Plausible because Reactor Building Exhaust Radiation Monitor 12 has failed downscale, which prevents it from causing an isolation, but in some circumstances SFP high range ARM could cause RBVS isolation.

Technical Reference(s): N1-OP-50B, C-19859-C Sheet 15

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-5

Question Source: Modified Bank - 2017 Cert #17

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 11/15/18 – Reworded 1st half of C and D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290002 K3.01
	Importance Rating	3.2

Reactor Vessel Internals

Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following: Reactor water level

Proposed Question: #65

The plant is operating at 100% power when a design basis loss of coolant accident occurs.

Which one of the following identifies where Reactor water level stabilizes, in accordance with the plant safety analysis?

A. Above -84"

B. Between -109" and -84"

C. Between -230" and -109"

D. Below -230"

Proposed Answer: D

Explanation: The design basis loss of coolant accident is a double-ended guillotine shear of a Recirc line discharge pipe. This accident results in Reactor water level lowering below the bottom of active fuel (-230") and adequate core cooling being maintained by Core Spray cooling.

Note: The question meets the K/A by testing knowledge of the effect that a malfunction of a Reactor vessel nozzle will have on Reactor water level. The DBA LOCA is a double-ended guillotine break of a Reactor vessel nozzle. By close inspection of the 290002 system in NUREG 1123, it has been determined that the intent of this system is to include the entire Reactor vessel ***and*** Reactor vessel internals. This includes Reactor vessel nozzles. Also note that NMP1 does not have jet pumps, which limits the potential questions that can be asked for this K/A when compared to other BWRs.

A. Incorrect – The DBA LOCA results in Reactor water level below -230". Plausible because above -84" would be required to ensure adequate core cooling based on submergence.
B. Incorrect – The DBA LOCA results in Reactor water level below -230". Plausible because between -109" and -84" is the N1-EOP-2 band used to ensure adequate core cooling based on steam cooling.
C. Incorrect – The DBA LOCA results in Reactor water level below -230". Plausible because this is the correct range for the majority of BWRs based on the Reactor vessel design including jet pumps.

Technical Reference(s): FSAR Chapter 15

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-101001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 11/19/18 – Added K/A match statement based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.4
	Importance Rating	3.3

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: #66

The plant is operating at 100% power.

Which one of the following identifies:

(1) the ***minimum*** number of licensed Unit 1 Reactor Operators that must be on shift, and

(2) whether one of these minimum required Reactor Operators may leave the Unit 1 Power Block without relief,

in accordance with OP-NM-103-102, Watch-Standing Practices at Nine Mile Point?

	(1) Minimum Number of Licensed Unit 1 Reactor Operators	(2) May One of The Minimum Required Reactor Operators Leave the Unit 1 Power Block Without Relief?
A.	2	Yes
B.	2	No
C.	3	Yes
D.	3	No

Proposed Answer: A

Explanation: The minimum number of licensed Unit 1 Reactor Operators that must be on shift with the plant at 100% power is 2. One of these ROs must remain within the Control Room ATC area. The other RO must normally remain within the Protected Area, but is allowed to leave the Unit 1 Power Block.

Note: The question meets the K/A by testing a shift staffing requirement (minimum number of ROs) and an individual operator responsibility related to this shift staffing requirement (whether one of these ROs may leave the power block without relief).

B. Incorrect – One of the required ROs may leave the Unit 1 Power Block. Plausible because these ROs are restricted on where they may go without relief, and the Unit 1 Power Block includes all the normal areas they are responsible for performing licensed duties.
C. Incorrect – The minimum number of ROs is 2. Plausible because the minimum number is 3 under other situations.
D. Incorrect – The minimum number of ROs is 2. Plausible because the minimum number is 3 under other situations. One of the required ROs may leave the Unit 1 Power Block. Plausible because these ROs are restricted on where they may go without relief, and the Unit 1 Power Block includes all the normal areas they are responsible for performing licensed duties.

Technical Reference(s): OP-NM-103-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

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 #	2.1.37
Importance Rating	4.3

Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Proposed Question: #67

The plant is operating at approximately 100% power and power maintenance is being performed to maintain steady-state, full-power conditions.

Which one of the following is the power maintenance goal as identified in N1-OP-43B, Normal Power Operations?

Maintain computer point...

A. E635 – CTP-2 HR AVG between 1846 to 1850 MWth.

B. E635 – CTP-2 HR AVG between 1840 to 1855 MWth.

C. H305 – 10 MINUTE AVERAGE OF CTPINST between 1846 to 1850 MWth.

D. H305 – 10 MINUTE AVERAGE OF CTPINST of CTPINST between 1840 to 1855 MWth.

Proposed Answer: A

Explanation: N1-OP-43B P&L 16.0 states, "When performing power maintenance during steady state full power conditions, the goal is to maintain the 2 hour averaged E635 – CTP-2 HR AVG point between 1846 to 1850 MWth."

B. Plausible – The band is 1846-1850 MWth. Plausible because 1840 MWth is a power level used in P&L 24.0 and 1855 MWth is a power level used in P&L 17.0.
C. Plausible – The computer point is E635, not H305. Plausible because H305 is a computer point discussed in P&L 16.0 and it is an averaged point.
D. Plausible – The computer point is E635, not H305. Plausible because H305 is a computer point discussed in P&L 16.0 and it is an averaged point. The band is 1846-1850 MWth. Plausible because 1840 MWth is a power level used in P&L 24.0 and 1855 MWth is a power level used in P&L 17.0.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-9

Question Source: Bank – 2017 Cert #70

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/15/18 – Revised noun name for H305 and wording of band, based on NRC comment.
PFI 11/29/18 – In the future, this will need slight revision based on changed implemented in November 2018, in light of the Unit 2 over power event. Unit 1 is creating a new 1 hour avg computer point that they will maintain 1846-1850 on. The 2 hour avg limit becomes 1849.5. I think...

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.43
	Importance Rating	4.1

Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Proposed Question: #68

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

- Heating from Feedwater Heater string 12 is lost.
- Reactor power is lowered to 85% per N1-SOP-1.1, Emergency Power Reduction.
- Feedwater temperatures are as follows:
  - A390 FW ENT RX TEMP W: 330°F
  - A392 FW ENT RX TEMP E: 331°F

Note: A portion of N1-SOP-16.1, Feedwater System Failures is provided on the following page.

Which one of the following describes the required control of Reactor power, in accordance with N1-SOP-16.1, Feedwater System Failures?

Reactor power…

A. may be maintained at the current level.

B. must be lowered to 75%. NO further power reduction is required.

C. must be lowered until either Feedwater temperatures return to band or Reactor power reaches 25%.

D. must be lowered to 25% regardless of Feedwater temperatures.

N1-SOP-16.1 - MINIMUM FW TEMP.

MWth	% Power	FW Temp	MWth	% Power	FW Temp
1850	100.0	350.0	1150	62.5	309.5
1840	99.5	349.5	1140	61.6	308.8
1830	98.9	348.9	1130	61.1	308.2
1820	98.4	348.4	1120	60.5	307.5
1810	97.8	347.8	1110	60.0	306.9
1800	97.3	347.3	1100	59.5	306.2
1790	96.8	346.8	1090	58.9	305.6
1780	96.2	346.2	1080	58.4	304.9
1770	95.7	345.7	1070	57.8	304.3
1760	95.1	345.1	1060	57.3	303.6
1750	94.6	344.6	1050	56.8	303.0
1740	94.1	344.1	1040	56.2	302.3
1730	93.5	343.5	1030	55.7	301.7
1720	93.0	343.0	1020	55.1	301.0
1710	92.4	342.4	1010	54.6	300.4
1700	91.9	341.9	1000	54.1	299.7
1690	91.4	341.4	990	53.5	299.0
1680	90.8	340.8	980	53.0	298.4
1670	90.3	340.3	970	52.4	297.7
1660	89.7	339.7	960	51.9	297.1
1650	89.2	339.2	950	51.4	296.4
1640	88.6	338.6	940	50.8	295.8
1630	88.1	338.1	930	50.3	295.1
1620	87.6	337.6	920	49.7	294.3
1610	87.0	337.0	910	49.2	293.2
1600	86.5	336.5	900	48.6	292.2
1590	85.9	335.9	890	48.1	291.2
1580	85.4	335.4	880	47.6	290.1
1570	84.9	334.9	870	47.0	289.1
1560	84.3	334.3	860	46.5	288.1
1550	83.8	333.8	850	45.9	287.0
1540	83.2	333.2	840	45.4	286.0
1530	82.7	332.7	830	44.9	285.0
1520	82.2	332.2	820	44.3	283.9
1510	81.6	331.6	810	43.8	282.9
1500	81.1	331.1	800	43.2	281.9
1490	80.5	330.5	790	42.7	280.8
1480	80.0	330.0	780	42.2	279.8
1470	79.5	329.5	770	41.6	278.8
1460	78.9	328.9	760	41.1	277.7
1450	78.4	328.4	750	40.5	276.7
1440	77.8	327.8	740	40.0	275.7
1430	77.3	327.3	730	39.5	274.6
1420	76.8	326.8	720	38.9	273.6
1410	76.2	326.2	710	38.4	272.6
1400	75.7	325.7	700	37.8	271.5
1390	75.1	325.1	690	37.3	270.5
1380	74.6	324.5	680	36.8	269.5
1370	74.1	323.9	670	36.2	268.4
1360	73.5	323.2	660	35.7	267.4
1350	73.0	322.6	650	35.1	266.4
1340	72.4	321.9	640	34.6	265.3
1330	71.9	321.2	630	34.1	264.3
1320	71.4	320.6	620	33.5	263.3
1310	70.8	319.9	610	33.0	262.2
1300	70.3	319.3	600	32.4	261.2
1290	69.7	318.6	590	31.9	260.2
1280	69.2	318.0	580	31.4	259.1
1270	68.6	317.3	570	30.8	258.1
1260	68.1	316.7	560	30.3	257.1
1250	67.6	316.0	550	29.7	256.0
1240	67.0	315.4	540	29.2	255.0
1230	66.5	314.7	530	28.6	254.0
1220	65.9	314.1	520	28.1	252.9
1210	65.4	313.4	510	27.6	251.9
1200	64.9	312.8	500	27.0	250.9
1190	64.3	312.1	490	26.5	249.8
1180	63.8	311.5	480	25.9	248.8
1170	63.2	310.8	470	25.4	247.8
1160	62.7	310.1			

Proposed Answer: C

Explanation: The given excerpt from N1-SOP-16.1 is a table of the minimum Feedwater temperatures allowed based on Reactor power. The given plant conditions show Feedwater temperatures entering the Reactor are below the minimum allowable (335°F for 85%). Therefore, N1-SOP-16.1 requires a Reactor power reduction. The power reduction must continue until either Feedwater temperatures are restored above the minimum allowable or Reactor power reaches 25%.

A. Incorrect – The given plant conditions show Feedwater temperatures entering the Reactor are below the minimum allowable (335°F for 85%). Therefore, N1-SOP-16.1 requires a Reactor power reduction. Plausible if the limits of the given table are misunderstood (minimum vs. maximum) or misinterpreted.
B. Incorrect – Reactor power will need to be reduced to less than 75%. Plausible because this is 25% below the initial power level and 25% is used in N1-SOP-16.1. Also plausible because the current Feedwater temperatures would first become SAT at 75% power, however Feedwater temperatures will lower with the power reduction, making a reduction to 75% not enough.
D. Incorrect – The power reduction may be stopped before 25% power, as long as Feedwater temperature limits can be satisfied. Plausible because Feedwater temperature will tend to lower further as power is reduced. Also plausible because N1-SOP-16.1 does discuss lowering power to 25%. Also plausible that operation without an entire Feedwater Heater string (vice loss of just an individual heater) would have an absolute power limit.

Technical Reference(s): N1-SOP-16.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-260000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/15/18 – Revised "become acceptable" in C and added more to choice D justification, based on NRC comment.
TRH 11/20/18 – Replaced cut-off temperature table based on NRC comment.
TRH 11/28/18 – Redacted note at bottom of table and verified current SOP revision, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.2
	Importance Rating	4.6

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Proposed Question: #69

A Reactor startup is in progress with the following:

- The next control rod in the startup sequence is to be moved from position 00 to 48.
- Initial and current SRM readings are as follows:

SRM	Initial Reading (cps)	Current Reading (cps)
11	240	700
12	270	840
13	220	730
14	260	750

Which one of the following describes the requirement, if any, to use notch withdrawal for this control rod based on SRM readings, in accordance with N1-OP-43A, Plant Startup?

Notch withdrawal…

A. is required at all positions.

B. is NOT required at any position.

C. is required from positions 04 to 16 only.

D. is required from positions 00 to 36 only.

Proposed Answer: B

Explanation: N1-OP-43A P&L 2.6 requires notch control rod movement from positions 00 to 36 once 3 count rate doublings are approached until criticality is achieved. The given SRM data shows two SRMs have tripled, but no SRMs are approaching 3 count rate doublings. Therefore, there is no requirement to use notch movement for the next control rod.

A. Incorrect – There is no requirement to use notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, notch withdrawal would be required, but from 00 to 36 only.
C. Incorrect – There is no requirement to use notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, notch withdrawal would be required, but from 00 to 36 only. 04 to 16 are the positions identified as having the highest control rod worth per N1-OP-43A P&L 2.2.
D. Incorrect – There is no requirement to use notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, notch withdrawal would be required from 00 to 36 only.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-10

Question Source: Bank - 2015 Cert #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 #	2.2.22
	Importance Rating	4.0

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #70

The plant is operating at 100% power when an unexpected transient occurs.

Which one of the following describes a resulting condition that will violate a Technical Specification Safety Limit?

A. Reactor water level drops to -20”

B. Reactor pressure rises to 1320 psig

C. Torus water temperature rises to 125°F

D. Minimum Critical Power Ratio (MCPR) lowers to 1.21

Proposed Answer: A

Explanation: Reactor water level <-10" violates Safety Limit 2.1.1.d.

B. Incorrect – This is not a Safety Limit violation. Plausible because it exceeds all Safety valve setpoints and is close to the Safety Limit of 1375 psig.
C. Incorrect – This is not a Safety Limit violation. Plausible because this does exceed the limits of 85°F, 110°F, and 120°F in LCO 3.3.2, as well as the HCTL curve at normal operating pressure.
D. Incorrect – This is not a Safety Limit violation. Plausible because MCPR less than 1.07 would be a Safety Limit violation and the given MCPR is an LCO violation.

Technical Reference(s): Technical Specifications 2.2.1 and 2.2.2, COLR

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-101001-RBO-14

Question Source: Bank - 2009 NRC #70

Question History: 2009 NRC #70

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/15/18 – Raised Torus water temperature to 125°F based on NRC comment.

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

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

Proposed Question: #71

The plant is operating 100% power when the power source to the Area Radiation Monitors (ARMs) de-energizes due to a spurious breaker trip.

Which one of the following describes the response of the Area Radiation Monitors (ARMs) and annunciator H1-4-8, Area Radiation Monitors?

The ARMs...

A. go into alarm. When the power source is re-energized, H1-4-8 automatically clears.

B. go into alarm. When the power source is re-energized, H1-4-8 does NOT automatically clear.

C. lose the ability to alarm. When the power source is re-energized, H1-4-8 automatically regains alarm capability.

D. lose the ability to alarm. When the power source is re-energized, H1-4-8 does NOT automatically regain alarm capability.

Proposed Answer: B

Explanation: I&C Bus 130 supplies power to the ARMs. Upon loss of I&C Bus 130, all the ARMs go into alarm and annunciator H1-4-8, AREA RADIATION MONITORS, alarms. Upon bus restoration, each ARM RESET pushbuttons must be depressed to clear the alarm.

A. Incorrect – Upon bus restoration, each ARM RESET pushbuttons must be depressed to clear the alarm. Plausible because power is automatically restored to each ARM as soon as I&C Bus 130 is re-energized.
C. Incorrect – Upon loss of I&C Bus 130, all the ARMs go into alarm. Plausible to lose alarm function because the ARMs do lose power. Upon bus restoration, each ARM RESET pushbuttons must be depressed to clear the alarm. Plausible because power is automatically restored to each ARM as soon as I&C Bus 130 is re-energized.
D. Incorrect – Upon loss of I&C Bus 130, all the ARMs go into alarm. Plausible to lose alarm function because the ARMs do lose power.

Technical Reference(s): N1-OP-50A, N1-272000-RBO-8

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-272000-RBO-8

Question Source: Bank – 2017 Cert #74

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:
TRH 11/15/18 – Revised stem and choices to get rid of reference to I&C Bus 130 based on NRC comment.
TRH 11/28/18 – Specified H1-4-8 in question and answer choices based on SRO comment.
TRH 11/28/18 – Fixed typo in D based on NRC comment.

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: #72

A transient is in progress with the following:

- Operators are required to enter a radiologically posted area in order to manually close a Primary Containment isolation valve.
- Radiation Protection (RP) has surveyed the area and determined that the highest general area dose rate in the area is 1500 mRem/hr.

Which one of the following describes the radiological requirements for the area and the entry, in accordance with RP-AA-460, Controls for High and Locked 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.	High Radiation Area, only	are
B.	High Radiation Area, only	are NOT
C.	Locked High Radiation Area	are
D.	Locked High Radiation Area	are NOT

Proposed Answer: D

Explanation: This area has a general area dose rate >1000 mRem/hr, therefore it must be posted as a Locked 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. Incorrect – The area must be posted as a Locked High Radiation Area due to general area dose rates >1000 mRem/hr. Plausible because the dose rate is near the threshold for High Radiation Area / Locked High Radiation Area. Operators do NOT need to have a continuous RP escort while in the area. Plausible because conditions are emergent and radiation levels are high.
B. Incorrect – The area must be posted as a Locked High Radiation Area due to general area dose rates >1000 mRem/hr. Plausible because the dose rate is near the threshold for High Radiation Area / Locked High Radiation Area.
C. Incorrect – Operators do NOT need to have a continuous RP escort while in the area. Plausible because conditions are emergent and radiation levels are high.

Technical Reference(s): RP-AA-460

Proposed references to be provided to applicants during examination: None

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

Question Source: Modified Bank - 2015 NRC #72

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:
TRH 11/20/18 – Replaced question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.14
	Importance Rating	3.8

Knowledge of general guidelines for EOP usage.

Proposed Question: #73

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

- At time 00:00, the N1-EOP-4, Primary Containment Control, entry condition for Torus temperature is exceeded.
- At time 01:00, the N1-EOP-4 entry condition for Torus water level is exceeded. The N1-EOP-4 entry condition for Torus temperature is also still exceeded.

Which one of the following describes the required usage of N1-EOP-4?

At time 00:00, ...

A. all legs of N1-EOP-4 must be entered concurrently. At time 01:00, all of N1-EOP-4 must be re-entered.

B. all legs of N1-EOP-4 must be entered concurrently. At time 01:00, just the Torus water level leg of N1-EOP-4 must be re-entered.

C. only the Torus temperature leg of N1-EOP-4 must be entered. At time 01:00, only the Torus water level leg of N1-EOP-4 must be entered.

D. only the Torus temperature leg of N1-EOP-4 must be entered. At time 01:00, the Torus water level leg of N1-EOP-4 must be entered and the Torus temperature leg must be re-entered.

Proposed Answer: A

Explanation: When an EOP entry condition is exceeded, the associated EOP is entered and all legs are entered concurrently. When an additional EOP entry condition is exceeded, the entire EOP is re-entered and all legs are re-entered.

B. Incorrect – Upon re-entry, all legs must be re-entered. Plausible since only the Torus water level entry condition was exceeded at time 01:00.
C. Incorrect – Upon entry, all legs are entered. Plausible since only one leg has a parameter exceeded an entry condition at time 0:00.
D. Incorrect – Upon entry, all legs are entered. Plausible since only one leg has a parameter exceeded an entry condition at time 0:00.

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 – 2017 Cert #75

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/15/18 – Added that Torus water temperature is still above EOP entry condition at time 01:00 based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.32
	Importance Rating	3.6

Knowledge of operator response to loss of all annunciators.

Proposed Question: #74

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

- Reactor power is being lowered to 75% using Recirc flow in preparation for a planned control rod pattern adjustment.
- You are about to begin N1-PM-M4, Monthly Service Water Pump Rotation.

Then, a loss of all Main Control Room annunciators occurs with the following:

- Additional personnel have been called in to assist with monitoring the plant but have NOT yet arrived.
- The Plant Process Computer has been verified to be operating properly.

Which one of the following describes the required control of the power reduction and the Service Water pump rotation, in accordance with N1-SOP-42, Loss of Annunciators?

A. The power reduction and Service Water pump rotation must be suspended.

B. The power reduction must be suspended, but the Service Water pump rotation may continue.

C. The power reduction and Service Water pump rotation may continue with no additional actions.

D. The power reduction and Service Water pump rotation may continue, but only once additional personnel are stationed to monitor parameters.

Proposed Answer: A

Explanation: N1-SOP-42 requires the following:

- Suspend all activities that are NOT absolutely essential for safe plant operation.
- Secure all surveillance/testing activities.
- Do NOT initiate any transients on the plant that are NOT absolutely essential for safe plant operation.

Since the power reduction and Service Water pump rotation are not emergent activities, they must be suspended.

B. Incorrect – The Service Water pump rotation must also be suspended. Plausible because the power reduction more broadly affects the plant, while the Service Water pump rotation is a more narrowly focused evolution.
C. Incorrect – The power reduction and Service Water pump rotation must be suspended. Plausible since the PPC has been verified to be working properly and it will provide some alarm function.
D. Incorrect – The power reduction and Service Water pump rotation must be suspended. Plausible because N1-SOP-42 does direct stationing additional personnel for plant monitoring.

Technical Reference(s): N1-SOP-42

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP42C01

Question Source: Bank - 2008 NRC #72

Question History: 2008 NRC #72

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 11/15/18 – Fixed typo based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.22
	Importance Rating	3.6

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Proposed Question: #75

A loss of coolant accident has resulted in the following:

- An RPV Blowdown has been performed due to exceeding the Pressure Suppression Pressure (PSP).
- Reactor pressure is 75 psig and stable.
- Reactor water level is -85" and slowly lowering with NO injection in service.
- Maintenance has just made one Core Spray pump and one Condensate pump available.
- The Primary Containment Pressure Limit (PCPL) is being challenged.

Which one of the following identifies the preferred pump to use for Reactor injection in this situation and which issue is higher priority, in accordance with the Emergency Operating Procedures (EOPs)?

	Preferred Reactor Injection Source	Higher Priority
A.	Core Spray	Prevent exceeding PCPL
B.	Core Spray	Maintaining adequate core cooling
C.	Condensate	Prevent exceeding PCPL
D.	Condensate	Maintaining adequate core cooling

Proposed Answer: B

Explanation: N1-EOP-2 contains the following step:

You cannot stay inside the Primary Containment Pressure Limit (Fig D)	Stop injection from sources outside the primary containment not needed for core cooling.

Core Spray injects from within the Primary Containment, while Condensate injection from outside the Primary Containment. Therefore, with PCPL being challenged, the preferred injection source is Core Spray. Both PCPL and adequate core cooling are currently being challenged. The priority is injection to maintain adequate core cooling.

A. Incorrect – The priority is injection to maintain adequate core cooling. Plausible because in other situations, maintaining Primary Containment integrity would take priority.
C. Incorrect – Core Spray is the preferred injection source. Plausible because Condensate is a cleaner source of water and more easily controlled. The priority is injection to maintain adequate core cooling. Plausible because in other situations, maintaining Primary Containment integrity would take priority.
D. Incorrect – Core Spray is the preferred injection source. Plausible because Condensate is a cleaner source of water and more easily controlled.

Technical Reference(s): N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Modified Bank – 2015 Cert #100

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295006 2.1.19
	Importance Rating	3.8

SCRAM

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

Proposed Question: #76

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

- Feedwater pump 13 spuriously de-clutches.
- Reactor water level lowers rapidly.
- A Reactor Operator places the Reactor Mode Switch in SHUTDOWN.
- The exact timing of these events was unclear due to other activities in the Control Room.
- The related plant computer typer data is provided on the following page.
- Given the following potentially applicable portion of the EAL matrix:

1	1
An automatic scram failed to shut down the reactor as indicated by reactor power > 6% **AND** Manual actions taken at the reactor control console (mode switch in shutdown, manual scram push buttons and ARI) failed to shut down the reactor as indicated by reactor power > 6%	An automatic scram failed to shut down the reactor **AND** Manual actions taken at the reactor control console (mode switch in shutdown, manual scram push buttons or ARI) successfully shut down the reactor as indicated by reactor power ≤ 6%

Which one of the following describes the need for EAL declaration and notification, in accordance with the Site Emergency Plan?

A. Declare an Alert and perform the associated Part 1 notification.

B. Declare a Site Area Emergency and perform the associated Part 1 notification.

C. NEITHER of these EALs require declaration and NO Part 1 notification is required.

D. NEITHER of these EALs require declaration, but a Part 1 notification is required for a Transitory Event.

Date	Time	Type	Point Name	Point Description	Status	Value
Today	39:07.8	DI	A013	FWP 13 CLUTCH HYD PRES LOW	L2 Alarm	LOW
Today	39:10.8	DI	A020	FW PMP 13 DIS PR	L2 Alarm	LOW
Today	39:14.6	AI	SPDS_RPVWL	RPV WATER LEVEL	Level 2 Low	64.331
Today	39:15.1	DI	W049	***RPS CH12/1 RX LO LVLD	L2 Alarm	LOW
Today	39:15.1	DI	W007	***RPS CH11/2RX LO LVL C	L2 Alarm	LOW
Today	39:15.6	DI	W048	***RPS CH12/2 RX LO LVLB	L2 Alarm	LOW
Today	39:15.6	DI	W006	***RPS CH11/1RX LO LVL A	L2 Alarm	LOW
Today	39:15.8	DI	W087	HPCI	L2 Alarm	YES
Today	39:19.6	AI	SPDS_RPVWL	RPV WATER LEVEL	Level 3 Low	50.948
Today	39:21.8	DI	J185	BELOW LOW PWR SET POINT	Rtn Normal	NO
Today	39:26.0	DI	H163	SPDS RX MODE RUN	Rtn Normal	NO
Today	39:26.0	DI	H161	SPDS RX MODE SHUTDOWN	L1 ALarm	YES
Today	39:26.3	DI	J183	CONTROL ROD DRIFT	L2 Alarm	TRUE
Today	39:26.3	DI	B009	CRD CNTL AIR PR	L2 Alarm	OFFN
Today	39:26.3	DI	B013	CRD DRIFT	L2 Alarm	DRIFT
Today	39:26.8	DI	W068	***RPS CH12 MAN RX TRIP	L2 Alarm	YES
Today	39:26.8	DI	W022	***RPS CH11 MAN RX TRIP	L2 Alarm	YES
Today	39:28.0	DI	B188	APRM 18 DOWN SCALE	L2 Alarm	TRIP
Today	39:28.3	DI	B187	APRM 17 DOWN SCALE	L2 Alarm	TRIP
Today	39:28.3	DI	B186	APRM 16 DOWN SCALE	L2 Alarm	TRIP
Today	39:28.3	DI	B185	APRM 15 DOWN SCALE	L2 Alarm	TRIP
Today	39:28.8	DI	B180	APRM 14 DOWN SCALE	L2 Alarm	TRIP
Today	39:28.8	DI	B179	APRM 13 DOWN SCALE	L2 Alarm	TRIP
Today	39:28.8	DI	B178	APRM 12 DOWN SCALE	L2 Alarm	TRIP
Today	39:28.8	DI	B177	APRM 11 DOWN SCALE	L2 Alarm	TRIP
Today	39:29.8	DI	B010	SCRAMDUMP VOL HI LVL	L2 Alarm	HIGH
Today	39:29.8	AI	H441	SPDS APRM	Rtn Normal	5.8047
Today	39:29.8	AI	H441	SPDS APRM	Rtn Normal	5.8047
Today	39:31.0	DI	B011	SCRAMDUMP VOL HIHI LVL	L2 Alarm	HIHI

Proposed Answer: A

Explanation: The plant computer typer data shows that an automatic scram was required at time 39:15 (all four RPS low Reactor water level channels in alarm low). However, there is no indication of control rod motion or Reactor power reduction in the next 10 seconds. At time 39:26, the manual scram occurs, control rods begin moving, and shortly thereafter Reactor power is downscale on APRMs. Therefore, the automatic scram failed to shutdown the Reactor, but manual action did insert control rods and result in APRM downscales. This requires declaration of an Alert per EAL SA3.1 and completion of the associated Part 1 notification.

B. Incorrect – An Alert declaration is required. Plausible because the Site Area Emergency would be correct if placing the Reactor Mode Switch in SHUTDOWN did not result in APRM downscales.

C. Incorrect – An Alert declaration is required. Plausible because the failure to scram only lasts approximately 12-13 seconds and is no longer in progress. Also plausible because this would be correct if the typer data showed rod motion and APRM downscales before the manual scram signal.

D. Incorrect – An Alert declaration is required. Plausible because the failure to scram only lasts approximately 12-13 seconds and is no longer in progress (however EAL bases specifically still require declaring this EAL even in this exact situation, whereas other conditions that occur briefly and then clear would just require notification of a transitory event).

Technical Reference(s): EP-AA-1013 Add.3, EPIP-EPP-01-EAL

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.06
	Importance Rating	3.3

Loss of Shutdown Cooling

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

Proposed Question: #77

The plant is shutdown with the following:

- Shutdown Cooling loop 11 is in service.
- Reactor water level is 78" and stable.
- Reactor pressure is 0 psig and slowly lowering.
- Reactor coolant temperature is 175°F and slowly lowering.

Then, a leak in Shutdown Cooling loop 11 occurs. The following conditions are present 25 minutes later:

- Shutdown Cooling has automatically isolated on high area temperature.
- Reactor water level reached a low of 45" and is now 57" and slowly rising.
- Reactor pressure is 17 psig and slowly rising.
- Reactor coolant temperature is 255°F and slowly rising.

Which one of the following identifies the HIGHEST emergency classification that is required, if any, in accordance with EPIP-EPP-01, Classification of Emergency Conditions at Unit 1?

A. NO emergency classification is required

B. Unusual Event

C. Alert

D. Site Area Emergency

Proposed Answer: C

Explanation: With the plant shutdown and Reactor coolant temperature initially <212°F, the plant is in Cold Shutdown (Mode 3 per EAL chart). This makes the "C" EALs applicable and the "S" EALs not applicable. Reactor pressure has risen more than 10 psig due to an unplanned loss of decay heat removal capability, therefore Alert CA4.1 is met.

A. Incorrect – Alert CA4.1 is met. Plausible because no EALs are met or exceeded on the EAL matrix for Modes 1 and 2.
B. Incorrect – Alert CA4.1 is met. Plausible because multiple Unusual Event EALs are related to this event, including UE CU4.1.
D. Incorrect – Alert CA4.1 is met. Plausible because an Alert is met and multiple SAE EALs are related and are possible if the Reactor water level transient were greater.

Technical Reference(s): EPIP-EPP-01-EAL

Proposed references to be provided to applicants during examination: Hot and Cold EAL Matrices (with Table F-1 Row D blocked out)

Note: EAL Matrix Table F-1 Row D is blocked out to avoid assisting question #98.

Learning Objective: NS-EPL001-TO-01

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(1)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295023 2.2.37
Importance Rating	4.6

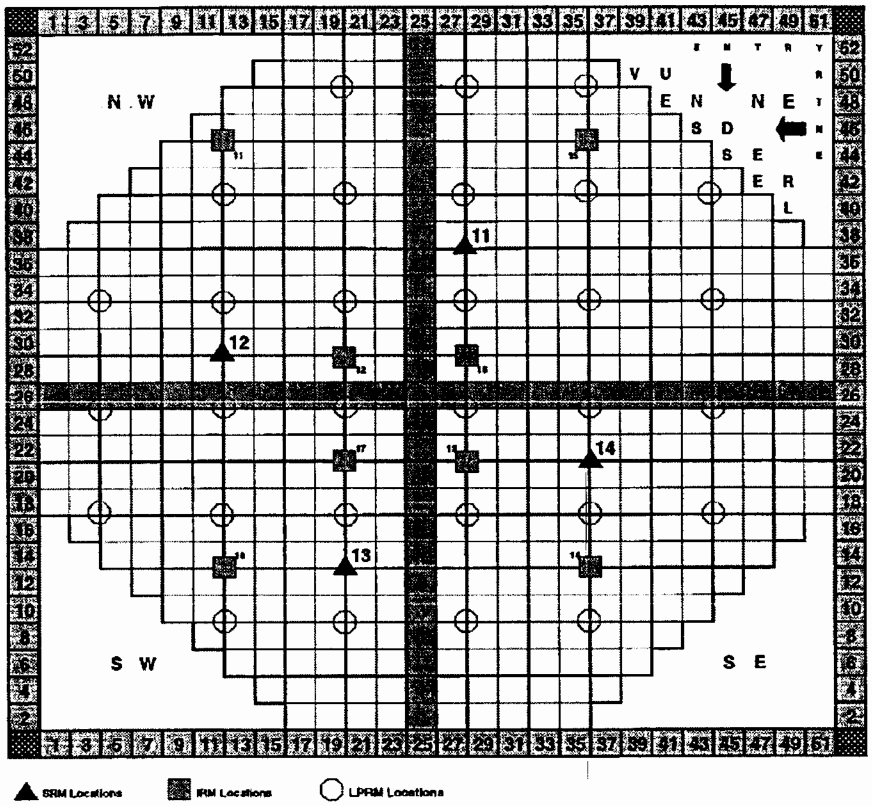
Refueling Accidents

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

Proposed Question: #78

The plant is shutdown for a refueling outage with the following:

- The Reactor is fully loaded with fuel.
- Core shuffle phase 1 is about to begin.
- The SRMs indicate as follows:
    - SRM 11 - 47 cps
    - SRM 12 - 50 cps
    - SRM 13 - 0 cps, with the DN SCL OR INOP light ON
    - SRM 14 - 2 cps



Which one of the following describes the allowable fuel movements, if any, in accordance with Technical Specifications?

Fuel movements…

A. are NOT allowed.

B. may be performed in any of the core quadrants.

C. may be performed in two of the core quadrants, only.

D. may be performed in three of the core quadrants, only.

Proposed Answer: C

Explanation: SRMs 13 and 14 are both inoperable due to count rate being below 3 cps. A count rate less than 3 cps is allowed only if it occurs during a spiral unload and count rate was greater than 3 cps at the start. With SRMs 13 and 14 inoperable, core alterations are NOT allowed in either of their respective core quadrants (SW and SE). Core alterations are allowed in the other two core quadrants, since each has an operable SRM in the quadrant and one adjacent quadrant.

Note: The question meets the K/A because SRMs are safety-related equipment and the question requires analyzing their behavior to determine their operability. Additionally, the SRMs are being tested in a situation where there function is to monitor for a Refueling Accident.

A. Incorrect – If SRMs 11 and 13 (or 12 and 14) were inoperable, no core alterations would be allowed because no quadrant would have both an operable SRM in and adjacent to the quadrant.
B. Incorrect – Only one SRM is fully downscale and TS 3.5.1 only requires three SRMs operable. However, SRM 14 is also inoperable per TS 3.5.3 and TS 3.5.3 requires an operable SRM in a core quadrant to allow core alterations.
D. Incorrect – SRM 14 is still indicating, however it is below the minimum count rate required for operability in this situation.

Technical Reference(s): Technical Specifications 3.5.1 and 3.5.3, N1-OP-34 P&L 14

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-234000-RBO-14

Question Source: Bank – 2015 Cert #82

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295026 EA2.01
	Importance Rating	4.2

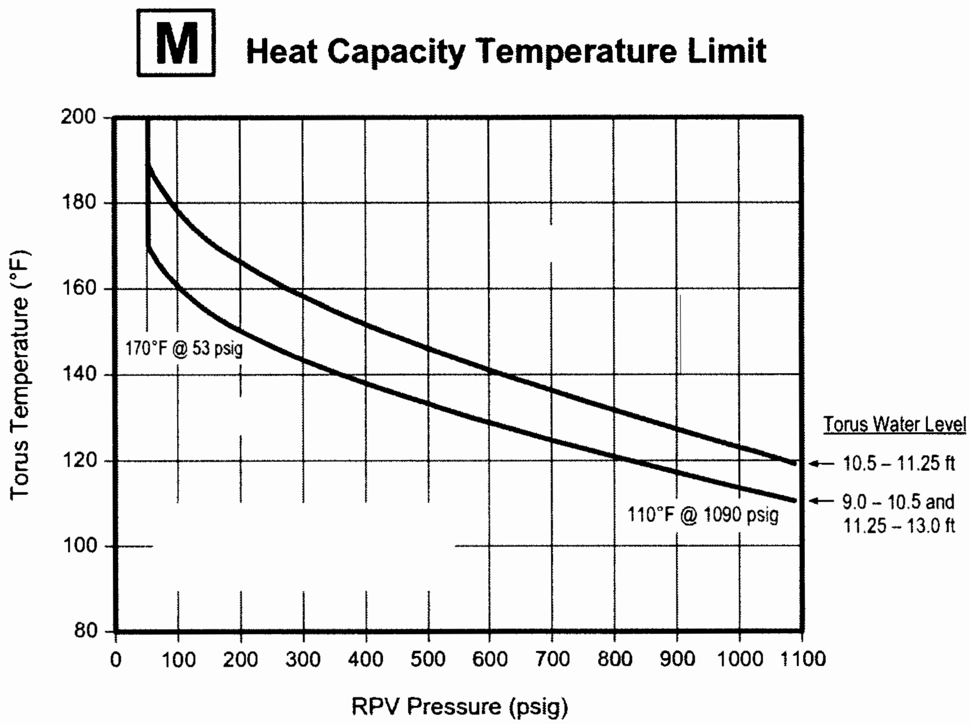
Suppression Pool High Water Temperature

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

Proposed Question: #79

An ATWS has occurred with the following conditions:

- Reactor power is 20%.
- Reactor water level is -50" and lowering.
- Turbine Bypass Valves have failed closed.
- Both Emergency Condensers are in service.
- ERVs are being manually opened to control Reactor pressure.
- Reactor pressure is 700 psig and stable.
- Torus water temperature is 130°F and rising.
- Torus water level is 11.1' and rising.
- Liquid Poison is being injected.
- Initial Liquid Poison tank level was 1450 gallons.
- Current Liquid Poison tank level is 1000 gallons.



Which one of the following indicates the NEXT action that is required, in accordance with the Emergency Operating Procedures?

A. Perform an RPV Blowdown due to HCTL violation

B. Raise Reactor water level to 53" to 95" to mix boron

C. Reduce Reactor pressure to stay below the applicable HCTL limit

D. Inject with Condensate/Feedwater to maintain Reactor water level -50" to -109"

Proposed Answer: C

Explanation: For a Torus water level of 11.1', the top curve on the HCTL graph is used. When Reactor pressure is 700 psig, the Torus temperature limit is ~137°F. The Reactor is at 20% power. Emergency Condensers take away ~6% of Reactor power. An excess of ~14% Reactor power is being sent to the Torus. The US should direct Reactor pressure to be lowered to stay below HCTL.

A. Incorrect – A blowdown is not required unless HCTL has been violated. This answer would be the correct answer if the lower curve is used on the HCTL graph. The lower curve is only used if Torus level is 9-10.5' or 11.25-13.0'.
B. Incorrect – Level is only raised in N1-EOP-3 when hot shutdown boron is injected. Hot shutdown boron is 600 gallons and only 450 gallons have been injected.
D. Incorrect – With Torus temp above 110°F, power above 6% and an ERV open, RPV water level must continue to be lowered. Plausible because this would be correct if Torus temperature were lower.

Technical Reference(s): N1-EOP-3, 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 – 2013 NRC #76

Question History: 2013 NRC #76

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295028 EA2.01
	Importance Rating	4.1

High Drywell Temperature

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

Proposed Question: #80

A steam leak in the Drywell has resulted in the following:

- The Reactor has been manually scrammed.
- All Containment Spray pumps have been placed in Pull-to-Lock.
- The following Containment parameters have occurred:

	08:01	08:02	08:03	08:04
Drywell Pressure (psig)	6.0	9.0	11.0	14.0
Torus Pressure (psig)	5.0	8.0	10.0	13.0
Drywell Temperature (°F)	225	250	276	302

Note: See the following page for the Containment Spray Initiation Limit curve.

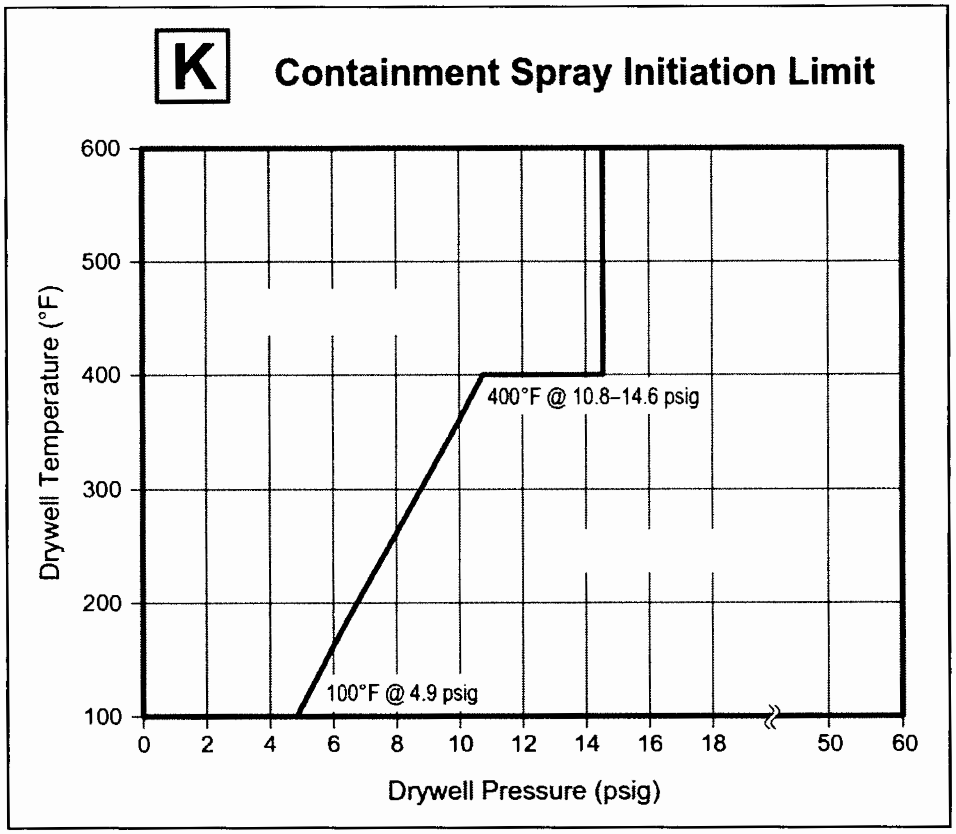
Which one of the following identifies the EARLIEST TIME at which Containment Spray can be initiated, in accordance with N1-EOP-4?

A. 08:01

B. 08:02

C. 08:03

D. 08:04


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

Proposed Answer: B

Explanation: Containment Spray can first be initiated at time 08:02, because this is the first time that the combination of Drywell pressure and temperature are on the OK TO SPRAY side of the Containment Spray Initiation Limit curve. Since Drywell temperature is above 150°F, the Drywell Temperature leg of N1-EOP-4 allows initiation of Containment Spray as soon as parameters are on the OK TO SPRAY side of the Containment Spray Initiation Limit curve.

A. Incorrect – Containment Spray cannot be initiated at this time because the combination of Drywell pressure and temperature are on the NO SPRAY side of the Containment Spray Initiation Limit curve. Plausible because Drywell temperature is above 150°F at this time, so Containment Sprays would be allowed if the operating point on the Containment Spray Initiation Limit curve were slightly to the right. Also plausible if the Containment Spray Initiation Limit curve were a softer warning, like the Containment Spray NPSH Limit curve, and not a hard requirement.
C. Incorrect – Containment Spray can first be initiated at the earlier time 08:02. Plausible because this would be correct if the operating point on the Containment Spray Initiation Limit curve were slightly farther to the left at time 08:02. Also plausible because Drywell temperature and Torus pressure are higher than at 08:02, and much closer to the next thresholds used in N1-EOP-4 (300°F, 13 psig)
D. Incorrect – Containment Spray can first be initiated at the earlier time 08:02. Plausible because 300°F is the Drywell temperature threshold used in N1-EOP-4 related to Containment Spray, and this is the first time Drywell temperature is above 300°F. Also plausible because this is the first time Torus pressure has reached the threshold of 13 psig.

Technical Reference(s): N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO #2

Question Source: Bank - 2009 NRC #77

Question History: 2009 NRC #77

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 2.4.41
	Importance Rating	4.6

High Offsite Radioactivity Release Rate

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #81

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

- Fuel movement was occurring in the Spent Fuel Pool in preparation for an upcoming outage.
- An irradiated fuel bundle was dropped and damaged in the Spent Fuel Pool.
- The following radiation monitors have indicated upscale for over an hour:
  - Refuel Floor Exhaust radiation monitors.
  - Refuel Bridge high range area radiation monitor.
  - Stack radiation monitors.
- Field survey teams have completed multiple surveys in the area.
- The surveys found the highest closed window dose rate of 150 mR/hr at the intersection of State Route 104 and County Route 29.
- The Offsite Dose Assessment team expects this release rate to continue for the next two (2) hours.

Which one of the following identifies the emergency classification that is required to be declared, in accordance with EP-CE-111, Emergency Classification and Protective Action Recommendations?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: C

Explanation: The given conditions meet or exceed multiple EALs, including the following:

- Unusual Event RU1.1 due to Stack radiation monitors above 300 cps.
- Unusual Event RU2.2 due to the Refuel Bridge high range radiation monitor indicating upscale.
- Alert RA1.1 due to Stack radiation monitors above 3.0E4 cps.
- Alert RA2.1 due to Refuel Bridge high range radiation monitor alarming due to damage to irradiated fuel.
- Site Area Emergency RS1.3 due to field survey indicating closed window dose rates >100mRem/hr beyond the Site Boundary.

No General Emergency EALs are met or exceeded, although RG1.3 is possible if radiation levels degrade further or are misinterpreted. Therefore, the highest emergency classification required is a Site Area Emergency.

A. Incorrect – A higher Site Area Emergency is required. Plausible because two Unusual Event EALs are exceeded.
B. Incorrect – A higher Site Area Emergency is required. Plausible because two Alert EALs are exceeded.
D. Incorrect – No General Emergency EALs are met or exceeded. Plausible because General Emergency RG1.3 is possible if radiation levels degrade further or are misinterpreted.

Technical Reference(s): EPIP-EPP-01-EAL

Proposed references to be provided to applicants during examination: Hot EAL Matrix ((with Table F-1 Row D blocked out)

Note: EAL Matrix Table F-1 Row D is blocked out to avoid assisting question #98.

Learning Objective: NS-EPL001-TO-01

Question Source: Bank – JAF 16-1 NRC #84

Question History: JAF 16-1 NRC #84

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	600000 2.1.7
	Importance Rating	4.7

Plant Fire On Site

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

Proposed Question: #82

The plant is shutdown following a scram from 100% power with the following sequence of events:

Time (hh:mm)	Conditions
00:00	• The following Main Fire Panel annunciators alarm: o 2-1-1-1, TURB. BLDG. 261 LOCAL PNL NO. 1 FIRE o 2-2-1-2, DIESEL FIRE PUMP #1 RUNNING o 2-2-2-2, ELECTRIC FIRE PUMP #1 STARTED • Fire Zone WD-8131, T1 XFMR Water Deluge System, is in alarm and indicates discharge is in progress.
00:03	• The Fire Brigade leader reports that there are visible flames coming from Transformer 1.
00:20	• The Fire Brigade leader reports that the fire is NOT yet under control.

Which one of the following describes the earliest time at which the fire can be considered "confirmed" and the required control of the Reactor, in accordance with OP-NM-201-005, Firefighting, and N1-SOP-21.1, Fire In Plant?

	Earliest Time Fire Can Be Considered "Confirmed"	Required Control of the Reactor
A.	00:00	Commence a rapid depressurization
B.	00:00	Maintain in a stable hot shutdown condition.
C.	00:03	Commence a rapid depressurization
D.	00:03	Maintain in a stable hot shutdown condition.

Proposed Answer: B

Explanation: Fire Zone WD-8131 is a water deluge system for Transformer 1 with automatic initiation capability. The given indications show that this zone has both detected a fire and started deluge flow. OP-NM-201-005 defines a confirmed fire as either (1) fire alarm/annunciator and suppression system activation accompanied by actual flow or discharge or (2) reported actual fire with flames showing. The first set of conditions for a confirmed fire is met at time 00:00. With the fire not under control after 15 minutes, override actions in N1-SOP-21.1 are required. This requires maintaining the Reactor in a stable hot shutdown condition.

Note: The question meets SRO level guidance by requiring the candidate to assess multiple abnormal plant conditions and then select a procedure section (maintain in hot shutdown vs. commence rapid depressurization) to mitigate the plant conditions. The question cannot be answered solely based on system knowledge, immediate operator actions, entry conditions for SOPs/EOPs, or overall strategy of a procedure (need to specifically assess plant conditions against specific criteria in a specific step within N1-SOP-21.2).

A. Incorrect – With the fire not under control after 15 minutes, override actions in N1-SOP-21.1 are required. This requires maintaining the Reactor in a stable hot shutdown condition. Plausible because other major abnormal procedures require rapid depressurization (eg. Station Blackout).

C. Incorrect – The fire is first considered "confirmed" at time 00:00. Plausible because some fires would not be considered confirmed until report from the Fire Brigade. With the fire not under control after 15 minutes, override actions in N1-SOP-21.1 are required. This requires maintaining the Reactor in a stable hot shutdown condition. Plausible because other major abnormal procedures require rapid depressurization (eg. Station Blackout).

D. Incorrect – The fire is first considered "confirmed" at time 00:00. Plausible because some fires would not be considered confirmed until report from the Fire Brigade.

Technical Reference(s): OP-NM-201-005, N1-SOP-21.1

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPIP28-CE-01

Question Source: Modified Bank – 2013 NRC #82

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295007 2.2.42
	Importance Rating	4.6

High Reactor Pressure

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

Proposed Question: #83

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

A document review reveals that the actual Emergency Condenser automatic initiation setpoints are set as follows:

Pressure Transmitter	Setpoint
36-08A	1068 psig
36-08B	1084 psig
36-08C	1077 psig
36-08D	1072 psig

Which one of the following describes the most restrictive requirement of this data on Technical Specifications (TS)?

A. Restore operability of the affected system within a maximum of 7 days, and perform the additional required surveillances.

B. Place a channel in one trip system in the tripped condition within a maximum of 24 hours.

C. Place a channel in one trip system in the tripped condition within a maximum of 6 hours.

D. Commence a normal shutdown within one hour, and be in cold shutdown within 10 hours.

Proposed Answer: B

Explanation: TS Table 3.6.2c requires two operable channels in each trip system, each with a setpoint of ≤1080 psig. The given data shows that one channel is set >1080 psig, while the other three channels are <1080 psig. There are no extra channels provided, therefore one of the required channels in one trip system is inoperable. Table 3.6.2c Note e applies. This requires placing the inoperable channel in the tripped condition within a maximum of 24 hours.

Note: The question meets the K/A by testing knowledge of high Reactor pressure setpoints that require entry into a Technical Specification and then testing application of the associated Technical Specification.

A. Incorrect – TS Table 3.6.2c Note e requires placing the inoperable channel in the tripped condition within a maximum of 24 hours. Plausible because this is the requirement from TS 3.1.3 for a single inoperable Emergency Condenser.
C. Incorrect – TS Table 3.6.2c Note e requires placing the inoperable channel in the tripped condition within a maximum of 24 hours. Plausible because TS Table 3.6.2c Note d also applies, which discusses a 6 hour allowance for making a channel inoperable for surveillance testing.
D. Incorrect – TS Table 3.6.2c Note e requires placing the inoperable channel in the tripped condition within a maximum of 24 hours. Plausible because this is the requirement from TS 3.1.3 for two inoperable Emergency Condensers, and would apply if the 24 hour clock expired without placing a channel in trip.

Technical Reference(s): Technical Specification 3.6.2

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.3 (pages 50-51) and 3.6.2 (with allowable values removed, Table 3.6.2b removed, Table 3.6.2.a Note (c) removed) (pages 194-204, 214-247a)

Note: Technical Specification Table 3.6.2a Note (c) is removed from the provided reference to prevent assisting in question 88. Technical Specification Table 3.6.2b is removed from the provided reference to prevent assisting in question 89.

Learning Objective: N1-207000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295033 EA2.03
	Importance Rating	4.2

High Secondary Containment Area Radiation Levels

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

Proposed Question: #84

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

- A malfunction during backwash of a Spent Fuel Pool Cooling filter is causing elevated radiation levels in the Reactor Building.
- Reactor Building Ventilation exhaust radiation monitors are indicating 25 mR/hr and slowly rising.
- The Reactor Building Continuous Air Monitor (CAM) is in alarm.
- The Stack radiation monitors are indicating 350 cps and slowly rising.
- Multiple Reactor Building Area Radiation Monitors (ARMs) are in alarm and indicate upscale.
- Radiation Protection reports the following general area survey results:
    - Reactor Building 261 east – 1.5 R/hr
    - Reactor Building 261 west – 0.8 R/hr
    - Reactor Building 281 east – 12.0 R/hr
    - Reactor Building 281 west – 9.5 R/hr

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

A. The Reactor may continue to operate at the current power level.

B. A Reactor shutdown is required, but a Reactor scram is NOT required.

C. A Reactor scram is required, but an RPV Blowdown is NOT required.

D. A Reactor scram and RPV Blowdown are required.

Proposed Answer: B

Explanation: N1-EOP-5 entry is required based on high Reactor Building ventilation exhaust radiation and high area radiation levels. The cause of the high radiation levels is a Spent Fuel Pool Cooling filter, therefore the discharge is from a non-primary system. This determines the flow path in N1-EOP-5. Two general areas (RB 281 east and west) are above Maximum Safe Values in the same parameter (radiation levels above 8 R/hr). Therefore, N1-EOP-5 step SC-7 requires shutting down the Reactor per N1-OP-43C. This is the normal shutdown procedure.

Note: The K/A requires interpreting the cause of high radiation conditions as it applies to a high radiation in the Secondary Containment ventilation. The question satisfies this K/A by giving a malfunction that causes high Reactor Building ventilation exhaust radiation, along with high area radiation levels, and requiring the candidate to interpret the cause/source of the high radiation conditions (SFPC filter malfunction) to determine the required course of action (procedural path in N1-EOP-5 based on non-primary system discharge).

A. Incorrect – A Reactor shutdown is required because two general areas are above the Maximum Safe Value for radiation (8 R/hr). Plausible because the source of the high radiation levels is not directly connected to the Reactor. Also plausible because if only one general area were above the Maximum Safe Value, continued Reactor operation would be allowed.
C. Incorrect – A Reactor shutdown is required, but a scram is not required because the source of the high radiation levels is a non-primary system. Plausible because this would be the correct answer if the discharge was from a primary system and only one general area radiation level was above the Maximum Safe Value.
D. Incorrect – A Reactor shutdown is required, but a scram and RPV Blowdown is not required because the source of the high radiation levels is a non-primary system. Plausible because this would be the correct answer if the discharge was from a primary system.

Technical Reference(s): N1-EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: Bank – 2017 Cert #83

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295036 EA2.03
	Importance Rating	3.8

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: Cause of the high water level

Proposed Question: #85

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

- Core Spray pump 121 is injecting to the Reactor.
- Reactor water level is -85" and stable.
- NO other injection sources are available.
- Containment Spray pump 122 is spraying the Drywell.
- Drywell pressure is 17 psig and rising slowly.
- Torus water level is 11' and rising slowly.
- NO other Containment Spray pumps are available.

Then, the following occur:

- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, is in alarm
- Computer point F190, SE RB CORNER RM WTR LVL HIGH, is in alarm
- An operator reports water level in the Southeast Corner Room is 5' and slowly rising

Which one of the following identifies:

- the pump that may be the source of leakage

  and

- the required control of this pump if it is the source of leakage,

in accordance with N1-EOP-5, Secondary Containment Control?

This rising water level may be caused by a leak from ___(1)___.

If a leak from this pump is causing the rising water level, it ___(2)___ required to be secured and isolated.

	(1)	(2)
A.	Core Spray pump 121	is
B.	Core Spray pump 121	is NOT
C.	Containment Spray pump 122	is
D.	Containment Spray pump 122	is NOT

Proposed Answer: B

Explanation: The southeast corner room contains Core Spray pump 121, therefore leakage from that pump could be causing the rising water level. With Reactor water level <-84", adequate core cooling is being challenged. Since no other injection sources are available, Core Spray pump 121 injection is required by N1-EOP-2. Therefore, N1-EOP-5 does not require securing and isolating the pump.

A. Incorrect – N1-EOP-5 does not require securing and isolating the pump. Plausible because this would be correct if adequate core cooling was assured with other injection sources.
C. Incorrect – Containment Spray pump 122 is not in the southeast corner room. Plausible because this would be correct for a leak in the northeast corner room. N1-EOP-5 does not require securing and isolating the pump. Plausible because this would be correct if adequate core cooling was assured with other injection sources.
D. Incorrect – Containment Spray pump 122 is not in the southeast corner room. Plausible because this would be correct for a leak in the northeast corner room.

Technical Reference(s): ARP H2-2-1, N1-OP-2, N1-EOP-2, N1-EOP-5

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: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	205000 A2.02
Importance Rating	2.7

Shutdown Cooling

Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low shutdown cooling suction pressure: Plant-Specific

Proposed Question: #86

The plant is shutdown with the following:

- Reactor pressure is 100 psig and stable.
- Reactor coolant temperature is 338°F and stable.
- Reactor water level is being controlled in a band of 70" to 80".
- Shutdown Cooling (SDC) loops 11 and 12 are operating.
- SDC loop 13 is in standby.

Then, the following occur:

- All available Reactor Recirc pumps trip and CANNOT be restarted.
- SDC loop 13 is being placed in service to raise the cooldown rate.

To mitigate the loss of Reactor Recirc pumps ***and*** place SDC loop 13 in service, which one of the following describes (1) the required control of Reactor water level and (2) the required control of the Recirculation loops, in accordance with N1-OP-43C and N1-OP-4?

	(1) Reactor water level must be raised to a minimum of...	(2) Required Control of Recirculation Loops
A.	0' on Flange level instruments.	The suction and discharge valves of at least two Recirculation loops must remain full open.
B.	0' on Flange level instruments.	All Recirculation pumps suction or discharge and discharge bypass valves must be closed.
C.	10.5' on Wide Range level instruments.	The suction and discharge valves of at least two Recirculation loops must remain full open.
D.	10.5' on Wide Range level instruments.	All Recirculation pumps suction or discharge and discharge bypass valves must be closed.

Proposed Answer: B

Explanation: The given conditions require implementation of N1-OP-4 sections F.2.0 and H.1.0. F.2.0 and H.1.0 each have their own requirements for control of Reactor water level. Under the current combination of Reactor coolant temperature and Reactor water level, starting a 3rd Shutdown Cooling pump could lead to inadequate pump NPSH, and is therefore not allowed. However, N1-OP-4 does allow starting the 3rd Shutdown Cooling pump at this Reactor coolant temperature if Reactor water level is raised high enough. The minimum Reactor water level to allow starting the 3rd SDC pump at this Reactor coolant temperature is 0' Flange level. With no Recirculation pumps in service, Reactor vessel thermal stratification and maintaining adequate Reactor water level indication are concerns addressed by both N1-OP-43C and N1-OP-4. Since no Recirculation pumps can be placed in service, thermal stratification will be prevented and adequate Reactor water level indication will be maintained by placing Shutdown Cooling in service, raising Reactor water level at least to the level of the Main Steam Lines (which is above the top end of the current control band of 95", at approximately 10.5' Wide Range), and closing either all the Recirculation loop suction valves or all the Recirculation loop discharge and discharge bypass valves.

A. Incorrect – All Recirculation loops must be isolated in this situation to prevent Shutdown Cooling flow from short-cycling the core. Plausible because N1-OP-43C normally requires maintaining the suction and discharge valves of at least two Recirculation loops full open.
C. Incorrect – 10.5' on Wide Range level instruments is not high enough to start the 3rd pump. Plausible because this is the level of the Main Steam Lines, which is a level used in N1-OP-4 related to keeping Reactor level instrumentation valid and preventing thermal stratification. All Recirculation loops must be isolated in this situation to prevent Shutdown Cooling flow from short-cycling the core. Plausible because N1-OP-43C normally requires maintaining the suction and discharge valves of at least two Recirculation loops full open.
D. Incorrect – 10.5' on Wide Range level instruments is not high enough to start the 3rd pump. Plausible because this is the level of the Main Steam Lines, which is a level used in N1-OP-4 related to keeping Reactor level instrumentation valid and preventing thermal stratification.

Technical Reference(s): N1-OP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-10

Question Source: Modified Bank – 2017 NRC #77

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	209001 A2.01
Importance Rating	3.4

Low Pressure Core Spray

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

Proposed Question: #87

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

Date	Time (hh:mm)	Condition(s)
12/1/18	0900	• Core Spray loop 111 is declared inoperable for performance of N1-ST-Q1A, CS 111 Pump, Valve, and SDC Water Seal Check Valve Operability Test.
12/1/18	1000	• Core Spray pump 111 trips immediately after being started. • The initial investigation reveals that the trip is caused by a breaker issue. • Further investigation is initiated to determine if there is a common-cause issue with the other Core Spray pump breakers.
12/1/18	1700	• Investigation reveals that there is a breaker issue that would also prevent Core Spray pump 112 from operating. • The same issue is NOT present in the breakers for Core Spray pumps 121 and 122.

Which one of the following describes the ***latest*** time that a plant shutdown can be initiated to comply with Technical Specifications?

A. 12/1/18 at 1100

B. 12/1/18 at 1800

C. 12/8/18 at 1000

D. 12/8/18 at 1800

Proposed Answer: B

Explanation: The most restrictive shutdown requirement is to initiate shutdown by 12/1/18 at 1800. At time 1700, Core Spray pump 112 must be declared inoperable due to the common-cause breaker issue. This makes both Core Spray pumps in loop 11 inoperable. This requires entering Technical Specification 3.1.4.d, which requires initiating a normal orderly shutdown within one hour (by 12/1/18 at 1800).

A. Incorrect – The most restrictive shutdown requirement is to initiate shutdown by 12/1/18 at 1800. Plausible because this would be the time if TS 3.1.4.d became applicable at time 1000, when the Core Spray pump first tripped and the common-cause investigation was initiated.

C. Incorrect – The most restrictive shutdown requirement is to initiate shutdown by 12/1/18 at 1800. Plausible because this is the original shutdown requirement based on Core Spray loop 111 being declared inoperable at time 0900.

D. Incorrect – The most restrictive shutdown requirement is to initiate shutdown by 12/1/18 at 1800. Plausible because this is the time if TS 3.1.4.c is applied at time 1700.

Technical Reference(s): C-18007-C, Technical Specification 3.1.4

Proposed references to be provided to applicants during examination: Technical Specification 3.1.4 (pages 54-56)

Learning Objective: N1-209001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	212000 2.4.46
	Importance Rating	4.2

Reactor Protection

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

Proposed Question: #88

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- Reactor power is mid-scale on IRM range 8.
- Reactor pressure is 750 psig and rising slowly.
- Annunciator F3-3-6, MSIV CLOSURE SCRAM BYPASS, is in alarm.

Which one of the following describes the current status of Technical Specifications with regard to the MSIV closure scram?

With regard to the MSIV closure scram, Technical Specifications...

A. require scramming the Reactor.

B. require inserting control rods, but a Reactor scram is NOT required.

C. require halting the startup, but do NOT require inserting control rods.

D. are currently satisfied. NO LCO entry is required.

Proposed Answer: B

Explanation: F3-3-6 should not be in alarm because Reactor pressure is above 600 psig. This indicates that the Reactor scram on MSIV position is not active when required by Technical Specifications. This affects both trip systems and makes trip capability not maintained. Technical Specification Table 3.6.2.a Note (o) applies. The first required action in Note (o) cannot be performed because trip capability is not maintained. Therefore Technical Specification 3.6.2a(1) applies. This requires inserting control rods, but does not require doing so with a manual Reactor scram. Control rod insertion using RMCS in a controlled manner is allowed. Additionally, requirements 2 and 3 in Note (o) are modified by an asterisk (*) that does not require putting both trip systems in trip, since it would cause the protective function (scram) to occur.

A. Incorrect – A Reactor scram is not required. Plausible because control rod insertion is required, a required scram is inoperable, and if TS Table 3.6.2.a Note (o) parts 2 and 3 were applied without regard to the asterisk (*), this would result in a manual scram being inserted.
C. Incorrect – Technical Specification 3.6.2 does require control rods to be inserted. Plausible because other conditions in Technical Specification 3.6.2 would not require control rod insertion, but would rather require placing a trip system in trip. Also plausible that halting the startup would be sufficient due to Reactor pressure still being relatively low (<850 psig)
D. Incorrect – F3-3-6 should not be in alarm. Plausible because this would be correct if Reactor pressure were lower. Also plausible because similar scrams are still bypassed at this point in the startup. If F3-3-6 were correctly in alarm, Technical Specifications would be fully satisfied.

Technical Reference(s): Technical Specification 3.6.2

Proposed references to be provided to applicants during examination: Technical Specification 3.6.2 (with allowable values removed, Table 3.6.2b removed, Table 3.6.2.a Note (c) removed)

Note: Technical Specification Table 3.6.2a Note (c) is removed from the provided reference to prevent assisting in this question. Technical Specification Table 3.6.2b is removed from the provided reference to prevent assisting in question 89.

Learning Objective: N1-212000-RBO-14

Question Source: Modified Bank - 2017 Cert #55

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	223002 A2.04
	Importance Rating	3.2

Primary Containment Isolation/Nuclear Steam Supply Shutoff

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Process radiation monitoring system failures

Proposed Question: #89

A plant shutdown is in progress with the following:

- Reactor power is 5%.
- The Reactor Mode Switch is in STARTUP.
- Vent and purge of the Drywell is in progress.

Then, OGESMS Stack radiation monitor RN10A fails downscale.

Which one of the following describes the resulting status of the Drywell vent and purge and the Offsite Dose Calculation Manual (ODCM) requirements based on this failure?

	Status of Drywell Vent and Purge	ODCM Requirements – The vent and purge...
A.	Remains in service	may continue.
B.	Remains in service	must be secured.
C.	Isolated	may be placed back in service after bypassing the isolation.
D.	Isolated	must remain isolated until the radiation monitor is fixed.

Proposed Answer: A

Explanation: There are two OGESMS Stack radiation monitors (RN-10A and RN-10B). Downscale failure of one of these monitors provides a trip signal to the Containment isolation for the "Big 8" Containment vent and purge valves. However, the logic for this isolation is two-out-of-two, so the vent and purge remains in service. ODCM Table 3.6.14-2 requires both monitors. Note (a) applies due to this failure. This note allows the vent and purge to continue since the channel is already in the tripped condition.

B. Incorrect – The vent and purge may continue. Plausible because one of the required channels is non-functional and there are only two channels.
C. Incorrect – The vent and purge remains in service. Plausible because the downscale failure does provide a trip signal and there are only two radiation monitors that input to the logic.
D. Incorrect – The vent and purge remains in service. Plausible because the downscale failure does provide a trip signal and there are only two radiation monitors that input to the logic.

Technical Reference(s): ARP F1-4-5, N1-OP-50B, ODCM Table 3.6.14-2

Proposed references to be provided to applicants during examination: ODCM 3.6.14

Learning Objective: N1-223002-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(4)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	259002 2.4.9
	Importance Rating	4.2

Reactor Water Level Control

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

Proposed Question: #90

A plant startup is in progress when a loss of coolant accident results in the following:

- Reactor water level is -109" and slowly lowering.
- Reactor pressure is 210 psig and slowly lowering.
- Both Emergency Condenser loops are in service.
- Core Spray pumps are injecting.
- Core Spray loop 11 flow is 150 x 10^4 lbm/hr.
- Core Spray loop 12 flow is 140 x 10^4 lbm/hr.
- NO other injection systems are available.
- NO ERVs have been opened.
- The Alternate Level Control Leg of N1-EOP-2, RPV Control, is being executed.

Which one of the following describes the required action, in accordance with N1-EOP-2?

A. Exit all EOPs and enter all SAPs.

B. Enter N1-EOP-8, RPV Blowdown.

C. Rapidly depressurize the Reactor using Turbine Bypass Valves.

D. Secure Emergency Condensers and attempt to stabilize Reactor pressure.

Proposed Answer: B

Explanation: Reactor water level is less than -84" and RPV Blowdown has not yet been performed, as evidenced by no ERVs being opened and the given place in N1-EOP-2. Reactor pressure is low enough for some Core Spray injection based on the low starting pressure of the Reactor startup and the loss of coolant accident, however more Core Spray flow is likely possible after Reactor pressure is lowered. Based on the current position in N1-EOP-2 and plant conditions, the required action is to enter N1-EOP-8, RPV Blowdown.

A. Incorrect – There is currently no requirement to exit all EOPs and enter all SAPs. Plausible because if an RPV Blowdown had already been performed, the combination of Reactor water level and Core Spray flows would require this action.
C. Incorrect – Rapidly depressurizing the Reactor using Turbine Bypass Valves is not allowed by OP-NM-101-111-1001 in this situation because it further reduces Reactor coolant inventory. Plausible because the pressure leg of N1-EOP-2 contains this allowance and OP-NM-101-111-1001 suggests its use under other conditions. Also plausible because this action would likely result in greater Core Spray flow.
D. Incorrect – Emergency Condensers are required to remain in service and ERVs are required to be opened to further lower Reactor pressure. Plausible because this is the Steam Cooling strategy that was recently revised out of the NMP1 EOPs which was used in similar situations with limited Reactor injection sources and Reactor water level <-109".

Technical Reference(s): N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	201002 A2.01
	Importance Rating	2.8

Reactor Manual Control

Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod movement sequence timer malfunctions

Proposed Question: #91

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

- A control rod sequence exchange is in progress.
- A control rod is required to be notch withdrawn from position 06 to position 08.
- A Reactor Operator gives the control rod a notch withdraw signal.
- The control rod settles at position 12.

Which one of the following describes the required action(s), in accordance with N1-OP-5, Control Rod Drive System?

A. Insert the control rod to position 08. Recirculation flow does NOT need to be lowered.

B. Lower Reactor power using Recirculation flow. Then, insert the control rod to position 08.

C. Leave the control rod at position 12 until Reactor Engineering provides direction. Recirculation flow does NOT need to be lowered.

D. Lower Reactor power using Recirculation flow. Leave the control rod at position 12 until Reactor Engineering provides direction.

Proposed Answer: A

Explanation: This malfunction requires performance of N1-OP-5 Section H.9.0, Control Rod(s) Mispositioned or Double Notched. For a control rod withdrawn beyond its intended position, this procedure requires moving the control rod to its intended position (08 in this case). No Reactor power reduction using Recirculation flow is required.

B. Incorrect – No Reactor power reduction using Recirculation flow is required. Plausible because in other situations, this would be required (control rod inserted more than 3 notches past intended position, control rod inserted past intended position and tips crossed). Also plausible because N1-SOP-5.2, Control Rod Drift, requires a Reactor power reduction to 85% using Recirculation flow in some situations.
C. Incorrect – The control rod is moved to position 08 prior to consulting with Reactor Engineering in this case. Plausible because in other situations, the control rod would be left as is until Reactor Engineering provides direction.
D. Incorrect – No Reactor power reduction using Recirculation flow is required. Plausible because in other situations, this would be required (control rod inserted more than 3 notches past intended position, control rod inserted past intended position and tips crossed). Also plausible because N1-SOP-5.2, Control Rod Drift, requires a Reactor power reduction to 85% using Recirculation flow in some situations. The control rod is moved to position 08 prior to consulting with Reactor Engineering in this case. Plausible because in other situations, the control rod would be left as is until Reactor Engineering provides direction.

Technical Reference(s): N1-OP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	201003 2.2.38
	Importance Rating	4.5

Control Rod and Drive Mechanism

Knowledge of conditions and limitations in the facility license.

Proposed Question: #92

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

- Control rod 06-39 is at position 48.
- Then, control rod 06-39 loses all position indication at all positions.
- Control rod 06-39 position CANNOT be determined by other means.

Which one of the following describes the required action, in accordance with N1-OP-5, Control Rod Drive System, and whether Technical Specification LCO entry is required?

A. Place administrative controls to prevent movement of control rod 06-39 with RMCS except in an emergency. Technical Specification LCO entry is required.

B. Place administrative controls to prevent movement of control rod 06-39 with RMCS except in an emergency. Technical Specification LCO entry is NOT required.

C. Fully insert control rod 06-39, disarm the control rod, and isolate the HCU. Technical Specification LCO entry is required.

D. Fully insert control rod 06-39, disarm the control rod, and isolate the HCU. Technical Specification LCO entry is NOT required.

Proposed Answer: C

Explanation: N1-OP-5 section H.10 provides the direction for response to a loss of rod position indication. Since all control rod position indication is lost for this control rod and its position cannot be determined by other means, subsection H.10.4 requires fully inserting the control rod, disarming the control rod, and isolating the HCU. With no rod position indication, the full-in reed switch will not provide the required input to the Remote Shutdown Panel "all rods in" light. This makes the "all rods in" light inoperable, therefore Technical Specification 3.6.13 LCO entry is required.

A. Incorrect – N1-OP-5 section H.10.4 requires fully inserting the control rod, not administratively preventing movement. Plausible that since position cannot be determined during movement and it is already fully withdrawn, the control rod could be left in the original position without much risk.
B. Incorrect – N1-OP-5 section H.10.4 requires fully inserting the control rod, not administratively preventing movement. Plausible that since position cannot be determined during movement and it is already fully withdrawn, the control rod could be left in the original position without much risk. Technical Specification 3.6.13 LCO entry is required. Plausible because LCO entry in the more obvious Technical Specification 3.1.1, Control Rod System, is NOT required.
D. Incorrect – Technical Specification 3.6.13 LCO entry is required. Plausible because LCO entry in the more obvious Technical Specification 3.1.1, Control Rod System, is NOT required.

Technical Reference(s): N1-OP-5, Technical Specifications 3.1.1 and 3.6.13

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.1 and 3.6.13

Learning Objective: N1-201002-RBO-14

Question Source: Bank – 2017 Cert #93

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	202001 2.2.36
	Importance Rating	4.2

Recirculation

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

Proposed Question: #93

The plant is operating at power with the following:

- RRMG 11 is taken out of service for maintenance.
- An RO applies a CLOSE signal to the RRP 11 discharge valve.
- The breaker for the valve motor operator trips after 15 seconds and CANNOT be reset.
- The RO successfully closes the RRP 11 suction valve.
- The RRP 11 suction valve motor breaker is racked out.
- Reactor power stabilizes at 85% after this evolution.

Which one of the following describes the ability to restore Reactor power, in accordance with Technical Specifications?

Reactor power...

A. must remain at or below 85%.

B. may be raised to a maximum of 90.5%.

C. may be raised to a maximum of 97.5%.

D. may be returned to 100%.

Proposed Answer: B

Explanation: RRP 11 discharge valve is stuck mid-position because it's stroke time is longer than 15 seconds. Technical Specification 3.1.7.e applies and provides Reactor power restrictions in this situation. With the suction valve closed, Reactor power is limited to 90.5% power until the suction, discharge, and discharge bypass valves are fully closed (with motor breakers locked open) and RRMG breaker removed. Since the RRP 11 discharge valve cannot be closed, these conditions cannot be met and Reactor power must be limited to 90.5%.

A. Incorrect – Reactor power may be raised. Plausible that no reactivity addition would be allowed because the given failure prevents establishing the required protections against a cold water reactivity addition required by Technical Specifications.
C. Incorrect – Reactor power may be raised to 90.5%. Plausible because there are various reactivity related precautions involving 2.5% power, and this is 2.5% below rated.
D. Incorrect – Reactor power is limited to 90.5%. Plausible because Reactor power could be raised to 100% if the discharge valve were closed only. Also plausible because closing the suction valve and racking out its breaker is part of the logic that would allow restoration of power to 100% (per TS 3.1.7.e.1) and this action alone does prevent flow through the given Recirc loop.

Technical Reference(s): Technical Specification 3.1.7

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.31
	Importance Rating	4.3

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

Proposed Question: #94

A plant startup is in progress with the following:

- Reactor power is 30%
- Feedwater pumps and Feedwater Blocking Valves are aligned as shown on the following page.

Which one of the following describes the most restrictive requirement, if any, based on this Feedwater system alignment, in accordance with Technical Specification (TS) 3.1.8, High Pressure Injection System?

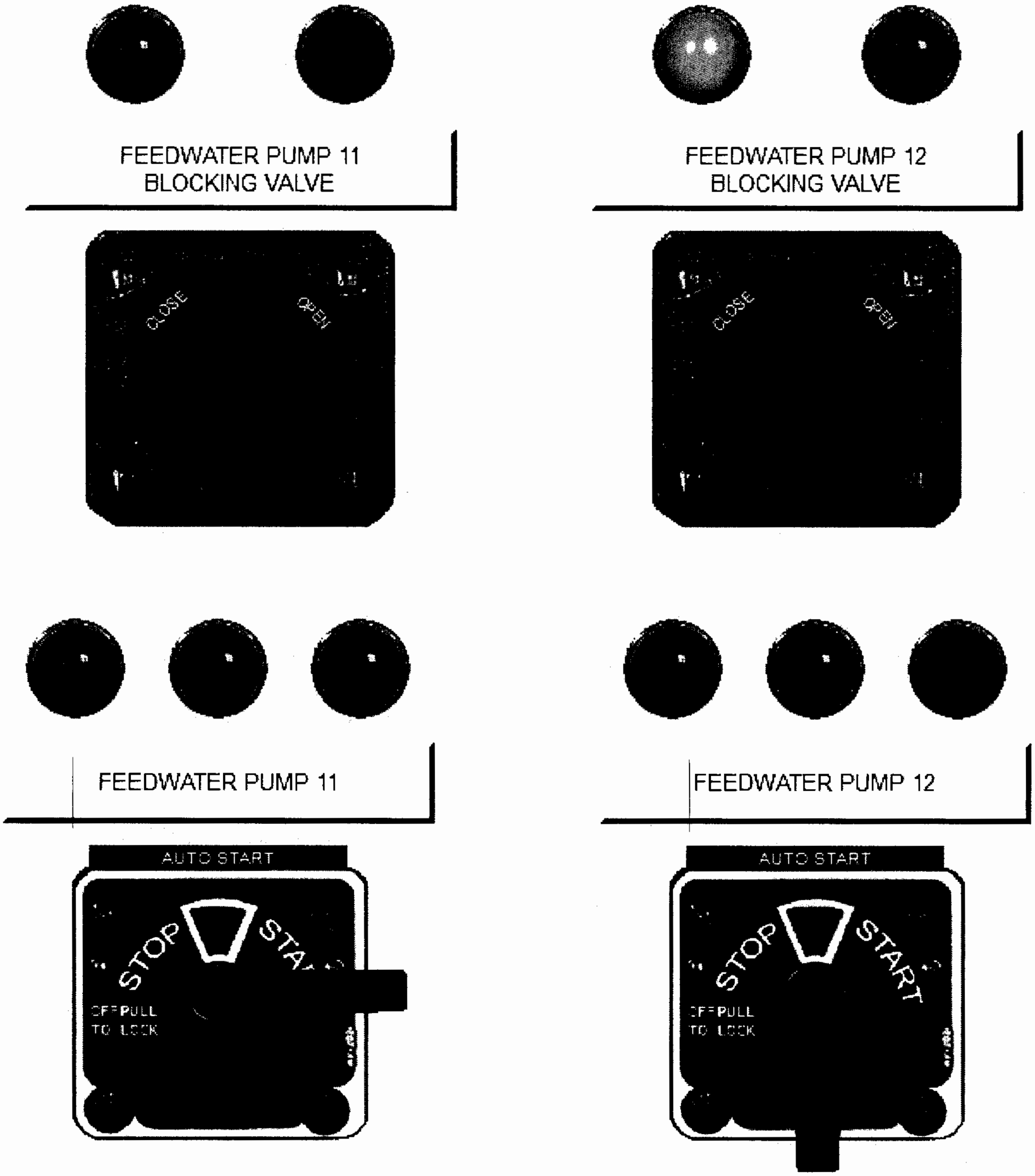
TS 3.1.8...

A. is fully satisfied and an LCO entry is NOT required.

B. requires a 15 day LCO based on the Feedwater pump lineup.

C. requires a 15 day LCO based on Feedwater Blocking Valve position.

D. requires a normal orderly shutdown be initiated within one hour.


FEEDWATER PUMP 11
BLOCKING VALVE
CLOSE
OPEN
FEEDWATER PUMP 12
BLOCKING VALVE
CLOSE
OPEN
FEEDWATER PUMP 11
AUTO START
STOP
OFF PULL
TO LOCK
FEEDWATER PUMP 12
AUTO START
STOP
START
OFF PULL
TO LOCK

Proposed Answer: D

Explanation: The given picture shows that HPCI train 11 is inoperable due to Feedwater pump 11 control switch being in PTL. Additionally, HPCI train 12 is inoperable due to Feedwater Blocking Valve 12 being closed. Therefore, TS 3.1.8.c requires initiating a normal orderly shutdown within one hour.

<u>Note:</u> The question satisfies SRO level guidelines, despite the answer being a less than one hour statement, because it cannot be answered ***solely*** by knowing a ≤1 hour action. Additional SRO level knowledge is required to determine operability of the HPCI system in a special circumstance that is covered by Technical Specification bases and a special note "below the line" in Technical Specifications.

A. Incorrect – TS 3.1.8.c requires initiating a normal orderly shutdown within one hour. Plausible because this would be correct if power were <25% and Feedwater pump 11 control switch was in Normal After Stop.
B. Incorrect – TS 3.1.8.c requires initiating a normal orderly shutdown within one hour. Plausible because this would be correct if power were <25% or if Feedwater Blocking Valve 12 were open.
C. Incorrect – TS 3.1.8.c requires initiating a normal orderly shutdown within one hour. Plausible because this would be correct if Feedwater pump 11 control switch was in Normal After Stop.

Technical Reference(s): N1-OP-43A, Technical Specification 3.1.8

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.40
	Importance Rating	3.9

Knowledge of refueling administrative requirements.

Proposed Question: #95

An outage is in progress with the following:

- Operators determine that SECONDARY CONTAINMENT is inoperable.
- The following evolutions are currently in progress:

Evolution (1) - Movement of new fuel bundles from the storage vault into the Spent Fuel Pool.

Evolution (2) - Reactor water level is being lowered using the RPV Bottom Head Drain path.

Which one of the following identifies whether these evolutions may continue while SECONDARY CONTAINMENT is inoperable, in accordance with N1-OP-34, Refueling Operations?

	Evolution (1)	Evolution (2)
A.	May continue	May continue
B.	May continue	Must be stopped
C.	Must be stopped	May continue
D.	Must be stopped	Must be stopped

Proposed Answer: B

Explanation: N1-OP-34 section D.33 requires SECONDARY CONTAINMENT integrity to be maintained whenever "recently irradiated fuel or an irradiated fuel cask is being handled in the SECONDARY CONTAINMENT and during operations with a potential for draining the reactor vessel (OPDRVs)". Lowering Reactor water level through the Reactor Bottom Head Drain path is defined as an OPDRV, and therefore, this evolution must be stopped. The movement of new fuel bundles into the Spent Fuel Pool may continue.

A. Incorrect – An Operation with the Potential to Drain the Reactor Vessel (OPDRV) is not allowed. Plausible because the loss of Secondary Containment integrity does not raise the risk of draining the vessel and this option does not involve fuel movement.

C. Incorrect – The movement of new fuel bundles into the Spent Fuel Pool may continue. Plausible because this is a sensitive evolution that moves nuclear material and movement of recently irradiated fuel would not be allowed. An Operation with the Potential to Drain the Reactor Vessel (OPDRV) is not allowed. Plausible because the loss of Secondary Containment integrity does not raise the risk of draining the vessel and this option does not involve fuel movement.

D. Incorrect – The movement of new fuel bundles into the Spent Fuel Pool may continue. Plausible because this is a sensitive evolution that moves nuclear material and movement of recently irradiated fuel would not be allowed.

Technical Reference(s): N1-OP-34

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288003-RBO-10

Question Source: Modified Bank – 2013 NRC #94

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.21
	Importance Rating	4.1

Knowledge of pre- and post-maintenance operability requirements.

Proposed Question: #96

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

Time (hhmm)	Condition(s)
1000	• Maintenance reports that all work is complete for an impeller replacement on Feedwater pump 11.
1400	• All tags have been cleared from the Feedwater pump 11 maintenance.
1500	• The required portions of the valve, electrical, and control switch lineups have been completed per N1-OP-16, Feedwater System Booster Pump to Reactor.
1600	• N1-ST-Q3, High Pressure Coolant Injection Pump and Check Valve Operability Test, has been commenced. • Feedwater pump 11 has been started and post-start checks are SAT.
1700	• Feedwater pump 11 has been secured and placed in the standby condition.
1800	• The Operations Review section of N1-ST-Q3 has been completed SAT.

Which one of the following identifies the ***earliest*** time that Feedwater pump 11 may be declared ***operable***, in accordance with MA-AA-716-012, Post Maintenance Testing?

A. 1500

B. 1600

C. 1700

D. 1800

Proposed Answer: D

Explanation: Technical Specification 3.1.8 requires Feedwater pump 11 to be able to deliver 3420 gpm at normal Reactor operating pressure. Since the maintenance activity included an impeller replacement, this capability must be proven as part of post maintenance testing prior to calling the pump operable. All required conditions for operability are not verified until the Operations Review section of N1-ST-Q3 is completed. Therefore, the earliest time Feedwater pump 11 can be declared operable is 1800.

A. Incorrect – The earliest time Feedwater pump 11 can be declared operable is 1800. Plausible because this is the time the pump is back in a standby lineup per the OP, and the pump is now ***available***.
B. Incorrect – The earliest time Feedwater pump 11 can be declared operable is 1800. Plausible because this is the first time when the ability of Feedwater pump 11 to run has been proven.
C. Incorrect – The earliest time Feedwater pump 11 can be declared operable is 1800. Plausible because this is the first time when Feedwater pump 11 has been returned to a standby condition (ready for start on a HPCI signal) after being verified able to run.

Technical Reference(s): Technical Specification 3.1.8 and bases, MA-AA-716-012, N1-ST-Q3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.7
	Importance Rating	3.6

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

Proposed Question: #97

An emergency is in progress with the following:

- An emergency exposure is being authorized for an emergent job to protect the Main Turbine from damage.
- A Radiation Work Permit (RWP) does not currently exist to support this emergency exposure.
- Radiation Protection reports that it will take approximately 30 minutes to process an RWP for this job.

Which one of the following identifies:

- the highest exposure limit that is allowed for this job, in accordance with EP-CE-113, Personnel Protective Actions, and
- the ability to perform the job without an RWP, in accordance with RP-AA-403, Administration of the Radiation Work Permit Program?

	Highest Emergency Exposure Allowed	Able to Perform Job Without an RWP?
A.	10 Rem	Yes
B.	10 Rem	No
C.	25 Rem	Yes
D.	25 Rem	No

Proposed Answer: A

Explanation: This job would be classified as protecting valuable property (Main Turbine), therefore EP-CE-113 allows a maximum of 10 Rem for the emergency exposure. RP-AA-403 does allow entry without an RWP to save plant equipment.

B. Incorrect – RP-AA-403 does allow entry without an RWP to save plant equipment. Plausible because an RWP is normally required for any entry into the RCA and this job is not for saving personnel or stopping a release.
C. Incorrect – EP-CE-113 allows a maximum of 10 Rem for this emergency exposure. Plausible because 25 Rem would be correct if the job were to save a life or protect large populations.
D. Incorrect – EP-CE-113 allows a maximum of 10 Rem for this emergency exposure. Plausible because 25 Rem would be correct if the job were to save a life or protect large populations. RP-AA-403 does allow entry without an RWP to save plant equipment. Plausible because an RWP is normally required for any entry into the RCA and this job is not for saving personnel or stopping a release.

Technical Reference(s): EP-CE-113, RP-AA-403

Proposed references to be provided to applicants during examination: None

Learning Objective: 245060 NS-EPL004-EO-02

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)(4)

Comments:

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.14
	Importance Rating	3.8

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: #98

Which one of the following describes a radiation monitor used in EOP Detail V, Core Damage Indications, and the associated threshold value for declaring that core damage is occurring, in accordance with the Emergency Operating Procedures?

	Radiation Monitor	Threshold for Declaring Core Damage is Occurring
A.	Offgas	≥ Hi-Hi Alarm
B.	Offgas	≥ 10 x Hi-Hi Alarm
C.	Drywell	≥ 80 R/hr
D.	Drywell	≥ 3000 R/hr

Proposed Answer: D

Explanation: EOP Detail V gives the following core damage indications:

V **Core Damage Indications**

Parameter	Value
Hydrogen concentration	≥ 4%
Drywell radiation	≥ 3000 R/hr
Reactor coolant activity	> 300 μCi/gm

Note: The question meets the K/A by testing knowledge of a radiation hazard (Offgas and Drywell radiation levels) that are expected to arise during an emergency condition (core damaging event).

A. Incorrect – Offgas radiation monitors are not used in Detail V. Plausible because Offgas radiation monitors are used as an indicator of fuel degradation in the EALs and in execution of N1-SOP-25.2.
B. Incorrect – Offgas radiation monitors are not used in Detail V. Plausible because Offgas radiation monitors are used as an indicator of fuel degradation in the EALs and in execution of N1-SOP-25.2.
C. Incorrect – The threshold used in Detail V is 3000 R/hr, not 80 R/hr. Plausible because 80 R/hr is the threshold for loss of the Reactor Coolant System barrier in the EALs

Technical Reference(s): N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Bank - 2017 Cert #96

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)(4)

Comments:

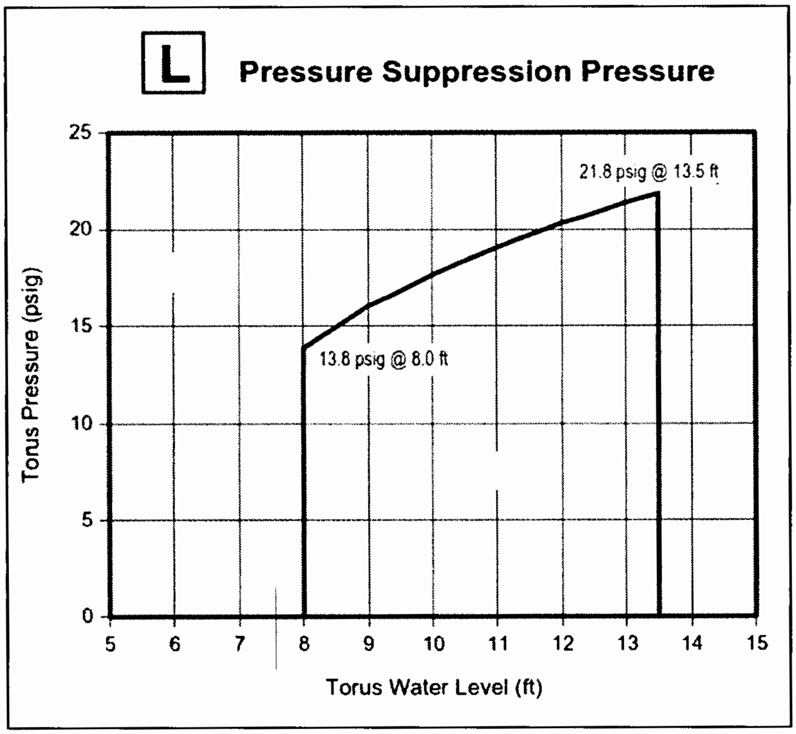
Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.17
	Importance Rating	4.3

Knowledge of EOP terms and definitions.

Proposed Question: #99

The plant is experiencing a transient with the following:

- N1-EOP-4, Primary Containment Control, has been entered.
- Torus pressure is 18.0 psig and rising slowly.
- Torus level is 11 feet and rising slowly.
- The order has been given to initiate Containment Spray per N1-EOP-1 Attachment 17.



Which one of the following describes when Containment Spray is defined to first be in service and the next required action once Containment Spray is in service?

Containment Spray is defined to first be in service when...

A. the first Containment Spray pump is running and up to rated flow. Next, an evaluation of pressure suppression capability must be performed.

B. two Containment Spray pumps are running and up to rated flow. Next, an evaluation of pressure suppression capability must be performed.

C. the first Containment Spray pump is running and up to rated flow. Next, an RPV Blowdown must be performed.

D. two Containment Spray pumps are running and up to rated flow. Next, an RPV Blowdown must be performed.

Proposed Answer: A

Explanation: When the Torus pressure is greater that the Pressure Suppression Pressure (PSP) upon entry into the Primary Containment Control EOP, all steps up to evaluating proximity to PSP are to be implemented prior to evaluating Torus Pressure against the PSP curve. Containment sprays are considered to be "in service" when one train of Containment Spray is initiated (the other additional loop of Containment Spray that is started is for Appendix J Water Seal requirements). Evaluation of Torus pressure in relation to the Pressure Suppression Pressure (PSP) curve is expected to occur once Containment Sprays are "in service."

B. Incorrect – Containment sprays are considered to be "in service" when one train of Containment Spray is initiated (the other additional loop of Containment Spray that is started is for Appendix J Water Seal requirements). Plausible because N1-EOP-1 Attachment 17 does require starting two Containment Spray pumps.
C. Incorrect – An RPV Blowdown is only performed once an evaluation of PSP is made after Containment Spray has been placed in service. Plausible because this would be correct if the direction to perform an RPV Blowdown based on PSP were an override step in N1-EOP-4.
D. Incorrect – Containment sprays are considered to be "in service" when one train of Containment Spray is initiated (the other additional loop of Containment Spray that is started is for Appendix J Water Seal requirements). Plausible because N1-EOP-1 Attachment 17 does require starting two Containment Spray pumps. An RPV Blowdown is only performed once an evaluation of PSP is made after Containment Spray has been placed in service. Plausible because this would be correct if the direction to perform an RPV Blowdown based on PSP were an override step in N1-EOP-4.

Technical Reference(s): N1-EOP-4, N1-EOP-1, OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank - 2010 NRC #99

Question History: 2010 NRC #99

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)(5)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	
K/A #	2.4.21
Importance Rating	4.6

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

Proposed Question: #100

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

- The Reactor has been shut down for 12 hours.
- Preparations are underway for Reactor head de-tensioning.
- Reactor coolant temperature is 190°F and slowly lowering.
- Shutdown Cooling loops 11 and 13 are in service.
- Shutdown Cooling loop 12 is unavailable due to an unexpected pump trip.
- Spent Fuel Pool Cooling loop 11 is in service.
- Reactor Water Cleanup is isolated and tagged out for maintenance.

Note: A portion of OU-NM-103-101, Shutdown Safety Management Program, Attachment 1, Unit 1 Key Safety Functions, is provided on the next page.

Which one of the following describes the current shutdown safety color code for Reactor decay heat removal, in accordance with OU-NM-103-101 and ER-AA-600-1023, Paragon Model Capability?

A. Green

B. Yellow

C. Orange

D. Red

Attachment 1, Unit 1 Key Safety Functions

Table 1: UNIT 1 REACTOR DECAY HEAT REMOVAL LESS THAN OR EQUAL TO 212 °F

Maintaining decay heat removal capability is a key safety function during shutdown conditions. An extended loss of the decay heat removal capability can result in a depletion of cooling water and uncovering of the fuel. Providing a Defense-In-Depth for decay heat removal capability, commensurate with the plant conditions identified in the outage schedule, can effectively enhance shutdown safety.

Systems Providing Reactor Decay Heat Removal	Minimum Support Systems	Other Possible Required Support Systems
• #11 Train Shutdown Cooling System	• 2 RBCLC Pumps / 2 RBCLC Heat Exchangers • Any combination of 2 SW Pumps OR ESW Pumps • SWP Intake Gates[2] • Instrument Air • PB 16B, PB 167[5], and DC Valve Board 12 • MG131 Trip Bus	• Diesel Fire Pump • Service Air tie to Instrument Air available
• #12 Train Shutdown Cooling System	• 2 RBCLC Pumps / 2 RBCLC Heat Exchangers • Any combination of 2 SW Pumps OR ESW Pumps • SWP Intake Gates[2] • Instrument Air • PB 17B, PB 167[5], and DC Valve Board 12 • MG131 Trip Bus	• Diesel Fire Pump • Service Air tie to Instrument Air available
• #13 Train Shutdown Cooling System	• 2 RBCLC Pumps / 2 RBCLC Heat Exchangers • Any combination of 2 SW Pumps OR ESW Pumps • SWP Intake Gates[2] • Instrument Air • PB 16B, PB 167[5], and DC Valve Board 12 • MG131 Trip Bus	• Diesel Fire Pump • Service Air tie to Instrument Air available
• #11 Train SFP Cooling with Cavity Flooded and Gates Out [1,4]	• 2 RBCLC Pumps / 2 RBCLC Heat Exchangers • Any combination of 2 SW Pumps OR ESW Pumps • SWP Intake Gates[2] • PB 16A • Instrument Air • See Attachment 1, Table 1B, for detailed list of equipment to protect	• Service Air tie to Instrument Air available
• #12 Train SFP Cooling with Cavity Flooded and Gates Out [1,4]	• 2 RBCLC Pumps / 2 RBCLC Heat Exchangers • Any combination of 2 SW Pumps OR ESW Pumps • SWP Intake Gates[2] • PB 17A • Instrument Air • See Attachment 1, Table 1B, for detailed list of equipment to protect	• Service Air tie to Instrument Air available

• Feed and Bleed [6]	• Reactor Inventory Control Systems per Table 3
	• Bleed using one of the following paths:
	  • Reactor Water Clean-up or bottom head drain with Waste Collector tank or condenser available to receive water or
	  • Lowering Reactor Cavity Level using Core Spray or
	  • Drain using the SFP[7]
		• DWEDT 11 or 12 and Pump

Attachment 1, Unit 1 Key Safety Functions (Continued)

Table 1: UNIT 1 REACTOR DECAY HEAT REMOVAL LESS THAN OR EQUAL TO 212 °F (Continued)

Systems Providing Reactor Decay Heat Removal	Minimum Support Systems	Other Possible Required Support Systems
• Reactor Water Clean-Up[5]	• 2RBCLC Pumps / 2 RBCLC Heat Exchangers • Any combination of 2SW or ESW Pumps • SWP Intake Gates • Instrument Air • PR 15B, PB-161B, PB171B DC Valve Board 12	
Shutdown Safety Criteria	• For Cold Shutdown condition, N=2 during first 72 hours after shutdown. Otherwise N=1. • For Refuel condition, N=1. • For Major Maintenance condition, N=0[3]	

1 Only if supported by Engineering Calculation for Decay Heat Load calculation and time delay after Reactor shutdown.

2 Any of the following of SWP Intake Gate combinations: (a) A, C Open; D, E Shut (b) B, C, Open, D, E Shut, (c) D, E Open, A, B, C Shut, (d) A and/or B Open, C Open, E throttled or Shut, D Shut

3 KSF NOT required and can be marked 'N/A' when Spent Fuel Pool Gates are installed. SFP Gates shall NOT be installed earlier than 185 hours after reactor shutdown (HARS).

4 Heat removal capacities of both SFC Loops operating simultaneously are additive. If decay heat load is greater than the capacity of an individual loop, then both loops may be operated in parallel so that their combined heat removal capacity is greater than the decay heat load and these loops together may count as an "N" or "+1" system.

5 PB-167 not required if SDC isolation is defeated.

6 N+1 or N+2 system only

7 This mode requires the gates out and the cavity flooded.

Proposed Answer: B

Explanation: With Reactor coolant temperature less than 212°F, the plant is in Cold Shutdown. During the first 72 hours after shutdown, the minimum number of system necessary for Reactor decay heat removal (N) is 2. Of the six possible Reactor decay heat removal systems, two are available (Shutdown Cooling loops 11 and 13) to satisfy N. Shutdown Cooling loop 12 is unavailable due to maintenance. SFP Cooling trains 11 and 12 are unavailable for Reactor decay heat removal because the SFP gates are still installed and the Reactor cavity is NOT yet flooded, as evidenced by preparations underway for Reactor head de-tensioning. RWCU is not available due to maintenance. Feed and bleed can be counted as N+1 , since N is already met by SDC trains. With the available number of systems for this key safety function equal to N+1, the shutdown safety color code is Yellow.

A. Plausible – If RWCU was available, or if the SFP gates were removed, the plant would be at N+2 or greater, which would result in a Green classification.
C. Plausible – The key safety function is at N, not <N, which would require an Orange classification if contingency plans were in place.
D. Plausible – If the key safety function was <N due to the unexpected pump trip, a Red classification would be required.

Technical Reference(s): OU-NM-103-101

Proposed references to be provided to applicants during examination: None

Learning Objective: NIP-OUT-01-CT-01

Question Source: Modified Bank - 2017 Cert #99

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)(1)

Comments: