Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK1.01
	Importance Rating	3.7

PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation

Proposed Question: #1

A plant startup is in progress with the following:

- The Reactor Mode Switch is in RUN.
- Reactor power is 10%.

Then, all Reactor Recirculation pumps trip.

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

A. A manual Reactor scram is required.

B. A manual Reactor scram is NOT required, but a plant shutdown is required.

C. Reactor operation can continue as long as NO indications of thermal-hydraulic instability occur.

D. Reactor operation can continue as long as core conditions satisfy the natural circulation line of the Power/Flow map.

Proposed Answer: A

Explanation: With the plant operating at power and a trip of one or more Reactor Recirculation pumps, N1-SOP-1.3 entry is required. With less than 3 operating Recirc loops, the procedure requires manually scramming the Reactor. Additionally, N1-SOP-1.5 entry is required due to the unplanned power change. With no Recirc pumps running, this procedure also requires manually scramming the Reactor.

B. Incorrect – A manual Reactor scram is required. Plausible because Reactor power is low to begin with and there are no indications of THI given. Also plausible because there are conditions in N1-SOP-1.5 that required a normal shutdown instead of a Reactor scram.
C. Incorrect – A manual Reactor scram is required. Plausible because one of the related conditions that requires a Reactor scram in N1-SOP-1.5 is presence of THI.
D. Incorrect – A manual Reactor scram is required. Plausible because position on the Power/Flow map is addressed in both N1-SOP-1.3 and N1-SOP-1.5 and there is a natural circulation line.

Technical Reference(s): N1-SOP-1.3, N1-SOP-1.5

Proposed references to be provided to applicants during examination: None

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

Question Source: Modified Bank - JAF 14-2 NRC #39

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 #	295001 AK1.01
	Importance Rating	3.5

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation

Proposed Question: #39

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP/HOT STANDBY.
- Reactor pressure is 50 psig and slowly rising.

Then, both Recirculation pumps trip.

Which one of the following describes the required control of the Reactor in accordance with AOP-8, Unexpected Change in Core Flow?

A. A manual Reactor scram is required.

B. A manual Reactor scram is NOT required, but a plant shutdown is required.

C. Reactor operation can continue as long as NO indications of thermal-hydraulic instability occur.

D. Reactor operation can continue as long as core conditions satisfy the natural circulation line of the Power-Flow map.

Proposed Answer: B

Explanation: If both Recirculation pumps trip and the Reactor Mode Switch is in STARTUP/HOT STANDBY, AOP-8 requires commencing a normal plant shutdown per OP-65 and achieving Mode 3 within 12 hours. The trip of both Recirculation pumps places the plant in natural circulation. AOP-8 does not allowed continued operation in natural circulation.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 2.4.16
	Importance Rating	3.5

PARTIAL OR COMPLETE LOSS OF AC POWER

Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines

Proposed Question: #2

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-RPV, 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-RPV, N1-EOP-RPV 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-RPV, 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-RPV.

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

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, in accordance with NMP1 PSTG/PSSAG Vol. I.

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): N1-EOP-B-HOT, N1-SOP-33A.1, N1-EOP-RPV, NMP1 PSTG/PSSAG Vol. I

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-EOP00C01 TO #1

Question Source: Bank – 2018 NRC #39

Question History: 2018 NRC #39

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AK3.01
	Importance Rating	3.9

PARTIAL OR COMPLETE LOSS OF DC POWER

Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: Load shedding

Proposed Question: #3

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

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

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

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

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

Proposed Answer: B

Explanation: During a Station Blackout, Static Battery Chargers are lost and Batteries begin to discharge. Per N1-OP-47A, B.1, load reductions are performed to preserve Battery capacity, specifically for Reactor instrumentation, Emergency Condenser control, and EDG starting.

A. Incorrect – N1-SOP-33A.2 requires load shedding MG Set 167 within 2 hours of event start. Plausible because MG Set 167 does operate from DC power during a Station Blackout.
C. Incorrect – Breakers on the SR Battery Boards have been bypassed. Fuses have been installed in their place in the back of the Battery Boards. Both the Negative and Positive legs to each load are fused. Plausible if believe lowering voltage will draw more current and cause trips.
D. Incorrect – PPC and annunciators are de-energized when MG 167 is tripped 2 hours into the 4 hour coping period. Plausible because these loads are maintained for a longer period of time than most other loads.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-10

Question Source: Bank - 2020 NRC #41

Question History: 2020 NRC #41

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AK1.03
	Importance Rating	3.8

MAIN TURBINE GENERATOR TRIP

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the MAIN TURBINE GENERATOR TRIP: Reactor level control

Proposed Question: #4

A plant startup is in progress with the following:

- Reactor power is 17%.
- The Main Generator has just been synchronized to the grid.
- Reactor water level is 72" and stable.
- Feedwater pump 12 is operating.
- Feedwater pumps 11 and 13 are secured.

Then, a Main Generator lockout results in the following:

- The Turbine trips.
- Reactor water level reaches a low of 68".
- Reactor pressure is 918 psig and stable on Turbine Bypass Valves.

Which one of the following describes the resulting status of Feedwater pump 11 and Feedwater flow control valve 12?

	Feedwater Pump 11	Feedwater Flow Control Valve 12
A.	Remains secured	Does NOT go into HPCI mode
B.	Remains secured	Goes into HPCI mode
C.	Automatically starts	Does NOT go into HPCI mode
D.	Automatically starts	Goes into HPCI mode

Proposed Answer: D

Explanation: The Turbine trip signal causes auto-initiation of the HPCI mode of Feedwater. The HPCI mode of Feedwater affects operation of Feedwater pumps and flow control valves. This causes Feedwater pump 11 to automatically start. This also causes Feedwater flow control valve 12 to go into HPCI mode.

A. Incorrect – Feedwater pump 11 automatically starts. Feedwater flow control valve 12 goes into HPCI mode. Plausible because Feedwater flow control valve 12 was already in service and Reactor water level remains above 53".
B. Incorrect – Feedwater pump 11 automatically starts. Plausible because this would be correct if Feedwater pump 12 flow rose to 1.9 Mlbm/hr without a Turbine trip signal.
C. Incorrect – Feedwater flow control valve 12 goes into HPCI mode. Plausible because Feedwater flow control valve 12 was already in service and Reactor water level remains above 53".

Technical Reference(s): N1-OP-16, ARP F4-4-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-05

Question Source: Modified Bank – 2013 NRC #30

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

HPCI

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

Proposed Question: #30

A plant startup is in progress with the following:

- Reactor power is 18%.
- Turbine startup is in progress.

Then, a Turbine overspeed signal results in the following:

- Annunciator F1-2-5, RPS CH 11 TURBINE TRIP, alarms.
- Annunciator F4-2-4, RPS CH 12 TURBINE TRIP, alarms.
- Reactor water level is 72" and stable.
- Reactor pressure is 918 psig and stable.

Which of the following annunciator(s) would also be in alarm?

(1) F1-2-1, RPS CH 11 AUTO REACTOR TRIP
(2) F4-2-8, RPS CH 12 AUTO REACTOR TRIP
(3) F4-4-1, HPCI MODE AUTO INITIATE
(4) A1-4-5, TURBINE STOP VALVES CLOSED

A. (4) only

B. (1) and (2) only

C. (3) and (4) only

D. (1), (2), (3), and (4)

Proposed Answer: C

Explanation: Turbine overspeed is a mechanical automatic Turbine trip signal, as evidenced by annunciators F1-2-5 and F4-2-4 being in alarm. Below approximately 40-45% power, a Turbine trip does not cause an automatic Reactor scram. Therefore, annunciators F1-2-1 and F4-2-8 would not be in alarm. The Turbine trip signal does cause Turbine stop valves to close (Annunciator A1-4-5). The subsequent generator trip will cause an electrical turbine trip signal and cause HPCI to auto-initiate (Annunciator F4-4-1) (3T-1, 3T-2). These functions are unaffected by the low power level.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AK2.03
	Importance Rating	3.9

SCRAM

Knowledge of the relationship between the SCRAM and the following systems or components: CRD hydraulic system

Proposed Question: #5

The plant is operating at 100% power with the in-service CRD flow control valve in automatic when a Reactor scram occurs.

Which one of the following describes the response of CRD total system flow and drive water pressure?

	CRD Total System Flow	CRD Drive Water Pressure
A.	Lowers	Lowers
B.	Lowers	Remains the same
C.	Rises	Lowers
D.	Rises	Remains the same

Proposed Answer: C

Explanation: When the scram occurs, CRD total system flow rises due to the additional flow paths from the charging water header to each of the HCUs. The flow control valve senses this higher flow and closes in an attempt to lower flow. Due to the arrangement of the system, this causes drive water pressure to lower.

A. Incorrect – Total system flow rises. Plausible because the flow control valve closes in an attempt to lower flow.
B. Incorrect – Total system flow rises. Plausible because the flow control valve closes in an attempt to lower flow. Drive water pressure lowers. Plausible because the drive water pressure control valve does not change position.
D. Incorrect – Drive water pressure lowers. Plausible because the drive water pressure control valve does not change position.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-03

Question Source: Bank – 2008 NRC #36

Question History: 2008 NRC #36

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 2.4.3
	Importance Rating	3.7

CONTROL ROOM ABANDONMENT

Ability to identify post-accident instrumentation

Proposed Question: #6

The Control Room has been evacuated and control has been established at Remote Shutdown Panel 11.

Given the following Primary Containment parameters:

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

Which of these parameters can be determined using installed indicators at Remote Shutdown Panel 11?

A. (1) only

B. (2) only

C. (3) only

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

Proposed Answer: A

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

Note: The question meets the K/A because it requires the candidate to identify post-accident instrumentation that would be used during a Control Room evacuation. 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 not have an installed indicator for Torus water level. Plausible because it does have an installed indicator for Torus water temperature.
C. 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.
D. Incorrect – Remote Shutdown Panel 11 does not have an installed indicator for Torus water level. Plausible because it does have an installed indicator for Torus water temperature. 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 3.6.11-1, Technical Specification Table 3.6.13-1, 1101-296000C01, N1-SOP-21.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-296000-RBO-5

Question Source: Modified Bank - 2018 NRC #57

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

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.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AA1.04
	Importance Rating	3.4

PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW)

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER (CCW): Reactor water cleanup system

Proposed Question: #7

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 95ºF 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

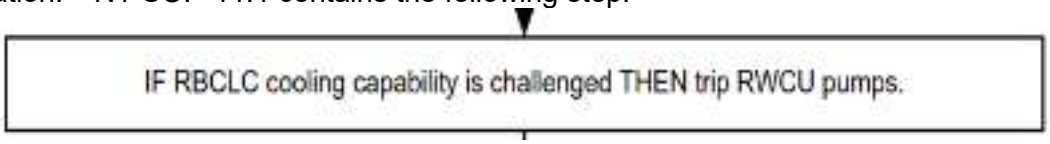
B. Secure Fuel Pool Cooling

C. Isolate RBCLC to the Drywell

D. Shutdown two Recirculation pumps

Proposed Answer: A

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



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 – 2018 NRC #34

Question History: 2018 NRC #34

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA2.01
	Importance Rating	4.1

PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR

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

Proposed Question: #8

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

- Instrument Air Compressor (IAC) 13 is running.
- IAC 11 is red-flagged with Loading Selector Switches in Position 1.
- IAC 12 is green-flagged with Loading Selector Switches in Position 2.

Then, an Instrument Air (IA) line break results in the following:

Time (mm:ss)	Condition(s)
00:10	IAC 13 loads
00:20	IAC 11 loads
00:30	IAC 12 starts and loads
00:45	Annunciator L1-4-7, INST AIR BACK – UP VALVE OPEN, alarms
00:57	Breathing Air Blocking Valve 114-02 closes Automatic Dryer Bypass valves 94-164 and 94-206 open

IA pressure has been continuously lowering throughout this transient.

Which one of the following describes Instrument Air pressure at time 00:38?

At time 00:38, Instrument Air pressure is between…

A. 98 and 101 psig

B. 93 and 98 psig

C. 90 and 93 psig

D. 85 and 90 psig

Proposed Answer: C

Explanation: IAC 12 is green-flagged as the standby compressor, therefore it starts and loads at an IA pressure of 93 psig, then attempts to control pressure 96-102 psig. Annunciator L1-4-7, INST AIR BACK – UP VALVE OPEN, alarms at 90 psig. Therefore, at time 00:38, Instrument Air pressure is between 90 and 93 psig.

A. Incorrect – At time 00:38, Instrument Air pressure is between 90 and 93 psig. Plausible because this would be correct for time 00:15.
B. Incorrect – At time 00:38, Instrument Air pressure is between 90 and 93 psig. Plausible because this would be correct for time 00:25.
D. Incorrect – At time 00:38, Instrument Air pressure is between 90 and 93 psig. Plausible because this would be correct for time 00:50.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-05

Question Source: Modified Bank - 2013 NRC #8

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Proposed Question: #8

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

- Instrument Air Compressor (IAC) 13 is running.
- IAC 11 is red-flagged with Loading Selector Switches in Position 1.
- IAC 12 is green-flagged with Loading Selector Switches in Position 2.

Then, an Instrument Air (IA) line break results in the following sequence of events:

Time (mm:ss)	Event
00:10	Instrument Air Compressor (IAC) 13 loads.
00:20	IAC 11 loads.
00:30	IAC 12 starts and loads.
00:45	Annunciator L1-4-7, INST AIR BACK – UP VALVE OPEN, alarms.
00:57	Breathing Air Blocking Valve 114-02 closes; Automatic Dryer Bypass valves 94-164 and 94-206 open.

IA pressure has still been continuously lowering throughout this transient.

Which one of the following describes Instrument Air pressure at time 00:25?

At time 00:25, Instrument Air pressure is between…

A. 98 and 101 psig.

B. 93 and 98 psig.

C. 90 and 93 psig.

D. 85 and 90 psig.

Proposed Answer: B

Explanation: IAC 11 is red-flagged as the backup compressor with Load Selector Switches in Position 1, therefore it loads on an IA pressure of 98#. IAC 12 is green-flagged as the standby compressor, therefore it starts and loads at an IA pressure of 93#, then attempts to control pressure 96#-102#.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.02
	Importance Rating	4.1

LOSS OF SHUTDOWN COOLING

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: RHR/shutdown cooling system flow

Proposed Question: #9

The plant is shutdown with the following:

- Shutdown Cooling (SDC) loops 11 and 12 have been placed in service.
- NO related valves have been gagged.

Then, a complete loss of Instrument Air occurs.

Which one of the following describes the effect, if any, on SDC flow through the core and RBCLC flow through the SDC heat exchangers?

	SDC Flow Through Core	RBCLC Flow Through SDC HX
A.	Remains the same	Remains the same
B.	Remains the same	Lowers
C.	Lowers	Remains the same
D.	Lowers	Lowers

Proposed Answer: D

Explanation: The instrument air system provides motive force to operate the electropneumatic flow control valves and the minimum flow control valves. It provides motive force for RBCLC inlet to SDC. On a loss of air pressure, the flow control valves close and the minimum flow control valves open. Also, the RBCLC blocking valve fails closed. Therefore, both flows will drop to zero.

A. Incorrect – SDC flow through the core lowers. Plausible because the SDC isolation and suction valves are all motor-operated, such that they remain open on loss of Instrument Air. Also plausible because some pneumatically controlled valves fail open or as-is on loss of pressure to maintain system operation.
B. Incorrect – SDC flow through the core lowers. Plausible because the SDC isolation and suction valves are all motor-operated, such that they remain open on loss of Instrument Air. RBCLC flow through the SDC heat exchangers lowers. Plausible because the valves used to set RBCLC flow through the SDC heat exchangers are manually operated. Also plausible because some pneumatically controlled valves fail open or as-is on loss of pressure to maintain system operation.
C. Incorrect – RBCLC flow through the SDC heat exchangers lowers. Plausible because the valves used to set RBCLC flow through the SDC heat exchangers are manually operated. Also plausible because some pneumatically controlled valves fail open or as-is on loss of pressure to maintain system operation.

Technical Reference(s): C-18018-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-8

Question Source: Bank – 2009 Cert #26

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AK3.03
	Importance Rating	3.7

REFUELING ACCIDENTS

Knowledge of the reasons for the following responses or actions as they apply to REFUELING ACCIDENTS: Ventilation isolation

Proposed Question: #10

A refueling outage is in progress with the following:

- An irradiated fuel bundle has been dropped in the Spent Fuel Pool.
- Reactor Building ventilation has automatically isolated due to a signal from Fuel Pool/Refuel Bridge High Range ARM (RN016C-10).

Which one of the following describes the reason for this isolation?

A. Minimize air currents above the Spent Fuel Pool

B. Avoid damage to Reactor Building ventilation fans

C. Control fission product release to the environment

D. Prevent personnel exposure in lower Reactor Building elevations

Proposed Answer: C

Explanation: Reactor Building ventilation isolates to control the release of fission products to the environment by ensuring they are routed to RBEVS.

A. Incorrect – The reason is to control fission product release to the environment. Plausible because RBEVS does have a lower flow rate than RBVS and higher air flow rate above the SFP could result in more spread of contamination.
B. Incorrect – The reason is to control fission product release to the environment. Plausible because the isolation does stop flow of contaminated air through the Reactor Building ventilation fans, avoiding any possible damage.
D. Incorrect – The reason is to control fission product release to the environment. Plausible that continued operation of Reactor Building ventilation could spread contamination throughout the building, which would be undesirable.

Technical Reference(s): Technical Specification 3.4.2 Objective and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EA1.12
	Importance Rating	4.2

HIGH DRYWELL PRESSURE

Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Suppression pool spray (Mark I, II)

Proposed Question: #11

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

- Reactor water level is -25" and lowering slowly.
- Reactor pressure is 400 psig and lowering.
- All four Containment Spray pumps are spraying the Containment.
- Drywell pressure is 3.2 psig and lowering slowly.

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

Secure Containment Spray…

A. now. The pump control switches are required to be left green-flagged.

B. now. The pump control switches are NOT required to be left green-flagged.

C. when Drywell pressure reaches 0 psig. When secured, the pump control switches are required to be left green-flagged.

D. when Drywell pressure reaches 0 psig. When secured, the pump control switches are NOT required to be left green-flagged.

Proposed Answer: B

Explanation: Drywell pressure of 3.5 psig is the threshold for securing Containment Sprays in N1-EOP-PC. Since Drywell pressure is below this value, Containment Sprays are required to be secured now. This may be accomplished by placing the pump control switches in PULL-TO-LOCK, specifically to prevent automatic Containment Spray operation. Containment Spray initiation is controlled manually while in N1-EOP-PC.

A. Incorrect – Pump control switches may be placed in PULL-TO-LOCK. Plausible because this does defeat auto-start capability and Drywell pressure is currently low enough that green-flagging the switches would stop the pumps.
C. Incorrect – Pump control switches may be placed in PULL-TO-LOCK. Plausible because this does defeat auto-start capability and Drywell pressure is currently low enough that green-flagging the switches would stop the pumps. Containment Spray is required to be secured now. Plausible because 0 psig is the threshold used in the SAPs.
D. Incorrect – Containment Spray is required to be secured now. Plausible because 0 psig is the threshold used in the SAPs.

Note: The question meets the K/A because, based on NMP1 design, the Containment Spray system encompasses both Suppression Pool (Torus) Spray and Drywell Spray. Containment Spray is a dedicated system (unlike RHR at other BWRs) and Torus and Drywell Spray are simultaneously initiated and secured.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank - 2017 Cert #58

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EK2.01
	Importance Rating	4.2

HIGH REACTOR PRESSURE

Knowledge of the relationship between the HIGH REACTOR PRESSURE and the following systems or components: RPS

Proposed Question: #12

A plant startup is in progress with the following:

- IRMs are on range 9.
- An open Turbine Bypass Valve rapidly fails closed.
- Reactor pressure rises rapidly to a peak of 1085 psig.
- Other Turbine Bypass Valves throttle open.
- Reactor pressure is restored to a normal value within 7 seconds of the start of the transient.

Which one of the following describes the resulting status of RPS, ERVs, and Emergency Condensers?

A. NO Reactor scram, ERV actuation, or Emergency Condenser initiation occurs.

B. The Reactor scrams. NO ERV actuation or Emergency Condenser initiation occurs.

C. The Reactor scrams. Two ERVs open. NO Emergency Condenser initiation occurs.

D. The Reactor scrams. Emergency Condensers initiate. NO ERV actuation occurs.

Proposed Answer: B

Explanation: The Reactor scrams based on pressure exceeding 1080 psig. There is no time delay on this signal. The lowest ERV actuation setpoint is 1090 psig, which is not reached. The Emergency Condensers initiate only if pressure is above 1080 psig for more than 12 seconds. In this question, pressure is only above normal for 7 seconds, so no Emergency condenser initiation occurs.

A. Incorrect – The Reactor scrams. Plausible because pressure goes only slightly above the scram setpoint for a short amount of time, and there are time delays on some related signals.
C. Incorrect – No ERVs open. Plausible because pressure is close to the lowest ERV initiation setpoint of 1090 psig.
D. Incorrect – ECs to not initiate. Plausible because pressure went above the setpoint of 1080 psig, but did not meet the 12 second time delay requirement

Technical Reference(s): N1-OP-40, N1-OP-13, N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: Bank - 2010 NRC #44

Question History: 2010 NRC #44

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EA1.09
	Importance Rating	4.1

SUPPRESSION POOL HIGH WATER TEMPERATURE

Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Safety/relief valves

Proposed Question: #13

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

- The Reactor has scrammed.
- MSIVs are closed.
- Reactor water level is 100" and stable.
- Reactor pressure is 975 psig and rising slowly.
- Torus water temperature is 115°F and rising slowly.
- All ERVs are currently closed.
- The Unit Supervisor has directed Reactor pressure controlled 800-1000 psig.

Which one of the following describes the allowed use of ERVs for pressure control under these conditions, in accordance with plant procedures?

Using ERVs for pressure control is currently…

A. NOT allowed due to Reactor water level.

B. NOT allowed due to Torus water temperature.

C. allowed. Cycle the same ERV repeatedly, as needed.

D. allowed. Cycle through all ERVs sequentially, as needed.

Proposed Answer: D

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

A. Incorrect – ERVs are currently allowed to be used for pressure control despite high Reactor water level. Plausible because this high Reactor water level results in restrictions on use of ECs and could result in carryover of moisture in the ERV discharge lines.
B. Incorrect – ERVs are currently allowed to be used for pressure control despite high Torus water temperature. Plausible because Torus water temperature is above the 110°F limit used in EOPs and use of ERVs will further raise Torus water temperature.
C. Incorrect – N1-EOP-RPV contains a note that directs opening ERVs per a pre-defined sequence, vice opening the same ERV multiple times. Plausible because Torus Cooling does discharge to one point in the Torus through the test return line, so this area will be cooled more directly.

Technical Reference(s): N1-EOP-RPV, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239002-RBO-10

Question Source: Modified Bank – 2023 NRC #23

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1	Written Examination Question Worksheet

Examination Outline Cross-Reference: Level RO

Tier #	1
Group #	2
K/A #	295013 AA2.02
Importance Rating	3.5

HIGH SUPPRESSION POOL TEMPERATURE

Ability to determine or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Localized heating/stratification

Proposed Question: #23

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

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

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

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

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

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

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

Proposed Answer: C

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

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 2.1.2
	Importance Rating	4.1

HIGH DRYWELL TEMPERATURE

Knowledge of operator responsibilities during any mode of plant operation

Proposed Question: #14

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

N1-EOP-PC 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. Incorrect – 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. Incorrect – 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. Incorrect – 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-PC, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank - 2018 NRC #40

Question History: 2018 NRC #40

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EK3.05
	Importance Rating	3.6

LOW SUPPRESSION POOL WATER LEVEL

Knowledge of the reasons for the following responses or actions as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool makeup system(s) operation

Proposed Question: #15

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

- A leak develops in the Torus.
- Torus water level is 10.5' and lowering.
- N1-EOP-PC, Primary Containment Control, has been entered.

Which one of the following describes a reason for using N1-EOP-1 Attachment 18, Raw Water to Torus Makeup, instead of Attachment 6, Torus Make-Up from Keep Fill?

A. Higher purity water

B. Higher makeup flow rate

C. Does NOT require starting a pump

D. Does NOT require valve manipulations

Proposed Answer: B

Explanation: One reason for using N1-EOP-1 Attachment 18 instead of Attachment 6 is that it has a higher makeup flow rate.

A. Incorrect – N1-EOP-1 Attachment 18 uses lower purity water (water directly from Lake Ontario vs. water from the CSTs). Plausible because this would be a reason for using Attachment 6 instead of Attachment 18.
C. Incorrect – N1-EOP-1 Attachment 18 requires starting a Containment Spray Raw Water pump. Plausible because this would be a reason for using Attachment 6 instead of Attachment 18.
D. Incorrect – N1-EOP-1 Attachment 18 requires multiple valve manipulations. Plausible because Attachment 6 does require valve manipulations. Also plausible because one advantage of Attachment 18 is that it does not require field actions to support valve manipulations (Attachment 6 requires field action to supply power to two valves).

Technical Reference(s): N1-EOP-PC, N1-EOP-1 Attachments 6 and 18, N1-OP-14, N1-OP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK1.01
	Importance Rating	4.7

REACTOR LOW WATER LEVEL

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the REACTOR LOW WATER LEVEL: Adequate core cooling

Proposed Question: #16

A loss of coolant accident has resulted in the following:

- Reactor water level is -94" and lowering slowly.
- Four (4) ERVs are open.
- Reactor pressure is 100 psig and lowering.
- Condensate pump 12 is injecting to the Reactor.
- NO other injection sources are available.

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

Adequate core cooling (ACC) is…

A. NOT currently being provided.

B. currently being provided. ACC will be first lost if Reactor water level reaches -109".

C. currently being provided. ACC will be first lost if Reactor water level reaches -126".

D. currently being provided. ACC will be first lost if Reactor water level reaches -156".

Proposed Answer: C

Explanation: The preferred method of ACC, Core Submergence, is not currently available because Reactor water level is below -84". However, ACC is still assured by Steam Cooling, which requires Reactor water level to be above -126". If Reactor water level reaches -126", this method of ACC will be lost. ACC by Spray Cooling is not available because no Core Spray pumps are available. Therefore, if Reactor water level reaches -126", ACC will no longer be assured.

A. Incorrect – Since Reactor water level is greater than -126", ACC is assured by Steam Cooling. Plausible because the preferred method of ACC, Core Submergence, is not currently available because Reactor water level is below -84".
B. Incorrect – ACC will first be lost if Reactor water level reaches -126". Plausible because -109" is the Reactor water level historically associated with Steam Cooling, however this level has recently been revised in the EOPs at NMP1.
D. Incorrect – ACC will first be lost if Reactor water level reaches -126". Plausible because -156" is the Reactor water level used in the N1-EOP-C1 Steam Cooling leg, which is not applicable since there is currently injection to the Reactor.

Technical Reference(s): N1-EOP-RPV, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: Bank – 2017 NRC #73

Question History: 2017 NRC #73

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK2.14
	Importance Rating	4.0

SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN

Knowledge of the relationship between the SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following systems or components: RPIS (BWR 2, 3, 4, 5)

Proposed Question: #17

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

- A Turbine trip causes a Reactor scram.
- All APRMs are unavailable.
- One control rod fails to insert and is at position 48.
- Two control rods partially insert and are at position 06.
- All other control rods fully insert.

Which one of the following describes the status of the Reactor shutdown, in accordance with the Emergency Operating Procedures?

The Maximum Subcritical Banked Withdrawal Position __(1)__ being exceeded.

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

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

Proposed Answer: B

Explanation: The Maximum Subcritical Banked Withdrawal Position of 04 is being exceeded. With more than one rod withdrawn beyond the Maximum Subcritical Banked Withdrawal Position, the Reactor is NOT considered shutdown under all conditions.

A. Incorrect – The Reactor is NOT considered shutdown under all conditions. Plausible because one rod full out is allowed by Technical Specification shutdown margin calculations and the other two rods are only withdrawn one notch beyond the Maximum Subcritical Banked Withdrawal Position. Also plausible because this would be correct if only one rod was out.
C. Incorrect – The Maximum Subcritical Banked Withdrawal Position of 04 is being exceeded. The Reactor is NOT considered shutdown under all conditions. Plausible because one rod full out is allowed by Technical Specification shutdown margin calculations and the other two rods are only withdrawn one notch beyond the Maximum Subcritical Banked Withdrawal Position.
D. Incorrect – The Maximum Subcritical Banked Withdrawal Position of 04 is being exceeded. Plausible because one rod full out is allowed by Technical Specification shutdown margin calculations and the other two rods are only withdrawn one notch beyond the Maximum Subcritical Banked Withdrawal Position.

Technical Reference(s): N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: Bank – NMP2 4/23 NRC #5

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EA2.05
	Importance Rating	3.6

HIGH OFFSITE RADIOACTIVITY RELEASE RATE

Ability to determine and/or interpret the following as they apply to HIGH OFFSITE RADIOACTIVITY RELEASE RATE: Emergency plan implementation

Proposed Question: #18

An unisolable Emergency Condenser tube leak results in the following sequence of events:

Time (hh:mm)	Event
08:14	The offsite release rate exceeds the Emergency Plan “Unusual Event” level.
09:05	The offsite release rate exceeds the Emergency Plan “Alert” level.
10:10	The offsite release rate exceeds the Emergency Plan “Site Area Emergency” level.
11:23	The offsite release rate exceeds the Emergency Plan “General Emergency” level.

Which one of the following is the ***first*** time that entry into N1-EOP-RR, Radioactivity Release Control, is required?

A. 08:14

B. 09:05

C. 10:10

D. 11:23

Proposed Answer: B

Explanation: N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05.

A. Incorrect – N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05. Plausible that entry would be required as soon as Emergency Plan entry occurred.
C. Incorrect – N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05. Plausible that entry would first be required one level higher in the Emergency Plan.
D. Incorrect – N1-EOP-RR entry is required when the offsite release rate is above the Emergency Plan “Alert” Level. The first time that this entry condition is met is 09:05. Plausible because the General Emergency level is used in N1-EOP-RR.

Technical Reference(s): N1-EOP-RR

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRRC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AA2.03
	Importance Rating	3.8

PLANT FIRE ON SITE

Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: Fire alarm

Proposed Question: #19

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

- Main Fire Panel Annunciator 2-1-1-3, TURB. BLDG 261 LOCAL PNL NO.3 FIRE, alarms.
- Fire Zone D-2263, TURB BLDG 289 RB CHARCOAL FILTERS, is in alarm.

Which one of the following describes the associated fire-fighting system and how it is initiated?

A. Manually initiated CO_2

B. Automatically initiated CO_2

C. Manually initiated water deluge

D. Automatically initiated water deluge

Proposed Answer: C

Explanation: This alarm is based on detectors in the RBEVS charcoal filters. These detectors initiate the given alarms, but do not automatically initiate fire suppression. The associated fire suppression system is a water deluge system that must be initiated manually by opening valves.

A. Incorrect – The associated fire suppression system is a water deluge system. Plausible because adding water to a charcoal filter is generally undesirable, such that use of CO_2 would be a likely alternative.
B. Incorrect – The associated fire suppression system is a water deluge system. Plausible because adding water to a charcoal filter is generally undesirable, such that use of CO_2 would be a likely alternative. The associated fire suppression system is manually initiated. Plausible that the same detectors that initiate the given alarms would also automatically initiate suppression, as in many other locations.
D. Incorrect – The associated fire suppression system is manually initiated. Plausible that the same detectors that initiate the given alarms would also automatically initiate suppression, as in many other locations.

Technical Reference(s): ARP 2-1-1-3, N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-8

Question Source: Bank – 2020 NRC #57

Question History: 2020 NRC #57

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AK3.01
	Importance Rating	3.8

GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES

Knowledge of the reasons for the following responses or actions as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Reactor and turbine trip criteria

Proposed Question: #20

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

- A grid disturbance is in progress.
- N1-SOP-33A.3, Major 115 KV Grid Disturbances, is being executed.
- N1-SOP-33B.1, Major 345 KV Grid Disturbances, is being executed.

Which one of the following grid conditions would be reason for an immediate Reactor scram?

A. 115 KV Line voltage of 113 KV

B. 115 KV Line voltage of 122 KV

C. 345 KV Line frequency of 59 Hz

D. 345 KV Line frequency of 62 Hz

Proposed Answer: D

Explanation: N1-SOP-33B.1 contains an override for scramming the Reactor and tripping the Main Turbine. This override is currently applicable since Turbine first stage bowl pressure is NOT in alarm when at 100% power. One condition that requires a scram is 345 KV line frequency variation of greater than ±1.9 Hz from the nominal frequency of 60 Hz. Therefore, a frequency of 62 Hz would be reason for an immediate Reactor scram.

A. Incorrect – N1-SOP-33A.3 does not require a scram based on this voltage. Plausible because voltage is a major parameter monitored in the procedure and the procedure does have various steps that are required based on 115 KV Line voltage.
B. Incorrect – N1-SOP-33A.3 does not require a scram based on this voltage. Plausible because voltage is a major parameter monitored in the procedure and the procedure does have various steps that are required based on 115 KV Line voltage.
C. Incorrect – N1-SOP-33B.1 does not require an immediate scram based on this frequency. Plausible because this frequency would start a 90 minute clock for a Reactor scram.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33B1C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 AK3.01
	Importance Rating	3.7

LOW REACTOR WATER LEVEL

Knowledge of the reasons for the following responses or actions as they apply to LOW REACTOR WATER LEVEL: Recirculation runback

Proposed Question: #21

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

- A Feedwater malfunction occurs.
- The Reactor scrams on low Reactor water level.
- Reactor water level reaches a low of 15" before recovering to the normal band.
- N1-SOP-1, Reactor Scram, is entered.
- N1-SOP-1 requires lowering Recirculation flow.

Which one of the following describes the reason for this Recirculation flow reduction?

Avoid prolonged operation in the Power/Flow Map…

A. Restricted Zone. Operation in this region could cause unseating of fuel bundles.

B. Reactor Internals Protection Region. Operation in this region could cause unseating of fuel bundles.

C. Restricted Zone. Operation in this region could cause core shroud damage in the event of a Main Steam line rupture.

D. Reactor Internals Protection Region. Operation in this region could cause core shroud damage in the event of a Main Steam line rupture.

Proposed Answer: D

Explanation: When operating at 100% power, Recirculation flow is greater than 43 Mlbm/hr. 43 Mlbm/hr is the lower bound of the Reactor Internals Protection Region of the Power/Flow Map. When the scram occurs, Reactor power rapidly lowers, but Recirculation flow stays at the original value. This places operation in the Reactor Internals Protection Region. N1-SOP-1 directs lowering Recirculation flow to 25-43 Mlbm/hr to exit the Reactor Internals Protection Region. The concern with operation in the Reactor Internals Protection Region is the potential for damage to the core shroud in the event of a Main Steam line break.

A. Incorrect – Operation following a scram from 100% power is in the Reactor Internals Protection Region, not the Restricted Zone. Plausible because the Restricted Zone is another undesirable area of the Power/Flow Map. The concern is the potential for damage to the core shroud in the event of a Main Steam line break. Plausible because higher Recirculation Flow causes more upward flow through the core, which would provide an upward force on fuel bundles.
B. Incorrect – The concern is the potential for damage to the core shroud in the event of a Main Steam line break. Plausible because higher Recirculation Flow causes more upward flow through the core, which would provide an upward force on fuel bundles.
C. Incorrect – Operation following a scram from 100% power is in the Reactor Internals Protection Region, not the Restricted Zone. Plausible because the Restricted Zone is another undesirable area of the Power/Flow Map.

Technical Reference(s): N1-SOP-1, P/F Map, N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 AK2.07
	Importance Rating	3.5

HIGH DRYWELL PRESSURE

Knowledge of the relationship between the HIGH DRYWELL PRESSURE and the following systems or components: Component cooling water

Proposed Question: #22

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 rising slowly.
- Drywell pressure is 1.2 psig and rising slowly.
- 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: Bank – 2018 NRC #49

Question History: 2018 NRC #49

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295014 AK1.09
	Importance Rating	4.4

INADVERTENT REACTIVITY ADDITION

Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the INADVERTENT REACTIVITY ADDITION: Reactor power

Proposed Question: #23

The plant is operating at 100% power when the extraction non-return valve to Feedwater Heater 135 trips closed.

Which one of the following describes Reactor power, as indicated by APRMs, two (2) minutes later?

Reactor power will be…

A. 0% due to an automatic scram

B. less than 100%, but no less than 90%

C. unchanged from the initial level

D. greater than 100%

Proposed Answer: D

Explanation: When the extraction non-return valve to Feedwater Heater 135 trips closed, the heater loses heating steam. Feedwater continues to flow through the tube side of the heater, but it receives less heat input. This causes Feedwater temperature into the Reactor to lower. This reduction in temperature, although small, results in a rise in Reactor power. Reactor power will be above 100%, but the power change is too small to challenge the flow-biased scram set point.

A. Incorrect – Reactor power will be above 100%. Plausible because this would be correct for a larger power transient that exceeded the flow-biased scram set point.
B. Incorrect – Reactor power will be above 100%. Plausible because this would be correct for other reactivity transients, such as a lowering Reactor water level.
C. Incorrect – Reactor power will be above 100%. Plausible that the other Feedwater heaters would compensate for or negate the effects of this isolated loss of heating steam.

Technical Reference(s): N1-SOP-16.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP16.1C01 EO-2

Question Source: Bank – 2010 Cert #59

Question History: 2005 NRC Retake #62

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 AA1.04
	Importance Rating	3.6

INADVERTENT CONTAINMENT ISOLATION

Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION: SGTS/FRVS

Proposed Question: #24

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

- Reactor Building Ventilation exhaust fan 11 is operating.
- Reactor Building Ventilation exhaust fan 12 is in standby.
- Reactor Building Emergency Ventilation (RBEVS) is in standby.

Then, Instrument Air is lost to just the following valves:

- 202-32, REACTOR BLDG EXHAUST ISOLATION VALVE 11
- 202-31, REACTOR BLDG EXHAUST ISOLATION VALVE 12

Which one of the following describes the response of RBEVS?

RBEVS…

A. automatically starts and has a suction path to the Reactor Building.

B. automatically starts, but does NOT have a suction path to the Reactor Building.

C. remains in standby. RBEVS can be manually started with a suction path to the Reactor Building.

D. remains in standby. RBEVS CANNOT be manually started with a suction path to the Reactor Building.

Proposed Answer: C

Explanation: The RBVS exhaust air flow path contains two IVs (202-31 and 202-32) in series. These valves fail closed on loss of Instrument Air, isolating the flow path for normal RBVS. The loss of exhaust flow will cause Reactor Building D/P to become less negative and possibly go positive. There is no automatic start of RBEVS in response to either the loss of exhaust flow, exhaust fan trip, or low building D/P. RBEVS takes suction on the RB just upstream of 202-31 and 202-32, therefore these valve closures do not make it unable to restore RB D/P.

A. Incorrect – RBEVS does not automatically start. Plausible that RBEVS would automatically start on the loss of exhaust flow, exhaust fan trip, or low building D/P.
B. Incorrect – RBEVS does not automatically start. Plausible that RBEVS would automatically start on the loss of exhaust flow, exhaust fan trip, or low building D/P.
D. Incorrect – RBEVS can be aligned with a suction path to the Reactor Building. Plausible that the closure of 202-31 and/or 202-32 would block the suction path for RBEVS since the tap off for RBEVS is just upstream of these valves. Also plausible because this would be correct if 202-36 failed closed.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295032 2.1.14
	Importance Rating	3.1

HIGH SECONDARY CONTAINMENT AREA TEMPERATURE

Knowledge of criteria or conditions that require plantwide announcements, such as pump starts, reactor trips, and mode changes

Proposed Question: #25

Given the following events:

(1) The Reactor is scrammed due to a high Reactor Building area temperature

(2) A local area evacuation is required due a chemical spill in Radwaste

(3) The Reactor reaches criticality during a startup

Which one of the following identifies the event that requires a plant-wide announcement and sounding the **Station** Alarm?

A. (1) only

B. (2) only

C. (3) only

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

Proposed Answer: A

Explanation: The Reactor being scrammed due to a high Reactor Building area temperature would require entry into N1-EOP-SC and N1-SOP-1. This requires a plant-wide announcement with the Station Alarm per OP-NM-101-111-0001.

B. Incorrect – Local area evacuation does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and sounding the related Evacuation Alarm.
C. Incorrect – Achieving criticality does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and is a very significant event.
D. Incorrect – Local area evacuation does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and sounding the related Evacuation Alarm. Achieving criticality does not require sounding the Station Alarm. Plausible because this does require a plant-wide announcement and is a very significant event.

Technical Reference(s): OP-NM-201-005, N1-SOP-34, N1-OP-43A, OP-NM-101-111-0001

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-SOP-6.1-CE-02

Question Source: Modified Bank – 2015 Cert #74

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

Proposed Question: #74

Given the following required announcements:

(1) Protected Area Evacuation announcement per CNG-EP-1.01-1009, Assembly, Evacuation, and Accountability.

(2) Medical Emergency announcement per EPIP-EPP-04, Personnel Injury or Illness.

(3) Fire Brigade Activation announcement per EPIP-EPP-28, Fire Fighting.

Which one of the following identifies which of these events require a plant-wide announcement and sounding the **Station** Alarm?

A. Event 1, only

B. Event 2, only

C. Event 3, only

D. Events 1, 2, and 3

Proposed Answer: B

Explanation: All three of these events require a plant announcement with an alarm, but only the Medical Emergency announcement uses the Station Alarm.

A. Plausible – A Protected Area Evacuation announcement requires a plant announcement and alarm, however it uses the Evacuation Alarm, not the Station Alarm.
C. Plausible – The Fire Brigade Activation announcement requires a plant announcement and alarm, however it uses the Fire Alarm, not the Station Alarm.
D. Plausible – All three of these events require a plant announcement with an alarm, but only the Medical Emergency announcement uses the Station Alarm.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295035 EA2.03
	Importance Rating	3.7

SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Lights and alarms

Proposed Question: #26

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): ARP L1-3-4, N1-OP-10, C-18013-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-3

Question Source: Bank - 2018 NRC #63

Question History: 2018 NRC #63

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 A2.01
	Importance Rating	3.6

SHUTDOWN COOLING SYSTEM

Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Recirculation loop high temperature

Proposed Question: #27

A plant shutdown is in progress with the following:

- Shutdown Cooling (SDC) loop 12 is in service.
- Reactor water level is 90".
- Reactor coolant temperature is 340°F.
- Reactor pressure is 115 psig.

Then, a spurious valve closure results in a loss of all Reactor Building Closed Loop Cooling (RBCLC) flow to the SDC heat exchangers.

Which one of the following describes how SDC will respond and the subsequent required control of Reactor Water Cleanup (RWCU) flow, in accordance with N1-SOP-6.1, Loss of Rx Cavity/Decay Heat Removal?

	SDC Response	Control of RWCU Flow
A.	SDC isolation valves (38-01, 38-02, 38-13) close and trip the SDC pump	Maximize
B.	SDC isolation valves (38-01, 38-02, 38-13) close and trip the SDC pump	Minimize
C.	SDC pump trips and SDC isolation valves (38-01, 38-02, 38-13) remain open	Maximize
D.	SDC pump trips and SDC isolation valves (38-01, 38-02, 38-13) remain open	Minimize

Proposed Answer: C

Explanation: With loss of RBCLC flow to the SDC heat exchangers, Reactor coolant temperature and pressure rise. The SDC pumps trip when Reactor coolant temperature reaches 350ºF. The isolation valves remain open. N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal.

A. Incorrect – The isolation valves remain open. Plausible because these valves cannot be opened until Reactor pressure is below 120 psig, but there is no isolation signal in the opposite direction.
B. Incorrect – The isolation valves remain open. Plausible because these valves cannot be opened until Reactor pressure is below 120 psig, but there is no isolation signal in the opposite direction. N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal. Plausible that RWCU flow would be lowered to prevent either thermal stratification or high NRHX outlet temperatures (with Reactor coolant temperature rising).
D. Incorrect – N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal. Plausible that RWCU flow would be lowered to prevent either thermal stratification or high NRHX outlet temperatures (with Reactor coolant temperature rising).

Technical Reference(s): N1-OP-4, N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: Bank – 2020 NRC #1

Question History: 2020 NRC #1

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K1.17
	Importance Rating	2.8

HIGH PRESSURE COOLANT INJECTION SYSTEM

Knowledge of the physical connections and/or cause and effect relationships between the HIGH PRESSURE COOLANT INJECTION SYSTEM and the following systems: Reactor protection system

Proposed Question: #28

The plant is operating at 100% power when RPS bus 11 de-energizes due to a sustained electrical fault.

Which one of the following describes the impact of this failure on HPCI?

A. Both Feedwater flow control valves 11 and 12 shift to HPCI mode.

B. Feedwater flow control valve 11 shifts to HPCI mode, but Feedwater flow control valve 12 does NOT.

C. NEITHER Feedwater flow control valve 11 NOR 12 shift to HPCI mode. Both Feedwater flow control valves 11 and 12 maintain the ability to shift to HPCI mode.

D. NEITHER Feedwater flow control valve 11 NOR 12 shift to HPCI mode. Feedwater flow control valve 11 loses the ability to shift to HPCI mode, but Feedwater flow control valve 12 does NOT.

Proposed Answer: C

Explanation: Feedwater level control system channel 11 is normally powered from RPS Bus 11, but automatically swaps to MG Set 167 on this loss. RPS Buses 11 and 12 supply power to the HPCI initiation logic. HPCI initiation is energize to function. Either relay 11K47 (RPS Bus 11) or 12K47 (RPS Bus 12) must energize for HPCI initiation. Only one of the two power supplies is required. With only a loss of RPS Bus 11, no HPCI initiation is received and both Feedwater flow control valves (FCVs) maintain the ability to shift to HPCI mode with just power from RPS Bus 12.

A. Incorrect – Neither FCV goes into HPCI mode. Plausible because some systems are de-energize to function.
B. Incorrect – Neither FCV goes into HPCI mode. Plausible because some systems are de-energize to function. Also plausible because just one relay is necessary to 11K47 (RPS Bus 11) or 12K47 (RPS Bus 12) initiate HPCI for both FCVs.
D. Incorrect – FCV 11 maintains the ability to shift to HPCI mode. Plausible because 11K47 cannot energize. Also plausible because the normal power supply to Feedwater level control channel 11 is RPS bus 11.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	207000 K6.07
	Importance Rating	3.5

ISOLATION (EMERGENCY) CONDENSER

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the ISOLATION (EMERGENCY) CONDENSER: AC electrical distribution

Proposed Question: #29

The plant is operating at 100% power when Powerboard 16B de-energizes due to a sustained electrical fault.

Which one of the following describes the impact of this failure on the Emergency Condensers (ECs)?

A. A Condensate Return Valve fails open

B. A Condensate Return Valve fails closed

C. A Steam Isolation Valve loses motive power while in the open position

D. A Steam Isolation Valve loses motive power while in the closed position

Proposed Answer: C

Explanation: Powerboard 16B supplies power to Powerboard 161B, which is the power supply to 39-10R, EC STM ISOLATION VALVE 121. This valve is open while in the standby lineup.

A. Incorrect – The Condensate Return Valves are unaffected. Plausible because they do fail open on loss of air or loss of both DC solenoids.
B. Incorrect – The Condensate Return Valves are unaffected. Plausible because they do fail open on loss of air or loss of both DC solenoids. Plausible that they would fail closed to prevent spurious system initiation.
D. Incorrect – With the plant at 100% power, ECs are in a standby lineup and the Steam Isolation Valves are open. Plausible because the Condensate Return Valves are closed.

Technical Reference(s): C-19437-C Sheet 6, C-18017-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K3.01
	Importance Rating	4.3

LOW PRESSURE CORE SPRAY SYSTEM

Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on the following systems or system parameters: Reactor water level

Proposed Question: #30

A loss of coolant accident occurs with the following:

- Reactor water level is -50” and lowering.
- Reactor pressure is 300 psig and lowering.
- Core Spray pump 111 is tagged out of service for motor replacement.
- Core Spray pump 121 has tripped and CANNOT be restarted.
- Powerboard 171B is de-energized due to a sustained electrical fault.

Which one of the following describes the availability of Core Spray loops 11 and 12 for Reactor water level control?

	Injection with Core Spray Loop 11	Injection with Core Spray Loop 12
A.	Available	Available
B.	Available	NOT available
C.	NOT available	Available
D.	NOT available	NOT available

Proposed Answer: A

Explanation: Powerboard 171B supplies power to Core Spray discharge inside IVs 40-10 (loop 11) and 40-09 (loop 12). Powerboard 161B supplies power to Core Spray discharge inside IVs 40-01 (loop 11) and 40-11 (loop 12). All of these valves are inside the Drywell and normally closed, such that they need electrical power to open and inject to the Reactor. Core Spray pump 111 is in loop 11 and Core Spray pump 121 is in loop 12. Loop 11 is still capable of injecting to the Reactor with pump 112 through inside IV 40-11. Loop 12 is still capable of injecting to the Reactor with pump 122 through inside IV 40-01.

B. Incorrect – Both Core Spray loops are still available for Reactor water level control. Plausible that the loss of Powerboard 171B, combined with the loss of Core Spray pump 111, would result in inability to inject with Core Spray loop 11.
C. Incorrect – Both Core Spray loops are still available for Reactor water level control. Plausible that the loss of Powerboard 171B, combined with the loss of Core Spray pump 111, would result in inability to inject with Core Spray loop 12.
D. Incorrect – Both Core Spray loops are still available for Reactor water level control. Plausible that the loss of Powerboard 171B, combined with the loss of Core Spray pumps 111 and 121, would result in inability to inject with both Core Spray loops.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K5.02
	Importance Rating	3.4

LOW PRESSURE CORE SPRAY SYSTEM

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the LOW PRESSURE CORE SPRAY SYSTEM: Abnormal differential pressure indication (leak detection)

Proposed Question: #31

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

- Annunciator K3-4-1, CORE SPRAY SPARGER DIFF PRESS, alarms.
- The alarm is determined to be due to a valid line break on Core Spray 11.

Which one of the following describes the location of the Core Spray line break?

A. Between the pumps and the discharge check valve (40-13)

B. Between the discharge check valve (40-13) and the discharge inside isolation valves (40-10 and 40-11)

C. Between the discharge inside isolation valves (40-10 and 40-11) and the core shroud

D. Inside the core shroud

Proposed Answer: C

Explanation: This alarm is caused by a change in pressure due to a leak between the injection line check valve and the core shroud.

A. Incorrect – This alarm would not detect a break in this location because closed valves prevent a change in pressure here from being detected. Plausible because this leak location is possible and would disrupt Core Spray flow in the same way as a leak in the correct location.
B. Incorrect – This alarm would not detect a break in this location because closed valves prevent a change in pressure here from being detected. Plausible because this leak location is possible and would disrupt Core Spray flow in the same way as a leak in the correct location.
D. Incorrect – This alarm would not detect a break in this location because there would be no pressure change (this section of piping is already pressurized to the same pressure as the surrounding environment). Plausible because this section of piping is directly connected to the piping that is monitored by the leak detection system.

Technical Reference(s): ARP K3-4-1, C-18007-C, C-18015-C, N1-209001-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-RBO-5

Question Source: Bank – JAF 18-1 NRC #32

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A1.05
	Importance Rating	2.9

STANDBY LIQUID CONTROL SYSTEM

Ability to predict and/or monitor changes in parameters associated with operation of the STANDBY LIQUID CONTROL SYSTEM including: Pump amps

Proposed Question: #32

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 0 psig.
- Liquid Poison pump 11 amps indicate upscale.
- The LIQUID POISON EXPL VALVE 11 CONTINUITY light is extinguished.
- The LIQUID POISON EXPL VALVE 12 CONTINUITY light is extinguished.
- NO change in Liquid Poison tank level has been observed.

Which one of the following describes the malfunction indicated by these conditions?

A. Seized pump rotor

B. Sheared pump shaft

C. Relief valve has lifted

D. Explosive valves failed to open

Proposed Answer: A

Explanation: The given indications show that the Liquid Poison pump 11 motor is drawing very high amps while the pump is not producing pressure. This indicates that the pump rotor is not spinning (locked or seized rotor).

B. Incorrect – A sheared pump shaft would result in lower than normal amps. Plausible because this would result in low discharge pressure and lack of lowering in tank level.
C. Incorrect – A lifting relief valve would not result in upscale amps or such low discharge pressure. Plausible because this would result in lack of lowering tank level and would have some effect on discharge pressure and/or pump amps.
D. Incorrect – Failure of explosive valves to open would not result in low discharge pressure or such high amps. Plausible because this would result in lack of lowering tank level and would have some effect on discharge pressure and/or pump amps.

Technical Reference(s): N1-OP-12, BC05Ir4_Motors

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO-6

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 A3.09
	Importance Rating	4.3

REACTOR PROTECTION SYSTEM

Ability to monitor automatic operation of the REACTOR PROTECTION SYSTEM including: System actuation

Proposed Question: #33

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All IRMs indicate approximately 30 out of 125 on Range 8.
- APRMs indicate 3%.
- Reactor pressure is 550 psig.

Then, the MSIVs close due to a spurious isolation signal, resulting in the following conditions:

- IRMs spike to 100 out of 125 on Range 8.
- APRMs spike to 7%.
- Reactor pressure spikes to 650 psig.

Which one of the following describes the RPS signal that initiates a Reactor scram, if any?

A. IRM Hi-Hi

B. APRM Hi-Hi

C. MSIV Closure

D. NO Reactor scram occurs

Proposed Answer: C

Explanation: IRMs do not cause a Reactor scram on a Hi-Hi signal because they remain below 117.5 out of 125 on Range 8. APRMs do not spike high enough to cause a Reactor scram given the startup conditions. MSIV Closure does cause a direct scram due to Reactor pressure being above 600 psig with MSIVs closed.

A. Incorrect – MSIV Closure initiates the Reactor scram. Plausible because this would be correct if IRMs exceeded 117.5. Also plausible because IRMs spike well above the normal control band.
B. Incorrect – MSIV Closure initiates the Reactor scram. Plausible because APRMs spike high enough to clear the downscales.
D. Incorrect – MSIV Closure initiates the Reactor scram. Plausible because this would be correct if Reactor pressure remained below 600 psig.

Technical Reference(s): ARPs F3-1-1, F3-3-1, F3-3-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: Modified Bank – 2020 NRC #43

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Proposed Question: #43

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All IRMs indicate approximately 30 out of 125 on Range 8
- APRMs indicate 3%
- Reactor pressure is 450 psig

Then, the MSIVs close due to a spurious isolation signal, resulting in the following conditions:

- IRMs spike to 120 out of 125 on Range 8
- APRMs spike to 9%
- Reactor pressure spikes to 550 psig

Which one of the following describes the RPS signal that initiates a Reactor scram, if any?

A. IRM Hi-Hi

B. APRM Hi-Hi

C. MSIV Closure

D. NO Reactor scram occurs

Proposed Answer: A

Explanation: IRMs cause a Reactor scram on a Hi-Hi signal due to exceeding 117.5 out of 125 on Range 8. APRMs do not spike high enough to cause a Reactor scram given the startup conditions. MSIV Closure does not cause a direct scram due to Reactor pressure being below 600 psig.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K3.01
	Importance Rating	4.2

INTERMEDIATE RANGE MONITOR SYSTEM

Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR SYSTEM will have on the following systems or system parameters: RPS

Proposed Question: #34

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All Intermediate Range Monitors (IRM) detectors are fully inserted and indicating mid-scale on range 2.

Then, IRM 15 fails upscale.

Which one of the following describes the response of the Reactor Protection System?

A. NO half scram is received

B. A half scram is received on RPS 11, only

C. A half scram is received on RPS 12, only

D. A full scram is received

Proposed Answer: C

Explanation: The upscale failure results in an upscale trip from IRM 15. The RPS logic is arranged such that an upscale trip from a single IRM causes a half scram. IRMs 11-14 input to RPS 11 and IRMs 15-18 input to RPS 12. Therefore, a half scram is received on RPS 12 only.

A. Incorrect – A half scram is received on RPS 12, only. Plausible that it would take more than 1 failed IRM to result in a half scram (there are 4 IRM inputs to RPS 11 and 4 IRM inputs to RPS 12). Also plausible because this would be correct if the Reactor Mode Switch was in RUN.
B. Incorrect – A half scram is received on RPS 12, only. Plausible that the odd numbered IRMs would all input to RPS 11.
D. Incorrect – A half scram is received on RPS 12, only. Plausible because this would be correct if IRM 11, 12, 13, or 14 also failed upscale or INOP.

Technical Reference(s): N1-OP-38B, ARP F2-3-6, C-19859-C Sheets 2 and 5

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:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A1.01
	Importance Rating	3.3

SOURCE RANGE MONITOR SYSTEM

Ability to predict and/or monitor changes in parameters associated with operation of the SOURCE RANGE MONITOR SYSTEM including: Detector position

Proposed Question: #35

A plant startup is in progress with the following:

- All IRMs are on Range 2.
- SRM 11 reads 85 cps with the detector fully inserted.
- SRM 12 reads 110 cps with the detector partially withdrawn.
- SRM 13 reads 95 cps with the detector partially withdrawn.
- SRM 14 reads 90 cps with the detector fully inserted.
- A control rod block is present.

Which one of the following identifies all of the SRMs that are causing the control rod block?

A. SRM 11 only

B. SRM 13 only

C. SRM 11 and 14

D. SRM 11, 13, and 14

Proposed Answer: B

Explanation: SRM indication less than 100 cps causes a rod block if the associated detector is not fully inserted. Therefore SRM 13 is causing a rod block based on not being fully inserted and <100 cps.

A. Incorrect – SRM 11 is not causing a rod block because it is fully inserted. Plausible because it is the lowest reading SRM and would be causing a rod block if it was not fully inserted.
C. Incorrect – SRMs 11 and 14 are not causing a rod block because they are fully inserted. Plausible because they are the lowest reading SRMs and would be causing a rod block if they were not fully inserted.
D. Incorrect – SRMs 11 and 14 are not causing a rod block because they are fully inserted. Plausible because they are the lowest reading SRMs and would be causing a rod block if they were not fully inserted.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Bank – 2013 NRC #37

Question History: 2013 NRC #37

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 A1.01
	Importance Rating	3.3

DC Electrical Distribution

Ability to predict and/or monitor changes in parameters associated with operation of the DC Electrical Distribution, including: Battery charging/discharging rate

Proposed Question: #36

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

- Static Battery Chargers 161A and 171B are in service.
- Powerboard 16B is inadvertently de-energized, then re-energized.

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

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

Proposed Answer: A

Explanation: Static Battery Charger 161A receives power from Powerboard 16B and charges Battery 11. Upon loss of Powerboard 16B, Battery 11 begins discharging. Approximately 90 seconds after Powerboard 16B is re-energized, Static Battery Charger 161A will automatically align itself to Battery 11 and restore the normal float charge. Static Battery Charger 171B receives power from Powerboard 17B and charges Battery 12. Therefore, Battery 12 is in the normal float charge throughout the transient.

B. Incorrect – Battery 12 is on the normal float charge throughout the transient. Plausible that Powerboard 16B would power one battery charger for each battery for added redundance/flexibility.
C. Incorrect – By 2 minutes, Battery 11 is back to charging. Plausible because Battery 11 was discharging. Also plausible that manual action would be required to restore charging.
D. Incorrect – By 2 minutes, Battery 11 is back to charging. Plausible because Battery 11 was discharging. Also plausible that manual action would be required to restore charging. Battery 12 is on the normal float charge throughout the transient. Plausible that Powerboard 16B would power one battery charger for each battery for added redundance/flexibility.

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: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K3.07
	Importance Rating	3.7

AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR

Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR will have on the following systems or system parameters: Rod block monitor system

Proposed Question: #37

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

- Rod line is 102%.
- APRM FLOW UNIT 12 signal fails downscale.
- Actual core flow remains constant.

Which one of the following describes the plant response?

A. NO rod block or half scram is received.

B. A rod block is received, but NO half scram is received.

C. A rod block and a half scram are received. A full scram is NOT received.

D. A rod block and full scram are received.

Proposed Answer: C

Explanation: APRM 12 flow unit provides the flow input to ARPMs 15-18, which all input to RPS channel 12 only. With their flow signal failed low, 100% Reactor power exceeds both their flow-biased rod block and scram setpoints. Therefore, both a rod block and a half scram on RPS channel 12 occur. A flow comparator rod block also occurs because the difference between flow channels 11 and 12 exceeds 6.8%.

A. Incorrect – A rod block and half scram occurs. Plausible because the flow-biased rod block and half scram would not be received for an upscale failure of the flow unit.
B. Incorrect – A half scram also occurs. Plausible because this would be correct for an upscale failure of the flow unit.
D. Incorrect – A full scram does not occur. Plausible that this flow unit would input to both some RPS 11 and some RPS 12 APRMs.

Note: The question meets the K/A because NMP1 does not have the same Rod Block Monitor (RBM) system as other BWRs, however it does have a design feature in the Control Room that is specifically called the "Rod Block Monitor". The Rod Block Monitor at NMP1 does just as its name implies – it monitors various rod blocks, including those from APRMs and Flow Comparators.

Technical Reference(s): N1-OP-38C, 5-loop Power/Flow Map, ARPs F4-1-8, F3-1-1, F3-2-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-11

Question Source: Bank – 2013 NRC #38

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 A4.01
	Importance Rating	4.5

AUTOMATIC DEPRESSURIZATION SYSTEM

Ability to manually operate and/or monitor the AUTOMATIC DEPRESSURIZATION SYSTEM in the control room: ADS valves

Proposed Question: #38

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

- Reactor water level is -20" and lowering slowly.
- Reactor pressure is 650 psig and lowering slowly.
- Drywell pressure is 10 psig and rising slowly.
- Annunciator F2-3-1, MAIN STM LINE AUTOMATIC DE-PRESS TIMING, came into alarm 30 seconds ago.
- The Unit Supervisor has directed bypassing ADS.
- A Reactor Operator has placed the Channel 12 ADS Inhibit keylock bypass switch in BYPASS.
- The Channel 11 ADS Inhibit keylock bypass switch CANNOT be placed in BYPASS due to a mechanical malfunction.

Which one of the following describes the status of ERVs two (2) minutes later?

A. NO ERVs are open

B. The three (3) primary ERVs are open, only

C. The three (3) secondary ERVs are open, only

D. All six (6) ERVs are open

Proposed Answer: B

Explanation: ADS channel 12 has successfully been bypassed, but channel 11 has not. ADS only needs one of these two channels to function to cause the three (3) primary ERVs to open after 111 seconds. Since the primary ERVs open, the three (3) backup ERVs remain closed.

A. Incorrect – The three (3) primary ERVs open. Plausible that both channels would be required to open ERVs. Also plausible because this would be correct prior to 111 seconds.
C. Incorrect – The three (3) primary ERVs open. Plausible because there are some malfunctions that would cause the primary ERVs to not open, in which case the backup ERVs open.
D. Incorrect – Only three (3) ERVs open. Plausible because both the primary and backup ERVs are capable of opening and EOPs require >3 ERVs to open for an RPV Blowdown.

Technical Reference(s): C-19859-C sheets 18, 18a, 24, 24a, 25, 25a, 26, 26a

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 A2.13
	Importance Rating	3.3

PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Leak detection system failures

Proposed Question: #39

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

- A failure of multiple Main Steam Line temperature detectors occurs.
- The failures result in an isolation signal closing the MSIVs.
- The isolation signal is sealed in and CANNOT be cleared.
- N1-SOP-1, Reactor Scram, and N1-EOP-RPV, RPV Control, are entered.
- You are directed to perform a Reactor cooldown.

Assuming the isolation signal remains sealed in, which one of the following describes the ability to:

- use Emergency Condensers for the cooldown

  and

- place Shutdown Cooling in service once the Shutdown Cooling pressure interlock clears?

	Emergency Condensers	Shutdown Cooling
A.	Available for cooldown	Will be available
B.	Available for cooldown	Will NOT be available
C.	NOT available for cooldown	Will be available
D.	NOT available for cooldown	Will NOT be available

Proposed Answer: A

Explanation: Failure of the Main Steam leak detection system (Main Steam tunnel temperature detectors) has caused an isolation of the MSIVs. This is a Vessel isolation. With the MSIVs closed, TBVs are unavailable for pressure control. This Vessel isolation also causes Emergency Condenser vents and drains to close. However, this does not prevent use of the Emergency Condensers for cooldown. Additionally, this Vessel isolation does not isolate Shutdown Cooling, therefore Shutdown Cooling will be available to be placed in service once the pressure interlock clears even if the Main Steam Line temperature isolation signal remains sealed it.

B. Incorrect – Shutdown Cooling will be available to be placed in service once the pressure interlock clears. Plausible because the Shutdown Cooling system isolates on a Vessel isolation signal from Lo-Lo Reactor water level, just not this Vessel isolation signal.
C. Incorrect – Emergency Condensers are available for cooldown. Plausible because some Emergency Condenser valves do close on this isolation signal. Also plausible because Emergency Condensers do isolate on their own, separate leak detection isolation signal.
D. Incorrect – Emergency Condensers are available for cooldown. Plausible because some Emergency Condenser valves do close on this isolation signal. Also plausible because Emergency Condensers do isolate on their own, separate leak detection isolation signal. Shutdown Cooling will be available to be placed in service once the pressure interlock clears. Plausible because the Shutdown Cooling system isolates on a Vessel isolation signal from Lo-Lo Reactor water level, just not this Vessel isolation signal.

Technical Reference(s): N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K6.09
	Importance Rating	3.2

PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUTOFF: Plant pneumatic systems

Proposed Question: #40

Which one of the following describes the effect of a complete loss of Instrument Air on the Main Steam Isolation Valves (MSIVs)?

A. The inboard MSIVs fail open

B. The inboard MSIVs fail closed

C. The outboard MSIVs fail open

D. The outboard MSIVs fail closed

Proposed Answer: D

Explanation: The outboard MSIVs are operated by Instrument Air. They fail closed on loss of Instrument Air.

A. Incorrect – The inboard MSIVs are not operated by Instrument Air. Plausible because the outboard MSIVs are operated by Instrument Air.
B. Incorrect – The inboard MSIVs are not operated by Instrument Air. Plausible because the outboard MSIVs are operated by Instrument Air. MSIVs fail closed on loss of Instrument Air. Plausible because many air-operated valves fail open on loss of the air supply.
C. Incorrect – MSIVs fail closed on loss of Instrument Air. Plausible because many air-operated valves fail open on loss of the air supply.

Technical Reference(s): N1-SOP-20.1, C-18002-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-8

Question Source: Bank - 2015 Cert #32

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 A3.01
	Importance Rating	4.1

SAFETY RELIEF VALVES

Ability to monitor automatic operation of the SAFETY RELIEF VALVES including: SRV operation after ADS actuation

Proposed Question: #41

Given the following indications for ERV 111:



Which one of the following describes the status of ERV 111 indications?

ERV 111…

A. has normal indications for a valve that cycled open and then closed

B. indicates open by acoustic monitor but closed by solenoid position

C. indicates open by solenoid position but closed by acoustic monitor

D. indicates open by both acoustic monitor and solenoid position

Proposed Answer: C

Explanation: The red light above the ERV control switch is illuminated, which indicates that the valve is open by solenoid position. The red light to the right of the ERV control switch is extinguished, which indicates that the valve is closed by acoustic monitor. These conditions would occur if an ERV received an open signal, such as from ADS, with Reactor pressure too low to open the valve.

A. Incorrect – The ERV indicates open by solenoid position. Plausible because this would be correct with different light indications.
B. Incorrect – The ERV indicates open by solenoid position and closed by acoustic monitor. Plausible because this would be correct with different light indications.
D. Incorrect – The ERV indicates open by solenoid position and closed by acoustic monitor. Plausible because this would be correct with different light indications.

Technical Reference(s): N1-OP-1

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: N1-218000-RBO-5

Question Source: Modified Bank - JAF 20-1 NRC #15

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Proposed Question: #15

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



Which one of the following describes the status of the SRV A indications?

SRV A...

A. has normal indications for a valve that cycled open then closed.

B. indicates open by acoustic monitor but closed by solenoid position.

C. indicates open by solenoid position but closed by acoustic monitor.

D. indicates open by both acoustic monitor and solenoid position.

Proposed Answer: B

Explanation: The SRV A white light being illuminated indicates the valve is open as sensed by the acoustic monitor. However, the green light being illuminated and the red light being extinguished indicate that the SRV A is not open based on solenoid position.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 2.1.45
	Importance Rating	4.3

SAFETY RELIEF VALVES

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

Proposed Question: #42

The plant is operating at 23% power during a startup, with the following conditions:

- Five Turbine Bypass Valves are open.
- The crew is preparing to roll the Turbine.
- Reactor pressure is 930 psig and stable.

Then, ERV 111 inadvertently opens and remains open.

Which one of the following describes:

- the ability of the Main Steam Line flow instruments to detect the flow through ERV 111

  and

- the response of Reactor pressure?

	The Main Steam Line flow instruments…	Reactor pressure…
A.	detect the steam flow through ERV 111	stabilizes slightly below 930 psig
B.	do NOT detect the steam flow through ERV 111	stabilizes slightly below 930 psig
C.	detect the steam flow through ERV 111	continually lowers as long as the ERV is open
D.	do NOT detect the steam flow through ERV 111	continually lowers as long as the ERV is open

Proposed Answer: A

Explanation: The ERVs tap off of the main steam lines after the flow restrictors. The flow restrictors are used to measure main steam line flow. Since the ERV flow passes through the flow restrictor, the ERV flow is measured as part of the total main steam line flow. The opening of the ERV lowers pressure at the turbine. EPR sensed pressure lowers, therefore the bypass valves are throttled closed. This action stabilizes pressure slightly below the initial value of 930 psig while ERV 111 is open.

B. Incorrect – ERV flow is measured as part of the total main steam line flow. Plausible if the ERV tapped off the main steam line upstream of the flow restrictor instead of downstream.
C. Incorrect – Reactor pressure stabilizes slightly below the initial value of 930 psig. Plausible because this would be correct if EHC did not sense the problem and adjust. Also plausible because this would be correct if the Reactor was shutdown.
D. Incorrect – ERV flow is measured as part of the total main steam line flow. Plausible if the ERV tapped off the main steam line upstream of the flow restrictor instead of downstream. Reactor pressure stabilizes slightly below the initial value of 930 psig. Plausible because this would be correct if EHC did not sense the problem and adjust. Also plausible because this would be correct if the Reactor was shutdown.

Technical Reference(s): C-18002-C Sheet 1, N1-SOP-1.4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-8

Question Source: Bank – 2017 Cert #24

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K2.03
	Importance Rating	3.1

REACTOR WATER LEVEL CONTROL SYSTEM

Knowledge of electrical power supplies to the following: Feedwater controllers

Proposed Question: #43

Which one of the following describes the normal power supply(ies) to the Feedwater Level Control RTP chassis?

A. I&C Bus 130

B. Computer Panel 167

C. RPS Buses 11 and 12

D. Battery Boards 11 and 12

Proposed Answer: C

Explanation: The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12.

A. Incorrect – The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12. Plausible because I&C Bus 130 supplies power to numerous instruments and controls.
B. Incorrect – The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12. Plausible because Computer Panel 167 supplies backup power to some Feedwater level control components.
D. Incorrect – The Feedwater Level Control RTP chassis receive normal power from RPS buses 11 and 12. Plausible because Battery Boards 11 and 12 supply power to many instruments and controls.

Technical Reference(s): N1-259002-RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	510000 K1.13
	Importance Rating	2.7

SERVICE WATER SYSTEM

Knowledge of the physical connections and/or cause and effect relationships between the Service Water System and the following systems: Feedwater system

Proposed Question: #44

The plant has been scrammed with the following:

- A loss of both Service Water pumps has occurred.
- Both Emergency Service Water (ESW) pumps have been started.

Which one of the following describes whether ESW is supplying cooling to Feedwater pumps 11 and 12?

ESW is…

A. NOT supplying cooling to Feedwater pumps 11 and 12

B. directly supplying cooling to Feedwater pumps 11 and 12

C. indirectly supplying cooling to Feedwater pumps 11 and 12 through TBCLC

D. indirectly supplying cooling to Feedwater pumps 11 and 12 through RBCLC

Proposed Answer: D

Explanation: Feedwater pumps 11 and 12 are directly cooled by RBCLC. RBCLC is normally cooled by Service Water. With Service Water pumps unavailable, ESW pumps are started to supply an alternate source of cooling water to RBCLC. Therefore, ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC.

A. Incorrect – ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC. Plausible because ESW is NOT supplying cooling to Feedwater pump 13 (it is cooled by TBCLC, which normally is cooled by Service Water, but does not get ESW flow).
B. Incorrect – ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC. Plausible because ESW flow is different than normal Service Water flow. Also plausible because Service Water does directly cool some components (including Feedwater pump area cooler).
C. Incorrect – ESW is indirectly cooling Feedwater pumps 11 and 12 through RBCLC. Plausible because some Feedwater components are normally cooled via Service Water through TBCLC (including Feedwater pump 13).

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-276000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 K2.04
	Importance Rating	2.9

STANDBY GAS TREATMENT SYSTEM

Knowledge of electrical power supplies to the following: Standby gas treatment system heaters

Proposed Question: #45

Which one of the following is the power supply to the Reactor Building Emergency Ventilation System (RBEVS) 10 KW heater?

A. Powerboard H16

B. Powerboard 151

C. Powerboard 153

D. Powerboard 167

Proposed Answer: D

Explanation: The power supply to the RBEVS 10 KW heater is Powerboard 167.

A. Incorrect – The power supply to the RBEVS 10 KW heater is Powerboard 167. Plausible because this is the power supply to multiple Reactor Building duct heaters.
B. Incorrect – The power supply to the RBEVS 10 KW heater is Powerboard 167. Plausible because this is the power supply to Reactor Building Ventilation supply fans.
C. Incorrect – The power supply to the RBEVS 10 KW heater is Powerboard 167. Plausible because this is the power supply to Reactor Building Ventilation exhaust fans.

Technical Reference(s): N1-261000-RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A3.03
	Importance Rating	3.8

AC ELECTRICAL DISTRIBUTION

Ability to monitor automatic operation of the AC ELECTRICAL DISTRIBUTION including: Load shedding

Proposed Question: #46

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

- The following pumps are running:
  - CRD pump 12
  - Liquid Poison pump 12
  - RBCLC pump 12

Then, a loss of offsite power occurs. Both Emergency Diesel Generators (EDGs) start and re-energize their respective powerboards.

Which one of the following describes the status of CRD pump 12, Liquid Poison pump 12, and RBCLC pump 12 one (1) minute after the EDGs re-energize their respective powerboards?

A. All three of these pumps are running.

B. CRD pump 12 and Liquid Poison pump 12 are running. RBCLC pump 12 is shut down.

C. Liquid Poison pump 12 and RBCLC pump 12 are running. CRD pump 12 is shut down.

D. All three of these pumps are shut down.

Proposed Answer: B

Explanation: 86-16 and 86-17 actuate on the loss of power to PBs 102 and 103 to initiate load shedding to reduce the loading on the diesels when they start and energize their busses. Liquid Poison pumps are NOT stripped, therefore Liquid Poison pump 12 is operating. CRD pumps are both started, even if initially in standby, therefore both CRD pumps 11 and 12 are operating. RBCLC pumps are stripped, therefore RBCLC pump 12 is NOT operating.

A. Incorrect – RBCLC pump 12 is NOT operating. Plausible because it was initially operating and its cooling function is important. An RBCLC pump is the first load restarted in N1-SOP-33A.1 while restoring from this transient.
C. Incorrect – RBCLC pump 12 is NOT operating. Plausible because it was initially operating and its cooling function is important. An RBCLC pump is the first load restarted in N1-SOP-33A.1 while restoring from this transient. CRD pump 12 is operating. Plausible because CRD pump 12 de-energizes during the transient before the EDGs re-energize their buses and load shedding does occur.
D. Incorrect – Liquid Poison pump 12 and CRD pump 12 are operating. Plausible because they do de-energize during the transient before the EDGs re-energize their buses and load shedding does occur.

Technical Reference(s): N1-OP-30, C-19409-C Sheet 1b

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-11

Question Source: Bank – 2017 Cert #25

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A4.01
	Importance Rating	3.7

AC ELECTRICAL DISTRIBUTION

Ability to manually operate and/or monitor the AC ELECTRICAL DISTRIBUTION in the control room: Breakers and disconnects

Proposed Question: #47

The plant is operating at 100% power when a Main Turbine trip occurs.

Which one of the following describes the resulting status of the Main Generator output breakers, R915 and R925, and disconnect MOD-18 three (3) minutes later?

	Main Generator Output Breakers R915 and R925	Disconnect MOD-18
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: A

Explanation: Both Main Generator output breakers automatically open on the Main Turbine trip. Both output breakers remain open until manual action is taken to reclose them. Main Generator disconnect MOD-18 also automatically opens on the Main Turbine trip and remains open until manual action is taken.

B. Incorrect – MOD-18 automatically opens and remains open. Plausible that it would remain closed since R915 and R925 being open separates the Main Generator from the grid.
C. Incorrect – R915 and R925 automatically open and remain open. Plausible that they would reclose to re-establish the preferred 345 KV grid alignment. Also plausible because N1-OP-31 contains direction to manually reclose these breakers to re-establish the preferred 345 KV grid alignment.
D. Incorrect – R915, R925, and MOD-18 automatically open and remain open. Plausible because the initiating signal is a Main Turbine trip, not a Main Generator lockout. Also plausible because it is desired to realign the system to re-establish the preferred 345 KV grid alignment.

Technical Reference(s): N1-SOP-31.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001

Question Source: Bank - JAF 18-1 NRC #46

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K4.01
	Importance Rating	3.5

UNINTERRUPTABLE POWER SUPPLY (AC/DC)

Knowledge of UNINTERRUPTABLE POWER SUPPLY (AC/DC) design features and/or interlocks that provide for the following: Transfer of power supplies

Proposed Question: #48

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

- Lines 1 and 4 de-energize.
- EDGs 102 and 103 re-energize their respective powerboards.

Which one of the following describes the response of UPS 172 to this electrical transient?

UPS 172…

A. loads are automatically transferred to the bypass transformer. UPS 172 loads remain on this source until manually re-transferred.

B. loads are automatically transferred to the bypass transformer. UPS 172 loads automatically transfer back after power restoration.

C. inverter is automatically supplied power from Battery Board 12. UPS 172 inverter remains supplied from this source until manually re-transferred.

D. inverter is automatically supplied power from Battery Board 12. UPS 172 inverter power supply automatically transfers back after power restoration.

Proposed Answer: D

Explanation: Loss of Lines 1 and 4 causes Powerboard 17B to de-energize. Powerboard 17B is the normal power supply for UPS 172, and also the power supply for the UPS 172 bypass transformer. With the loss of the normal AC input to UPS 172, Battery Board 12 automatically supplies power to the inverter. When normal AC power is restored (automatically after EDG start and synchronization), the UPS 172 inverter automatically swaps back to the normal power source.

A. Incorrect – UPS 172 inverter automatically receives power from Battery Board 12. Plausible because the bypass transformer is another alternate power supply that will power loads in some situations. UPS 172 automatically transfers back to normal power supply. Plausible that the transfer back would take manual action to ensure the AC source is stable before the transfer.
B. Incorrect – UPS 172 inverter automatically receives power from Battery Board 12. Plausible because the bypass transformer is another alternate power supply that will power loads in some situations.
C. Incorrect – UPS 172 automatically transfers back to normal power supply. Plausible that the transfer back would take manual action to ensure the AC source is stable before the transfer.

Technical Reference(s): N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-5

Question Source: Bank – 2013 NRC #47

Question History: 2013 NRC #47

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K4.03
	Importance Rating	2.9

DC ELECTRICAL DISTRIBUTION

Knowledge of DC ELECTRICAL DISTRIBUTION design features and/or interlocks that provide for the following: Ground detection

Proposed Question: #49

Which one of the following describes the ground detection for Battery 14?

A. Voltmeter located on Battery Board 14

B. Voltmeter located in the Main Control Room

C. Indicating lights located on Battery Board 14

D. Indicating lights located in the Main Control Room

Proposed Answer: A

Explanation: The ground detector for Battery 14 is a voltmeter located on Battery Board 14.

B. Incorrect – There is no ground detection indication for Battery 14 in the Main Control Room. Plausible because there is for Batteries 11 and 12.
C. Incorrect – The indication is a voltmeter. Plausible because this is another style of ground detector and there are many other plant indications that use indicating lights.
D. Incorrect – The indication is a voltmeter. Plausible because this is another style of ground detector and there are many other plant indications that use indicating lights. There is no ground detection indication for Battery 14 in the Main Control Room. Plausible because there is for Batteries 11 and 12.

Technical Reference(s): ARP A3-2-2, N1-OP-47A, N1-263000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 A4.07
	Importance Rating	3.4

EMERGENCY GENERATORS (DIESEL/JET)

Ability to manually operate and/or monitor the EMERGENCY GENERATORS (DIESEL/JET) in the control room: Transfer of control to local/remote panels

Proposed Question: #50

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

- Annunciator A4-2-6, DSL GEN 102-103 ISO. SW. LOCAL R1012-R1013 SW. LOCKOUT, alarms.
- Investigation reveals that the EDG 102 selector switch is in LOCAL START at the EDG 102 engine control panel.

While this switch is in this position, which one of the following describes:

(1) the ability to manually start and load EDG 102 in the Main Control Room

and

(2) whether EDG 102 will start and load on a valid undervoltage signal?

	(1) Can EDG 102 be manually started and loaded in the Main Control Room?	(2) Will EDG 102 automatically start and load on a valid undervoltage signal?
A.	Yes	Yes
B.	Yes	No
C.	No	Yes
D.	No	No

Proposed Answer: D

Explanation: Placing the EDG 102 selector switch in LOCAL START at the EDG 102 engine control panel disables the ability to start EDG 102 in the Main Control Room. This switch position also disables field flashing for EDG 102. This prevents the EDG from being able to be loaded in any condition.

A. Incorrect – EDG 102 cannot be manually started and loaded from the Main Control Room with this switch position. Plausible that the switch would only serve to enable to local start pushbutton, but would not disable remote starting.
B. Incorrect – EDG 102 cannot be manually started and loaded from the Main Control Room with this switch position. Plausible that the switch would only serve to enable to local start pushbutton, but would not disable remote starting.
C. Incorrect – EDG 102 will not start and load on a valid undervoltage signal with this switch position. Plausible that the switch would only affect local/remote manual starting/loading of the EDG and not automatic starting/loading.

Technical Reference(s): ARP A4-2-6, N1-OP-45, C-19410-C sheets 3 and 10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 291001 K1.04
	Importance Rating	2.8

INSTRUMENT AIR SYSTEM

Valves

The failed-valve positions for different operators (open, closed, and as-is positions; spring-loaded valves; hydraulically/pneumatically controlled valves; electric motor-driven valves)

Proposed Question: #51

The Reactor is scrammed with the following:

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

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

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

Proposed Answer: B

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278000-RBO-8

Question Source: Bank – 2017 NRC #6

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A2.08
	Importance Rating	3.5

COMPONENT COOLING WATER SYSTEM

Ability to (a) predict the impacts of the following on the COMPONENT COOLING WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Residual heat removal system heat exchanger tube leak

Proposed Question: #52

A plant shutdown is in progress with the following:

- Reactor pressure is 100 psig.
- Reactor coolant temperature is 337°F.
- Shutdown Cooling (SDC) loop 11 is in service.

Then, a tube leak develops in the SDC 11 heat exchanger.

Which one of the following describes:

- the resulting flow of leakage

  and

- the ability to isolate this tube leak, in accordance with N1-OP-4, Shutdown Cooling System?

	Leakage is from…	Leak Isolation
A.	SDC into RBCLC	All three SDC loops must be isolated
B.	SDC into RBCLC	SDC loop 11 can be isolated without isolating SDC loops 12 and 13
C.	RBCLC into SDC	All three SDC loops must be isolated
D.	RBCLC into SDC	SDC loop 11 can be isolated without isolating SDC loops 12 and 13

Proposed Answer: B

Explanation: The SDC heat exchangers are cooled by RBCLC. Normal supply pressure from RBCLC pumps is 80-90 psig. With Reactor pressure at 100 psig, SDC system pressure will be above RBCLC supply pressure. Therefore, leakage will be from SDC into RBCLC. This will cause rising temperature in RBCLC, rising rad levels in RBCLC, and discharge from the RBCLC overflow line near the CLC makeup tank. N1-OP-4 section H.3 provides guidance for isolating a SDC heat exchanger tube leak. This procedure section allows isolation of just SDC loop 11 without isolating SDC loops 12 and 13.

Note: The equivalent to an RHR heat exchanger tube leak at NMP Unit 1 is a SDC heat exchanger tube leak.

A. Incorrect – SDC loop 11 can be isolated without isolating SDC loops 12 and 13 per OP-4 section H.3. Plausible because use of certain isolation valves would result in isolation of all three SDC loops (38-01, 38-02, 38-13, 70-53). Also plausible because numerous conditions do isolate SDC with these valves (low Reactor water level, high SDC area temperature, manual Vessel isolation using E-console switches).
C. Incorrect – Under the given conditions, leakage is from SDC into RBCLC. Plausible because, for some systems, leakage would be in the other direction. Also plausible because leakage may be in the other direction if Reactor pressure was lower. SDC loop 11 can be isolated without isolating SDC loops 12 and 13 per OP-4 section H.3. Plausible because use of certain isolation valves would result in isolation of all three SDC loops (38-01, 38-02, 38-13, 70-53). Also plausible because numerous conditions do isolate SDC with these valves (low Reactor water level, high SDC area temperature, manual Vessel isolation using E-console switches).
D. Incorrect – Under the given conditions, leakage is from SDC into RBCLC. Plausible because, for some systems, leakage would be in the other direction. Also plausible because leakage may be in the other direction if Reactor pressure was lower.

Technical Reference(s): ARP K3-2-1, N1-OP-4 section H.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 K2.01
	Importance Rating	3.5

RECIRCULATION SYSTEM

Knowledge of electrical power supplies to the following: Recirculation pumps

Proposed Question: #53

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

- Reactor Recirculation pump (RRP) 15 is shut down.
- The other four RRPs are operating.

Then, Powerboard 11 de-energizes due to a sustained electrical fault.

Which one of the following identifies the resulting status of RRPs?

	RRP 11	RRP 12	RRP 13	RRP 14
A.	Running	NOT running	Running	NOT running
B.	Running	Running	NOT running	NOT Running
C.	NOT running	Running	NOT running	Running
D.	NOT running	NOT running	Running	Running

Proposed Answer: D

Explanation: RRPs 11 and 12 are powered from Powerboard 11. RRPs 14 and 15 are powered from Powerboard 12. RRP 13 is powered from Powerboard 101. The loss of Powerboard 11 results in RRPs 11 and 12 NOT running, RRPs 13 and 14 running.

A. Incorrect – RRPs 11 and 12 are NOT running and RRP 14 is running. Plausible because this would be correct if the loss tripping RRPs 12 and 14.
B. Incorrect – RRPs 11 and 12 are NOT running and RRP 13 is running. Plausible because this would be correct if the loss tripping RRPs 13 and 14.
C. Incorrect – RRP 12 is NOT running and RRP 13 is running. Plausible because this would be correct if the power supply alignment was different (RRPs 11 and 13 on PB 11, RRPs 12 and 14 on PB 12).

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	216000 A1.05
	Importance Rating	3.3

NUCLEAR BOILER INSTRUMENTATION

Ability to predict and/or monitor changes in parameters associated with operation of the NUCLEAR BOILER INSTRUMENTATION including: Lights and alarms

Proposed Question: #54

The plant is experiencing a transient with the following:

- Annunciator F1-2-8, CORE LEVEL TORUS TEMP. MONITOR SYS. 11 TROUBLE, alarms.
- Annunciator F4-2-1, CORE LEVEL TORUS TEMP. MONITOR SYS. 12 TROUBLE, alarms.
- The Fuel Zone channel 11 and 12 displays begin flashing.

Which one of the following describes what is indicated by these conditions?

A. Reactor water level is below the usable range of Fuel Zone instruments

B. Reactor water level is above the usable range of Fuel Zone instruments

C. Fuel Zone instruments are unreliable due to the status of reference leg temperature

D. Fuel Zone instruments are unreliable due to the status of Reactor Recirculation pumps

Proposed Answer: C

Explanation: The given annunciators and flashing indicators indicate that Fuel Zone instruments are unreliable due to reference leg temperatures above the saturation temperature for the current Reactor pressure (indicates potential for reference leg flashing).

A. Incorrect – The given conditions indicate Fuel Zone instruments are unreliable due to reference leg temperature. Plausible because being below usable range would also be of concern and is called out in EOP Detail A.
B. Incorrect – The given conditions indicate Fuel Zone instruments are unreliable due to reference leg temperature. Plausible because being above usable range would also be of concern and is called out in EOP Detail A.
D. Incorrect – The given conditions indicate Fuel Zone instruments are unreliable due to reference leg temperature. Plausible because status of Reactor Recirculation pumps does affect Fuel Zone indicators and is called out in EOP Detail A.

Technical Reference(s): ARPs F1-2-8 and F4-2-1, N1-216000-RBO-5, EOP Detail A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-216000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	223001 K1.16
	Importance Rating	3.5

PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES

Knowledge of the physical connections and/or cause and effect relationships between the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and the following systems: Containment and drywell atmosphere monitoring system

Proposed Question: #55

A loss of coolant accident has resulted in the following:

- Drywell pressure is 12 psig and rising slowly.
- Reactor water level is 20" and stable.

Which one of the following describes:

- the signal that is currently causing an isolation of the H_2O_2 monitors

  and

- the switches that would need to be manipulated to restore the H_2O_2 monitors under these conditions?

	Signal Currently Causing Isolation	Switches Manipulated to Restore
A.	High Drywell pressure	CAD CHANNEL 11 and CAD CHANNEL 12 RPS BY-PASS
B.	High Drywell pressure	AUTO VESSEL ISOL CH 11 and AUTO VESSEL ISOL CH 12 BYPASS
C.	Low Reactor water level	CAD CHANNEL 11 and CAD CHANNEL 12 RPS BY-PASS
D.	Low Reactor water level	AUTO VESSEL ISOL CH 11 and AUTO VESSEL ISOL CH 12 BYPASS

Proposed Answer: A

Explanation: The H_2O_2 monitors are currently isolated due to Drywell pressure above 3.5 psig. Per N1-OP-9 and N1-EOP-PC, to recover the H_2O_2 monitors it is necessary to un-isolate the system by bypassing the Containment Isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches.

B. Incorrect – AUTO VESSEL ISOL CH 11 and CH 12 BYPASS switches do not need to be manipulated to restore H_2O_2 monitors. Plausible because these switches are used to bypass other isolations.
C. Incorrect – Reactor water level is not currently causing an isolation of the H_2O_2 monitors. Plausible because this would be correct if Reactor water level was below 5" and Drywell pressure was below 3.5 psig.
D. Incorrect – Reactor water level is not currently causing an isolation of the H_2O_2 monitors. Plausible because this would be correct if Reactor water level was below 5" and Drywell pressure was below 3.5 psig. AUTO VESSEL ISOL CH 11 and CH 12 BYPASS switches do not need to be manipulated to restore H_2O_2 monitors. Plausible because these switches are used to bypass other isolations.

Technical Reference(s): N1-EOP-PC, N1-OP-9, N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: Modified Bank – 2013 NRC #27

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Proposed Question: #27

A loss of coolant accident has resulted in the following:

- Reactor water level is -60 inches and rising slowly.
- Drywell pressure is 12 psig and rising slowly.

Which one of the following describes the availability of the H_2O_2 monitors and what operator actions are REQUIRED in accordance with N1-OP-9?

The H_2O_2 monitors are…

A. available and lined up. Verify proper operation at H_2O_2 Monitor Control Panel 11 (TB 291).

B. isolated. Bypass the isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches.

C. isolated. Bypass the isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches AND AUTO VESSEL ISOL CH 11 and CH 12 switches.

D. NOT available. Direct Chemistry to manually sample the Containment using N1-ECP-209 OR N1-ECP-210.

Proposed Answer: B

Explanation: The H_2O_2 monitors isolate with Drywell pressure above 3.5 psig. Per N1-OP-9, Sect. H.3, and EOP-4, to recover the H2 and O2 sampling systems it is necessary to un-isolate the system by bypassing the Containment Isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 A3.07
	Importance Rating	3.9

RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE

Ability to monitor automatic operation of the RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE including: Pump start

Proposed Question: #56

A loss of coolant accident occurs with the following:

- Drywell pressure is 4 psig and rising slowly.
- Reactor water level is 0" and stable.
- Reactor pressure is 920 psig and stable.
- NO actions are taken in the EOPs.

Which one of the following describes the status of Containment Spray and Containment Spray Raw Water pumps one (1) minute later?

	Containment Spray Pumps	Containment Spray Raw Water Pumps
A.	Running	Running
B.	Running	NOT running
C.	NOT running	Running
D.	NOT running	NOT running

Proposed Answer: B

Explanation: The Containment Spray pumps automatically start due to the combination of Drywell pressure >3.5 psig and Reactor water level <+5". The Containment Spray Raw Water pumps do not automatically start. Further manual action is required to initiate Containment Spray Raw Water flow to the heat exchangers.

A. Incorrect – Containment Spray Raw Water pumps do not automatically start. Plausible that these pumps would start on the same signal as the Containment Spray pumps to ensure heat removal.
C. Incorrect – Containment Spray pumps automatically start. Plausible because this would be correct if Reactor water level was above +5". Also plausible because Reactor water level is above the ADS setpoint of -10". Containment Spray Raw Water pumps do not automatically start. Plausible that these pumps would start earlier than Containment Spray pumps to ensure the lake side of the heat exchanger was pressurized first.
D. Incorrect – Containment Spray pumps automatically start. Plausible because this would be correct if Reactor water level was above +5". Also plausible because Reactor water level is above the ADS setpoint of -10".

Technical Reference(s): N1-OP-14, SDBD-203

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	233000 K5.03
	Importance Rating	3.3

FUEL POOL COOLING/CLEANUP

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the FUEL POOL COOLING/CLEANUP: Spent fuel decay heat generation

Proposed Question: #57

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

- Annunciator L1-4-5, FUEL POOL ANNUNCIATOR, alarms.
- An Operator in the field reports annunciator SFP-1-2, FUEL POOL PUMPS INLET TEMP, is in alarm.
- The Operator also reports Spent Fuel Pool temperature is 102°F and rising slowly.

Which one of the following describes:

- the most restrictive Spent Fuel Pool temperature limitation that must be observed, in accordance with N1-OP-6, Fuel Pool Filtering and Cooling System,

  and

- the need to enter N1-EOP-SC, Secondary Containment Control?

	Most Restrictive Spent Fuel Pool Temperature Limitation	N1-EOP-SC Entry
A.	110°F	Required
B.	110°F	NOT required
C.	140°F	Required
D.	140°F	NOT required

Proposed Answer: A

Explanation: During power operation, Spent Fuel Pool temperature must be maintained less than 110ºF. N1-EOP-SC entry is required because Spent Fuel Pool temperature is above 100ºF.

B. Incorrect – N1-EOP-SC entry is required because Spent Fuel Pool temperature is above 100ºF. Plausible because this would be correct if Spent Fuel Pool temperature was slightly lower. Also plausible because N1-EOP-SC entry would not be required if the plant was shutdown.
C. Incorrect – During power operation, Spent Fuel Pool temperature must be maintained less than 110ºF. Plausible because 140ºF is the limit during an outage with the Spent Fuel Pool in communication with the Reactor coolant.
D. Incorrect – During power operation, Spent Fuel Pool temperature must be maintained less than 110ºF. Plausible because 140ºF is the limit during an outage with the Spent Fuel Pool in communication with the Reactor coolant. N1-EOP-SC entry is required because Spent Fuel Pool temperature is above 100ºF. Plausible because this would be correct if Spent Fuel Pool temperature was slightly lower. Also plausible because N1-EOP-SC entry would not be required if the plant was shutdown.

Technical Reference(s): N1-OP-6, N1-EOP-SC

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-233000-RBO-10

Question Source: Modified Bank - 2013 NRC #93

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level SRO
Tier # 2
Group # 2
K/A # 233000 2.4.31
Importance Rating 4.1

Fuel Pool Cooling/Cleanup

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #92

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

- Annunciator L1-4-5, FUEL POOL ANNUNCIATOR, alarms.
- An Operator in the field reports annunciator SFP-1-2, FUEL POOL PUMPS INLET TEMP, is in alarm.
- The Operator also reports Spent Fuel Pool temperature is 100°F and rising slowly.

Which one of the following describes the most restrictive Spent Fuel Pool temperature limitation that must be observed to avoid violation of license condition?

Spent Fuel Pool temperature must be maintained less than or equal to…

A. 110°F based on Secondary Containment licensing basis.

B. 110°F based on Spent Fuel Pool licensing basis.

C. 140°F based on Secondary Containment licensing basis.

D. 140°F based on Spent Fuel Pool licensing basis.

Proposed Answer: A

Explanation: During power operation, Spent Fuel Pool temperature must be maintained less than 110°F per the licensing basis for the Secondary Containment.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 K6.08
	Importance Rating	3.5

MAIN AND REHEAT STEAM SYSTEM

Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the MAIN AND REHEAT STEAM SYSTEM: Main condenser vacuum

Proposed Question: #58

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

- A loss of Main Condenser vacuum occurs.
- The Reactor is manually scrammed.
- The Reactor Mode Switch is placed in SHUTDOWN.
- Reactor water level reaches a low of 12” before recovering to the normal band.
- Main Condenser vacuum is 6” Hg.

Which one of the following describes the resulting status of MSIVs?

MSIVs…

A. remain open

B. are closed due to a low Reactor water level signal, only

C. are closed due to a low Main Condenser vacuum signal, only

D. are closed due to both low Reactor water level and Main Condenser vacuum signals

Proposed Answer: C

Explanation: The MSIVs close due to Main Condenser vacuum <7" Hg. The MSIVs do not receive an isolation signal on Reactor water level because Reactor water level remained above the 5" isolation setpoint.

A. Incorrect – The MSIVs are closed due to Main Condenser vacuum <7" Hg. Plausible because this would be correct if Main Condenser vacuum was slightly higher.
B. Incorrect – The MSIVs do not receive an isolation signal on Reactor water level because Reactor water level remained above the 5" isolation setpoint. Plausible because this would be correct if Reactor water went slightly lower and Main Condenser vacuum was slightly higher.
D. Incorrect – The MSIVs do not receive an isolation signal on Reactor water level because Reactor water level remained above the 5" isolation setpoint. Plausible because this would be correct if Reactor water went slightly lower.

Technical Reference(s): N1-SOP-40.2

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(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 A2.02
	Importance Rating	3.8

MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS

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

Proposed Question: #59

The plant is operating at 100% power when a large leak causes a complete loss of Main Turbine lube oil pressure.

Which one of the following describes:

- the impact of this loss on the Main Turbine

  and

- the ability to mitigate this impact using plant procedures?

The Main Turbine…

A. automatically trips. Reactor pressure can be controlled using the Bypass Opening Jack.

B. automatically trips. Reactor pressure CANNOT be controlled using the Bypass Opening Jack.

C. does NOT automatically trip. The Main Turbine can be manually tripped from the Main Control Room.

D. does NOT automatically trip. The Main Turbine CANNOT be manually tripped from the Main Control Room.

Proposed Answer: B

Explanation: The Main Turbine lube oil system supplies control oil for the Main Turbine Control system (MHC). With a loss of all lube oil pressure, the Main Turbine automatically trips due to the loss of MHC control oil pressure. This loss of control oil pressure will also result in loss of automatic Reactor pressure control on Turbine Bypass Valves. The Bypass Opening Jack also requires control oil pressure to open the Turbine Bypass Valves, so this method of Reactor pressure control is not available to the operators.

A. Incorrect – Reactor pressure CANNOT be controlled using the Bypass Opening Jack. Plausible that the Bypass Opening Jack would mechanically jack open Turbine Bypass Valves and not require control oil pressure.
C. Incorrect – The Main Turbine automatically trips. Plausible that the loss of control oil pressure would result in a loss of trip function rather than an automatic trip.
D. Incorrect – The Main Turbine automatically trips. Plausible that the loss of control oil pressure would result in a loss of trip function rather than an automatic trip. Also plausible that this would disable Control Room turbine trip pushbuttons and require tripping the Main Turbine from the front standard.

Technical Reference(s): N1-OP-31, N1-247000-RBO-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-247000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 K4.01
	Importance Rating	3.9

FEEDWATER SYSTEM

Knowledge of FEEDWATER SYSTEM design features and/or interlocks that provide for the following: Automatic start of the RFPs

Proposed Question: #60

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

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

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

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

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

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

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

Proposed Answer: C

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

A. Incorrect – Condensate pumps 11 and 12, Feedwater Booster pumps 11 and 12, and Feedwater pump 11 remain off. Plausible because all pumps were running prior to the loss of power and Reactor water level is relatively low.
B. Incorrect – Condensate pumps 11 and 12, Feedwater Booster pumps 11 and 12, and Feedwater pump 11 remain off. Plausible because this would be correct if the control switches for the preferred components were in PTL.
D. Incorrect – Condensate pump 13, Feedwater Booster pump 13, and Feedwater pump 12 start. Plausible because some HPCI components are prevented from automatically restarting.

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 – 2015 NRC #17

Question History: 2015 NRC #17

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	268000 2.2.43
	Importance Rating	3.0

RADWASTE SYSTEM

Knowledge of the process used to track inoperable alarms

Proposed Question: #61

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

- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, is alarming every few minutes due to a faulty input.
- It is desired to disable the faulty annunciator input per S-ARP-001, Disabling Nuisance Annunciators.
- It is determined that a lead must be lifted to disable the annunciator.

Which one of the following describes the proper tag to be used on this annunciator, in accordance with in accordance with S-ARP-001?

A. Caution Tag

B. Information Tag

C. Special Condition Tag

D. Temporary Configuration Change Tag

Proposed Answer: D

Explanation: S-ARP-001 provides guidance to eliminate the source of a nuisance alarm. The process is carried out by OP-AA-108-101 and utilizes a Temporary Configuration Change Tag for the annunciator.

A. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because Caution Tags are commonly used on the Control Room boards for off-normal controls or indications.
B. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because Information Tags are commonly used on the Control Room boards for off-normal controls or indications.
C. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because a Special Condition Tag is another type of tag used at the facility for other processes.

Technical Reference(s): S-ARP-001

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-OJTTPE-SRO-OPAA10

Question Source: Bank – JAF 17-2 NRC #42

Question History: JAF 17-2 NRC #42

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	271000 A4.09
	Importance Rating	3.0

OFFGAS SYSTEM

Ability to manually operate and/or monitor the OFFGAS SYSTEM in the control room: Offgas system controls / components

Proposed Question: #62

The plant is operating at 100% power when a fuel defect results in the following annunciators:

- H1-1-7, OFF GAS HIGH RADIATION
- H1-2-7, OFF GAS RAD MON 11-12 FILTER DP SAMPLE FLOW
- H1-3-7, MN CONDR OFF GAS TIMER STARTED

The Off Gas rad monitors are confirmed to have tripped on high-high level.

Which one of the following describes the resulting status of:

- 76-12 and 76-13, Interstage Blocking Valves

  and

- 77-03, Stack Blocking Valve,

one (1) minute later?

	76-12 and 76-13, Interstage Blocking Valves	77-03, Stack Blocking Valve
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: B

Explanation: High-high Offgas radiation causes 77-03 to close immediately. The associated Offgas timers are set to 0, so no time delay occurs. 76-12 and 76-13 are unaffected by these conditions and remain open.

A. Incorrect – 77-03 is closed. Plausible that the timer would be set such that the valve would still be open after one minute.
C. Incorrect – 76-12 and 76-13 are unaffected by these conditions and remain open. Plausible because they do automatically close on high pressure or temperature. Also plausible because they also block the flow of gases from the Main Condenser to the Stack. 77-03 is closed. Plausible that the timer would be set such that the valve would still be open after one minute.
D. Incorrect – 76-12 and 76-13 are unaffected by these conditions and remain open. Plausible because they do automatically close on high pressure or temperature. Also plausible because they also block the flow of gases from the Main Condenser to the Stack.

Technical Reference(s): ARP H1-3-7, N1-OP-25

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-271000-RBO-5

Question Source: Modified Bank – 2013 NRC #62

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference: Level RO
Tier # 2
Group # 2
K/A # 271000 K6.09
Importance Rating 3.4

Offgas

Knowledge of the effect that a loss or malfunction of the following will have on the OFFGAS SYSTEM: Fuel cladding integrity

Proposed Question: #62

The plant is operating at 100% power when a fuel defect results in the following annunciators:

- H1-1-7, OFF GAS HIGH RADIATION
- H1-2-7, OFF GAS RAD MON 11-12 FILTER DP SAMPLE FLOW
- H1-3-7, MN CONDR OFF GAS TIMER STARTED

The Off Gas rad monitors are confirmed to have tripped on high-high level.

Which one of the following describes the resulting automatic action(s) and the required manual action?

	Automatic Action(s)	Required Manual Action
A.	Stack Blocking Valve, 77-03, closes.	Immediately scram the reactor due to rapidly lowering Main Condenser vacuum.
B.	Interstage Blocking Valves, 76-12 and 76-13, close.	Immediately scram the reactor due to rapidly lowering Main Condenser vacuum.
C.	Stack Blocking Valve, 77-03, closes.	Perform an emergency power reduction to attempt to control radiation levels and stabilize Main Condenser vacuum.
D.	Interstage Blocking Valves, 76-12 and 76-13, close.	Perform an emergency power reduction to attempt to control radiation levels and stabilize Main Condenser vacuum.

Proposed Answer: C

Explanation: BV 77-03 closes on high-high offgas radiation. The vacuum loss following closure of 77-03 is not rapid. Sufficient volume exists in the offgas system piping to continue to remove non-condensable gases from the main condenser for a period of time, during which vacuum will slowly degrade. N1-SOP-25.1 and 25.2 direct an emergency power reduction to control radiation/vacuum.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	290003 K5.04
	Importance Rating	3.7

CONTROL ROOM VENTILATION

Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the CONTROL ROOM VENTILATION: Control room habitability

Proposed Question: #63

Which one of the following identifies when manual initiation of Control Room Emergency Ventilation System (CREVS) is required, in accordance with N1-OP-49, Control Room Ventilation System?

A. High Drywell pressure or Low Low Reactor Water level at Unit 1

B. High Drywell pressure or Low Low Reactor Water level at Unit 2

C. High total Iodine concentration at Unit 1, or Unit 2 enters N2-EOP-RR, Radioactivity Release Control

D. High total Iodine concentration at Unit 2, or Unit 1 enters N1-EOP-RR, Radioactivity Release Control

Proposed Answer: C

Explanation: N1-OP-49 requires manual initiation of CREVS if either of the following occur:
- Total iodine radioactivity concentration greater than 9.47 µCi/gm (T.S. 3.2.4)
- Unit 2 enters RR (Radioactivity Release) EOP

Note: The question meets the K/A because an operational implication of Control Room habitability is knowing when CREVS is required to be manually initiated.

A. Incorrect – These conditions cause automatic initiation of CREVS, therefore manual initiation is not required. Plausible because these conditions require automatic initiation.
B. Incorrect – LOCA signals at Unit 2 do not require manual initiation of CREVS. Plausible because these signals at Unit 1 cause automatic initiation and there are conditions at Unit 2 that required manual initiation of CREVS at Unit 1.
D. Incorrect – The applicable Units are backwards. Plausible because this is correct if the Units are reversed.

Technical Reference(s): N1-OP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288003-RBO-10

Question Source: Bank – 2017 Cert #71

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.21
	Importance Rating	3.5

Ability to verify that a copy of a controlled procedure is the proper revision

Proposed Question: #64

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

- A working copy of an Operating Procedure was prepared for an evolution one week ago.
- The evolution was placed on hold, but will now be performed on the current shift.

Which one of the following identifies an electronic database that may be used to verify that the working copy of the procedure is the currently approved revision, in accordance with HU-AA-104-101, Procedure Use and Adherence?

A. FCMS

B. eSOMs

C. Maximo

D. PassPort

Proposed Answer: A

Explanation: HU-AA-104-101 requires using the Controlled Document Register for verifying current procedure revision. RM-AA-102 lists the Controlled Document Register as FCMS.

B. Incorrect – FCMS is the correct electronic database for this task. Plausible because this is another electronic database in use at the plant that is relevant to Operations and has numerous functions.
C. Incorrect – FCMS is the correct electronic database for this task. Plausible because this is another electronic database in use at the plant that is relevant to Operations and has numerous functions.
D. Incorrect – FCMS is the correct electronic database for this task. Plausible because this is another electronic database in use at the plant that is relevant to Operations and has numerous functions. Also plausible because PassPort is the Controlled Document Register at some Constellation plants.

Technical Reference(s): HU-AA-104-101, RM-AA-102

Proposed references to be provided to applicants during examination: None

Learning Objective: N-AN-OP-HU-LP-OBJ-29

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.32
	Importance Rating	3.8

Ability to explain and apply system precautions, limitations, notes, or cautions

Proposed Question: #65

Which one of the following identifies the limitation on the timing of positive Reactivity additions, in accordance with N1-OP-43B, Normal Power Operations?

Between positive reactivity additions, a minimum wait period of…

A. one (1) minute must be observed. This limitation applies at Reactor power levels between 5% and 10%, only.

B. three (3) minutes must be observed. This limitation applies at Reactor power levels between 5% and 10%, only.

C. one (1) minute must be observed. This limitation applies at Reactor power levels of 98% or greater, only.

D. three (3) minutes must be observed. This limitation applies at Reactor power levels of 98% or greater, only.

Proposed Answer: D

Explanation: N1-OP-43B contains the following requirement:

“When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3-minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.”

A. Incorrect – The minimum wait period is 3 minutes. Plausible because 1 minute is of the same magnitude. This restriction only applies at Reactor powers of 98% or greater. Plausible because positive reactivity additions must be performed in a cautious manner at all power levels. Also plausible because 5-10% power is high on IRMs with limited void feedback.
B. Incorrect – This restriction only applies at Reactor powers of 98% or greater. Plausible because positive reactivity additions must be performed in a cautious manner at all power levels. Also plausible because 5-10% power is high on IRMs with limited void feedback.
C. Incorrect – The minimum wait period is 3 minutes. Plausible because 1 minute is of the same magnitude.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-OP43AC01-EO-05

Question Source: Bank – NMP2 11/23 NRC #65

Question History: NMP2 11/23 NRC #65

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

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: #66

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 single notch withdrawal for this control rod based on SRM readings, in accordance with N1-OP-43A, Plant Startup?

Single 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 single 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 single notch movement for the next control rod.

A. Incorrect – There is no requirement to use single notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, single notch withdrawal would be required, but from 00 to 36 only.
C. Incorrect – There is no requirement to use single notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, single 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 single notch movement for the next control rod. Plausible because if 3 count rate doublings were approached, single 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 – 2018 NRC #69

Question History: 2018 NRC #69

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.38
	Importance Rating	3.6

Knowledge of conditions and limitations in the facility license

Proposed Question: #67

The plant has experienced a transient with the following:

- Peak Reactor pressure was 1255 psig.
- Lowest Reactor water level was -30".

Which one of the following describes whether Reactor pressure and Reactor water level violated Safety Limits (SLs), in accordance with Technical Specifications?

	Reactor Pressure	Reactor Water Level
A.	Violated a SL	Violated a SL
B.	Violated a SL	Did NOT violate a SL
C.	Did NOT violate a SL	Violated a SL
D.	Did NOT violate a SL	Did NOT violate a SL

Proposed Answer: C

Explanation: The peak Reactor pressure remained below the Safety Limit of 1375 psig. The lowest Reactor water level violated the Safety Limit of -10".

Note: The question meets the K/A because Technical Specifications, including Safety Limits, are part of the facility license (Appendix A).

A. Incorrect – Reactor pressure did not exceed a Safety Limit. Plausible because this would be correct if Reactor pressure rose higher to above 1375 psig. Also plausible because Reactor pressure is well above the 1080 psig limit in LSSS 2.1.2.c and slightly above the design pressure of 1250 psig.
B. Incorrect – Reactor pressure did not exceed a Safety Limit. Plausible because this would be correct if Reactor pressure rose higher to above 1375 psig. Also plausible because Reactor pressure is well above the 1080 psig limit in LSSS 2.1.2.c and slightly above the design pressure of 1250 psig. Reactor water level violated the Safety Limit of -10". Plausible because it remained well above top of active fuel (-84").
D. Incorrect – Reactor water level violated the Safety Limit of -10". Plausible because it remained well above top of active fuel (-84").

Technical Reference(s): Technical Specifications 2.1.1 and 2.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-TECSPEC01-EO-3.0

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

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

Ability to control radiation releases

Proposed Question: #68

A loss of coolant accident has resulted in the following:

- Primary Containment venting is required due to approaching the Primary Containment Pressure Limit (PCPL).
- Torus water level is 12.5' and stable.

Which one of the following describes the path required to be used to vent the Containment and the reason why, in accordance with N1-EOP-PC, Primary Containment Control?

Vent from the…

A. Drywell to more quickly reduce Containment pressure

B. Torus to minimize cycling of the Torus to Drywell vacuum breakers

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

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

Proposed Answer: D

Explanation: N1-EOP-PC requires venting from the Torus as long as Torus water level is below 27'. The reason is to better scrub fission products from the Containment atmosphere before release.

A. Incorrect – Venting should be from the Torus. Plausible because Drywell pressure is expected to be higher than Torus pressure under LOCA conditions, which would result in a faster pressure reduction if the Drywell was vented.
B. Incorrect – The reason is not to minimize cycling of vacuum breakers. Plausible because lowering Torus pressure first would minimize cycling of vacuum breakers and cycling vacuum breakers does raise the risk of them not re-seating properly, which would be undesirable.
C. Incorrect – Venting should be from the Torus. Plausible because Torus water level is elevated above the EOP entry condition of 11.25', and if it was above 27', then Drywell venting would be required.

Technical Reference(s): N1-EOP-PC, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank - NMP2 2017 NRC #70

Question History: NMP2 2017 NRC #70

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

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

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

Proposed Question: #69

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

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

Given the following possible actions:

(1) Discuss the alarm with the Unit Supervisor before receipt of the alarm
(2) Review the associated ARP before and/or after receipt of the alarm
(3) Hang a flagging tool on the alarm window after receipt of the alarm

Which one of the following identifies which of these actions are requirements to classify the alarm as an “Expected Alarm”, in accordance with OP-AA-103-102, Watchstanding Practices?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: A

Explanation: OP-AA-103-102 section 2.3 classifies an “Expected Alarm” as “an alarm that occurs in the normal course of equipment operation or testing shall be considered expected if the alarm is discussed with the responsible Control Room Supervisor and the Annunciator Response Procedure/Alarm Response Card has been reviewed before receipt.” Section 4.5.5.2 requires reviewing the ARP either before or after the alarm is received the first time the alarm comes in on a shift. Section 4.5.5.3 requires informing the Unit Supervisor of the alarm prior to actuation.

B. Incorrect – Reviewing the ARP is required. Plausible that the ARP wouldn’t need to be review since the operator understands why it is going to be received ahead of time. Hanging a flagging tool is not required. Plausible that some visual marking would be required, such as for a disabled annunciator.
C. Incorrect – Discussing the alarm with the Unit Supervisor is required. Plausible that the Unit Supervisor would only need to be informed once the alarm is received that it is expected. Hanging a flagging tool is not required. Plausible that some visual marking would be required, such as for a disabled annunciator.
D. Incorrect – Hanging a flagging tool is not required. Plausible that some visual marking would be required, such as for a disabled annunciator.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-SOP-5.1-CE-02

Question Source: Bank – NMP2 2014 NRC #72

Question History: PB 2017 NRC #74

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	4
	Group #	
	K/A #	292005 K1.10
	Importance Rating	3.3

Control Rods

State the purpose of flux shaping

Proposed Question: #70

The plant is operating at 80% power for a control rod sequence exchange.

Which one of the following states a primary reason for establishing specific control rod patterns to shape neutron flux?

A. Equalize burn up of control rods

B. Minimize local power peaks in the core

C. Prevent conversion of uranium into plutonium in the core

D. Reduce excessive localized Reactor vessel neutron irradiation

Proposed Answer: B

Explanation: One of the primary reasons for flux shaping is to minimize local power peaks in the core. This aids in satisfying thermal limits, preventing damage to the fuel, and burning up the fuel more evenly.

A. Incorrect – Equalizing burn up of control rods is not a primary reason for flux shaping. At 100% power, most control rods are fully withdrawn. Even with flux shaping, the flux profile is weighted towards the bottom center of the core, resulting in higher control rod burnup for the rods in this area, which are typically the rods that are partially inserted to shape flux. Plausible because having a more even flux profile would cause more even burn up of control rods than if the flux profile was more concentrated.
C. Incorrect – Part of the control rod sequencing strategy is to create conditions early in cycle that actually promote conversion of uranium into plutonium in the core. Plausible because there are generally restrictions on plutonium production due to nuclear weapons proliferation concerns.
D. Incorrect – Reactor vessel neutron irradiation is not a primary reason for flux shaping. Plausible because Reactor vessel neutron irradiation is a long term concern for the Reactor vessel's metal strength/embrittlement.

Technical Reference(s): BR0SIr4_Control_Rods

Proposed references to be provided to applicants during examination: None

Learning Objective: BR0SIr4_Control_Rods 10

Question Source: Bank – NRC GFES B1557 & B1656

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(1)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292004 K1.10
	Importance Rating	3.2

REACTIVITY COEFFICIENTS

Define the void coefficient of reactivity

Proposed Question: #71

The plant is operating at 50% power.

Which one of the following describes the effect of the void coefficient of reactivity if Recirculation flow is ***raised***?

If Recirculation flow is **raised**, the void coefficient of reactivity will cause…

A. positive reactivity to be added due to lower moderator density

B. positive reactivity to be added due to higher moderator density

C. negative reactivity to be added due to lower moderator density

D. negative reactivity to be added due to higher moderator density

Proposed Answer: B

Explanation: As Recirculation flow is raised, the moderator spends less time in the core, resulting in boiling occurring higher in the fuel channels. This results in fewer voids in the Reactor, or higher moderator density. The increase in moderation results in addition of positive reactivity.

A. Incorrect – Moderator density rises. Plausible because the change is due to a change in moderator density and density would lower given lowering Recirculation flow.
C. Incorrect – Positive reactivity is added. Plausible because negative reactivity would be added given lowering Recirculation flow. Moderator density rises. Plausible because the change is due to a change in moderator density and density would lower given lowering Recirculation flow.
D. Incorrect – Positive reactivity is added. Plausible because negative reactivity would be added given lowering Recirculation flow.

Technical Reference(s): BR08Ir4_Operational_Physics

Proposed references to be provided to applicants during examination: None

Learning Objective: BR08Ir4_Operational_Physics 21

Question Source: Bank – Exelon GFES Bank Reactor Theory #15

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Comments:

Form 4.2-1 Written Examination Question Worksheet

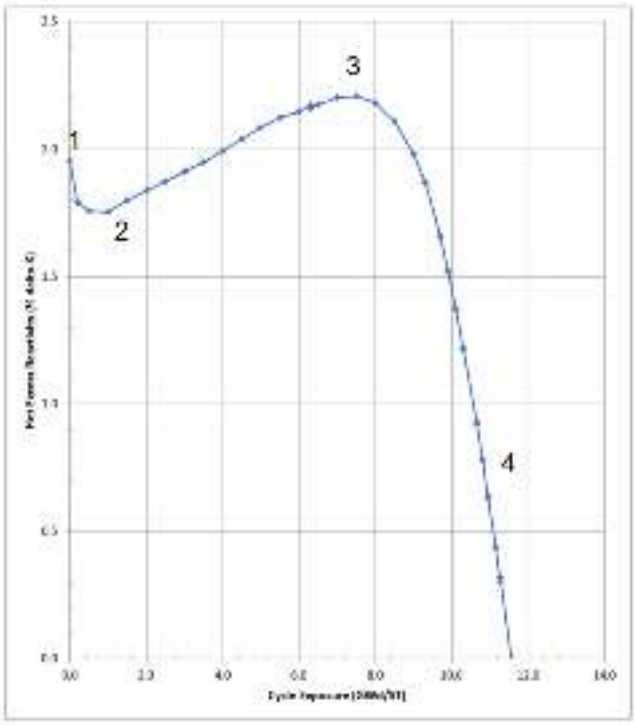
Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	292007 K1.03
	Importance Rating	2.7

FUEL DEPLETION AND BURNABLE POISONS

Given a curve of K-effective versus core age, state the reasons for maximum, minimum, and inflection points

Proposed Question: #72

Given the following drawing of K_{eff} versus core age from the Nine Mile Point Unit 1 Cycle Management report:



Which one of the following describes the major cause for the change in K_{eff} from point 2 to point 3?

A. Depletion of fuel

B. Depletion of control rods

C. Burnout of burnable poisons

D. Burnout of fission product poisons

Proposed Answer: C

Explanation: From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. The burnable poisons deplete faster than the fuel, raising K_{eff}.

A. Incorrect – From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. Plausible because depletion of fuel is the major cause for the change in K_{eff} from point 3 to 4.
B. Incorrect – From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. Plausible because the neutron absorbers in control rods do deplete over time.
D. Incorrect – From point 2 to 3, the major cause for the change in K_{eff} is the burnout of burnable poisons. Plausible because the buildup of fission product poisons is the major cause for the change in K_{eff} from point 1 to 2. Also plausible because fission product poisons do burnout in some conditions.

Technical Reference(s): BR07Ir4_Fuel_Depletion, Cycle Management Report

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: BR07Ir4_Fuel_Depletion Objective 3

Question Source: Bank – NMP2 Nov 2023 NRC #72

Question History: NMP2 Nov 2023 NRC #72

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(1)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	4
	Group #	
	K/A #	293003 K1.16
	Importance Rating	2.8

STEAM

Define the following term: Subcooling

Proposed Question: #73

The plant is operating at 100% power when extraction steam to a Feedwater heater isolates.

Which one of the following describes:

(1) the resulting change in Feedwater inlet temperature to the Reactor

and

(2) the amount of subcooling of the Feedwater entering the Reactor?

	Feedwater Inlet Temperature	Amount of Subcooling
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.	Lowers	Lowers

Proposed Answer: C

Explanation: Loss of extraction steam to a Feedwater heater results in less heating of the Feedwater before it enters the Reactor. This results in a lower Feedwater inlet temperature, which equates to more subcooling.

A. Incorrect – Feedwater inlet temperature lowers. Plausible because this would be correct for restoring extraction steam to a Feedwater heater.
B. Incorrect – Feedwater inlet temperature lowers and subcooling rises. Plausible because this would be correct for restoring extraction steam to a Feedwater heater.
D. Incorrect – Subcooling rises. Plausible if the definition of subcooling is not understood.

Technical Reference(s): BT03Ir4_Steam

Proposed references to be provided to applicants during examination: None

Learning Objective: BT03Ir4_Steam 11.o

Question Source: Bank – Exelon Fundamentals Thermo #2

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(14)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	293006 K1.05
	Importance Rating	3.3

FLUID STATICS AND DYNAMICS

Explain operational implications of fluid/water hammer

Proposed Question: #74

The plant has experienced a scram with the following:

- Reactor water level is 100” and stable.
- Emergency Condensers are NOT in service.

Which one of the following identifies the operational concern with manually initiating an Emergency Condenser under these conditions?

A. Water hammer

B. Thermal shock

C. Tube uncovering

D. Valve pressure binding

Proposed Answer: A

Explanation: With Reactor water level above 95”, the concern with initiating an Emergency Condenser is system damage due to water hammer.

B. Incorrect – The concern is water hammer. Plausible because the concern is a shock from a pressure transient. Also plausible because thermal shock is a concern in other applications in the plant.
C. Incorrect – The concern is water hammer. Plausible because tube uncovering is a concern related to other Emergency Condenser conditions (low shell level).
D. Incorrect – The concern is water hammer. Plausible because valve pressure binding is a concern related to other Emergency Condenser conditions (closing steam IVs at normal Reactor pressure).

Technical Reference(s): N1-OP-13, N1-EOP-HC attachment 9

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-207000-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(14)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	RO
	Tier #	4
	Group #	
	K/A #	293010 K1.06
	Importance Rating	4.1

BRITTLE FRACTURE AND VESSEL THERMAL STRESS

State the operational concerns of uncontrolled cooldown

Proposed Question: #75

A normal plant shutdown and cooldown is in progress in accordance with N1-OP-43C, Plant Shutdown.

Which one of the following is the cooldown rate limit and the basis for this limit?

	Cooldown Rate Limit	Basis for Limit
A.	25°F/hr	Avoid excessive stress to the Reactor vessel
B.	25°F/hr	Avoid degradation of Recirculation pump seals
C.	100°F/hr	Avoid excessive stress to the Reactor vessel
D.	100°F/hr	Avoid degradation of Recirculation pump seals

Proposed Answer: C

Explanation: N1-OP-43C and the Pressure and Temperature Limits Report (PTLR) require maintaining the cooldown rate ≤ 100°F/hr. This limit is based on avoiding excessive stress to the Reactor vessel.

A. Incorrect – The limit is 100°F/hr. Plausible because the PTLR gives a limit of 25°F/hr during hydrostatic and class 1 leak testing.
B. Incorrect – The limit is 100°F/hr. Plausible because the PTLR gives a limit of 25°F/hr during hydrostatic and class 1 leak testing. This limit is based on avoiding excessive stress to the Reactor vessel. Plausible because Recirculation pump seals would also be affected by changes in Reactor coolant temperature and degradation of Recirculation pump seals is a concern related to loss of coolant.
D. Incorrect – This limit is based on avoiding excessive stress to the Reactor vessel. Plausible because Recirculation pump seals would also be affected by changes in Reactor coolant temperature and degradation of Recirculation pump seals is a concern related to loss of coolant.

Technical Reference(s): BT10Ir4_Brittle_Stress, N1-OP-43C, PTLR, Technical Specification 3.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: BT10Ir4_Brittle_Stress 11

Question Source: Bank – Exelon Fundamentals Thermo #56

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(14)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295001 AA2.13
	Importance Rating	3.9

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: Thermal limits

Proposed Question: #76

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

- Two Reactor Recirculation pumps (RRPs) trip.
- N1-SOP-1.3, Recirc Pump Trip at Power, is entered
- The associated RRP discharge valves are closed in accordance with N1-SOP-1.3.
- Reactor power is now approximately 86%.

Which one of the following describes:

- the status of Reactor power, in accordance with N1-SOP-1.3,

  and

- the need to apply a thermal limit penalty, in accordance with Technical Specifications and/or the Core Operating Limits Report?

	Reactor Power	Thermal Limit Penalty
A.	Must be lowered	Required
B.	Must be lowered	NOT required
C.	May be raised	Required
D.	May be raised	NOT required

Proposed Answer: C

Explanation: Technical Specification 3.1.7.e requires Reactor power to be limited to 90% when operating with only three Reactor Recirculation pumps in service. With power currently at 86%, it can be raised. Technical Specification 3.1.7.e also requires referencing the COLR for APLHGR and MCPR values for three loop operation. The COLR requires a thermal limit penalty to be applied for MCPR while in three loop operation.

A. Incorrect – Reactor power is allowed to be as high as 90%. Plausible because this would be correct if Reactor power was >90% or if the limit was lower.
B. Incorrect – Reactor power is allowed to be as high as 90%. Plausible because this would be correct if Reactor power was >90% or if the limit was lower. A thermal limit penalty must be applied. Plausible because the MAPLHGR multiplier for three loop operation is currently 1.00, meaning there is no actual penalty on that thermal limit, despite it being referenced by Technical Specification 3.1.7.
D. Incorrect – A thermal limit penalty must be applied. Plausible because the MAPLHGR multiplier for three loop operation is currently 1.00, meaning there is no actual penalty on that thermal limit, despite it being referenced by Technical Specification 3.1.7.

Technical Reference(s): N1-SOP-1.3, Technical Specification 3.1.7, COLR

Proposed references to be provided to applicants during examination: Technical Specification 3.1.7, COLR

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

Question Source: Bank – 2008 NRC #93

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	202001 2.2.22		
	Importance Rating			4.7

(K&A Statement) Equipment Control - Knowledge of limiting conditions for operations and safety limits.(Recirc)

Proposed Question: SRO 93

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

- Reactor Recirc Pump 14 is secured and isolated
- Reactor Recirc Pump 11 develops a failure of the high pressure mechanical seal
- Reactor Recirc Pump 11 will be removed from service and isolated

Which one of the following is true for these conditions?

A. Power must be reduced to less than 90% and thermal limits verified

B. The reactor must be manually scrammed to prevent exceeding MCPR limits

C. Power can be restored to 100% provided the MCPR limit is increased by 0.02

D. Power must be reduced to less than 90.5% within ten minutes to prevent exceeding the APLHGR limits

Proposed Answer: A.

Explanation (Optional):

A. Correct – Based on the conditions, the plant will be in three loop operation. In accordance with TS 3.1.7.e, when operating with three recirculation loops and the two remaining loops isolated or unisolated, the reactor may operate at 90% of full power. Both APLHGR and MCPR limits must be adjusted per the Core Operating Limits report.

B. Incorrect – Three loop operation is permitted. A MCPR penalty must be applied to thermal limits. Nothing in the question implies that MCPR is violated. Even if MCPR were being violated, 15 minutes is allowed per Tech Specs to take action to restore operation to within limits.

C. Incorrect – Tech Specs limit 3 loop operation to 90%.

D. Incorrect – Power must be less than 90% for 3 loop operation. 90.5% is based on 4 loop operating limits.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295003 2.4.47
	Importance Rating	4.2

PARTIAL OR COMPLETE LOSS OF AC POWER

Ability to diagnose and recognize trends in an accurate and timely manner using the appropriate control room reference material (reference potential)

Proposed Question: #77

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

- Annunciator A8-1-3, 115 KV. BUS LOW VOLTAGE, alarms.
- N1-SOP-33A.3, Major 115 KV Grid Disturbances, is entered.
- Investigation reveals 115 KV system voltage momentarily dropped to a low of 105 KV on all three phases.
- 115 KV system voltage has recovered slightly and is now 109 KV and stable on all three phases.
- Power Control reports that the Low Voltage Contingency Computer is NOT available.

Which one of the following describes the need to enter an LCO in:

- Technical Specification 3.1.8, High Pressure Coolant Injection,

  and

- Technical Specification 3.6.3, Emergency Power Sources,

in accordance with N1-SOP-33A.3?

	Technical Specification 3.1.8 LCO Entry	Technical Specification 3.6.3 LCO Entry
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: A

Explanation: N1-SOP-33A.3 contains multiple overrides based on status of 115 KV system voltage and status of the low voltage post contingency alarm. With voltage below 109.3 KV, Arrow C applies. With this procedure section entered, HPCI, Line 1, and Line 4 must be declared inoperable. This requires entering LCOs in both Technical Specifications 3.1.8 and 3.6.3.

B. Incorrect – Technical Specification 3.6.3 must be entered. Plausible because voltage has recovered back towards the limit. Also plausible because Lines 1 and 4 are still connected to the plant.
C. Incorrect – Technical Specification 3.1.8 must be entered. Plausible because voltage is just below the 109.3 KV threshold associated with this Technical Specification entry. Also plausible because supported systems are usually not declared inoperable solely due to degradation of the supporting system (cascading Tech Specs).
D. Incorrect – Technical Specification 3.1.8 must be entered. Plausible because voltage is just below the 109.3 KV threshold associated with this Technical Specification entry. Also plausible because supported systems are usually not declared inoperable solely due to degradation of the supporting system (cascading Tech Specs). Technical Specification 3.6.3 must be entered. Plausible because voltage has recovered back towards the limit. Also plausible because Lines 1 and 4 are still connected to the plant.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP33A3C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2 & 5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295005 AA2.06
	Importance Rating	2.9

MAIN TURBINE GENERATOR TRIP

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

Proposed Question: #78

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

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

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

This is…

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

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

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

D. NOT the correct Reactor response. Enter N1-EOP-RPV, RPV Control, and direct a manual Reactor scram.

Proposed Answer: B

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

A. Incorrect – A manual Reactor scram is not required. Plausible because a Reactor scram would be required if Reactor power was above 35%.
C. Incorrect – This is the correct response to a Turbine trip <35% power. Plausible because an automatic scram would be expected if >35% power. Also plausible that Reactor power should not be rising if the Feedwater temperature effect is not understood. Also plausible because N1-SOP-1.5 entry is warranted for the rising Reactor power.
D. Incorrect – This is the correct response to a Turbine trip <35% power. Plausible because an automatic scram would be expected if >35% power. Also plausible that Reactor power should not be rising if the Feedwater temperature effect is not understood. Also plausible because this would be the correct response if >35% power and the Reactor failed to scram.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP31.1C01 EO-2

Question Source: Bank – 2015 NRC #78

Question History: 2015 NRC #78

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295024 2.4.22
	Importance Rating	4.4

HIGH DRYWELL PRESSURE

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

Proposed Question: #79

The plant has experienced an earthquake with the following:

- Reactor water level is 50” and lowering slowly.
- Drywell pressure is 4 psig and rising slowly.
- Spent Fuel Pool water level is 3” below the level of the weirs and slowly lowering.

Which one of the following describes the required prioritization, in accordance with N1-EOP-B-HOT, NMP1 Emergency Operating Procedures Basis Document, and OP-NM-101-111-1001, Transient Mitigation Guidelines?

The EOP Director…

A. must make Reactor injection the priority

B. must make Containment Control actions the priority

C. must make makeup to the Spent Fuel Pool the priority

D. is allowed to set the priority based on plant conditions

Proposed Answer: D

Explanation: N1-EOP-B-HOT and OP-NM-101-111-1001 both specify that there is no pre-determined prioritization for equipment use between EOPs. EOPs are to be implemented concurrently and priorities are set as determined by the EOP Director based on plant conditions.

A. Incorrect – Reactor injection does NOT always take priority over Containment Control activities and Spent Fuel Pool makeup. Plausible because adequate core cooling, if challenged, is given priority over other concerns in the EOPs.
B. Incorrect – Containment Control activities do NOT always take priority over Reactor injection and Spent Fuel Pool makeup. Plausible there are situations in the SAPs where Containment Control activities are given the priority.
C. Incorrect – Spent Fuel Pool makeup does NOT always take priority over Reactor injection and Containment Control activities. Plausible because low Spent Fuel Pool level can lead to a direct release to the environment, whereas the Reactor is inside a Primary Containment.

Technical Reference(s): N1-EOP-B-HOT, OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-TMG-EO-01

Question Source: Modified Bank – 2020 NRC #100

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference: Level SRO
Tier # 3
Group #
K/A # 2.4.22
Importance Rating 4.4

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

Proposed Question: #100

The plant has experienced an earthquake with the following:

- The Diesel Fire pump is the only available pump.
- Firefighting activities are in progress.
- Reactor water level is 50" and slowly lowering.
- Spent Fuel Pool water level is 3" below the level of the weirs and slowly lowering.

Which one of the following describes the prioritization for use of the Diesel Fire pump, in accordance with NER-1M-095, NMP1 Emergency Operating Procedures and Severe Accident Procedures Basis Document, and OP-NM-101-111-1001, Transient Mitigation Guidelines?

The EOP Director…

A. is allowed to set the priority for use of the Diesel Fire pump based on plant conditions.

B. must make lining up the Diesel Fire pump for Reactor injection the priority.

C. must make lining up the Diesel Fire pump for Spent Fuel Pool makeup the priority.

D. must make maintaining the Diesel Fire pump lined up to the Fire Water header the priority.

Proposed Answer: A

Explanation: NER-1M-095 and OP-NM-101-111-1001 both specify that there is no pre-determined prioritization for pump use or between EOPs. EOPs are to be implemented concurrently and priorities are set as determined by the EOP Director based on plant conditions.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295028 2.4.14
	Importance Rating	4.5

HIGH DRYWELL TEMPERATURE

Knowledge of general guidelines for emergency and abnormal operating procedures usage

Proposed Question: #80

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

- Annunciator H1-4-1, R BUILDING COOLING WATER PRESS TEMP MAKEUP FLOW, alarms.
- A malfunction of RBCLC temperature control has occurred.
- RBCLC supply temperature is 97°F and rising slowly.
- N1-SOP-11.1, RBCLC Failure, is entered.
- Drywell average temperature is 151°F and rising slowly.

Which one of the following describes how to implement N1-EOP-PC, Primary Containment Control, and N1-SOP-11.1?

A. Enter N1-EOP-PC. Continue performing N1-SOP-11.1. In the event of a conflict between the procedures, N1-SOP-11.1 is the overriding document.

B. Enter N1-EOP-PC. Continue performing N1-SOP-11.1. In the event of a conflict between the procedures, N1-EOP-PC is the overriding document.

C. Exit N1-SOP-11.1 and enter N1-EOP-PC. N1-SOP-11.1 is re-entered at the step in-progress after exiting N1-EOP-PC.

D. Exit N1-SOP-11.1 and enter N1-EOP-PC. N1-SOP-11.1 entry conditions are re-evaluated after exiting N1-EOP-PC.

Proposed Answer: B

Explanation: N1-EOP-PC is entered due to Drywell average temperature above 150oF. There is no requirement to exit SOPs when EOPs are entered. Both procedures are executed concurrently. The EOPs are higher-tiered documents than SOPs. Therefore, in the event of a conflict between the procedures, the EOP must be followed.

A. Incorrect – N1-EOP-PC is the overriding document. Plausible because N1-SOP-11.1 was entered first and is event-specific.
C. Incorrect – There is no requirement to exit SOPs when EOPs are entered. Both procedures are executed concurrently. Plausible because the EOP is a higher-tiered document.
D. Incorrect – There is no requirement to exit SOPs when EOPs are entered. Both procedures are executed concurrently. Plausible because the EOP is a higher-tiered document.

Technical Reference(s): N1-EOP-PC, N1-SOP-11.1, N1-EOP-B-HOT section 1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank – NMP2 2017 NRC #84

Question History: NMP2 2017 NRC #84

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295030 EA2.02
	Importance Rating	3.8

LOW SUPPRESSION POOL WATER LEVEL

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

Proposed Question: #81

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

- Reactor power is 6%.
- Reactor water level has been lowered in accordance with N1-EOP-C5, RPV Control - ATWS.
- Reactor pressure is 750 psig and slowly rising.
- Reactor pressure is being controlled between 700 and 900 psig with Emergency Condensers and ERVs.
- Torus water temperature is 117°F and slowly rising.
- N1-EOP-PC, Primary Containment Control, has been entered.
- Torus makeup is in service in accordance with N1-EOP-1 Attachment 18, Raw Water to Torus Makeup.
- Torus water level is 8.1 feet and lowering slowly.

Which one of the following describes the required control of Reactor pressure, in accordance with the EOPs?

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

B. An RPV Blowdown is required based on the Torus Level leg of N1-EOP-PC.

C. An RPV Blowdown is required based on the Torus Temperature leg of N1-EOP-PC.

D. Reactor pressure must be maintained in a lower Reactor pressure band, but an RPV Blowdown is NOT required.

Proposed Answer: B

Explanation: The Torus Level leg of N1-EOP-PC requires an RPV Blowdown if you cannot maintain Torus water level above 8 feet. Since the highest capacity Torus makeup method is already in service and Torus water level is 8.1 feet and lowering, Torus water level cannot be maintained above 8 feet. This requires an RPV Blowdown.

A. Incorrect – An RPV Blowdown is required. Plausible because, using the 9.0' HCTL curve, this would be correct.
C. Incorrect – An RPV Blowdown is not required based on Torus temperature. Plausible because this would be correct for higher Torus water levels or if the Good/Bad sides of the HCTL curves were reversed.
D. Incorrect – An RPV Blowdown is required. Plausible because this would be correct for some of the higher Torus water level HCTL curves.

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

Proposed references to be provided to applicants during examination: HCTL curve

Note: The HCTL curve to be provided is located at the end of this file and is to be included in the rest of the reference package supplied to the students.

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2015 Cert #75

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295037 EA2.06
	Importance Rating	4.3

SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN

Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure

Proposed Question: #82

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

- Multiple control rods are fully withdrawn.
- Reactor power is 12% and stable.
- Reactor water level is being controlled -84” to -41” using Feedwater.
- Reactor pressure is 750 psig and lowering on Turbine Bypass Valves (TBVs).

Then, the following occurs:

- An RPV Blowdown becomes required due to Secondary Containment temperatures.
- Emergency Condensers (ECs) are initiated.

Which one of the following describes the required control of Reactor pressure, in accordance with N1-EOP-C2, RPV Blowdown?

A. Use TBVs to lower Reactor pressure. Reduce Reactor pressure to as low as possible.

B. Use TBVs to lower Reactor pressure. Control ECs and TBVs to maintain Reactor pressure above 350 psig but as low as possible.

C. Open 4 ERVs to lower Reactor pressure. Then, reduce Reactor pressure to as low as possible.

D. Open 4 ERVs to lower Reactor pressure. Then, control ECs and ERVs to maintain Reactor pressure above 350 psig but as low as possible.

Proposed Answer: B

Explanation: With multiple control rods still fully withdrawn, the Reactor will not stay shutdown without boron, therefore the ATWS leg of N1-EOP-C2 must be executed. This leg requires lowering Reactor pressure to as low as possible above 350 psig using ECs and TBVs.

A. Incorrect – Reactor pressure must be maintained above 350 psig. Plausible because N1-EOP-C2 requires lowering Reactor pressure to as low as possible under non-ATWS conditions.
C. Incorrect – 4 ERVs are not opened. Plausible because this would be correct under non-ATWS conditions. Reactor pressure must be maintained above 350 psig. Plausible because N1-EOP-C2 requires lowering Reactor pressure to as low as possible under non-ATWS conditions.
D. Incorrect – 4 ERVs are not opened. Plausible because this would be correct under non-ATWS conditions.

Technical Reference(s): N1-EOP-C2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPC2C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295007 2.4.6
	Importance Rating	4.7

HIGH REACTOR PRESSURE

Knowledge of emergency and abnormal operating procedures major action categories

Proposed Question: #83

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

- A Station Blackout occurs.
- All control rods insert.
- N1-SOP-33A.2, Station Blackout, is entered.
- N1-EOP-RPV, RPV Control, is entered due to low Reactor water level and high Reactor pressure.
- Both Emergency Condensers automatically initiate.
- Reactor pressure is 1000 psig and lowering slowly.

Which one of the following describes the required control of Reactor pressure, in accordance with plant procedures?

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

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

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

D. Maintain both Emergency Condensers in continuous service.

Proposed Answer: D

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

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

Note: The question meets SRO level guidelines for 10 CFR 55.43(b)(5) because:

- It cannot be answered solely from systems knowledge (need knowledge of N1-SOP-33A.2, N1-EOP-RPV, and how to coordinate them)
- It cannot be answered solely by knowing immediate operator actions (there is an immediate operator action to place both ECs in service at the beginning of N1-SOP-33A.2, but further knowledge is required to understand continued control of ECs, including subsequent steps in N1-SOP-33A.2 andN1-EOP-RPV, as well as information in discussion section of N1-SOP-33A.2 and EOP bases)
- It cannot be answered solely by knowing the entry conditions for SOPs/EOPs
- It cannot be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure
- It requires:
    - Assessment of plant conditions (station blackout) and then deciding correct procedure execution (remove ECs or not, depressurize or not, adhere to cooldown rate limits or not)
    - Knowledge of administrative procedures (N1-EOP-B-HOT, N2-SOP-33A.2 discussion) that specify hierarchy, implementation, or coordination of abnormal (N2-SOP-33A.2) and emergency (N1-EOP-RPV) procedures

Technical Reference(s): N1-SOP-33A.2, N1-EOP-RPV, N1-EOP-B-HOT

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank – 2017 NRC #80

Question History: 2017 NRC #80

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295022 2.2.22
	Importance Rating	4.7

LOSS OF CONTROL ROD DRIVE PUMPS

Knowledge of limiting conditions for operation and safety limits

Proposed Question: #84

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

- CRD pump 11 is in service.
- Annunciator F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, alarms.
- Charging Water pressure is 900 psig and stable.
- CRD system flow is $10x10^3$ lbm/hr (20 gpm) and stable.
- An Operator reports from the field that a leak has developed from the discharge flange of CRD pump 11.

Which one of the following describes the need to enter an LCO in:

- Technical Specification 3.1.1, Control Rods,

  and

- Technical Specification 3.1.6, Control Rod Coolant Injection?

	Technical Specification 3.1.1 LCO Entry	Technical Specification 3.1.6 LCO Entry
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: C

Explanation: Technical Specification 3.1.6 and the associated bases require both CRD pumps to be able to supply at least 50 gpm. The given leak is preventing CRD pump 11 from being able to produce this flow. Therefore, CRD pump 11 must be declared inoperable and LCO entry is required in Technical Specification 3.1.6. Technical Specification 3.1.1 includes multiple requirements related to operation of the CRD system, including limitations on stuck control rods, control rod withdrawal, control scram times, and control rod accumulators. However, the given conditions do not exceed or violate any requirements in Technical Specification 3.1.1 or the associated bases. Therefore, LCO entry in Technical Specification 3.1.1 is not required.

A. Incorrect – LCO entry in Technical Specification 3.1.1 is not required. Plausible because Charging Water header pressure is below the 940 psig required accumulator pressure (LCO entry would be required if Annunciator F3-2-5 alarmed and accumulator pressures were less than 940 psig).
B. Incorrect – LCO entry in Technical Specification 3.1.1 is not required. Plausible because Charging Water header pressure is below the 940 psig required accumulator pressure (LCO entry would be required if Annunciator F3-2-5 alarmed and accumulator pressures were less than 940 psig). LCO entry is required in Technical Specification 3.1.6. Plausible because CRD pump 11 is still running and capable of providing significant flow.
D. Incorrect – LCO entry is required in Technical Specification 3.1.6. Plausible because CRD pump 11 is still running and capable of providing significant flow.

Technical Reference(s): ARPs F3-1-5 and F3-2-5, Technical Specifications 3.1.1 and 3.1.6 and associated bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-14

Question Source: Modified Bank – 2010 Cert #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures (Loss of CRD pumps).

Proposed Question: SRO Question # 85

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

- CRD pump 11 is in service
- Annunciator F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, is in alarm
- Charging Water pressure lowers and stabilizes at 900 psig
- CRD system flow lowers and stabilizes at 10×10^3 lbm/hr (20 gpm)
- An operator reports from the field that a leak has developed from the discharge flange of CRD pump 11

Which one of the following describes the entry requirements for SOP-5.1, Loss of CRD, and the operability of CRD pump 11 per Technical Specification 3.1.6, Control Rod Drive Pump Coolant Injection?

Entry into SOP-5.1 is __(1)__. CRD pump 11 is __(2)__ per Technical Specification 3.1.6.

		(1)	(2)
A.		Required due to low Charging Water pressure	Operable
B.		Required due to low Charging Water pressure	Inoperable
C.		NOT required unless CRD system flow lowers below 10 gpm	Operable
D.		NOT required unless CRD system flow lowers below 10 gpm	Inoperable

Proposed Answer: B

Explanation (Optional):

A.		Incorrect - TS 3.1.6 bases require a CRD pump to produce at least 50 gpm flow for operability. With CRD pump 11 only providing a system flow of 20 gpm, it must be declared inoperable.
B.		Correct - Annunciator F3-1-5 is an entry condition for SOP-5.1. TS 3.1.6 bases require a CRD pump to produce at least 50 gpm flow for operability. With CRD pump 11 only providing a system flow of 20 gpm, it must be declared inoperable.
C.		Incorrect - Annunciator F3-1-5 is an entry condition for SOP-5.1. TS 3.1.6 bases require a CRD pump to produce at least 50 gpm flow for operability. With CRD pump 11 only providing a system flow of 20 gpm, it must be declared inoperable.
D.		Incorrect - Annunciator F3-1-5 is an entry condition for SOP-5.1.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295029 EA2.03
	Importance Rating	3.6

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: #85

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

- Reactor water level is 30" and rising slowly.
- Reactor pressure is 350 psig and lowering slowly.
- Core Spray is injecting to the Reactor.
- Containment Sprays are in service per N1-EOP-1 Attachment 17, Auto or Manual Initiation of Containment Spray.
- Drywell pressure is 5 psig and lowering slowly.
- Water is being pumped from the Torus to Radwaste per N1-EOP-1 Attachment 15, Torus Water to Waste Collector.
- Torus water level is 13.6' and rising slowly.
- Torus water temperature is 125°F and rising slowly.

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

A. Perform an RPV Blowdown

B. Vent the Primary Containment

C. Secure Containment Sprays and initiate Torus Cooling

D. Stop injection from Core Spray not needed for adequate core cooling

Proposed Answer: A

Explanation: Torus water level is above 13.5' and continues to rise with Torus pump down (N1-EOP-1 attachment 15) already in progress. This requires an RPV Blowdown per the Torus level leg of N1-EOP-PC.

B. Incorrect – Venting the Primary Containment is not required. Plausible if the candidate believed the concern on Torus water level reaching 13.5' had to do with operability of Primary Containment venting, which is of concern at higher containment water levels.
C. Incorrect – Securing Containment Sprays is not required. Plausible because Containment Sprays are required to be secured when Drywell pressure lowers further, and Torus temperature is elevated, which makes Torus Cooling appropriate.
D. Incorrect – There is no need to secure injection from Core Spray because it injects from the Torus. Plausible because this would be correct if the injection sources was from outside of the Containment, such as Feedwater.

Note: The question meets the K/A because Containment water level includes Torus water level (at Containment water levels <30') and Drywell water level (at Containment water levels >30').

Technical Reference(s): N1-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Modified Bank – 2020 NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

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: #85

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

- Reactor water level is 30" and rising.
- Reactor pressure is 450 psig and lowering.
- Feedwater and CRD are injecting to the Reactor.
- Containment Sprays are in service.
- Drywell pressure is 5 psig and lowering.
- Torus water level is 13.4' and slowly rising.
- Torus water temperature is 125°F and slowly rising.

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

A. Perform an RPV Blowdown.

B. Vent the Primary Containment.

C. Secure Containment Sprays and initiate Torus Cooling.

D. Stop injection from Feedwater and CRD not needed for adequate core cooling.

Proposed Answer: D

Explanation: Torus water level is approaching 13.5'. Per N1-EOP-4, this requires stopping injection into the RPV from sources outside the Primary Containment (which includes Feedwater and CRD) not needed for adequate core cooling. With Reactor water level at 30" and rising, adequate core cooling is currently assured and there is more injection than needed to maintain adequate core cooling, therefore at least some of the injection from Feedwater and CRD can be stopped to help mitigate the rise in Torus water level.

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	205000 A2.05
	Importance Rating	4.0

SHUTDOWN COOLING SYSTEM

Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: System isolation

Proposed Question: #86

The plant is shutdown with the following:

- Shutdown Cooling (SDC) loop 11 is in service.
- Reactor coolant temperature is 140°F and slowly lowering.

Then, 38-01, REACTOR SHUTDOWN COOLING SUPPLY ISOL VAL 11, spuriously closes.

- SDC pump 11 trips.
- N1-SOP-6.1, Loss of Rx Cavity Level / Decay Heat Removal, has been entered.
- All attempts to re-open 38-01 have been unsuccessful.

Which one of the following describes:

- the ability to place an alternate loop of Shutdown Cooling in service while 38-01 is closed

  and

- a required action,

in accordance with N1-SOP-6.1?

	Can an alternate loop of SDC be placed in service while 38-01 is closed?	Required Action
A.	Yes	Verify Secondary Containment is established
B.	Yes	Secure all Water Inventory Control (WIC) activities
C.	No	Verify Secondary Containment is established
D.	No	Secure all Water Inventory Control (WIC) activities

Proposed Answer: C

Explanation: 38-01 is in the common suction path for all 3 SDC cooling loops. There is no alternate or parallel suction path. Therefore, with 38-01 closed, an alternate loop of SDC cannot be placed in service. N1-SOP-6.1 requires Secondary Containment be set given the loss of decay heat removal.

A. Incorrect – An alternate loop of SDC cannot be placed in service with 38-01 closed. Plausible that there would be an alternate or parallel suction path. Also plausible because this would be correct for closure of 38-03.
B. Incorrect – An alternate loop of SDC cannot be placed in service with 38-01 closed. Plausible that there would be an alternate or parallel suction path. Also plausible because this would be correct for closure of 38-03. It is not required to secure all WIC activities. Plausible because maintaining Reactor cavity inventory maximizes the available heat capacity for removal of decay heat.
D. Incorrect – It is not required to secure all WIC activities. Plausible because maintaining Reactor cavity inventory maximizes the available heat capacity for removal of decay heat.

Technical Reference(s): C-18018-C, N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1010-SOP6.1C01 EO-2

Question Source: Modified Bank – 2009 NRC #78

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

QUESTION	78

The plant is in a refueling outage with Core Shuffle Phase II in progress, when the following events occur:

- Shutdown Cooling (SDC) is lost
- Actual temperature measurements indicate Reactor water temperature is projected to exceed 140°F

Which one of the following is required, in accordance with N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal?

A.	Perform a time to boil estimation.
B.	Verify Secondary Containment established.
C.	Secure all Operations with the Potential for Draining the Reactor Vessel (OPDRVs).
D.	Return any core component being transferred to the nearest storage location in Spent Fuel Pool OR Core.

K&A #	295021 AA2.04
Importance Rating	3.6
QUESTION 78	
K&A Statement:	Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water temperature.
Justification:	
A.	Correct – Per N1-SOP-6.1 if the safety assessment level for DHR <N+1 and temperature is predicted to exceed 140F, then perform a time to boil.
B.	Incorrect because the procedure under pool or vessel inventory lowering requires this action. But plausible if the candidate is not familiar with actions in N1-SOP-6.1.
C.	Incorrect but plausible because N1-SOP-6.1 also covers actions for loss of vessel inventory while refueling.
D.	Incorrect because the procedure under pool or vessel inventory lowering requires this action. But plausible if the candidate is not familiar with actions in N1-SOP-6.1.

References:	N1-SOP-6.1	Student Ref:	None
Learning Objective:	N/A		
Question source:	NMP1		
Question History:	NRC 2004 Exam		

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	212000 A2.18
	Importance Rating	3.9

REACTOR PROTECTION SYSTEM

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: SCRAM air header low pressure

Proposed Question: #87

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

- Annunciator L1-3-7, INST AIR SYSTEM, is in alarm.
- Computer point A193, IA DRYERS OR FILTERS, is in alarm due to both high differential pressure across the filters and low pressure downstream of the filters.
- Instrument Air Header pressure indicates 100 psig and stable in the Control Room.
- Annunciator F3-3-2, CRD CONTROL AIR PRESSURE HI-LO, is in alarm.
- Scram Air Header pressure indicates 59 psig and slowly lowering.
- NO other abnormal CRD or RPS conditions have been observed yet.

Which one of the following describes:

- the potential impact of the low Scram Air Header pressure

  and

- whether a Reactor scram is currently required,

in accordance with N1-SOP-20.1, Instrument Air Failure?

	Potential Impact	Is a Reactor scram currently required?
A.	Loss of scram capability	Yes
B.	Loss of scram capability	No
C.	Control rod drifts or scrams	Yes
D.	Control rod drifts or scrams	No

Proposed Answer: D

Explanation: The concern with lowering Scram Air Header pressure is control rod drifts or scrams. N1-SOP-20.1 requires scramming the Reactor one or more control rods drift or scram. Since no other abnormal CRD or RPS conditions have been observed, a Reactor scram is not currently required.

A. Incorrect – The concern with lowering Scram Air Header pressure is control rod drifts or scrams. Plausible that the low air pressure would cause water leaking into the SDV and not draining properly, which could lead to loss of scram function due to hydraulic lock. A Reactor scram is not currently required. Plausible because Scram Air Header pressure is below the alarm setpoint, below the pressure listed in the N1-SOP-20.1 discussion section, and below the value requiring a scram in previous revisions of N1-SOP-20.1.
B. Incorrect – The concern with lowering Scram Air Header pressure is control rod drifts or scrams. Plausible that the low air pressure would cause water leaking into the SDV and not draining properly, which could lead to loss of scram function due to hydraulic lock.
C. Incorrect – A Reactor scram is not currently required. Plausible because Scram Air Header pressure is below the alarm setpoint, below the pressure listed in the N1-SOP-20.1 discussion section, and below the value requiring a scram in previous revisions of N1-SOP-20.1.

Note: The question meets SRO-level guidelines because the 2nd half of the question requires the student to use knowledge of non-immediate action steps in N1-SOP-20.1 to decide when execution of N1-SOP-1 is or is not required (procedure selection).

Technical Reference(s): N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP20.1C01 EO-2

Question Source: Modified Bank – 2023 NRC #79

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR

Ability to interpret and execute procedure steps

Proposed Question: #79

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

- Annunciator L1-3-7, INST AIR SYSTEM, is in alarm.
- Computer point A193, IA DRYERS OR FILTERS, is in alarm due to both high differential pressure across the filters and low pressure downstream of the filters.
- Annunciator F3-3-2, CRD CONTROL AIR PRESSURE HI-LO, is in alarm.
- Scram Air Header pressure indicates 65 psig and slowly lowering.
- Instrument Air Header pressure indicates 100 psig and stable in the Control Room.

Which one of the following describes the procedure to be entered and the required action?

	Procedure to Be Entered		Required Action
A.	N1-SOP-5.1, Loss of Control Rod Drive		Scram the Reactor if Scram Air Header pressure reaches 60 psig
B.	N1-SOP-5.1, Loss of Control Rod Drive		Scram the Reactor if one or more control rods drift or scram
C.	N1-SOP-20.1, Instrument Air Failure		Scram the Reactor if Scram Air Header pressure reaches 60 psig
D.	N1-SOP-20.1, Instrument Air Failure		Scram the Reactor if one or more control rods drift or scram

Proposed Answer: D

Explanation: N1-SOP-5.1 covers multiple different CRD system malfunctions, however lowering Instrument Air and Scram Air Header pressure requires entry into N1-SOP-20.1, not N1-SOP-5.1. N1-SOP-20.1 does not require a scram unless one or more control rods drift or scram

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215003 2.2.21
	Importance Rating	4.1

INTERMEDIATE RANGE MONITOR SYSTEM

Knowledge of pre- and post-maintenance operability requirements

Proposed Question: #88

Preparations are underway for a plant startup with the following:

- All control rods are full in.
- The Reactor Mode Switch is in SHUTDOWN.
- Maintenance was performed on IRMs during the shutdown.
- Post-maintenance testing reveals:
  - The upscale trip setpoint for IRM 13 is too high.
  - All other IRM trip setpoints are SAT.

Which one of the following describes the impact of these conditions on the startup, in accordance with Technical Specifications?

Control rod withdrawal may…

A. NOT begin until IRM 13 is fixed.

B. begin before IRM 13 is fixed. If IRM 13 is NOT fixed before SRMs reach three count rate doublings, then control rod withdrawal must be stopped.

C. begin before IRM 13 is fixed. If IRM 13 is NOT fixed within 6 hours, then control rod withdrawal must be stopped.

D. begin before IRM 13 is fixed. IRM 13 does NOT need to be fixed in order to complete the startup.

Proposed Answer: D

Explanation: N1-OP-43A, Plant Startup, requires verifying the requirements of Technical Specification Table 3.6.2.a for IRMs prior to commencing a startup. Technical Specification Table 3.6.2.a requires at least 3 operable IRMs in each trip system. With only IRM 13 inoperable, this requirement is met. Therefore, control rod withdrawal may begin and there is no requirement to fix IRM in order to complete the startup.

A. Incorrect – Control rod withdrawal may begin with IRM 13 inoperable. Plausible because this would be correct if another IRM in the same trip system was inoperable or if the requirement was for 4 operable IRMs per trip system.
B. Incorrect – IRM 13 does NOT need to be fixed in order to complete the startup. Plausible because IRMs are required to be operable for a startup and three count rate doublings is related to a control rod withdrawal requirement in N1-OP-43A.
C. Incorrect – IRM 13 does NOT need to be fixed in order to complete the startup. Plausible because IRMs are required to be operable for a startup and 6 hours is a related time requirement in Technical Specification 3.6.2 note (o).

Technical Reference(s): N1-OP-43A, Technical Specification 3.6.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	218000 2.2.19
	Importance Rating	3.4

AUTOMATIC DEPRESSURIZATION SYSTEM

Knowledge of maintenance work order requirements

Proposed Question: #89

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

- A failure has occurred in the ADS Channel 11 Inhibit keylock bypass switch.
- The failure is causing ADS Channel 11 to be bypassed.
- Maintenance is preparing to remove the switch and install a new one.
- Operations has determined a tagout will be placed to support the work.

Which one of the following describes:

- whether this job can be performed as “Tool Pouch Maintenance”, in accordance with MA-AA-716-003, Tool Pouch / Minor Maintenance,

  and

- whether a formal risk assessment needs to be performed, in accordance with OP-AA-107, Integrated Risk Management?

	Can this job be performed as “Tool Pouch Maintenance”?	Is a formal risk assessment required to be performed?
A.	Yes	Yes
B.	Yes	No
C.	No	Yes
D.	No	No

Proposed Answer: C

Explanation: This job cannot be performed as “Tool Pouch Maintenance” because it meets two of the criteria that exclude use of this classification: it is a job to restore Tech Spec equipment to an operable status and it is a power block activity which requires a clearance. This job does need a risk assessment to be performed, since it does not meet any of the exclusion criteria in OP-AA-107.

A. Incorrect – This job cannot be performed as “Tool Pouch Maintenance”. Plausible because it is replacement of a single component that can be accessed from the exterior of a panel. Also plausible because the component is already causing inoperability, such that the maintenance itself will not make the system become inoperable.
B. Incorrect – This job cannot be performed as “Tool Pouch Maintenance”. Plausible because it is replacement of a single component that can be accessed from the exterior of a panel. Also plausible because the component is already causing inoperability, such that the maintenance itself will not make the system become inoperable. A risk assessment must be performed. Plausible because the system is already inoperable due to the component malfunction and the work is isolated to a single system and component.
D. Incorrect – A risk assessment must be performed. Plausible because the system is already inoperable due to the component malfunction and the work is isolated to a single system and component.

Technical Reference(s): MA-AA-716-003, WC-AA-106, OP-AA-107

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-OJTTPE-SRO-WCAA10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262002 2.2.36
	Importance Rating	4.2

UNINTERRUPTABLE POWER SUPPLY (AC/DC)

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

Proposed Question: #90

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

Time (hh:mm)	Event
08:00	• UPS 172A is supplying RPS Bus 12. • UPS 172B is tagged out for maintenance.
12:00	• UPS 172A de-energizes due to a sustained electrical fault.
12:15	• RPS Bus 12 is re-energized from I&C Bus 130A.

Which one of the following describes the status of Technical Specification 3.6.3, Emergency Power Sources, during these events?

Technical Specification 3.6.3 LCO entry is ***first*** required at time…

A. 08:00. After time 12:15, this LCO can be exited.

B. 08:00. After time 12:15, this LCO entry must remain in effect.

C. 12:00. After time 12:15, this LCO can be exited.

D. 12:00. After time 12:15, this LCO entry must remain in effect.

Proposed Answer: D

Explanation: When UPS 172B is tagged out, there is no need to enter an LCO in Technical Specification 3.6.3. When both UPS 172A and 172B become unavailable (at time 12:00) Technical Specification 3.6.3 LCO entry is required. Once the RPS Bus is transferred to I&C Bus 130A, TS LCO 3.6.3 entry is still required due to loss of emergency power source (N1-OP-40 P&L D.10.a).

A. Incorrect – When UPS 172B is tagged out, there is no need to enter an LCO in Technical Specification 3.6.3. Plausible because this does reduce the available power sources to the RPS bus. At time 12:15, TS LCO 3.6.3 entry is still required. Plausible because I&C Bus 130A re-energizes RPS Bus 12 and all functions are restored.
B. Incorrect – When UPS 172B is tagged out, there is no need to enter an LCO in Technical Specification 3.6.3. Plausible because this does reduce the available power sources to the RPS bus.
C. Incorrect – At time 12:15, TS LCO 3.6.3 entry is still required. Plausible because I&C Bus 130A re-energizes RPS Bus 12 and all functions are restored.

Technical Reference(s): N1-OP-40, Technical Specification 3.6.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-14

Question Source: Modified Bank – 2017 Cert #89

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

UPS (AC/DC)

Ability to explain and apply all system limits and precautions.

Proposed Question: #89

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

- UPS 172A is supplying RPS Bus 12.
- Then, Annunciator A3-1-3, RPS UPS 172 TROUBLE, alarms.
- No half scram occurs.
- Computer point G184, UPS 172A SYSTEM ____TRBL, is received.
- An operator is dispatched to UPS 172A and reports the following local indications:
  - UPS 172A output voltage is 105 volts.
  - UPS 172A output frequency is 55 hertz.
- Then, RPS Bus 12 is manually transferred to I&C Bus 130A.

Which one of the following describes the Technical Specification impact for these events, if any?

A. No Technical Specification LCO entries are required.

B. LCO entry is required into Technical Specification 3.6.3, but NOT Technical Specification 3.6.12.

C. LCO entry is required into Technical Specification 3.6.12, but NOT Technical Specification 3.6.3.

D. LCO entry is required into both Technical Specifications 3.6.3 and 3.6.12.

Proposed Answer: D

Explanation: The UPS has low output voltage and frequency. The undervoltage and underfrequency protection for the RPS Bus has failed and this requires entry into TS LCO 3.6.12.b. Once the RPS Bus is transferred to I&C Bus 130A, TS LCO 3.6.3 entry is required due to loss of emergency power source (N1-OP-40 P&L D.10.a).

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	272000 2.3.12
	Importance Rating	3.7

Radiation Monitoring System

Knowledge of radiological safety principles and 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, or alignment of filters

Proposed Question: #91

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 required control of the Reactor, in accordance with the Emergency Operating Procedures?

A. An emergency power reduction is required, but a Reactor shutdown is NOT required.

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-SC 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-SC. 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-SC requires shutting down the Reactor per N1-OP-43C. This is the normal shutdown procedure.

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. 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. This would be the correct answer if the discharge was from a primary system.

Technical Reference(s): N1-EOP-SC

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPSCC01 EO-2

Question Source: Bank – 2017 Cert #83

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4 & 5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	2
	K/A #	256000 A2.01
	Importance Rating	3.9

CONDENSATE SYSTEM

Ability to (a) predict the impacts of the following on the CONDENSATE SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Pump trips

Proposed Question: #92

The plant is operating at 10% power during a startup with the following:

- Condensate pump 13 is operating.
- Condensate pumps 11 and 12 are in standby.

Then, Condensate pump 13 trips.

- Condensate pumps 11 and 12 automatically start.
- Reactor water level is 72” and stable.

Which one of the following describes the need for LCO entry in Technical Specification 3.1.8, High Pressure Coolant Injection?

LCO entry is…

A. required

B. NOT required because Reactor power is below 25%

C. NOT required because Condensate pump 13 is NOT a HPCI component

D. NOT required because any two Condensate pumps satisfy minimum operability requirements

Proposed Answer: A

Explanation: Condensate pump 13 is one of the two Condensate pumps associated with the HPCI mode of Condensate/Feedwater operation. With Reactor power at 10%, Reactor pressure is above 110 psig. Technical Specification 3.1.8 requires all HPCI-related pumps to be operable with Reactor pressure above 110 psig. Therefore, LCO 3.1.8.b entry is required.

B. Incorrect – LCO 3.1.8.b entry is required. Plausible because Technical Specification 3.1.8 allows keeping on Feedwater pump blocking valve closed up to 25% power.
C. Incorrect – LCO 3.1.8.b entry is required. Plausible because this would be correct for Condensate pump 12.
D. Incorrect – LCO 3.1.8.b entry is required. Plausible because only two Condensate pumps are required (but they are specific ones). Also plausible because all three Condensate pumps are capable of providing the necessary flow for the Feedwater pumps.

Technical Reference(s): N1-OP-16, Technical Specification 3.1.8

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-256000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	290002 2.4.21
	Importance Rating	4.6

REACTOR VESSEL AND INTERNALS

Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions

Proposed Question: #93

The plant has experienced a loss of coolant accident from 100% power with the following:

- N1-EOP-RPV, RPV Control, and N1-EOP-PC, Primary Containment Control, are being implemented.
- N1-EOP-C2, RPV Blowdown, has been performed due to low Reactor water level.
- All available Reactor injection systems have been maximized.
- NO indications of core damage have been observed.
- Core Spray loop 11 flow is 250 x 10^4 lbm/hr and stable.
- Core Spray loop 12 flow is 230 x 10^4 lbm/hr and stable.
- Reactor pressure is 50 psig and stable.
- Reactor water level is -129” and slowly lowering.

Which one of the following describes the required procedure implementation?

A. Remain in N1-EOP-RPV and N1-EOP-PC.

B. Exit N1-EOP-RPV and N1-EOP-PC.
Enter N1-SAP-1, RPV and Primary Containment Injection.
Entry into N1-SAP-2, RPV, Containment, and Radioactivity Release Control, is NOT required.

C. Exit N1-EOP-RPV and N1-EOP-PC.
Enter N1-SAP-2, RPV, Containment, and Radioactivity Release Control.
Entry into N1-SAP-1, RPV and Primary Containment Injection, is NOT required.

D. Exit N1-EOP-RPV and N1-EOP-PC.
Enter N1-SAP-1, RPV and Primary Containment Injection, and N1-SAP-2, RPV, Containment, and Radioactivity Release Control.

Proposed Answer: A

Explanation: Reactor water level below -126” indicates that adequate core cooling is challenged, However, since both Core Spray loop flows are above 180 x 10^4 lbm/hr, adequate core cooling from Spray Cooling exists. Therefore, SAP entry is NOT required and the crew must remain in N1-EOP-RPV and N1-EOP-PC.

B. Incorrect – SAP entry is not required. Plausible because adequate core cooling is challenged. Also plausible that SAPs would be entered individually based on specific plant conditions, as the EOPs are.
C. Incorrect – SAP entry is not required. Plausible because adequate core cooling is challenged. Also plausible that SAPs would be entered individually based on specific plant conditions, as the EOPs are.
D. Incorrect – SAP entry is not required. Plausible because this would be correct if either Core Spray loop flow was below 180 x 10^4 lbm/hr.

Note: The question meets the K/A because 290002, Reactor Vessel and Internals, includes the Reactor vessel, Reactor water level, and the Reactor core. The question gives a failure of the Reactor vessel (in the form of a loss of coolant accident), low Reactor water level, and asks about the status of cooling to the Reactor core.

Technical Reference(s): N1-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: Modified Bank – 2023 NRC #83

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.23
	Importance Rating	4.4

Ability to perform general or normal operating procedures during any plant condition

Proposed Question: #94

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

- A Technical Specification shutdown becomes required at time 12:00.
- The Technical Specification LCO states, "a normal orderly shutdown shall be initiated within one hour, and the reactor shall be in the cold shutdown condition within 10 hours."

Which one of the following describes the ***latest*** required timing of the shutdown in order to meet the 10-hour time requirement, in accordance with N1-OP-43C, Plant Shutdown?

At the ***latest***, insert a manual Reactor scram by approximately ___(1)___ and have Shutdown Cooling in service by approximately ___(2)___.

	(1)	(2)
A.	13:00	19:00
B.	13:00	20:30
C.	15:00	19:00
D.	15:00	20:30

Proposed Answer: C

Explanation: N1-OP-43C provides guidance on how to implement a 10-hour LCO shutdown in order to meet the time requirement. The procedure states, “When required to shutdown and cooldown to meet a 10-hour LCO requirement, it is necessary to perform a manual scram at approximately 3 hours into the LCO and have shutdown cooling in service approximately 7 hours into the LCO if cold shutdown is to be achieved in 10 hours.” Therefore, it is required to insert a manual Reactor scram by approximately 15:00 (12:00 + 3 hours) and to have Shutdown Cooling in service by approximately 19:00 (12:00 + 7 hours), at the latest.

A. Incorrect – It is required to insert a manual Reactor scram by approximately 15:00. Plausible because 13:00 is when the shutdown is required to be initiated by. Also plausible because this would provide extra time to perform the cooldown and balance of plant shutdown.
B. Incorrect – It is required to insert a manual Reactor scram by approximately 15:00. Plausible because 13:00 is when the shutdown is required to be initiated by. Also plausible because this would provide extra time to perform the cooldown and balance of plant shutdown. It is required to have Shutdown Cooling in service by approximately 19:00. Plausible because 20:30 would provide 1.5 hours to complete the cooldown with Shutdown Cooling, which would allow enough time to keep the cooldown rate <100°F/hr.
D. Incorrect – It is required to have Shutdown Cooling in service by approximately 19:00. Plausible because 20:30 would provide 1.5 hours to complete the cooldown with Shutdown Cooling, which would allow enough time to keep the cooldown rate <100°F/hr.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-OP-43C-CT-01

Question Source: Bank – 2009 Cert #95

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2 & 5)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.42
	Importance Rating	3.4

Knowledge of new and spent fuel movement procedures (SRO Only)

Proposed Question: #95

A refueling outage is in progress with the following:

- Core shuffle is in progress.
- A four cell section of the core is ***displayed on the following page***.

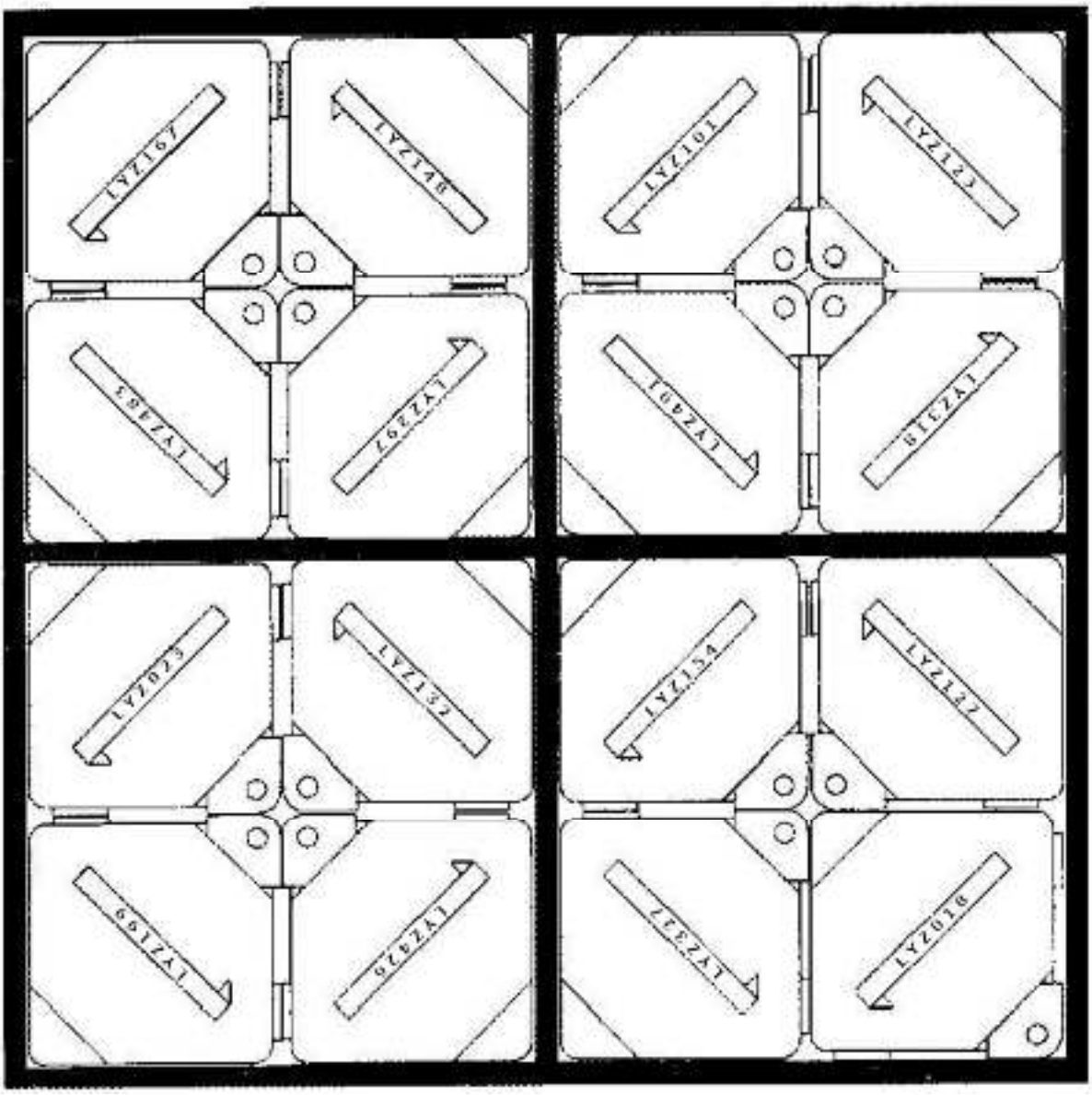
Which one of the following describes the status of this portion of the core and the required action(s), if any, in accordance with N1-FHP-25, General Description of Fuel Moves?

A. The fuel is loaded correctly. Core shuffle may continue with no additional required actions.

B. A discrepancy exists in the fuel loading. Core shuffle may continue without interruption as long as the discrepancy is fixed prior to core verification.

C. A discrepancy exists in the fuel loading. The discrepancy must be immediately fixed and then fuel movements may continue with permission from the Shift Manager.

D. A discrepancy exists in the fuel loading. Fuel movements must be immediately stopped. Completion of a formal evaluation is required before fuel movement may resume.


LYZ167
LYZ148
LYZ082
LYZ297
LYZ101
LYZ123
LYZ401
LYZ319
LYZ025
LYZ132
LYZ109
LYZ276
LYZ154
LYZ127
LYZ237
LYZ036

Proposed Answer: D

Explanation: The fuel assembly in the lower right corner is oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell, the position of the channel fastener, and the orientation of the text on the handle. N1-FHP-25 defines this as a Fuel Movement Discrepancy (FMD). The procedure requires fuel movement to be stopped and requires completion of a formal evaluation prior to resuming fuel movements.

A. Incorrect – There is a discrepancy in the fuel loading. Plausible because 15 of the 16 assemblies are loaded correctly and the overall shape is symmetrical.
B. Incorrect – Core shuffle cannot continue with the given error. Plausible because no significant adverse consequence would be expected from this fuel movement discrepancy until Reactor operation recommenced.
C. Incorrect – The discrepancy cannot be immediately fixed without a formal evaluation. Plausible because the issue would be fairly simple to fix.

Technical Reference(s): N1-FHP-25

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: S-101002-RBO-9

Question Source: Bank – NMP2 2021 NRC #95

Question History: 2013 NRC #80

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(6)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.6
	Importance Rating	3.6

Knowledge of the process for making changes to procedures

Proposed Question: #96

A Temporary Change is being processed for an operating procedure per AD-AA-101, Processing of Procedures and T&RMs.

Which one of the following describes:

(1) the individual(s) in Operations that are allowed to approve the Temporary Change for initial use,

and

(2) when full review, approval, and authorization of the Temporary Change must be completed by,

in accordance with AD-AA-101?

	(1)	(2)
A.	Any qualified SRO	14 days
B.	Any qualified SRO	31 days
C.	Operations Director, only	14 days
D.	Operations Director, only	31 days

Proposed Answer: A

Explanation: AD-AA-101 allows any qualified SRO to perform the initial approval of a Temporary Change. The full review, approval, and authorization must be completed within 14 days.

B. Incorrect – 14 days are allowed for full review, approval, and authorization. Plausible because 31 days is a common completion time and of similar magnitude as 14 days.
C. Incorrect – Any qualified SRO may perform the initial approval. Plausible because many processes require Operations Director approval.
D. Incorrect – Any qualified SRO may perform the initial approval. Plausible because many processes require Operations Director approval. 14 days are allowed for full review, approval, and authorization. Plausible because 31 days is a common completion time and of similar magnitude as 14 days.

Technical Reference(s): AD-AA-101, AD-AA-101-F-10

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-OJTTPE-SRO-ADAA10

Question Source: Bank – JAF 20-1 NRC #96

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.37
	Importance Rating	4.6

Ability to determine operability or availability of safety-related equipment (SRO Only)

Proposed Question: #97

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

- It is discovered that a surveillance has NOT been completed on time for a Technical Specification required system.
- The surveillance frequency is quarterly (92 days).
- The surveillance was last performed 120 days ago.
- A risk evaluation has been performed and the associated impact is being managed.

Which one of the following describes the requirement for this missed surveillance, in accordance with Technical Specifications?

A. The associated LCO must be declared NOT met at this time.

B. Complete the surveillance within a maximum of 24 hours from the time of discovery or then the associated LCO must be declared NOT met.

C. Complete the surveillance within a maximum of 23 days from the time of discovery or then the associated LCO must be declared NOT met.

D. Complete the surveillance within a maximum of 92 days from the time of discovery or then the associated LCO must be declared NOT met.

Proposed Answer: D

Explanation: Surveillance Requirement (SR) Applicability 4.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, 92 days), whichever is greater. Therefore, up to 92 days are allowed to perform the missed surveillance before being required to declare the LCO not met.

A. Incorrect – SR 4.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met. Plausible because the surveillance is currently over the 125% limit of SR 4.0.2.
B. Incorrect – SR 4.0.3 allows the longer of 24 hours or the specified frequency (92 days). Plausible because in some cases 24 hours would be the limit for a missed surveillance.
C. Incorrect – SR 4.0.3 allows the longer of 24 hours or the specified frequency (92 days). 23 days is based on the grace period (25% of 92 days) allowed by SR 4.0.2.

Technical Reference(s): Technical Specification Surveillance Requirement 4.0.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-TECSPEC01-EO-6.0

Question Source: Bank – NMP2 4/23 NRC #97

Question History: NMP2 4/23 NRC #97

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.6
	Importance Rating	3.8

Ability to approve liquid or gaseous release permits

Proposed Question: #98

A loss of coolant accident has resulted in the following:

- An RPV Blowdown has been performed due to exceeding the Pressure Suppression Pressure.
- Reactor pressure is 75 psig and stable.
- Reactor water level is being controlled 53” to 95” using Condensate.
- Core Spray injection has been secured by closing the inside isolation valves.
- NO other injection systems are available.
- Drywell pressure is 45 psig and slowly rising.
- Torus pressure is 42 psig and slowly rising.
- Torus water level is 17 feet and slowly rising.
- Chemistry reports that venting of the Primary Containment will exceed release rate limits.

Which one of the following describes the required control of Condensate injection and Primary Containment venting, in accordance with the Emergency Operating Procedures?

A. Continue injection with Condensate. Direct Primary Containment venting.

B. Continue injection with Condensate. Do NOT direct Primary Containment venting.

C. Transition injection from Condensate to Core Spray. Direct Primary Containment venting.

D. Transition injection from Condensate to Core Spray. Do NOT direct Primary Containment venting.

Proposed Answer: C

Explanation: The given values of Torus pressure and Torus water level show that the Primary Containment Pressure Limit (PCPL) is about to be exceeded. N1-EOP-RPV requires stopping injection from sources outside the Primary Containment not needed for core cooling if you cannot stay below PCPL. Reactor water level is well above top of active fuel, Core Spray is available for injection, and Condensate injects water from outside the Primary Containment. Therefore, injection must be transitioned from Condensate to another system, such as Core Spray. N1-EOP-PC requires venting the Primary Containment even if release rate limits will be exceeded.

Note: This question meets the K/A by testing the operator's knowledge of when to approve a release through Primary Containment venting in excess of normal release rate limits.

A. Incorrect – Condensate injection must be terminated because it injects from outside the Primary Containment and PCPL is being challenged. Plausible because PCPL has not yet been exceeded. Also plausible because this would be correct if other injection systems were not available and/or adequate core cooling were challenged.
B. Incorrect – Condensate injection must be terminated because it injects from outside the Primary Containment and PCPL is being challenged. Plausible because PCPL has not yet been exceeded. Also plausible because this would be correct if other injection systems were not available and/or adequate core cooling were challenged. Primary Containment venting is required. Plausible because PCPL has not yet been exceeded. Also plausible because release rates will be exceeded.
D. Incorrect – Primary Containment venting is required. Plausible because PCPL has not yet been exceeded. Also plausible because release rates will be exceeded.

Technical Reference(s): N1-EOP-RPV, N1-EOP-PC

Proposed references to be provided to applicants during examination: PCPL curve

Note: The PCPL curve to be provided is located at the end of this file and is to be included in the rest of the reference package supplied to the students.

Learning Objective: 1101-EOPPCC01 EO-2

Question Source: Bank – 2015 Cert #100

Question History: JAF 17-2 NRC #99

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.29
	Importance Rating	4.4

Knowledge of the emergency plan implementing procedures

Proposed Question: #99

An emergency is in progress with the following:

- An Alert has been declared.
- A condition requires a worker to enter a high radiation area.
- The required dose to perform the job is 17 Rem.

Which one of the following describes the ability to authorize this exposure for:

- protecting valuable plant equipment

  and

- life-saving activities

in accordance with EP-AA-113, Personnel Protective Actions?

This exposure…

A. can be authorized for protecting valuable plant equipment and can be authorized for life-saving activities.

B. can be authorized for life-saving activities, but NOT for protecting valuable plant equipment.

C. CANNOT be authorized unless the emergency is upgraded to a higher declaration.

D. CANNOT be authorized for either reason regardless of the emergency declaration.

Proposed Answer: B

Explanation: EP-AA-113 allows emergency dose authorization of up to 10 Rem for protecting valuable plant equipment. A dose authorization of 25 Rem is allowed for life-saving activities. Therefore, the given job that requires 17 Rem can be authorized for life-saving activities, but not for protecting valuable plant equipment.

A. Incorrect – The given dose is too high for protecting valuable plant equipment. Plausible because an emergency dose up to 10 Rem could be authorized for this purpose.
C. Incorrect – Emergency dose authorization is allowed at the Alert level. Plausible because this is the 2nd lowest declaration level and some E-plan actions do not happen until higher levels (eg. PARs, KI).
D. Incorrect – The given dose can be authorized for life-saving activities. Plausible because this is above the max dose for other activities and is well above the normal federal limit.

Technical Reference(s): EP-AA-113

Proposed references to be provided to applicants during examination: None

Learning Objective: S100-EPL001C07

Question Source: Bank – NMP2 11/23 NRC #100

Question History: JAF 17-2 NRC #98

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Form 4.2-1 Written Examination Question Worksheet

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.51
	Importance Rating	4.0

Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required)

Proposed Question: #100

The plant has experienced a loss of coolant accident and N1-EOP-RPV, RPV Control, is being executed.

Which one of the following indications would require exiting N1-EOP-RPV, in accordance with N1-EOP-RPV?

A. Drywell radiation at 400 R/hr

B. Reactor coolant activity at 500 µCi/gm

C. Drywell hydrogen concentration at 1.2%

D. Field survey at the site boundary at 200 mR/hr

Proposed Answer: B

Explanation: EOP Detail V, Core Damage Indications, defines the parameters and values used to determine if core damage is occurring in N1-EOP-RPV. If core damage is occurring, N1-EOP-RPV must be exited. Of the given indications, Reactor coolant activity at 500 µCi/gm is the one that exceeds a value in Detail V and requires exiting N1-EOP-RPV.

A. Incorrect – This does not require exiting N1-EOP-RPV. Plausible because Drywell radiation ≥1100 R/hr would require exiting N1-EOP-RPV.
C. Incorrect – This does not require exiting N1-EOP-RPV. Plausible because Drywell hydrogen concentration ≥4% would require exiting N1-EOP-RPV.
D. Incorrect – This does not require exiting N1-EOP-RPV. Plausible because this does indicate significant release of radioactivity from the plant and is above the value for declaration of a Site Area Emergency.

Technical Reference(s): N1-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOPRPVC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments: