

**March 2014 River Bend Station
NRC Initial License Examination
Reactor Operator**

QUESTION 1 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group #	1
K/A #	295001	AA1.05	IR 3.3

Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation:

AA1.05 Recirculation flow control system

Proposed Question:

Due to a failure of Reactor Recirculation Pump "B" an unexpected entry into the Restricted Region of the Power to Flow Map has occurred. The CRS has directed immediate exit of the Restricted Region using the Recirculation Flow Control System.

To comply with this direction, the ATC operator should toggle B33-K603A in the (1) direction using (2) detent.

- A. (1) open; (2) slow
- B. (1) open; (2) fast
- C. (1) close; (2) slow
- D. (1) close; (2) fast

Proposed Answer: A.

Explanation:

- A. Correct- Flow should be raise to exit the region. Lowering flow would result in further instability. Despite the word immediate, the FCVs are operated in slow detent to maintain control of the evolution.
- B. Part 1 is correct, but the FCVs are operated in slow detent to maintain control of the evolution.
- C. Toggling to the close direction would result in further instability. The region must be exited by raising flow. Part 2 is correct.
- D. Toggling to the close direction would result in further instability. The region must be exited by raising flow. Despite the word immediate, the FCVs are operated in slow detent to maintain control of the evolution.

Technical Reference(s): SOP-0003 Rev 311 Pg 37 of 100; AOP-0024 Rev 25 Pg 8 of 16

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053 Obj 1j, 7e & RLP-OPS-AOP024 Obj 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒3

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 2 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295003	G.2.1.20	IR 4.6

Ability to interpret and execute procedure steps during a partial or complete loss of AC power.

Proposed Question:

A loss of offsite power has occurred. All 3 diesel generators have started and are supplying associated buses.

Which of the following lists the immediate operator actions for the applicable Abnormal Operation Procedure (AOP)?

- A. Manually initiate Reactor Core Isolation Cooling (RCIC). Bypassing of interlocks is only allowed by the Emergency Operating Procedures.
- B. Manually initiate RCIC AND bypass high temperature isolation interlocks.
- C. Manually initiate High Pressure Core Spray (HPCS). Bypassing of interlocks is only allowed by the Emergency Operating Procedures.
- D. Manually initiate HPCS, AND defeat the high water level isolation interlock.

Proposed Answer: B.

Explanation:

- A. Initiation of RCIC is correct, however the AOP also directs bypassing high temperature interlocks.
- B. Correct – AOP-0004, Loss of Offsite Power directs Initiation of RCIC and bypassing of high temperature isolation interlocks.
- C. Although the availability of the Div 3 DG keeps HPCS as an available injection source, the AOP directs the initiation of RCIC and bypassing of high temperature interlocks.
- D. Although the availability of the Div 3 DG keeps HPCS as an available injection source, the AOP directs the initiation of RCIC and bypassing of high temperature interlocks. The bypassing of the high water level isolation interlock is only authorized by the Emergency Operating Procedures.

Technical Reference(s): AOP-0004, Rev 50, Pg 4 of 72.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0523 Obj 2, 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.10

Comments: Although the recall of immediate operator actions would normally be memory or fundamental knowledge, the cognitive level of this question is comprehension or analysis since the examinee must evaluate the conditions to determine which AOP is applicable.

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QUESTION 3 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295004	AA1.02	IR 3.8

Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of D.C. Power:
AA1.02 Systems necessary to assure safe plant shutdown

Proposed Question:

The plant is recovering from a loss of coolant accident. Residual Heat Removal (RHR) pump "A" is running in the Suppression Pool Cooling mode when the pump breaker experiences a loss of DC control power.

RHR "A" pump will _____.

- A. trip and all indicating lights will be extinguished.
- B. remain running with all indicating lights extinguished.
- C. trip but all only the white indicating light will be extinguished.
- D. remain running with only the white indicating light extinguished.

Proposed Answer: B.

Explanation:

- A. The pump will not trip on loss of DC control power. The pump must be manually tripped locally.
- B. Correct – The pump will remain running without trip capability and all indicating lights will lose power.
- C. The pump will not trip, and all lights will be extinguished.
- D. The pump will remain running, but all indicating lights will be extinguished.

Technical Reference(s): R-STM-0305 Rev 6, Pg 13 of 37; AOP-0014 Rev 23 Pg 19 of 74

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0605 Obj 10b

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 4 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295005 AK2.08		IR 3.2

Knowledge of the interrelations between Main Turbine Generator Trip and the following:
AK2.08 A.C. electrical distribution

Proposed Question:

Following an automatic Turbine-Generator Trip from 100% power, a (1) transfer will occur resulting in NPS-SWG1A being supplied from (2) .
(Assume all systems work as designed)

- A. (1) fast; (2) RTX-XSR1E, Preferred Station Transformer
- B. (1) slow; (2) RTX-XSR1E, Preferred Station Transformer
- C. (1) fast; (2) STX-XNS1A, Normal Station Transformer
- D. (1) slow; (2) STX-XNS1A, Normal Station Transformer

Proposed Answer: A.

Explanation:

- A. Correct -With all systems working as designed as stated in the stem, a FAST transfer will occur allowing a transfer of power while maintaining continuity without a loss of loads. During the transfer, the loads transfer from the Normal Station Transformer which were supplied by the Main Generator to the Preferred Station Transformers which are supplied by an offsite power source. Had this been a manual trip of the output breakers, a slow transfer would have occurred.
- B. A slow transfer only occurs if the fast transfer is unsuccessful (or if the output breakers are manually opened). Part 2 is correct.
- C. Part 1 is correct, but Part 2 is incorrect because the Normal Station Transformers will not longer have power when the Main Generator trips.
- D. A slow transfer only occurs if the fast transfer is unsuccessful (or if the output breakers are manually opened) and the Normal Station Transformers will not longer have power when the Main Generator trips.

Technical Reference(s): R-STM-0300 Rev 26 Pg 16-17 of 105,

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0300 Obj 3d, 14c, 18a

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments: Original question used NNS-SWG1A; changed to NPS and changed transformers to reflect NPS from XSR1C to E and XNS1C to A

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QUESTION 5 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group #	1
K/A #	295006 AK1.03		IR 3.7

Knowledge of the operational implications of the following concepts as they apply to SCRAM:
AK1.03 Reactivity control

Proposed Question:

To ensure sufficient time is allowed for full control rod insertion, the scram signal from the ____ (1) ____ is present ____ (2) ____ to prevent premature scram reset.

- A. (1) RPS manual pushbuttons; (2) for 10 seconds
- B. (1) RPS manual pushbuttons; (2) until bypassed with a keylock switch
- C. (1) Mode Switch; (2) for 10 seconds
- D. (1) Mode Switch; (2) until bypassed with a keylock switch

Proposed Answer: C.

Explanation:

- A. The signal which ensure full insertion of control rods during a scram is present for 10 seconds after placing the mode switch in shutdown; not the RPS manual pushbuttons.
- B. The signal which ensure full insertion of control rods during a scram is only present for 10 seconds after placing the mode switch in shutdown; not the RPS manual pushbuttons.
- C. Correct – The reactor mode switch in shutdown provides a redundant scram signal. After 10 seconds, this signal is automatically bypassed to allow resetting the scram. During the initial 10 seconds, the scram can not be reset ensuring adequate shutdown margin by allowing all control rods to fully insert.
- D. Part 1 is correct, but the signal is only present for 10 seconds.

Technical Reference(s): R-STM-0508 Rev 6 Pg 15 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 5c

Question Source: Bank # December 2010 NRC Exam #5

Question History: Last NRC Exam December 2010 NRC Exam #5

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.6

Comments: **Appeared on one of last 2 NRC exams (1 of 3)**

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QUESTION 6 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295016	G2.4.34	IR 4.2

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects related to Control Room Abandonment.

Proposed Question:

Due to a fire in the Main Control Room, transfer of control has been established at the Division 1 Remote Shutdown Panel in accordance with AOP-0031, SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM.

Following the transfer which modes of Safety Relief Valve operation will still be functional?

- A. Automatic Depressurization and Relief
- B. Manual and Automatic Depressurization
- C. Relief and Safety
- D. Safety and Manual

Proposed Answer: D.

Explanation:

- A. Once control has been transferred to the RSS panel, ADS and Relief will no longer function.
- B. Manual is correct, but once control is transferred to the RSS panel, ADS will no longer function.
- C. Safety is correct, but once control is transferred to the RSS panel, Relief will no longer function.
- D. Correct – After control is transferred, the SRVs will still operate in the Safety mode and the valves will still be able to be open in Manual. During a fire, power to ENB-PNL02A(B) is de-energized but once transfer of control is complete the SRVs receive power from ENB-PNL03A(B).

Technical Reference(s): AOP-0031 Rev 322 Pg 27 of 122

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0200 Obj 6

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 8 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295019 AK2.02		IR 2.9

Knowledge of the interrelations between Partial or Complete Loss of Instrument Air and the following:
AK2.02 Component cooling water

Proposed Question:

During a loss of instrument air, MWS-AOV134, CCP SURGE TANK MAKEUP VALVE fails (1) and the CCP-TVX128, HEAT EXCHANGER BYPASS VALVE fails (2) .

- A. (1) open; (2) open
- B. (1) open; (2) closed
- C. (1) closed; (2) closed
- D. (1) closed; (2) open

Proposed Answer: C.

Explanation:

- A. MWS-AOV134 will fail closed and CCP-TVX128 will also fail closed.
- B. MWS-AOV134 will fail closed. Part 2 is correct.
- C. Correct – Both valves will fail closed on loss of air.
- D. Part 1 is correct. CCP-TVX128 will fail closed to force flow through the heat exchangers for maximum cooling.

Technical Reference(s): PID-09-01A, PID-09-01B and PID-09-15B

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj 3d, 4a, 5e

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 9 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295021	AK3.05	IR 3.6

Knowledge of the reasons for the following responses as they apply to Loss of Shutdown Cooling:

AK3.05 Establishing alternate heat removal flowpaths

Proposed Question:

The plant has been in a refueling outage for 10 days when the operating shutdown cooling pump trips. Plant procedures require establishing an alternate heat removal flowpath in order to _____.

- A. remove sensible heat present due to extended plant operation
- B. provide flow to temperature monitoring instrumentation
- C. avoid vessel chemistry concerns due to stagnation from low flow conditions
- D. remove decay heat generated by the decay of fission products

Proposed Answer: D.

Explanation:

- A. Although a concern in the first 48 hours after shutdown, at 10 days into the outage, decay heat is the main concern.
- B. Although temperature monitoring is important, removal of decay heat is the prime concern.
- C. Loss of SDC does not affect flow through RWCU therefore this would not cause a concern regarding vessel chemistry.
- D. Correct - Decay heat removal capability must be maintained due to the continuous generation of heat caused by the heat of fission products. Insufficient removal of heat will result in overheating of the core and clad damage.

Technical Reference(s): AOP-00051 Rev 312 Pg 3 of 30

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0543 Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.14

Comments:

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QUESTION 10 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295023	AK1.03	IR 3.7

Knowledge of the operational implications of the following concepts as they apply to Refueling Accidents:
AK1.03 Inadvertent criticality

Proposed Question:

During refueling, the initial indication of inadvertent criticality will first be observed by the (1) who will (2) in accordance with AOP-0027, FUEL HANDLING MISHAPS.

- A. (1) Refuel SRO; (2) direct the fuel handler to move the bundle to a safe location.
- B. (1) ATC operator; (2) insert a reactor scram.
- C. (1) Refuel platform driver; (2) move the bundle to a safe location.
- D. (1) Spotter; (2) notify the ATC to insert a reactor scram.

Proposed Answer: B.

Explanation:

- A. See B.
- B. Correct - First signs of inadvertently criticality will be rising count rates on the SRMs which are monitored by the ATC operator during fuel handling. AOP-0027 directs inserting a reactor scram if inadvertent criticality is observed.
- C. See B.
- D. See B.

Technical Reference(s): AOP-0027 Rev 28, Pg 5 of 24

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP027 Obj. 4, 5, & 9

Question Source: Bank; Oct 2012 Audit #9

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41b.10

Comments:

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QUESTION 11 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group #	1
K/A #	295024	EK3.03	IR 3.6

Knowledge of the reasons for the following responses as they apply to High Drywell Pressure:
EK3.03 Containment venting: Mark III

Proposed Question:

A leak in the drywell has resulted in elevated pressure in the primary containment.

In accordance with EOP-0002, PRIMARY CONTAINMENT CONTROL, containment venting is required BEFORE containment pressure reaches 30 psig with allowance to exceed release rate limits in order to prevent _____.

- A. damage to Auxiliary Building HVAC ductwork
- B. exceeding the code allowable stresses in the SRV tailpipe
- C. containment damage resulting in a large uncontrolled release of radioactivity due to the inability to open ventilation dampers above this pressure
- D. containment damage resulting in a large uncontrolled release of radioactivity due to exceeding the pressure suppression pressure.

Proposed Answer: C.

Explanation:

- A. This containment pressure leg of EOP-0002 has guidance to avoid damage to the AB ductwork, but the direction is to stop venting when containment pressure can not be maintained below 2 psig.
- B. The SRV tailpipe stress limit is based on high suppression pool level of 21'3".
- C. Correct- The Primary Containment Pressure Limit of 30 psig requires venting containment to avoid exceeding this limit. Above this value, containment vent dampers (HVR-AOD127, HVR-AOD128) may be unable to open against this high pressure condition. The inability to lower containment pressure can lead to containment failure and an uncontrolled release of radioactive material.
- D. The pressure suppression pressure is a function of suppression pool level and is used to determine when emergency depressurization is required, not emergency venting of containment.

Technical Reference(s): R-STM-0057 Rev 4 Pg 16 of 69

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-HLO-517 Obj 2.5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments: Validation team recommended to change distractor B from "exceeding the containment design pressure limit" to current reading based on 30 psig being inclusive of 15 psig.

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QUESTION 13 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295026	EA1.01	IR 4.1

Ability to operate and/or monitor the following as they apply to Suppression Pool High Water Temperature:

EA1.01 Suppression pool cooling

Proposed Question:

Following an ATWS, both loops of Residual Heat Removal are in the Suppression Pool Cooling mode with maximum flow through the heat exchangers.

- Reactor power <5%
- RPV Level -50 inches, slowly raising to normal band
- RPV Pressure 0 psig
- Suppression Pool Level 19 feet 11 inches
- Suppression Pool Temp 140°F
- RHR A in Sup Pool Cooling @ 5400 gpm, SWP flow @ 5500 gpm
- RHR B in Sup Pool Cooling @ 5600 gpm, SWP flow @ 5700 gpm

Both divisions of Standby Service Water are in service due to a loss of Normal Service Water.

Based on these conditions, which of the following should be of concern to the operator?

- A. RHR B system flow has exceeded limits.
- B. SWP flow has exceeded limits.
- C. RHR pumps may experience air entrainment due to vortex limit concerns.
- D. RHR pumps may experience cavitation due to NPSH concerns.

Proposed Answer: A.

Explanation:

- A. Correct-RHR shell side flow limit is 5550 gpm.
- B. SWP flow is less than 5800 gpm per loop limit.
- C. Vortex limit concerns are at <10 feet SP Level
- D. NPSH concerns begin at 160°F

Technical Reference(s): SOP-0031 Rev 320 Precaution 2.4.5; EOP-0001, Rev 26 Caution 5

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0511 Obj 6; RLP-STM-0204 Obj 8

Question Source: Bank; April 2010 Audit #13 Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b7

Comments:

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QUESTION 14 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295027	G.2.4.4	IR 4.5

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

Given the following plant conditions:

- | | |
|---|------------------------------|
| • Reactor power | 0% (all rods in) |
| • Reactor level | +33 inches |
| • Reactor pressure | 890 psig |
| • Drywell pressure | 1.5 psid |
| • Drywell temperature | 138°F |
| • Containment temperature | 88°F |
| • Containment pressure | 0.35 psig |
| • Corrected Annulus Differential Pressure | -4.5 inches H ₂ O |

Based on the above conditions, which of the following Emergency Operating Procedures require entry?

- A. EOP-1 only
- B. EOP-1 and EOP-2
- C. EOP-2 only
- D. EOP-2 and EOP-3

Proposed Answer: C.

Explanation:

- A. EOP-1 not applicable. RPV Level >9.7 inches, RPV Press.<1094.7 psig, Drywell pressure<1.68 psid, All rods in.
- B. EOP-1 not applicable. RPV Level >9.7 inches, RPV Press.<1094.7 psig, Drywell pressure<1.68 psid, All rods in.
- C. Correct – Containment pressure is above the 0.3 psig entry condition.
- D. EOP-3 is not applicable. Annulus pressure<3.0 inches H₂O.

Technical Reference(s): EOP-1 Rev 26, EOP-2 Rev 15, EOP-3 Rev 16

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-512 Obj 4; RLP-HLO-514 Obj 4; RLP-HLO-515 Obj 3

Question Source: Bank # RBS-NRC-10

Question History: Last NRC Exam February 2003

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 15 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group #	1
K/A #	295028	EK1.01	IR 3.5

Knowledge of the operational implications of the following concepts as they apply to High Drywell Temperature:

EK1.01 Reactor water level measurement

Proposed Question:

EOP-001A is being implemented due to ATWS conditions. Reactor power is above 5%. The following parameters are indicated in the Main Control Room:

- | | |
|---------------------------|-------------|
| • Reactor Pressure | 450 psig |
| • Wide Range Level | -135 inches |
| • Upset Range Level | +6 inches |
| • Shutdown Range Level | +10 inches |
| • Narrow Range Level | 0 inches |
| • Fuel Zone Range Level | -180 inches |
| • Drywell Temperature | 310°F |
| • Containment Temperature | 165°F |

Which of the following Reactor Level instruments are valid for the current conditions?

- A. Fuel Zone and Upset Range only
- B. Upset Range and Wide Range only
- C. Fuel Zone and Wide Range only
- D. All Reactor Level instruments are invalid

Proposed Answer: C.

Explanation:

- A. Fuel Zone use is allowed, but Upset Range is only valid down to 53 inches due to DW above 300°F.
- B. Wide Range use is allowed, but Upset Range is only valid down to 53 inches due to DW above 300°F.
- C. Correct – Fuel Zone is valid above -310 inches and Wide Range is valid above -142 inches.
- D. Fuel Zone and Wide Range are both above the Minimum Indicated Water Level for the provided temperatures and Part 1 of Caution 1 is satisfied due to the provided RPV Pressure and Drywell Temperature being in the Safe Zone of the RPV Saturation Temperature Curve.

Technical Reference(s): EOP-0001, Rev 26 Caution 1

Proposed references to be provided to applicants during examination: **EOP-0001 Caution 1**

Learning Objective: R-LPOPS-HLO-511 Obj 6

Question Source: Bank – RBS-NRC-558

Question History: Last NRC Exam February 1999

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 16 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group #	1
K/A #	295031	EK3.01	IR 3.9

Knowledge of the reasons for the following responses as they apply to Reactor Low Water Level:

EK3.01 Automatic depressurization system actuation

Proposed Question:

The reason for actuation of the Automatic Depressurization System is to_____.

- A. lower reactor pressure to allow recovery of reactor water level by low pressure injection systems
- B. lower the driving head in the reactor pressure vessel to reduce inventory loss during a Loss of Coolant Accident.
- C. limit the release of fission products due to isolation valve leak-by after a Loss of Coolant Accident
- D. preclude exceeding the ASME Code limit for peak pressure in the reactor coolant pressure boundary

Proposed Answer:

A.

Explanation:

- A. Correct – The ADS system is designed to lower reactor pressure during a small break LOCA coincident with failure of High Pressure Core Spray to allow low pressure ECCS to restore reactor water level.
- B. Although lowering reactor pressure does reduce inventory loss during a LOCA, this is not the reason for ADS actuation.
- C. Although lowering reactor pressure does limit the release of fission products due to reduced driving head, this is not the reason for ADS actuation.
- D. The safety relief valves (which are used as part of ADS) provide the stated function in this distracter, but this is not the reason for ADS actuation.

Technical Reference(s): RLP-STM-0202 Rev 2 Pg 4 of 36

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0202 Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 17 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295037	EA2.03	IR 4.3

Ability to determine and/or interpret the following as they apply to Scram Condition Present and Reactor Power above APRM Downscale or Unknown:
EA2.03 SBLC tank level

Proposed Question:

During ATWS conditions, the CRS has directed injection with Standby Liquid Control (SLC). If the initial level of the SLC tank is 3350 gallons, at what SLC tank level would you report to the CRS that Hot Shutdown Boron Weight has been injected?

- A. 1769 gallons
- B. 1819 gallons
- C. 2674 gallons
- D. 2724 gallons

Proposed Answer: C.

Explanation:

- A. This is the correct value for Cold Shutdown Boron Weight; stem asked for Hot Shutdown Boron Weight.
- B. This is an interpolated value for Cold Shutdown Boron Weight. Additionally, the stem asked for Hot Shutdown Boron Weight.
- C. Correct - Using OSP-0053 Hard Card, initial tank level of 3300 gallons should be used based on note above table. Hot Shutdown Boron Weight is 69 lbs, so first column is used yielding 2674 gallons.
- D. This is an interpolated value for Hot Shutdown Boron Weight. Interpolation is not allowed. The lower tank value must be used as annotated in the note above the table.

Technical Reference(s): R-STM-0201 Rev 7 Pg 7 of 37; OSP-0053 Rev 20 Pg 46 of 71

Proposed references to be provided to applicants during examination: **OSP-0053 Rev 20 Pg 46 of 71**

Learning Objective: RLP-STM-0201 Obj 1a

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒2

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 18 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295038 EK2.04		IR 3.9

Knowledge of the interrelations between High Off-Site Release Rate and the following:

EK2.04 Stack-gas monitoring system: Plant-specific

Proposed Question:

Which of the following radiation monitors is utilized in determining when a High Offsite Radioactive Release is occurring that requires entry into the Emergency Operating Procedures?

- A. RMS-RE16, CONTAINMENT POST ACCIDENT MONITOR
- B. RMS-RE118, TURBINE BLDG VENT
- C. RMS-RE124, COND DEMIN OFFGAS VENTILATION
- D. RMS-RE125, MAIN PLANT EXHAUST

Proposed Answer: D.

Explanation:

- A. RMS-RE125, 5A(B) & 6A(B) are used in determining entry into EOP-0003 due to High Offsite Release.
- B. RMS-RE125, 5A(B) & 6A(B) are used in determining entry into EOP-0003 due to High Offsite Release.
- C. RMS-RE125, 5A(B) & 6A(B) are used in determining entry into EOP-0003 due to High Offsite Release.
- D. Correct – RMS-RE125 in Alarm condition indicates a High Offsite Release and requires entry into EOP-0003.

Technical Reference(s): EOP-0003 Rev 16 ; EIP-2-001 Rev 24 Pg 21 of 158

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-515 Obj 2

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.11

Comments: Validation team suggested changing distractor A from RE110, Aux Bldg to one that would not enter EOPs for any reason; RMS-RE110 does have an EOP entry level.

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QUESTION 19 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group #	1
K/A #	600000	AA1.06	IR 3.0

Ability to operate and/or monitor the following as they apply to Plant Fire On Site:
AA1.06 Fire Alarm

Proposed Question:

Firewatch personnel have identified a fire and operated a fire pull station, FPS-4CB in the HCU East Area of the Reactor Building 114' el. Manipulation of this pull station will alarm (1) and (2) result in actuation of a suppression system.

- A. (1) locally only; (2) will
- B. (1) locally only; (2) will not
- C. (1) locally and in the Main Control Room, (2) will not
- D. (1) locally and in the Main Control Room, (2) will

Proposed Answer: C.

Explanation:

- A. Annunciation will also occur in the MCR. There is no suppression associated with this pull station.
- B. Annunciation will also occur in the MCR. Part 2 is correct.
- C. Correct – Alarms will sound locally as well as in the MCR. There is no suppression associated with this pull station.
- D. Part 1 is correct. There is no suppression associated with this pull station.

Technical Reference(s): R-STM-0250, Rev 6 Pg 47 of 65 ; Pre-Fire Strategy RB-114-004, Rev 3 Pg 2 of 9

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0250 Obj 3

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒4

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 21 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295002 AK3.09		IR 3.2

Knowledge of the reasons for the following responses as they apply to Loss of Main Condenser Vacuum: AK3.09 Reactor power reduction

Proposed Question:

During periods of deteriorating condenser vacuum, reactor power is reduced in accordance with AOP-0005, LOSS OF MAIN CONDENSER VACUUM/TRIP OF CIRCULATING WATER PUMP, in order to avoid _____.

- A. explosive hydrogen concentrations in the main condenser
- B. a cooldown rate of the turbine rotor in excess of 125°F per hour
- C. high airborne levels due to loss of SJAE loop seals
- D. turbine blade damage due to buffeting

Proposed Answer: D.

Explanation:

- A. Operation of the SJAE and Offgas system remove non-condensable from the condenser preventing a buildup of hydrogen.
- B. A higher condenser pressure would cause heating of the turbine, not accelerate the cooldown.
- C. Loop seals are of sufficient height to not be affected by the change in pressure.
- D. Correct – As Turbine exhaust backpressure rise, excess windage will be experienced resulting in buffeting and heat of the last stage turbine blading.

Technical Reference(s): R-STM-0110 Rev 10, Pg 15 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP005 Obj 6

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.5

Comments:

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QUESTION 22 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295007	G.2.2.44	IR 4.2

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

Proposed Question:

The plant was operating at 100% power when a Main Turbine trip occurred. On H13-P601, the SRV LOW-LOW SET (LLS) LOGIC indication lights (both A & B) are all illuminated.

Based on the above conditions, the next opening of Safety Relief Valves due to high reactor pressure will occur at (1) psig. If the LLS LOGIC RESET pushbutton is depressed, the next opening of Safety Relief Valves will occur at (2) psig.

- A. (1) 1133; (2) 1133
- B. (1) 1133; (2) 1063
- C. (1) 1063; (2) 1143
- D. (1) 1063; (2) 1133

Proposed Answer: D.

Explanation:

- A. 1133 is incorrect because the status lights indicate that LLS is active making the initial SRV opening pressure 1063 psig. Part 2 is correct.
- B. 1133 is incorrect because the status lights indicate that LLS is active making the initial SRV opening pressure 1063 psig. Part 2 is incorrect because depressing the pushbutton restores the initial SRV opening setpoint to 1133 psig.
- C. Part 2 is correct. 1143 psig is the opening pressure for subsequent SRV openings, not the initial opening following reset of the LLS logic.
- D. Correct – All LLS illuminated indicates that LLS has actuated therefore the initial opening of SRVs will occur at 1063 psig. After the RESET pushbutton is depressed, the initial setpoint of 1133 psig will be restored.

Technical Reference(s): R-STM-0109 Rev 12 Pg 33 of 95 and Table 3

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109 Obj 4, 10

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments: Validation team suggested changing the verb tense in the stem to current version (initial opening to next opening).

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QUESTION 23 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295008	AK1.01	IR 4.0

Knowledge of the operational implications of the following concepts as they apply to High Containment Temperature (Mark III Containment Only):
AK1.01 Containment pressure: Mark III

Proposed Question:

Following a Loss of Coolant Accident (LOCA), drywell bypass leakage has resulted in the following conditions:

- Reactor Level -80 inches, stable
- HVR-UC1A & 1B running
- Containment Temperature 105°F, slowly rising
- Containment Pressure 0.85 psig, slowly rising

Which of the following actions can be performed to address the rising containment pressure?

- A. Normal containment venting with HVR-FN14
- B. Normal containment venting with Standby Gas Treatment
- C. Restoration of chilled water (HVN) to the containment unit coolers
- D. Alignment of Service Water (SWP) to the containment unit coolers

Proposed Answer: D.

Explanation:

- A. Although venting with HVR-FN14 will lower containment pressure, it can not be performed below Level 2.
- B. Although venting with Standby Gas Treatment will lower containment pressure, it can not be performed below Level 2.
- C. HVN can not be restored due to Level 2 isolation signal present.
- D. Correct - The only method of lowering containment pressure with the given conditions is to supply SWP as cooling water to the containment unit coolers.

Technical Reference(s): AOP-0003 Rev 33 Pg 15,18 of 22; SOP-0059 Rev 34 Pg 44 of 83

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0403 Obj 4.2

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.5

Comments:

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QUESTION 24 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295015 AA2.02		IR 4.1

Ability to determine and/or interpret the following as they apply to Incomplete Scram:
AA2.02 Control rod position

Proposed Question:

A reactor scram, with a failure of all control rods to fully insert has occurred coincident with a failure of the Full Core Display.

In accordance with SOP-0071, ROD CONTROL AND INFORMATION SYSTEM, individual control rod position information is determined at _____.

- A. either Rod Action Control System (RACS) cabinet
- B. RACS Cabinet 1 only
- C. RACS Cabinet 2 only
- D. Rod Gang Drive System (RGDS) cabinet

Proposed Answer: A.

Explanation:

- A. Correct – Rod position information can be determined from either H13-P651 or H13-P652.
- B. Information concerning rod position can be obtained from both RACS cabinets.
- C. Information concerning rod position can be obtained from both RACS cabinets.
- D. Information concerning rod position can be obtained from both RACS cabinets, but not from RGDS.

Technical Reference(s): SOP-0071 Rev 29 Pg 13 of 49; R-STM-0500 Rev 4 Pg 31 of 46

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0500 20b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.6

Comments:

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QUESTION 25 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295029 EK2.06		IR 3.4

Knowledge of the interrelations between High Suppression Pool Water Level and the following:
EK 2.06 SRVs and discharge piping

Proposed Question:

The highest suppression pool level where operation of a Safety Relief Valve will not result in exceeding the code allowable stresses in the SRV tail pipe is _____.

- A. 21' 6"
- B. 21' 3"
- C. 21' 0"
- D. 20' 0"

Proposed Answer: B.

Explanation:

- A. 21'6" is the Maximum Pressure Suppression Primary Containment Water Level
- B. Correct – 21'3" defines the SRV Tail Pipe Level Limit as described in EPSTG0002.
- C. 21'0" is the level which requires a reactor scram per EOP-0002.
- D. 20'0" is the upper level of the Suppression Pool normal band.

Technical Reference(s): R-STM-0057 Rev 4 Pg 12, 16 of 69, EPSTG*0002 App A, A-55

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-517 Obj 3.21

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.9

Comments:

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QUESTION 26 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295036	EA1.03	IR 2.8

Ability to operate and/or monitor the following as they apply to Secondary Containment High Sump/Area Water Level:
EA1.03 Radwaste

Proposed Question:

Following a plant scram, EOP-0003, SECONDARY CONTAINMENT AND RADIOACTIVE RELEASE CONTROL, has been entered due to high level in DFR-TK5A and TK5B, AUXILIARY BLDG FLOOR DRAIN SUMPS.
No operator action has been taken to address the sump level issue.

Based on the above conditions, level in the _____ will rise as pumps associated with the 2 tanks automatically run to lower level in the sumps.

- A. Suppression Pool
- B. Waste Collector Tank
- C. Floor Drain Collector Tank
- D. Regenerant Waste Tank

Proposed Answer: C.

Explanation:

- A. Contents of these tank can be pumped back to the suppression pool, but manual operator action is required to open DFR-MOV146. The stem states that no operator action has been taken.
- B. WC Tank receives equipment drain waste. Tank 5A and 5B pump to Floor Drains in Radwaste.
- C. Correct – With suppression pool pumpback closed (no operator action taken), the content of these tanks is pumped to the Floor Drain Collect tank in Radwaste.
- D. RW tanks receive water from Chem labs, sample panels and URC systems.

Technical Reference(s): R-STM-0609 Rev 5 Pg 14 of 69.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0609 Obj 2b, 3e, 4b, 7a

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 27 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	1	Group # 2	
K/A #	500000	EK3.03	IR 3.0

Knowledge of the reasons for the following responses as they apply to High Primary Containment Hydrogen Concentrations:
EK3.03 Operation of hydrogen and oxygen recombiners

Proposed Question:

The Hydrogen Recombiners are secured when containment hydrogen concentrations exceed the Hydrogen Deflagration Overpressure Limit (HDOL) because _____.

- A. insufficient amounts of are oxygen available to support recombination
- B. the heat produced from recombination at this concentration will overload cooling capability of containment unit coolers
- C. the hydrogen igniters alone are a sufficient method of hydrogen removal
- D. the recombiner may become an ignition source leading to rapid uncontrolled combustion

Proposed Answer: D.

Explanation:

- A. See "D".
- B. See "D".
- C. See "D".
- D. Correct - When the HDOL is exceeded, this indicates that containment can be over pressurized if a deflagration event were to occur. To avoid deflagration all potential ignition sources are removed (igniters, recombiners) and containment purge is the only authorized method of hydrogen removal.

Technical Reference(s): EPSTG*0002 App A, A-15 ; App B, B-14-13 & B-14-23

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-517 Obj 2.4

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 28 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	203000	K5.02	IR 3.5

Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: Injection Mode (Plant Specific):
K5.02 Core cooling methods

Proposed Question:

A plant transient has resulted in the following conditions:

RPV Level -180 inches

RHR A is the only available injection source and is injecting 5000 gpm.

Based on the given conditions, the adequate core cooling method in effect is _____

- A. submergence.
- B. steam cooling with injection.
- C. steam cooling without injection.
- D. spray cooling.

Proposed Answer: B.

Explanation:

- A. Adequate core cooling by submergence required level $>-162''$,
- B. Correct. With injection present, the fuel can be cooled by steam at level above $186''$.
- C. At $-180''$, level is above the minimum zero injection reactor water level limit ($-200''$), but this level is only valid with no injection into the core. Per the stem, RHR A is injecting 5000 gpm.
- D. Only HPCS/LPCS can provide spray cooling if they are injecting at least 5000 gpm. RHR A injects through a nozzle not a spray sparger.

Technical Reference(s): EOP-0001 Rev 26 ACC Chart

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-511 Obj.3, 4

Question Source: Bank; Nov 2010 Audit #28

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒ 2

10 CFR Part 55 Content: 55.41b.5

Comments:

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QUESTION 29 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	205000	K2.01	IR 3.1

Knowledge of the electrical power supplies to the following:
K2.01 Pump motors

Proposed Question:

The power supply to Residual Heat Removal (RHR) Pump "B" is _____.

- A. ENS-SWG1A
- B. ENS-SWG1B
- C. NNS-SWG1A
- D. NNS-SWG1B

Proposed Answer: B.

Explanation:

- A. ENS-SWG1A is the power supply to RHR Pump "A"
- B. Correct – ENS-SWG1B is the power supply to RHR "B"
- C. Although a 4160VAC bus, this bus is non-safety related.
- D. Although a 4160VAC bus, this bus is non-safety related.

Technical Reference(s): SOP-0031, Rev 320 Pg 141 of 187

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204 Obj. 11c

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 30 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	209001	A2.06	IR 3.2

Ability to (a) predict the impacts of the following on the Low Pressure Core Spray System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.06 Inadequate system flow

Proposed Question:

A small break LOCA has resulted in a High Drywell Pressure initiation signal. All ECCS pumps have started. RHR A has been aligned in the Suppression Pool Cooling mode but no other operator action has been taken. The unit operator observes the following:

- | | |
|---|------------------|
| • Reactor Pressure | 750 psig |
| • Suppression Pool Level | 15 feet 3 inches |
| • Suppression Pool Temp | 101°F |
| • E21-MOVF011, LPCS MIN FLOW VLV | CLOSED |
| • E21-MOVF012, LPCS TEST RETURN VLV | CLOSED |
| • E12-MOVF024A, RHR PUMP A TEST RET TO SUP PL | OPEN |

(1) What is the impact of continued operation in this condition and (2) What action should the operator take to mitigate this condition?

- A. (1) LPCS Pump degradation due to inadequate flow; (2) Open the minimum flow valve
- B. (1) LPCS Pump degradation due to inadequate flow; (2) Open the Test Return to the Suppression Pool.
- C. (1) ECCS pump cavitation due to inadequate net positive suction head; (2) Raise suppression pool level.
- D. (1) ECCS pump cavitation due to inadequate net positive suction head; (2) Maximize suppression pool cooling with all available systems.

Proposed Answer: A.

Explanation:

- A. Correct – The LPCS is dead-headed. Per P&L 2.2 after 60 seconds of operation in this condition pump degradation will be experienced. The minimum flow must be opened to establish flow.
- B. Part 1 is correct, but the Test Return to the Suppr Pool can't be opened due to the RHR A alignment. Ref. P&L 2.15
- C. Although suppression pool temperature is elevated and suppression pool level is below normal, cavitation is not expected to occur at temperatures below 160°F and suppression pool levels above 10 feet,
- D. See "C",

Technical Reference(s): SOP-0032 Rev 22 Page 3 and 4 of 29

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205 Obj 5, 8

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.5 Comments:

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QUESTION 31 Rev 0

Examination Outline Cross-Reference:	Level	RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
	Tier #	2	Group # 1
	K/A #	209001 A4.07	IR 2.7

Ability to manually operate and/or monitor in the control room:
A4.07 Fill pump (LPCS)

Proposed Question:

The unit operator observed the following indications on H13-P601 for E21-C002 LPCS/RHR DIV 1 LINE FILL PUMP:

Green Light	ILLUMINATED
Amber Light	ILLUMINATED
Red Light	OFF

Following receipt of these indications, which of the following alarms should the operator anticipate?

- A. LPCS/LPCI A INJ VALVE RPV PRESS LOW AND
LPCS INJECTION LINE PRESSURE HI/LOW
- B. RHR PUMP A DISCH PRESSURE HI/LOW AND
LPCS PUMP E21-C001 IN MAUAL OVERRIDE
- C. LPCS INJECTION LINE PRESSURE HI/LOW AND
RHR PUMP A DISCH PRESSURE HI/LOW
- D. LPCS PUMP E21-C001 IN MAUAL OVERRIDE AND
LPCS/LPCI A INJ VALVE RPV PRESS LOW

Proposed Answer: C.

Explanation:

- A. The second alarm is correct, but the first alarm would not be present as a result of a tripped line fill pump.
- B. First alarm is correct, but second would not be present as a result of a tripped line fill pump, the amber light would be lit if the main pump was in manual override which adds to the plausibility of this distractor.
- C. Correct - The indications provide are that of a tripped line fill pump. This condition will result in a low pressure condition in the injection lines of LPCS and RHR A which have a common line fill system.
- D. Neither of these alarms would be present as a result of a tripped line fill pump., the amber light would be lit if the main pump was in manual override which adds to the plausibility of this distractor.

Technical Reference(s): ARP-601-20 Rev 305 Pg 19 of 38 & ARP-601-21 Rev 312 Pg 31 of 79

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205 Obj 6, 7, 11f, 19

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 32 Rev 0

Examination Outline Cross-Reference:

Level RO ☒ SRO ☐
Tier # 2 Group # 1
K/A # 209002 K1.04 IR 3.8

Knowledge of the physical connections and/or cause-effect relationships between High Pressure Core Spray System (HPCS) and the following:
K1.04 HPCS diesel generator: BWR-5,6

Proposed Question:

The High Pressure Core Spray (HPCS) Diesel Generator is running and paralleled to offsite power in support of STP-309-0613, DIVISION III DIESEL GENERATOR 24 HOUR RUN.

A loss of offsite power occurs with a concurrent Loss of Coolant Accident (LOCA).

Which of the following represents the resulting status of the individual High Pressure Core Spray system components and closure time of the HPCS Pump Breaker?

E22-ACB01, HPCS D/G OUTPUT BRKR	E22-ACB02, HPCS PUMP BRKR (breaker closure time)	E22-MOVF012, HPCS MIN FLOW VALVE TO SUPPRESSION POOL
A. Closed	immediately	Closed
B. Closed	30 seconds	Open
C. Open	immediately	Closed
D. Open	30 seconds	Open

Proposed Answer: A.

Explanation:

- A. Correct – Output was initially closed and will remain closed. The pump breaker will close immediately and the minimum flow valve will be closed because flow will be maintained through the injection valve.
- B. Load shedding does not occur because bus voltage was never lost. Additionally, 30 second sequencer applies to SWP-P2C which supplies cooling to the Div 3 DG. HPCS pump breaker closure is immediate.
- C. DG output breaker remains closed to supply the bus due to the loss of offsite power.
- D. See C; Additionally, 30 second sequencer applies to SWP-P2C which supplies cooling to the Div 3 DG. HPCS pump breaker closure is immediate.

Technical Reference(s): R-STM-0309H Rev 12, Pg 4, 48, 49 of 78

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0203 Obj 13d, RLP-STM-0309H Obj 15f

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.8

Comments:

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QUESTION 33 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	211000	A1.09	IR 4.0

Ability to predict and/or monitor changes in parameters associated with operating the Standby Liquid Control System controls including:
A1.09 SBLC system lineup

Proposed Question:

During an ATWS condition, Standby Liquid Control "A" was initiated with SLC tank level initially at 3800 gallons. 10 minutes later the following conditions exist:

SLC Tank Level	3800 gallons
SLC discharge pressure	1400 psig

SLC Pump A was secured and SLC Pump B was initiated. 10 minutes later the following conditions exist:

SLC Tank Level	3800 gallons
SLC discharge pressure	1400 psig

Which of the following is responsible for both sets of indications above? (Assume a single failure)

- A. Squib valve failure
- B. Suction valve failure
- C. Injection line manual isolation valve left closed following maintenance
- D. Test tank inlet valve left closed following maintenance

Proposed Answer: C.

Explanation:

- A. A single squib valve failure would not affect both subsystems.
- B. A single suction valve failure would not affect both subsystems.
- C. Correct – The closure of the common injection line manual isolation would produce the indicated parameters due to "dead-heading" the system.
- D. The test tank inlet valve is normally closed and therefore would not produce the given indications.

Technical Reference(s): PID 27-16A

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0201 Obj 2, 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.5

Comments:

**March 2014 River Bend Station
NRC Initial License Examination
Reactor Operator**

QUESTION 34 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	212000	K3.06	IR 4.0

Knowledge of the effect that a loss or malfunction of the Reactor Protection System will have on the following:
K3.06 Scram air header solenoid operated valves

Proposed Question:

The plant is operating at 100% power when a loss of Reactor Protection System (RPS) "A" occurs. RPS "B" is unaffected and remains energized.

This results in the 145 scram pilot valves (1) and the backup scram valves (2) .

- A. (1) repositioning;
 (2) de-energizing
- B. (1) remaining unchanged;
 (2) remaining de-energized
- C. (1) repositioning;
 (2) energizing
- D. (1) remaining unchanged;
 (2) remaining energized

Proposed Answer: B.

Explanation:

- A. The scram pilot valve will not reposition on a single solenoid being de-energized and the backup scram valves are normal de-energized.
- B. Correct – The "A" solenoid on the scram pilot valves will de-energize but the pilot valve will not reposition as long as the "B" solenoid is still energize. The backup scram valve are normally de-energize and will remain as such with the loss of only RPS A.
- C. The scram pilot valve will not reposition on a single solenoid being de-energized and the backup scram valves require loss of both RPS A and B to energize.
- D. Part 1 is correct, but the backup scram valves are normally de-energize and will remain de-energize with the loss of only one RPS bus.

Technical Reference(s): R-STM-0508 Rev 6 Pg 20-22 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 8d

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.6

Comments:

**March 2014 River Bend Station
NRC Initial License Examination
Reactor Operator**

QUESTION 35 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 1	
K/A #	212000	G.2.1.28	IR 4.1

Knowledge of the purpose and function of major Reactor Protection System components and controls.

Proposed Question:

With the Reactor Mode Switch in the RUN position, which of the following represents the status of the indicated interlocks?

	IRM Hi Flux <u>Scram</u>	MSIV Closure <u>Scram</u>	RPV Hi Water Level <u>Scram</u>
A.	Enabled	Bypassed	Bypassed
B.	Bypassed	Enabled	Bypassed
C.	Enabled	Bypassed	Enabled
D.	Bypassed	Enabled	Enabled

Proposed Answer: D.

Explanation:

- A. IRM Scram is not enabled in RUN, nor is the MSIV Closure is not bypassed, nor is the high level scram bypassed.
- B. IRM scram is bypassed and the MSIV closure scram is enabled, but the high level scram is not bypassed.
- C. IRM scram is not enabled in RUN, nor is the MSIV scram bypassed, but the high level scram is enabled.
- D. Correct – The IRM scram is bypassed in RUN and the MSIV closure scram is enabled, and the high water level scram is enabled.

Technical Reference(s): R-STM-0508 Rev 6 Pg 57 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 3b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.6

Comments:

**March 2014 River Bend Station
NRC Initial License Examination
Reactor Operator**

QUESTION 36 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	215003	K5.01	IR 2.6

Knowledge of the operational implications of the following concepts as they apply to Intermediate Range Monitor (IRM) System:
K5.01 Detector operation

Proposed Question:

During a plant startup the high voltage power supply for IRM 'D' fails and drops to 0 volts. The Reactor Mode Switch is in the START / HOT STANDBY position and all other IRMs are OPERABLE.

How will the RC&IS and RPS respond to the change in IRM Channel "D" detector supply voltage?

- A. BOTH a control rod withdrawal block and a RPS half scram signal are generated.
- B. ONLY a control rod withdrawal block signal is generated.
- C. ONLY a RPS half scram signal is generated
- D. NEITHER a control rod withdrawal block NOR a RPS half scram signal is generated

Proposed Answer: A.

Explanation:

- A. Correct – Both the IRM Rod Block and RPS trip signals are generated due to low voltage to the detector.
- B. RPS trip signal will also be generated.
- C. RC&IS Rod Block will also be generated.
- D. With the Mode Switch in RUN this would be correct, but with the Mode Switch in Start/Hot Stby both RPS trip and RC&IS rod block will be generated.

Technical Reference(s): R-STM-0503 Rev 7 Pg 100-101 of 111

Proposed references to be provided to applicants during examination: None

Learning Objective: R-STM-0503 Rev 7 Obj 13

Question Source: Bank # RBS-NRC-1

Question History: Last NRC Exam February 2003

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.2

Comments:

**March 2014 River Bend Station
NRC Initial License Examination
Reactor Operator**

QUESTION 37 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	215004	K6.01	IR 3.2

Knowledge of the effect that a loss or malfunction of the following will have on the Source Range Monitor (SRM) System:
K6.01 RPS

Proposed Question:

A plant startup is in progress and the reactor is nearing criticality when RPS Motor Generator Set A generator output breaker trips.

What is the status of the SRM A indicating lights on H13-P680?

- A. The UPSC TRIP, ALARM OR INOP, and DNSC lights will ALL be lit.
- B. ONLY the ALARM OR INOP, and DNSC lights will be lit.
- C. ONLY the ALARM OR INOP light will be lit.
- D. NONE of the indicating lights will be lit.

Proposed Answer: A.

Explanation:

- A. Correct - All indicating lights illuminate on loss of power.
- B. All indicating lights illuminate on loss of power.
- C. All indicating lights illuminate on loss of power.
- D. All indicating lights illuminate on loss of power.

Technical Reference(s): AOP-0010 Rev 19, Step 2.1 Pg 3 of 21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503 Obj 7, 35

Question Source: Bank # RBS-NRC-908

Question History: Last NRC Exam Sept 2004 #35

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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NRC Initial License Examination
Reactor Operator**

QUESTION 38 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	215005	K1.09	IR3.6

Knowledge of the physical connections and/or cause-effect relationships between Average Power Range Monitor/Local Power Range Monitor System and the following:
K1.09 Reactor recirculation system: BWR-5,6

Proposed Question:

The Average Power Range Monitors (APRM) Upscale Thermal trip is biased by a Reactor Recirculation flow signal generated from differential pressure signals measured across _____.

- A. the core plate
- B. the recirculation loop elbows
- C. calibrated jet pumps
- D. all jet pumps

Proposed Answer: B.

Explanation:

- A. Although core flow can be calculated from core plate differential pressure, this signal is not input to the APRMs.
- B. Correct – The Recirculation Flow Monitoring Subsystem consists of flow converters which receive a differential pressure signal provided from elbow taps on both recirculation loops.
- C. Although core flow can be calculated from calibrated jet pump flows, this signal is not input to the APRMs.
- D. Although core flow can be calculated from total jet pump flow, this signal is not input to the APRMs..

Technical Reference(s): R-STM-0503 Rev 7 Pg 61 of 111

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503 Obj 27

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

**March 2014 River Bend Station
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QUESTION 39 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	217000	A3.03	IR 3.7

Ability to monitor automatic operations of the Reactor Core Isolation Cooling System (RCIC) including:
A3.03 System pressure

Proposed Question:

The plant is recovering from a loss of Offsite Power. Reactor Core Isolation Cooling (RCIC) is injecting in the reactor vessel as efforts are made to cool down the reactor pressure vessel to Cold Shutdown conditions.

What is the expected response of RCIC parameters as reactor pressure reduction begins?

<u>System flow</u>	<u>Discharge Pressure</u>	<u>Turbine Speed</u>
A. Lowers	rises	lowers
B. Remains constant	rises	rises
C. Rises	lowers	rises
D. Remains constant	lowers	lowers

Proposed Answer: D.

Explanation:

- A. Flow remains constant at 600 gpm, Discharge pressure lowers as reactor pressure lowers, Turbine speed does lower as reactor pressure lowers.
- B. Flow does remain constant, but discharge pressure lowers as reactor pressure lowers and turbine speed lowers because less work is required to maintain the same flowrate at lower pressure.
- C. Flow remains constant at 600 gpm, Discharge pressure does lower, but turbine speed lowers because less work is required to maintain the same flowrate at lower pressure.
- D. Correct- Flow remains constant at 600 gpm, discharge pressure lowers as reactor pressure lowers and turbine speed lowers because less work is required to maintain the same flowrate at lower pressure.

Technical Reference(s): R-STM-0209 Rev 10 Pg 10 of 52

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209 Obj 12

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒ 4

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 40 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	218000	K6.07	IR 3.4

Knowledge of the effect that a loss or malfunction of the following will have on the Automatic Depressurization System:

K6.07 Primary containment instrumentation.

Proposed Question:

A LOCA has occurred inside the drywell; however, the "A" high drywell pressure input into Automatic Depressurization System (ADS) logic channel "A" failed low. Conditions are as follows:

- Drywell pressure is 5 psid
- RHR "A", "B" and "C" are operating in the LPCI mode
- Reactor water level is at Level 2 and decreasing

Upon reaching Level 1, _____.

- A. ADS valves will NOT open
- B. ADS valves will open immediately
- C. ADS valves will open after a 105 second time delay
- D. ADS valves will open after a 5 minute + 105 second time delay

Proposed Answer: C.

Explanation:

- A. A single instrument failure will not prevent the ADS system from actuating on a valid signal. The Div 2 solenoid will energize to open all ADS valves.
- B. The logic will not be satisfied immediately. The 105 second timer must be timed out to complete the logic.
- C. Correct-The failed instrument will not prevent the logic from being complete since the Div 2 solenoid will energize to open the ADS valve.
- D. Even though the 5 minute timer does bypass the drywell pressure signal, the Div 2 logic is not affected by the Div 1 failure, so the expiration of the 5 minute timer is not required to open the valves.

Technical Reference(s): R-STM-0202 Rev 2 Pg 12 of 36

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0202 Obj 12.6

Question Source: Bank # 2007 NRC Exam # 41

Question History: Last NRC Exam RBS 2007 NRC Exam

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

**March 2014 River Bend Station
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QUESTION 41 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	223002	K4.08	IR 3.3

Knowledge of Primary Containment Isolation System/Nuclear Steam Supply Shut-Off design feature(s) and/or interlocks which provide for the following:
K4.08 Manual defeating of selected isolations during specified emergency conditions.

Proposed Question:

During an ATWS, the following conditions exist:

- RPV Level -65 inches, steady
- Drywell pressure 1.52 psid, lowering

The CRS has directed the defeating of instrument air isolation interlocks. This task is accomplished by _____.

- A. depressing the Inboard and Outboard isolation seal-in reset pushbuttons
- B. placing the associated keylock switch to Emergency
- C. removing the associated relay
- D. lifting associated leads

Proposed Answer: B.

Explanation:

- A. Due to level being <-43 inches, the logic can not be reset with the reset pushbuttons.
- B. Correct – EOP Enclosure 16 provides guidance under these conditions to place the associated keylock switch to Emergency.
- C. Although some Enclosures defeat isolation interlocks by removing relays, the IAS interlock is bypassed via a keylock switch.
- D. See C.

Technical Reference(s): EOP-0005 Rev 314 Pg 59 of 139

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-516 Obj 1

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 42 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	239002 A4.02		IR 3.6

Ability to manually operate and/or monitor in the control room:
A4.02 Tail pipe temperatures

Proposed Question:

With the plant operating at 100% power, the ADS/SRV VALVE LEAKING annunciator alarms in the Main Control Room.

Which of the following ranges of temperatures include the tailpipe temperature that the unit operator would expect to see on B21-R614 Temperature Recorder for the open Safety Relief Valve?

- A. 500°F - 575°F
- B. 425°F - 499°F
- C. 350°F - 426°F
- D. 275°F - 349°F

Proposed Answer: D.

Explanation:

- A. See "D".
- B. See "D".
- C. See "D".
- D. Correct - At 100% power, reactor pressure is slightly over 1000 psia. Since expansion through throttle valve is assumed to be an isenthalpic event, using a Mollier diagram and from 1000 psia on the saturation curve moving on the line of constant enthalpy to the left towards atmospheric conditions yields at temperature of approximately 320°F.

Technical Reference(s): Mollier Diagram

Proposed references to be provided to applicants during examination: **Steam Tables and Mollier Diagram**

Learning Objective: RLP-STM-0109 Obj 26a

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 43 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	259002	K4.10	IR 3.4

Knowledge of Reactor Water Level Control System design feature(s) and/or interlocks which provide for the following:

K4.10 Three element control (main steam flow, reactor feedwater flow, and reactor water level provide input)

Proposed Question:

With the plant operating at 100% power, a single steam flow signal input to the Feedwater Level Control (FWLC) System fails downscale. If the FWLC system is in 3 Element control, how will actual vessel level and feedwater regulating valve position respond. Assume no operator action has been taken.

<u>Actual Level</u>	<u>Valve Position</u>
A. Rise	Open
B. Rise	Close
C. Lower	Open
D. Lower	Close

Proposed Answer: D.

Explanation:

A. See "D".

B. See "D".

C. See "D".

D. Correct - With a steam flow input downscale, the Steam/Feed Summer will sense higher feed flow than steam flow and will send a close signal to the regulating valves in an attempt to stabilize level as a result of this actual level will lower.

Technical Reference(s): R-STM-0107 Rev 27 Pg 71 of 105

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107B Obj 14d

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 44 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	259002	G.2.1.32	IR 3.8

Ability to explain and apply system limits and precautions of Reactor Water Level Control System.

Proposed Question:

During power operations, the following Feedwater System (FWS) conditions exist:

- | | |
|----------------------------|-------------------|
| • FWS-P1A | RUNNING, 355 amps |
| • FWS-P1B | OFF |
| • FWS-P1C | OFF |
| • Master Level Controller | AUTO |
| • FWS Regulating Valve A | AUTO, 90% open |
| • FWS Regulating Valve B | MANUAL, 20% open |
| • FWS Regulating Valve C | MANUAL, 0% open |
| • Startup Regulating Valve | AUTO, 100% open |

Based on the above conditions, which of the following should be of concern?

- A. Operating FWS pump amperage loading indicates excessive load on the pump.
- B. Startup FWS Regulating Valve has exceeded flow limitations.
- C. Operating FWS pump amperage loading indicates a minimum flow valve malfunction.
- D. In service FWS Regulating Valve has inadequate margin for valve modulation.

Proposed Answer: A.

Explanation:

- A. Correct - Per SOP-0009 P&L 2.14, full load current should be limited to 311 amps, however operation above 311 has been evaluated and pre-approved as long as amps are limited to less than 350 amps; FWS-P1A is above the limit.
- B. Per SOP-0009, P&L 2.18 10% limit only applicable when startup reg valve is controlling RPV level alone.
- C. Per SOP-0009, P&L 2.9 minimum flow valve malfunction is indicated if <200 amps
- D. Per SOP-0009, P&L 2.22, FWS Regulating Valve position should be limited to ≤92% open to allow an adequate margin for valve modulation while maintaining reactor level.

Technical Reference(s): SOP-0009 Rev 62, Pg 5-8 of 99

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107A Obj 9 & RLP-STM-0107B Obj 11

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.4

Comments:

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QUESTION 45 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	261000	A1.02	IR 3.1

Ability to predict and/or monitor changes in parameters associated with operating the Standby Gas Treatment System controls including:
A1.02 Primary containment pressure

Proposed Question:

Standby Gas Treatment Train 'A' is running in the Containment Purge lineup.
Containment pressure is stable.

Which of the following describes the effect on containment pressure if the 'A' Standby Gas Treatment Train STOP pushbutton is depressed?

- HVR-FN8, HIGH VOL CONTMT PURGE
 - HVR-FN13, LOW VOL CONTMT PURGE
- A. Containment Pressure will rise due to supply air from HVR-FN8
- B. Containment Pressure will rise due to supply air from HVR-FN13
- C. Containment Pressure will remain stable as HVR-FN8 trips
- D. Containment Pressure will remain stable as HVR-FN13 trips

Proposed Answer: A.

Explanation:

- A. Correct - When GTS runs in the containment purge lineup, its associated supply fan is HVR-FN8. Both these fans are 12,500 scfm capacity fans. If GTS A is secured, HVR-FN8 will continue to supply 12,500 scfm air to containment resulting in rising pressure.
- B. HVR-FN13 is the low volume supply fan for containment purge and would not be run in conjunction with GTS.
- C. HVR-FN8 will not trip when GTS is stopped.
- D. HVR-FN13 would not be run in conjunction with GTS, nor would it trip if its associated exhaust fan (HVR-FN14) were to be secured when in the Containment Low Volume purge lineup.

Technical Reference(s): R-STM-0257 Rev 5, pg 13 of 28, R-STM-0403 Rev 8, pg 18-21 of 50.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0257 Obj 7 & 11c

Question Source: Bank; December 2010 NRC Exam # 46

Question History: Last NRC Exam December 2010

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41b.5

Comments: **Appeared on one of last 2 exams (3 of 3).**

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QUESTION 46 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	262001	K3.01	IR 3.5

Knowledge of the effect that a loss or malfunction of the A.C. Electrical Distribution will have on the following:
K3.01 Major system loads

Proposed Question:

A failure of NPS-SWG1B has occurred while operating at 90% power.

Which of the following lists the condensate and feedwater pumps that remain running after this event?

- A. Condensate pump B; Feedwater pumps B and C
- B. Condensate pumps A and C; Feedwater pump A
- C. Condensate pump B; Feedwater pumps A and C
- D. Condensate pumps A and B; Feedwater pump C

Proposed Answer: B.

Explanation:

- A. Condensate B and Feedwater B and C would all be tripped on a loss of NPS-SWG1B.
- B. Correct – Condensate A and C as well as Feedwater A are all powered from NPS-SWG1A and would remain running on a loss of NPS-SWGR1B.
- C. Condensate B and Feedwater C would both trip on loss of NPS-SWGR1B. Feedwater A would remain running.
- D. Condensate B and Feedwater C would both trip on loss of NPS-SWGR1B. Condensate A would remain running.

Technical Reference(s): R-STM-0107 Rev 27 Pg 7 of 101 and R-STM-0104 Rev 7 Pg 13 of 65

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-107A Obj 4a and RLP-STM-0104 Obj 4

Question Source: Bank # RBS-NRC-362

Question History: Last NRC Exam January 1994

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41b.7

Comments:

**March 2014 River Bend Station
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QUESTION 47 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	262002 A3.01		IR 2.8

Ability to monitor automatic operations of the Uninterruptable Power Supply (A.C./D.C.) including:

A3.01 Transfer from preferred to alternate source.

Proposed Question:

ENB-INV01B has experienced a condition which caused power to be supplied from the alternate AC source through the static switch. How will this condition be detected?

- A. An annunciator will alarm in the main control room and status light indication for the static switch at the inverter will illuminate.
- B. An annunciator will alarm in the main control room and status light indication for the static switch on H13-P808 will illuminate.
- C. This condition must be detected by the control building operator based on static switch indication at the inverter and reported to the control room due to no annunciator for this condition.
- D. An alarm light will illuminate at the inverter and status light indication for the static switch on H13-P808 will illuminate.

Proposed Answer: A.

Explanation:

- A. Correct - An ENB-INV01B alarm will annunciate in the MCR indicating inverter trouble. Local verification at the inverter will identify the specific trouble condition as static switch swap.
- B. There is no method of verifying the static switch condition from H13-P808.
- C. Although static switch position can be detected by the local operator, a control room annunciator will still alarm.
- D. There is no method of verifying the static switch condition from H13-P808.

Technical Reference(s): R-STM-0300 Rev 26 Figure 7 ; ARP-808-87A, Rev 24 Pg 10 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0300 Obj 5, 12

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 48 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	263000	A1.01	IR 2.5

Ability to predict and/or monitor changes in parameters associated with operating the D.C. Electrical Distribution controls including:
A1.01 Battery charging/discharging rate

Proposed Question:

A 125 volt DC bus has experienced a trip of its battery charger supply breaker. Efforts to restore the battery charger to service have been unsuccessful. Assuming battery loading remains unchanged, predict the change in bus voltage under these conditions.

Voltage will _____.

- A. decrease in a linear fashion
- B. slowly lower and then drop sharply
- C. stair step down as individual cells are depleted
- D. decrease sharply and then level off

Proposed Answer: B.

Explanation:

- A. See "B".
- B. Correct – Battery voltage will decrease slowly then drop sharply as battery voltage reversal occurs. Reference SER 3-99.
- C. See "B".
- D. See "B".

Technical Reference(s): RPPT-STM-0305-ILO Rev 2, Slide 95 ; SER 3-99

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0305 Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.8

Comments:

**March 2014 River Bend Station
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QUESTION 50 Rev 0

Examination Outline Cross-Reference:

Level	RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 1
K/A #	264000 K1.04	IR 3.2

Knowledge of the physical connections and/or cause-effect relationships between Emergency Generators (Diesel/Jet) and the following:
K1.04 Emergency generator cooling water system.

Proposed Question:

The plant has just experienced a loss of all offsite electrical power sources. The diesel generator status is as follows:

- Division 1 tagged out for super outage
- Division 2 tripped during start
- Division 3 running and supplying associated electrical bus

Under these conditions, cooling water is being supplied to the operating diesel generator by (1) and returns to the Standby Cooling Tower through (2) .

- A. (1) SWP-P2A; (2) SWP-MOV55A, STBY CLG TOWER 1 INLET
- B. (1) SWP-P2A ; (2) SWP-AOV599, STBY CLG TWR INLET
- C. (1) SWP-P2C; (2) SWP-MOV55A, STBY CLG TOWER 1 INLET
- D. (1) SWP-P2C ; (2) SWP-AOV599, STBY CLG TWR INLET

Proposed Answer: D.

Explanation:

- A. SWP-P2A would normally supply cooling water to the Division 1 loop but due to the loss of power and the diesel generator being tagged out, SWP-P2A cannot supply water to the Div 1 loop. Additionally, due to the loss of Div 1 power SWP-MOV55A cannot open.
- B. SWP-P2A would normally supply cooling water to the Division 1 loop but due to the loss of power and the diesel generator being tagged out, Part 2 is correct.
- C. Part 1 is correct, but due to the loss of Div 1 power SWP-MOV55A cannot open.
- D. Correct - Under normal conditions, any Standby Service Water Pump can supply the Div 3 DG but under these conditions, SWP-P2A, B, and D are all without power. SWP-P2C which is supplied by the Div 3 diesel is the only available SWP pump. Due to the loss of Div 1 and Div 2 power, SWP-AOV599 will open to supply a return path back to the SBCT since SWP-AOV55A can not open with Div 1 de-energized.

Technical Reference(s): R-STM-0118, Rev 24 Pg 23 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0118 Obj 10a, 13h

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7 Comments:

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QUESTION 51 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	264000	A2.06	IR 3.4

Ability to (a) predict the impacts of the following on the Emergency Generators (Diesel/Jet); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.06 Opening normal and/or alternate power to emergency bus.

Proposed Question:

During normal plant operation, the control switch for the Division 1 4160 VAC bus, ENS-ACB06, NORMAL SUPPLY BREAKER is taken to the tripped position.

- (1) What is the impact of this condition, and
(2) What actions should be taken as a result?
- A. (1) ENS-AC04, ALTERNATE SUPPLY BREAKER, automatically closes to supply the bus.
(2) Transfer Division 2 to its Alternate supply to ensure divisional separation.
- B. (1) The bus will remain de-energized.
(2) Depress the Div 1 Diesel Generator Emergency Start pushbutton.
- C. (1) The Diesel Generator starts and ties to the bus.
(2) Dispatch an operator to the local diesel control panel to monitor Diesel Generator operation.
- D. (1) The Diesel Generator starts and runs unloaded.
(2) Manually load the diesel to supply the bus.

Proposed Answer: C.

Explanation:

- A. The alternate supply breaker will not close on undervoltage. The DG will start.
- B. The bus will not remain de-energize since the DG will start.
- C. Correct - The opening of ENS-ACB06 will cause a bus undervoltage signal which will start the associated diesel generator. The DG output breaker will close to supply the bus when sufficient voltage is established. SOP-0053 directs local monitoring of the diesel generator at the local control panel.
- D. The DG starts and loads due to the undervoltage signal generated when the normal supply breaker was opened.

Technical Reference(s): R-STM-0309S Rev 14 Pg 6 of 117 ; SOP-0053 Rev 328 Pg 43 of 115.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0309S Obj 5a, 6d, 14b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7, b.10

Comments:

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QUESTION 52 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 1	
K/A #	300000 K2.02		IR 3.0

Knowledge of the electrical power supplies to the following:
K2.02 Emergency air compressor

Proposed Question:

The power supply to LSV-C3A, PENETRATION V LEAKAGE CONT AIR COMPRESSOR, is _____.

- A. EHS-MCC2L
- B. EHS-MCC2D
- C. EHS-MCC8A
- D. EHS-MCC8B

Proposed Answer: A.

Explanation:

- A. Correct – EHS-MCC2L is a Div 1 safety related power source in the Auxiliary Bldg.
- B. EHS-MCC2D is a Div 2 safety related power source. LSV-C3A is a Div 1 component.
- C. EHS-MCC8A is a Div 1 safety related power source, but is located in the Control Bldg.
- D. EHS-MCC8B is a Div 2 safety related power source and is located in the Control Bldg.

Technical Reference(s): SOP-0034 Rev 13 Pg 34 of 45

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0208 Obj 7b

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 53 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	1
K/A #	400000	A4.01	IR 3.1

Ability to manually operate and/or monitor in the control room:
A4.01 CCW indications and control

Proposed Question:

The plant is operating at rated conditions with no equipment out of service. The Reactor Plant Component Cooling Water (CCP) System is in normal operation with CCP-P1A and P1B running. CCP-P1C is in standby.

RPCCW SYSTEM LOW HEADER PRESSURE has alarmed on H13-P870-55. An investigation revealed that the CCP header pressure transmitter CCP-PT127 has failed low. No other alarms or automatic actions have occurred.

What automatic feature failed to function?

- A. Trip of the running CRD pump.
- B. Start of the standby pump CCP-P1C.
- C. Initiation of both Standby Service Water divisions.
- D. Isolation of cooling water to the CCP heat exchangers.

Proposed Answer: B.

Explanation:

- A. This occurs on either CCP vital loop signal <56 psig. A single transmitter failure will not cause this action.
- B. Correct
- C. This occurs on when both CCP vital loop signals are <56 psig. A single transmitter failure will not cause this action.
- D. Extreme low system pressure will automatically isolate safety and non-safety related portions of SSP, but will not isolate cooling water to the HX's.

Technical Reference(s): R-STM-0115, Rev 6 Pg 9 of 35 ; ARP-P870-55-F04, Rev 14

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj. 4

Question Source: Bank # RBS-NRC-405

Question History: Last NRC Exam RBS 2004 #53

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41b.7

Comments:

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QUESTION 54 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	201001	K2.05	IR 4.5

Knowledge of electrical power supplies to the following:

K2.05 Alternate rod insertion valve solenoids: Plant-Specific

Proposed Question:

The feeder breaker to BYS-PNL02A2 has just tripped. Which of the following valves will be unable to operate due to the loss of its power source?

- A. Backup Scram Valves
- B. Safety Relief Valves
- C. Alternate Rod Insertion valves
- D. CMS H2 analyzer sample valves

Proposed Answer: C.

Explanation:

- A. The backup scram valves are powered by ENB-PNL02A or ENB-PNL02B.
- B. Safety Relief Valves utilizes divisional solenoids powered by ENB-PNL02A or ENB-PNL02B.
- C. Correct – ARI valves energized to vent with power supplied from BYS-PNL02A2.
- D. CMS-H2 analyzer sample points are supplied by SCM-PNL01A(B).

Technical Reference(s): R-STM-0052 Rev 9 Pg 28 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0052 Obj 3e

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 55 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	201005 A2.12	IR 3.7	

Ability to (a) predict the impacts of the following on the Rod Control and Information System (RCIS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
A2.12 Rod uncoupled: BWR-6

Proposed Question:

An uncoupled control rod has the potential to lead to a (1) . If this were to occur, it would be corrected by utilizing plant procedures to (2) .

- A. (1) control rod drop accident; (2) fully insert the affected control rod.
- B. (1) control rod drop accident;
(2) perform an individual rod scram to attempt re-coupling of the rod.
- C. (1) control rod drift event;
(2) flush the collet piston by performing an individual rod scram.
- D. (1) control rod drift event; (2) fully insert the affected control rod.

Proposed Answer: A.

Explanation:

- A. Correct - A rod drop accident occurs when a rod becomes uncouple from its mechanism, sticks in the core and at some later point becomes free and drops out of the core. AOP-0061 requires fully inserting the affected control rod when a control rod drop accident occurs.
- B. Part 1 is correct. The rod should be driven in using the INSERT pushbutton, not by scrambling.
- C. A control rod drift event is the result of stuck collet piston, directional control valve failure, scram valve leakage, or excessive cooling water pressure/flow, but not from being uncoupled. Part 2 is the correct action for a stuck collet piston, but not for an uncoupled rod leading to a rod drop accident.
- D. A control rod drift event is the result of stuck collet piston, directional control valve failure, scram valve leakage, or excessive cooling water pressure/flow, but not from being uncoupled. Part 2 is correct.

Technical Reference(s): AOP-0061, Rev 7 Pg 6 of 8

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPHLO-0549 Obj 4; RLP-STM-0052 Obj 7, 9b

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.5/b.10

Comments:

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QUESTION 56 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	202002	A1.08	IR 3.4

Ability to predict and/or monitor changes in parameters associated with operating the Recirculation Flow Control System controls including:
A1.08 Recirculation FCV position

Proposed Question:

While operating at 100% power, a feedwater pump trip resulted in a Recirculation Flow Control Valve (FCV) Runback on both FCVs.

The ATC operator is preparing to reset the runback condition by lowering the output demand of the Recirculation Loop Controllers (B33-K603A(B)) in accordance with the Alarm Response Procedure.

How will the FCV positions be affected by the ATC operator's actions?

FCV position will begin to close _____.

- A. after % M/A ERROR is zero AND the Cavitation Interlock Reset pushbutton is depressed.
- B. after % LIMITER ERROR is zero AND negative deviation is observed on the % SERVO ERROR meter.
- C. after % LIMITER ERROR is zero AND the Cavitation Interlock Reset pushbutton is depressed.
- D. as soon as the B33-K603A(B) controller is toggled to the left.

Proposed Answer: B.

Explanation:

A. See "B"

B. Correct - A FCV runback signal present applies a 60% drive flow upper limit to FCV position. When the controller is initially toggled to the left, no valve motion will occur. As the control output continues to lower, valve position will begin to close when controller output demand matches the current valve position. At this point a negative servo deviation will be observed and the valve will begin to close. Depressing of the cavitation interlock is only necessary to allow opening the valve, not closing the valve.

C. See "B"

D. See "B"

Technical Reference(s): R-STM-0053 Rev 13 Pg 37-38, 41 of 76; ARP-680-04 Rev 26 Pg 8 of 87

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053 Obj 2c, 2j

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.6 Comments:

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QUESTION 57 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	204000 A4.04		IR 2.8

Ability to manually operate and/or monitor in the control room(Reactor water cleanup):
A4.04 Heat exchanger operation

Proposed Question:

The Reactor Water Cleanup Non-Regenerative Heat Exchangers have experienced a reduction in cooling water flow. Which of the following should the operator expect to see if this condition is not corrected?

- A. G33-MOVF004, PUMP SUCTION CNMT OUTBOARD ISOL, isolation when Filter Demineralizer inlet temperature reaches 140°F.
- B. G33-MOVF001, PUMP SUCTION DRYWELL INBOARD ISOL, isolation when Filter Demineralizer inlet temperature reaches 140°F.
- C. G33-MOVF004, PUMP SUCTION CNMT OUTBOARD ISOL, isolation when Filter Demineralizer inlet temperature reaches 130°F.
- D. G33-MOVF001, PUMP SUCTION DRYWELL INBOARD ISOL, isolation when Filter Demineralizer inlet temperature reaches 130°F.

Proposed Answer: A.

Explanation:

- A. Correct – A reduction in CCP cooling water flow will cause temperature at the outlet of the Non-Regenerative Heat Exchanger (or inlet to the Filter Demineralizer) to rise. When temperature reaches 140°F, G33-MOVF004 will isolate.
- B. Although G33-MOVF001 isolates the same penetration as G33-MOVF004, only F004 will receive an isolation signal due to high temperature at the filter demineralizer inlet.
- C. At 130°F an alarm will annunciate, but the isolation will not occur until 140°F.
- D. At 130°F an alarm will annunciate, but the isolation will not occur until 140°F.

Technical Reference(s): R-STM-0601 Rev 8 Pg 15 of 51 ; AOP-0003, Rev 33 Pg 17 of 22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0601 Obj 3b, 5e, 7b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 58 Rev 0

Examination Outline Cross-Reference:

Level RO ☒ SRO ☐
Tier # 2 Group # 2
K/A # 219000 A3.01 IR 3.3

**Ability to monitor automatic operations of the RHR/LPCI: Torus/Suppression Pool Cooling Mode including:
A3.01 Valve operation**

Proposed Question:

Residual Heat Removal (RHR) Pump "A" is operating in the suppression pool cooling mode with the following valve alignment:

E12-MOV24A, RHR PUMP A TEST RTN TO SUP PL	OPEN
E12-MOV64A, RHR PUMP A MIN FLOW TO SUP PL	CLOSED
E12-MOV48A, RHR A HX BYPASS VALVE	CLOSED

Which of the following reflects the status of the above valves if a small break LOCA results in receipt of a High Drywell Pressure signal?

<u>E12-MOV24A</u>	<u>E12-MOV64A</u>	<u>E12-MOV48A</u>
A. OPEN	CLOSED	OPEN
B. CLOSED	OPEN	CLOSED
C. CLOSED	OPEN	OPEN
D. OPEN	CLOSED	CLOSED

Proposed Answer: C.

Explanation:

A. See "C".

B. See "C".

C. Correct – E12-MOV24A receives a close signal on High Drywell pressure,
E12-MOV64A will open due to low flow (Test Return and Injection Valve closed),
E12-MOV48A receives a close signal on High Drywell pressure.

D. See "C".

Technical Reference(s): R-STM-0204 Rev 10 Pg 40, 48 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204 Obj 6, 9

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.8

Comments:

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QUESTION 59 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	233000	K4.06	IR 2.9

Knowledge of Fuel Pool Cooling and Clean-Up design feature(s) and/or interlocks which provide for the following:

K4.06 Maintenance of adequate pool level.

Proposed Question:

The design feature of the Fuel Pool Cooling and Cleanup System (SFC) which prevents an uncontrolled loss of upper pool level in the event of a pipe break is/are _____.

- A. anti-siphon devices which can NOT be overridden.
- B. anti-siphon devices which can be overridden to allow controlled draining of the pool.
- C. containment isolation due to low pool level.
- D. containment isolation due to high temperatures in SFC equipment areas.

Proposed Answer:

B.

Explanation:

- A. Anti-siphon device is correct; however, the device may be overridden to allow normal draining of the pools to support vessel disassembly and refueling activities.
- B. Correct – The anti-siphon device is simply a hole drilled in the top of SFC piping which serves to break the siphon effect if water level begins lowering due to a pipe break.
- C. The Suppression Pool Cooling and Cleanup System isolates due to low level in the upper pool, but the SFC system only isolates on Level 2 or High Drywell Pressure of 1.68 psid.
- D. The SFC cooling pump will trip on high room temperature, but containment isolation will only occur on Level 2 or High Drywell Pressure of 1.68 psid.

Technical Reference(s): R-STM-0602 Rev 7 Pg 37 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0602 Obj 2e

Question Source: New

Question History:

Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☒3

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 60 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	256000	K3.02	IR 3.2

Knowledge of the effect that a loss or malfunction of the Reactor Condensate System will have on the following:
K3.02 CRD hydraulics system

Proposed Question:

The plant is operating at 100% power when a malfunction during a Condensate Demineralizer evolution results in a total loss of the Condensate System.

How does this condition affect the Control Rod Drive (CRD) Hydraulics System?

The operating CRD pump will....

- A. auto trip and can not be restored.
- B. auto trip but can be restored following valve manipulation in the field.
- C. remain in service with no change in available Net Positive Suction Head.
- D. remain in service with a reduction in available Net Positive Suction Head.

Proposed Answer: D.

Explanation:

- A. CRD pump will remain in service with suction from the Condensate Storage Tank (CST).
- B. Both the Condensate and CST suction sources are aligned to the CRD pump suction. No valve manipulation is required, and therefore the pump will not trip.
- C. The pump will remain in service, however, the loss of Condensate will reduce available NPSH.
- D. Correct – The pump will remain in service with a reduction in NPSH due to the loss of one of suction sources.

Technical Reference(s): R-STM-0052 Rev 9, Pg 46 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0052 Obj 5a, 5b, 15a

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.7

Comments:

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QUESTION 61 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	271000	K5.07	IR 2.7

Knowledge of the operational implications of the following concepts as they apply to Offgas System: K5.07 Radioactive decay

Proposed Question:

The first indication of insufficient radioactive decay in the Offgas Charcoal Adsorbers is _____.

- A. Elevated Post Treat Radiation Levels
- B. Elevated Pre Treat Radiation Levels
- C. Elevated Area Radiation Levels
- D. Elevated Turbine Building Exhaust Radiation Levels

Proposed Answer: A.

Explanation:

- A. Correct – The charcoal adsorbers are provided to delay the radioactive gases in the Offgas process flow to allow for sufficient time for radioactive decay to occur. If this does not occur these gases will be released prior decay which will result in elevated radiation levels at the Post Treatment Radiation Monitors.
- B. The Pre Treatment Radiation monitor sample point is upstream of the charcoal adsorbers so elevated radiation levels would not be sensed at this point if insufficient radioactive decay was occurring in the charcoal adsorber.
- C. If the radioisotope were being released from the adsorbers prior to decay this would not cause elevated area radiation levels
- D. The offgas system discharges to the main plant exhaust stack, not the turbine building exhaust stack.

Technical Reference(s): R-STM-0606 Rev 8 Pg 10 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0606 Obj 3k, 13

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.4

Comments:

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QUESTION 62 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group #	2
K/A #	286000	K1.03	IR 2.9

Knowledge of the physical connections and/or cause-effect relationships between Fire Protection System and the following:

K1.03 Reactor water level.

Proposed Question:

In order to inject Fire Water into the reactor in accordance with EOP-0005 Enclosure 7, INJECTION INTO THE RPV WITH FIRE SYSTEM, the Fire Protection System must first be aligned to the _____ systems in order to have a flowpath to the reactor vessel.

- A. Service Water and Residual Heat Removal "A"
- B. Condensate Makeup & Transfer (CNS) and Residual Heat Removal "A"
- C. Service Water and Residual Heat Removal "B"
- D. Condensate Makeup & Transfer (CNS) and Residual Heat Removal "B"

Proposed Answer: C.

Explanation:

- A. Service Water is correct, but there is no interface between SWP and RHR A to provide a path for Fire Protection Water (FPW) to the vessel.
- B. Although CNS can be provided to the vessel via RHR A per EOP Enclosure 6, there is no interface to inject FPW to the vessel via CNS and RHR A.
- C. Correct – In EOP Enclosure 7, FPW is aligned to the SWP header and then the RHR B flowpath which allows injection of SWP to the RPV is used to provide FPW to the RPV. This lineup is only available in the RHR B loop, not RHR A.
- D. Although CNS can be provided to the vessel via RHR B per EOP Enclosure 6, there is no interface to inject FPW to the vessel via CNS and RHR B.

Technical Reference(s): R-STM-0250 Rev 6 Pg 19 of 65 ; EOP-0005, Enclosure 7 Rev 314

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0250 Obj 2; RLP-STM-0204 Obj 13b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.4

Comments:

**March 2014 River Bend Station
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QUESTION 63 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	288000	G.2.1.30	IR 4.4

Ability to locate and operate components, including local controls. (Plant Ventilation)

Proposed Question:

After associated isolations are bypassed in accordance with EOP-0005 Enclosure 21, to perform Emergency Containment Venting, the operator must manipulate HVR-AOD127, CONTNT PURGE RTN ISOL and HVR-AOD-128, CONTNT RTN INBD ISOL located on control room panel (1) as well as CPP-MOV105, H2 PURGE FAN DISCH VALVE TO ANNULUS located on (2) .

- A. (1) H13-P863; (2) H13-P852 control room back panel
- B. (1) H13-P863; (2) CPP-PNL102 on the 171'el of the Aux Bldg.
- C. (1) H13-P601; (2) CPP-PNL102 on the 171'el of the Aux Bldg.
- D. (1) H13-P601; (2) H13-P852 control room back panel

Proposed Answer: B.

Explanation:

- A. Part 1 is correct but CPP-MOV105 control switch is located on CPP-PNL102 in the Aux Bldg. H13-P852 contains isolation logic which must be bypassed to allow opening of HVR air operated dampers which is also performed in this Enclosure.
- B. Correct – HVR-AOD127 & 128 control switches are located on H13-P863. CPP-MOV105 control switch is located in the Auxiliary Building 171'el on CPP-PNL102.
- C. H13-P601 contains controls for safety related systems, however the controls for HVR-AOD127 & 128 are located on H13-P863. Part 2 is correct.
- D. H13-P601 contains controls for safety related systems, however the controls for HVR-AOD127 & 128 are located on H13-P863. CPP-MOV105 control switch is located on CPP-PNL102 in the Aux Bldg. H13-P852 contains isolation logic which must be bypassed to allow opening of HVR air operated dampers which is also performed in this Enclosure.

Technical Reference(s): EOP-0005 Enclosure 21 Rev 314

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0403 Obj 7, 14; RLP-STM-0057 Obj 23

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments:

**March 2014 River Bend Station
NRC Initial License Examination
Reactor Operator**

QUESTION 64 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	290001	K6.04	IR 3.9

Knowledge of the effect that a loss or malfunction of the following will have on the Secondary Containment:
K6.04 Primary containment system

Proposed Question:

During a Loss of Coolant Accident (LOCA) leakage from the Primary Containment steel structure would be directed into the ____ (1) ____ where fission products are ____ (2) ____.

- A. (1) Annulus; (2) discharged to the main plant exhaust by annulus pressure control
- B. (1) Annulus; (2) filtered and discharged to the main plant exhaust by Standby Gas Treatment
- C. (1) Auxiliary building; (2) removed by Emergency containment venting as directed by EOP Enclosures
- D. (1) Auxiliary Building; (2) allowed adequate decay time until Normal Containment venting can be performed

Proposed Answer: B.

Explanation:

- A. Annulus is correct, but annulus pressure control isolates during a LOCA.
- B. Correct-The Annulus(Shield Building) completely surrounds the steel containment structure. If the steel structure were to experience leakage, it would be into this area. This area is maintained at a negative pressure during normal operation by Annulus Pressure Control, but during LOCA conditions, APC isolates and SGTS starts to maintain the annulus pressure negative and filters the air drawn from the annulus prior to sending it to the Main Plant Exhaust Stack.
- C. Although the Auxiliary Building is considered part of Secondary Containment, leakage from the steel structure would enter the annulus, not the Auxiliary Building.
- D. Although the Auxiliary Building is considered part of Secondary Containment, leakage from the steel structure would enter the annulus, not the Auxiliary Building.

Technical Reference(s): R-STM-0057 Rev 4 Pg 5, of 69; R-STM-0403 Rev 8 Pg 5 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0057 Obj 29.1; RLP-STM-0403 Obj 12

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.41.b.9

Comments:

**March 2014 River Bend Station
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QUESTION 65 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	2	Group # 2	
K/A #	290002	K1.06	IR 3.1

Knowledge of the physical connections and/or cause-effect relationships between Reactor Vessel Internals and the following:

K1.06 HPCS: Plant-Specific

Proposed Question:

The RPV internal component which functions as part of a single system to keep the peak fuel cladding temperature $\leq 2200^{\circ}\text{F}$ in the event of a small break LOCA is the _____.

- A. High Pressure Core Spray sparger
- B. Low Pressure Core Spray sparger
- C. Jet Pump Riser
- D. Low Pressure Coolant Injection nozzle

Proposed Answer: A.

Explanation:

- A. Correct - The HPCS system is designed to maintain $\text{PCT} \leq 2200^{\circ}\text{F}$ during a small break LOCA. The injection point in the vessel is through a spray sparger.
- B. Although LPCS has a spray sparger above the core, it can not maintain $\text{PCT} \leq 2200^{\circ}\text{F}$ during a small break LOCA.
- C. The jet pumps throat and diffuser are required to maintain 2/3 core coverage during a LOCA, but the riser has no effect in maintaining core coverage during a LOCA.
- D. LPCI systems can not maintain $\text{PCT} \leq 2200^{\circ}\text{F}$ during a small break LOCA.

Technical Reference(s): R-STM-0050 Rev 5 Pg 9 of 46; R-STM-0203 Rev 8 Pg 4 of 38

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0050 Obj 3d

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.3

Comments:

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QUESTION 66 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	Cond of Ops
K/A #	G.2.1.2		IR 4.1

Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question:

In accordance with EN-OP-115, Conduct of Operations, which of the following is NOT the responsibility of a licensed reactor operator (RO)?

- A. Coordinate the activities of shift non-license personnel.
- B. Place the plant in a safe condition when faced with unexpected or uncertain conditions
- C. Ensure minimum staffing requirements are met at the beginning of each shift
- D. Take direction for control manipulations from the person having Control Room Command Function

Proposed Answer: C.

Explanation:

- A. Coordination field activities of non-licensed operators is a responsibility of the RO.
- B. When faced with uncertain or unexpected plant conditions, the RO has the responsibility to place the plant in a safe condition including power reduction or plant shutdown.
- C. Correct - Minimum staffing verification is the responsibility of the Shift Manager.
- D. Taking direction for control manipulation from the individual with the Control Room Command Function is an RO responsibility

Technical Reference(s): EN-OP-115 Rev 14 Pg 14 of 89

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206 Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 67 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	Cond of Ops
K/A #	G.2.1.15		IR 2.7

Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, operations memos, etc

Proposed Question:

The mechanism Operations Managements uses to communicate operational requirements of a continuing nature to the operations crews is (1) , which are to be reviewed by watch standers (2) .

- A. (1) night orders; (2) quarterly
- B. (1) night orders; (2) each shift
- C. (1) standing orders; (2) quarterly
- D. (1) standing orders; (2) each shift

Proposed Answer: D.

Explanation:

- A. Night orders are used for short duration, not items of a continuing nature. Watch standers are required to review standing orders every shift.
- B. Night orders are used for short duration, not items of a continuing nature.
- C. Standing orders is correct, but the review must occur every shift. Quarterly applies to the periodic audit performed for continued applicability.
- D. Correct - Standing orders can be used by Ops Mgmt to communicate items of continuing nature to Ops crews whereas, night order are for short duration. Watch standers are to review standing orders every shift in eSoms.

Technical Reference(s): EN-OP-112 Rev 1 Pg 5 & 9 of 11

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-402 Obj 3

Question Source: New Question History: Last NRC Exam None

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 68 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	Cond of Ops
K/A #	G.2.1.44		IR 3.9

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

Proposed Question:

Select the action required by Technical Specifications if communication between the At the Controls Operator and the refuel platform is lost during fuel handling in the reactor vessel.

- A. Verify operability of the refueling interlocks
- B. Establish communications with the upper IFTS operator
- C. Insert a control rod block
- D. Suspend core alterations

Proposed Answer: D.

Explanation:

- A. Although refueling interlocks operability is required during fuel movement, the interlocks do not prevent the refueling team from continuing core alterations during a loss of communications.
- B. Although the IFTS operator is stationed on the refuel floor, he is not stationed on the platform and therefore establishing communications with the IFTS operator does not meet requirements.
- C. Although inserting a rod block prevents the control room from adding positive reactivity to the core, it would not prevent the refueling team from continuing core alterations which is forbidden when communications is lost.
- D. Correct - TRM 3.9.11 and FHP-0001 both require immediate suspension of core alts if communications is lost between the main control room and the Refueling Platform personnel during core alterations.

Technical Reference(s): TRM 3.9.11 Condition A. and FHP-0001 Rev 34 Pg 28 of 40

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0055 Obj 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 69 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	Equip Control
K/A #	G.2.2.6		IR 3.0

Knowledge of the process for making changes to procedures.

Proposed Question:

In accordance with RBNP-001, DEVELOPMENT AND CONTROL OF RBS PROCEDURES, a COMMENT Procedure Action Request (PAR) may be used for which of the following?

- A. To make minor changes that result in a change of intent.
- B. To correct typographical errors to a procedure that is currently being implemented.
- C. To suggest future improvements or enhancements.
- D. To change acceptance criteria.

Proposed Answer: C.

Explanation:

- A. A procedure REVISION is required for changes that result in a change of intent.
- B. Typographical changes to a procedure being implemented require an EDITORIAL CHANGE.
- C. Correct-COMMENT PARs recommend improvements or enhancements to procedures.
- D. A change of acceptance criteria also results in a change of intent, so would require a REVISION.

Technical Reference(s): RBNP-001 Rev 35 Pg 19 of 43.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-202 Obj 1

Question Source: Bank # RBS-OPS-1667

Question History: Last NRC Exam RBS NRC April 2010

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 70 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	Equip Control
K/A #	G. 2.2.14		IR 3.9

Knowledge of the process for controlling equipment configuration or status.

Proposed Question:

In accordance with OSP-0038, PROTECTIVE TAGGING GUIDELINES, how are temporary blank flanges which have been installed in a system to support maintenance activities controlled to ensure removal upon system restoration?

The blank flange is _____.

- A. Logged in the Main Control Room Log
- B. Logged in the Manipulated Device/Operations Hold Tag Log
- C. CAUTION tagged
- D. DANGER tagged

Proposed Answer: D.

Explanation:

A. See "D"

B. See "D"

C. See "D"

D. Correct - Per step 3.2.6 of OSP-0038, temporary blank flanges must be danger tagged.

Technical Reference(s): OSP-0038 Rev 35 Pg. 6 of 24

Proposed references to be provided to applicants during examination: None

Learning Objective: R-QC-HLO-RO; R-QC-HLO-SROI; R-QC-HLO-SROU

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 71 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	Rad Control
K/A #	G.2.3.12		IR 3.2

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question:

Which of the following activities require Containment Purge operation to control airborne radiological activity in the containment building?

- A. Reactor Core Isolation Cooling system manual startup
- B. Alternate Decay Heat Removal system startup
- C. Suppression Pool Cooling startup
- D. Fuel Pool Purification filter backwash

Proposed Answer: A.

Explanation:

- A. Correct - SOP-0035 Step 4.2.3 requires Containment Purge operation prior to commencing RCIC manual startup.
- B. Although the 3 distractors have the potential to affect radiological conditions in the containment, only RCIC startup procedurally requires operation of containment purge.
- C. See "B"
- D. See "B"

Technical Reference(s): SOP-0035, Rev 46 Pg 13 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209 Obj 10b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.12

Comments:

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QUESTION 72 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	Rad Control
K/A #	G.2.3.14		IR 3.4

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question:

As compared to normal water chemistry, Hydrogen Water Chemistry operation results in (1) Nitrogen-16 carry-over into the steam lines and (2) dose rates in steam affected area.

- A. (1) greater; (2) lower
- B. (1) less; (2) lower
- C. (1) greater; (2) higher
- D. (1) less; (2) higher

Proposed Answer: C.

Explanation:

- A. Part 1 is correct. Part 2 is incorrect. Dose rates are higher due to N16 carryover.
- B. Both parts are incorrect.
- C. Correct - N16 carry over is greater in an excess H₂ environment due to the formation of NH₃. NH₃ is more likely to carry over with the steam unlike water soluble NO₂ and NO₃ which are formed in an excess O₂ condition. N16 gammas are a significant dose contributor in steam affected areas.
- D. Part 1 is incorrect. Part 2 is correct

Technical Reference(s): R-STM-0127 Rev 12 Pg 37 of 68

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0127 Obj 2, 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒ 3

10 CFR Part 55 Content: 55.41.b.12

Comments: Operating Experience: CR-RBS-2002-00195

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QUESTION 73 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	E-Plan
K/A #	G.2.4.5		IR 3.7

Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question:

Of the procedure types listed below, which one would provide guidance for the notification of state and local agencies in the event of a Fuel Handling accident that resulted in a radioactive release?

- A. Fuel Handling Procedures
- B. Emergency Implementing Procedures
- C. Emergency Operating Procedures
- D. Radiation Section Procedures

Proposed Answer: B.

Explanation:

- A. Although fuel handling procedures would be in effect, they do not provide guidance for notification of state and local agencies.
- B. Correct. The EIPs provide guidance for the classification of events and the subsequent notification to state and local agencies.
- C. The EOPs will provide guidance for certain radioactive release events, but do not provide guidance for the notification to state and local agencies.
- D. RSPs do not provide guidance regarding notification to state and local agencies during a fuel handling accident resulting in a radioactive release.

Technical Reference(s): EIP-2-006 Rev 40 Pg 2 of 18

Proposed references to be provided to applicants during examination: None

Learning Objective: LEC-EP-022.13 Obj 1 check this

Question Source: Bank # 2008 NRC Exam #75

Question History: Last NRC Exam RBS 2008 NRC Exam

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 74 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	E-Plan
K/A #	G.2.4.13		IR 4.0

Knowledge of crew roles and responsibilities during EOP usage.

Proposed Question:

During Emergency Operating Procedure (EOP) implementation, which of the following is a responsibility of both RO & SRO licensed operators?

- A. Answering Decision Steps to determine applicable path.
- B. Evaluation of Override Steps to determine when an alternate action is required.
- C. Determining when an EOP may be exited.
- D. Identification of EOP entry conditions.

Proposed Answer: D.

Explanation:

- A. This action is performed by the SRO.
- B. This action is performed by the SRO.
- C. This action is performed by the SRO.
- D. Correct – Identification of EOP entry conditions is the responsibility of all licensed personnel.

Technical Reference(s): OSP-0022 Rev 57 Pg 52 of 77

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206 Obj 4

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments:

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QUESTION 75 Rev 0

Examination Outline Cross-Reference:

Level		RO <input checked="" type="checkbox"/>	SRO <input type="checkbox"/>
Tier #	3	Group #	E-Plan
K/A #	G.2.4.37		IR 3.0

Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question:

During an event requiring implementation of the emergency plan, the shift manager assumes the role of _____.

- A. Emergency Director
- B. EOF (Emergency Operations Facility) Manager
- C. OSC (Operations Support Center) Manager
- D. Emergency Plant Manager

Proposed Answer: A.

Explanation:

- A. Correct – Per EIP-2-002, the Shift Manager assumes the role of ED until properly relieved or until the emergency situation is terminated. The other three positions are manned by the ERO using off-shift personnel.
- B. See "A"
- C. See "A"
- D. See "A"

Technical Reference(s): EIP-2-007 Rev 25 Pg 3 of 9; EIP-2-020 Rev 36 Pg 4 of 58

Proposed references to be provided to applicants during examination: None

Learning Objective: RCBT-EP-NEONCO, Obj

Question Source: Bank RBS 2008 Audit #74

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments: The original question was not an RO question; replaced with current question and re-validated.

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QUESTION 77 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group #	1
K/A #	295006	AA2.01	IR 4.6

Ability to determine and/or interpret the following as they apply to SCRAM:
AA2.01 Reactor power

Proposed Question:

A reactor scram has just occurred resulting in the following conditions:

- | | |
|----------------------|--------------------------|
| • RPV Level | 12 inches, rising |
| • RPV Pressure | 1050 psig |
| • Set Point Set Down | Active |
| • Control Rod Status | 40 rods remain withdrawn |
| • Reactor Power | 3% power |

Which of the following represents the appropriate procedure execution for the current conditions?

- A. Enter EOP-0001, Transition to EOP-0001A and maintain the Reactor Recirculation Pumps in service.
- B. Enter EOP-0001, Transition to EOP-0001A and trip the Reactor Recirculation Pumps.
- C. Enter and remain in EOP-0001 and maintain the Reactor Recirculation Pumps in service.
- D. Enter and remain in EOP-0001 and trip the Reactor Recirculation Pumps.

Proposed Answer: A.

Explanation:

- A. Correct – Based on stem conditions, EOP-0001 is entered (Level 3) and due to control rods remaining withdrawn, transition to EOP-0001A is required. Due to reactor power being <5%, the recirculation pumps are maintain in service due to minimal power reduction effect from securing them at <5% power.
- B. It is not appropriate to trip the recirculation pumps during an ATWS with power <5%.
- C. Even though power is <5% and there is no EOP-0001 entry condition due to the ATWS, EOP-0001 is entered due to Level 3 and due to control rods remaining withdrawn, transition to EOP-0001A is required.
- D. Even though power is <5% and there is no EOP-0001 entry condition due to the ATWS, EOP-0001 is entered due to Level 3 and due to control rods remaining withdrawn, transition to EOP-0001A is required.

Technical Reference(s): EOP-0001 Rev 26; EOP-0001A Rev 26

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-0512 Obj 4, 7

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒3

10 CFR Part 55 Content: 55.43.b.5 Comments:

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QUESTION 78 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group #	1
K/A #	295018	G.2.4.45	IR 4.3

Ability to prioritize and interpret the significance of each annunciator or alarm. (Partial or Complete Loss of Component Cooling Water)

Proposed Question:

The plant is currently in an ATWS condition. The transient began with the receipt of the following alarms:

- REACTOR PLANT COMPNT CLG PUMPS AUTO TRIP
- RPCCW SYSTEM LOW HEADER PRESSURE
- CRD PUMP A OR B AUTO TRIP

Which of the following should the CRS direct to facilitate control rod insertion?

- A. ARP-601-22-A1, CRD Pump A or B Auto Trip
- B. AOP-0011 Section 5.3, Align SSW to the CRD Pump Bearing Coolers
- C. EOP-0005 Enclosure 12, Defeating RPS & ARI Interlocks
- D. SOP-0002 Section 5.1, Alternating CRD Pumps

Proposed Answer: B.

Explanation:

- A. This procedure would have the operator start the standby CRD pump; this action cannot be performed without CCP header pressure.
- B. Correct – Aligning Standby Service Water to the CRD Pumps allows restarting of the system in order to facilitate driving control rods.
- C. This Enclosure is normally installed during ATWS conditions to allow draining of the Scram Discharge Volume to allow insertion of another scram signal. In this condition however, with no CRD pump running, the scram can not be reset (hydraulically).
- D. This procedure action cannot be performed without CCP header pressure.

Technical Reference(s): AOP-0011 Rev 19 Pg 7 of 8

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP011 Obj 5b, 6b, 6c

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments: In its original form, it could be argued that 3 answers would have worked, though the priority would have been on a single answer.

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QUESTION 79 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group #	1
K/A #	295025	EA 2.06	IR 3.8

Ability to determine and/or interpret the following as they apply to High Reactor Pressure:
EA2.06 Reactor water level

Proposed Question:

During a plant transient, a loss of ALL high pressure injection has occurred. The following conditions exist:

- RPV Level -186 inches, slowly lowering
- RPV Pressure 1035 psig
- Drywell Pressure 0.28 psid

Which of the following should the CRS direct?

- A. Enter STP-050-0700, RCS PRESSURE/TEMPERATURE LIMITS VERIFICATION to conduct a controlled cooldown not to exceed 100°F/hour cooldown rate.
- B. Enter Steam Cooling leg of EOP-0001 and stabilize reactor pressure.
- C. Enter Emergency Depressurization leg of EOP-0001 and open 7 Automatic Depressurization System (ADS) valves.
- D. Exit EOP-0001 and enter EOP-0004, RPV Flooding.

Proposed Answer: C.

Explanation:

- A. This would be the normal course of action per the EOPs for pressure control if a level control problem did not exist, but the due to level control issues, pressure must be rapidly lowered to allow injection of low pressure systems.
- B. There is no indication that low pressure systems are unavailable. With level below -143", all LP ECCS systems are running, EOP-0001 ALC-5 is answered YES so the Steam Cooling leg is not entered.
- C. Correct – ALC-5 is answered YES, since all LP ECCS systems start on -143". Per ALC-6 when level can not be restored and maintained above -186" ED is required.
- D. Entry into RPV Flooding is required when level can not be determined. This condition does not exist.

Technical Reference(s): EOP-0001 Rev 26

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-512 Obj 4, 7

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 80 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group # 1	
K/A #	295025	G.2.4.21	IR 4.6

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (High Reactor Pressure)

Proposed Question:

The Technical Specification Bases for the Reactor Recirculation System response to a high reactor pressure signal is to _____

- A. protect the Reactor Recirculation system from excessive thermal gradient between the steam dome and the recirculation loops.
- B. protect the pumps from cavitation due to the decrease in feedwater flow following a reactor scram.
- C. insert additional negative reactivity to account for the reduced amount of negative reactivity inserted during the first few feet of control rod travel following a turbine trip at the end of core life.
- D. insert negative reactivity during ATWS conditions by increasing voids to counteract the pressure increase maintaining peak RPV pressure below applicable ASME codes.

Proposed Answer: D.

Explanation:

- A. The thermal shock interlocks are start permissives and do not trip the Recirc System.
- B. This describes the bases for the cavitation interlock transfer for the Recirc Pumps. The cavitation interlock is initiated by low feedwater flow, not a high reactor pressure signal.
- C. This describes the bases of the End of Cycle Recirc Pump Trip interlocks. EOC-RPT is initiated from a turbine trip, not a high reactor pressure signal.
- D. At 1153 psig, the Recirc Pumps trip to off to add negative reactivity by increasing core voiding due to the reduction in core flow. This limits the peak pressure in the RPV to maintain below ASME codes. This setpoint is above the normal pressure scram therefore pressures at this level would be present if a failure to scram would occur.

Technical Reference(s): Technical Specifications Bases 3.3.4.2

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053 Obj 2e, 11

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.2

Comments:

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QUESTION 81 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group #	1
K/A #	295030	G.2.4.50	IR 4.0

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
(Low Suppression Pool Water Level)

Proposed Question:

During Mode 1 plant operation, the following annunciators are received:

- H13-P808/83A/F02 SUPPRESSION POOL LEVEL LOW
- H13-P808/84A/F02 SUPPRESSION POOL LEVEL LOW

Corresponding level indicators reads slightly less than 19'6".

Which of the following procedures should the CRS direct for the above conditions?

- A. AOP-0003, AUTOMATIC ISOLATIONS AND SOP-0091, Section 6.2 Shutdown of Containment Pool Cooling System Loop A(B) due to SFC isolation.
- B. SOP-0035, REACTOR CORE ISOLATION COOLING, Section 5.6 Manual Swap of Suppression Pool and CST Suction Valves to maintain system operability.
- C. SOP-0031, HIGH PRESSURE CORE SPRAY, Section 5.3 Manual Swap of Suppression Pool and CST Suction Valves to maintain system operability.
- D. AOP-0003, AUTOMATIC ISOLATIONS AND SOP-0140, Section 7.1 Suppression Pool Cooling and Cleanup Shutdown due to SPC isolation.

Proposed Answer: D.

Explanation:

- A. Part 1 is correct, but SFC will not isolate. This system selected as a plausible distractor due RHS also isolating on Dryer Storage Pool Level, a component of the SFC system.
- B. RCIC LCO entry is not required on SP Level. Additionally, RCIC is normally aligned to the CST. It is not prudent to aligned RCIC to the Sup Pool when it is experiencing low level conditions.
- C. HPCS LCO entry is not required on SP Level. Additionally, HPCS is normally aligned to the CST. It is not prudent to aligned HPCS to the Sup Pool when it is experiencing low level conditions.
- D. Correct – At 19'6", RHS-AOV62, 63, 64 will isolate tripping the Suppression Pool Cooling and Cleanup System. AOP-0003 is appropriate to verify the isolations occurred and SOP-0104 Section 7.1 is appropriate to place the SPC system in the shutdown lineup.

Technical Reference(s): AOP-0003 Rev 33 Pg 14, 20 if 22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0656 Obj 3d,4b,6,10

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 82 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group #	1
K/A #	295031	EA 2.03	IR 4.2

Ability to determine and/or interpret the following as they apply to Reactor Low Water Level:
EA2.03 Reactor pressure

Proposed Question:

ATWS procedures are currently being implemented following a loss of 2 Feedwater pumps.

- Emergency Depressurization is required
- All injection systems except for Boron, RCIC and CRD are terminated
- 7 ADS SRVs are opened.

When should the CRS direct the commencing of injection into the RPV?

- A. When level drops below -186 inches.
- B. When reactor water level can not be determined.
- C. When the reactor pressure has lowered to the Minimum Steam Cooling Pressure (MSCP)
- D. When reactor pressure lowers below the Heat Capacity Temperature Limit (HCTL)

Proposed Answer: C.

Explanation:

A. See "C".

B. See "C".

C. Correct-The EOPs require termination of injection prior to ED to prevent the uncontrolled rapid injection of cold water into the vessel. Injection is maintained terminated until adequate core cooling no longer exists. As pressure lowers, ACC is maintained as long as reactor pressure remains above the Minimum Steam Cooling Pressure. At pressures above this value, the steam flow provides sufficient cooling to maintain ACC. Once reactor pressure drops below this value, injection must be restored.

D. See "C".

Technical Reference(s): EOP-0001A Rev 26, RLA-20, RLA-21

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-513 Obj 5

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 83 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295009	G.2.4.6	IR 4.7

Knowledge of EOP mitigation strategies. (Low Reactor Water Level)

Proposed Question:

A Station Blackout has occurred concurrent with an overspeed trip of Reactor Core Isolation Cooling. Current conditions are:

- Level -188", slowly lowering
- Pressure 450 psig

Based on the above conditions, which of the following actions should the CRS take?

- A. Enter Steam Cooling section of the EOPs and direct Enclosure 7, INJECTION IN TO THE RPV WITH FIRE SYSTEM
- B. Enter Emergency Depressurization section of the EOPs and direct Enclosure 6, INJECTION INTO THE RPV WITH CONDENSATE TRANSFER
- C. Enter Steam Cooling section of the EOPs and direct Enclosure 6, INJECTION INTO THE RPV WITH CONDENSATE TRANSFER
- D. Enter Emergency Depressurization section of the EOPs and direct Enclosure 7, INJECTION IN TO THE RPV WITH FIRE SYSTEM

Proposed Answer: A.

Explanation:

- A. Correct – With no injection available and level below -162" (TAF), Steam Cooling is required. In Steam Cooling, Emergency Depressurization is required at -200". As level slowly lowers from -188" to -200", the fuel is being cooled by steam flow. This time is utilized to align an injection system. Based on initial conditions, the only available injection is Fire Water (FPW) which is aligned per Enclosure 7. Once FPW is aligned the RPV will be depressurized to allow injection from this low pressure system.
- B. It would be inappropriate to ED with no injection available while level is still above -200". Condensate Transfer System would not be available under the given conditions.
- C. Part 1 is correct, but the Condensate Transfer System would not be available under the given conditions.
- D. It would be inappropriate to ED with no injection available while level is still above -200". The fuel has adequate core cooling if maintained above -200". This time should be used to align an injection system.

Technical Reference(s): EOP-0001, Rev 26 ALC-9

Proposed references to be provided to applicants during examination: None

Learning Objective: R-LPOPS-HLO-512 Obj 7

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 84 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295020	AA 2.02	IR 3.4

Ability to determine and/or interpret the following as they apply to Inadvertent Containment Isolation:
AA2.02 Drywell/containment temperature

Proposed Question:

Numerous alarms have annunciated in the Main Control Room coincident with the isolation of all Division 2 containment isolation valves. The plant has remained at 100% power. The following conditions exist:

- | | |
|---|-----------------|
| • Div 2 scram solenoids | De-energized |
| • H13-P680 Div 2 APRM, IRM, SRM status lights | All lit |
| • H13-P601 CRVICS Div 2 & 3 Status light | Amber light lit |
| • Drywell temperature | 138°F, rising |
| • Containment temperature | 87°F, rising |

Which of the following procedures should the CRS direct to address rising temperatures in the containment and drywell?

- A. SOP-0059, Containment HVAC System
- B. SOP-0060, Drywell Cooling System
- C. AOP-0003, Automatic Isolations
- D. AOP-0010, Loss of One RPS Bus

Proposed Answer: D.

Explanation:

- A. See "D".
- B. See "D".
- C. See "D".
- D. Correct – The given indications are that of an inadvertent containment isolation caused by the loss of the Div 2 RPS bus. The correct action to address this condition is to transfer power to the alternate source, reset the isolation and restore the containment and drywell cooling systems to service. This guidance is provided in AOP-0010, Loss of One RPS Bus.

Technical Reference(s): AOP-0010 Rev 19 Pg 10-11 of 21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0529 Obj 3 & 6

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 85 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	1	Group # 2	
K/A #	295034	EA 2.01	IR 4.2

Ability to determine and/or interpret the following as they apply to Secondary Containment Ventilation High Radiation:
EA2.01 Ventilation radiation levels

Proposed Question:

A leak in the Reactor Core Isolation Cooling (RCIC) room has resulted in the following radiation monitor status:

- | | |
|---|--------|
| • RMS-RE11A & B, REACTOR BLDG ANNULUS VENT | GREEN |
| • RMS-RE125, MAIN PLANT EXHAUST DUCT | YELLOW |
| • RMS-RE110, AUXILIARY BUILDING VENTILATION | RED |

Which of the following should the CRS direct based on the current radiation monitor status?

- A. AOP-0003, AUTOMATIC ISOLATIONS
- B. SOP-0059 Section 5.2, Standby Gas Manual Startup
- C. SOP-0065 Section 6.0 Auxiliary Building Ventilation Shutdown
- D. OSP-0053 Hard Card for Auxiliary Building Manual Isolation

Proposed Answer: D.

Explanation:

- A. There are no automatic actions associated with the one radiation monitor that is in ALARM condition.
- B. Although Standby Gas will be started up, this section will not perform the requirement to manually isolate the Auxiliary Building.
- C. Although the Auxiliary Building Ventilation will be shutdown, this section will not perform the steps required in align SGTS to the annulus.
- D. Correct – Unlike most radiation monitors, RMS-RE110 has no associated automatic actuations. Plant procedures require manual isolation of the Auxiliary Building as well as alignment of Standby Gas Treatment taking suction from the Auxiliary Building and the Annulus and exhausting to the Main Plant Stack. The OSP-0053 Hard Card provides direction to accomplish these actions.

Technical Reference(s): OSP-0053 Rev 20 Attachment 21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0409 Obj 8c, 9

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 87 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	2	Group #	1
K/A #	218000	G.2.2.40	IR 4.7

Ability to apply Technical Specifications for a system. (Automatic Depressurization System)
--

Proposed Question:

During a High Pressure Core Spray system outage while in Mode 1, it is determined that B21-F051D (Automatic Depressurization System (ADS) valve) is also inoperable.

What are the Tech Spec implications for this condition?

- A. Restore the ADS valve to operable status within 14 days
- B. Restore either ADS or HPCS to operable status within 72 hours
- C. Be in Mode 3 in 12 hours and reduce steam dome pressure to <100 psig in 36 hours
- D. Within 1 hour initiate action to place the unit in Mode 2 within 7 hours, Mode 3 in 36 hours and Mode 4 within 37 hours.

Proposed Answer: D.

Explanation:

- A. This is the Required Action for an inoperable ADS valve without HPCS being inoperable.
- B. This is the Required Action for ADS inoperable coincident with LOW Pressure Core Spray inoperable.
- C. This is the Required Action for multiple ADS valves inoperable. Plausible due to its similarity to LCO 3.0.3
- D. Correct – The inoperability of an ADS coincident with HPCS requires entry into LCO 3.0.3.

Technical Reference(s): Technical Specification 3.5.1, 3.0.3

Proposed references to be provided to applicants during examination: **TS 3.5.1 No Bases**

Learning Objective: RLP-STM-0109 Obj 16; RLP-STM-0203 Obj 14

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒2

10 CFR Part 55 Content: 55.43.b.2

Comments:

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QUESTION 88 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	2	Group #	1
K/A #	239002 A2.06		IR 4.3

Ability to (a) predict the impacts of the following on the Relief/Safety Valves; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations:
A2.06 Reactor high pressure

Proposed Question:

Following a turbine trip, reactor pressure peaked at 1150 psig resulting in a reactor scram.

(1) How does this condition affect the operation of the Safety Relief Valves and (2) select the procedure the CRS should direct to return the SRVs to pre-transient operation?

- A. (1) SRVs will operate with alternate setpoints to reduce pressure loading in containment by preventing multiple actuations in rapid succession. (2) Direct performance of ARP-601-19 Alarm Response Procedure for both Low Low Set logic trip systems.
- B. (1) SRVs will operate with alternate setpoints to reduce pressure loading in containment by preventing multiple actuations in rapid succession. (2) Direct performance of SOP-0011 Section 6.3, SVV System Shutdown.
- C. (1) SRVs will not operate due to actuation of "Smart Fire" Relay. (2) Direct performance of ARP-601-19 Alarm Response Procedure for both Low Low Set logic trip systems.
- D. (1) SRVs will not operate due to actuation of "Smart Fire" Relay (2) Direct performance of SOP-0011 Section 6.3, SVV System Shutdown.

Proposed Answer: A.

Explanation:

- A. Correct – Upon actuation of the first SRV pressure trip unit, Low Low Set is activated which lowers the opening setpoint of 2 SRVs and the closing setpoint of 5 SRVs allowing the associated SRVs to open earlier and stay open longer, minimizing the number of cycles which reduced the pressure loading on containment. To return the SRVs to their normal setpoints, the Alarm Response Procedure is utilized to reset the associated logic.
- B. Part 1 is correct, but the ARP for the LLS Seal in annunciators must be performed.
- C. Smart Fire relay not activated until 1397 psig. Part 2 is correct.
- D. Smart Fire relay not activated until 1397 psig. The ARP for LLS Seal in annunciators must be performed.

Technical Reference(s): R-STM-0109 Rev 12 Pg 12-14 of 95; ARP-601-19 Rev 34 Pg 93-94 of 94

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109 Obj 4c, 5, 27f

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5 Comments:

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QUESTION 89 Rev 0

Examination Outline Cross-Reference:

Level	RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	2	Group # 1
K/A #	262002 A2.02	IR 2.7

Ability to (a) predict the impacts of the following on the Uninterruptable Power Supply (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.02 Over voltage

Proposed Question:

During the performance of STP-302-0102, Power Distribution System Operability Check, the Unit Operator has reported a voltage reading of 123.2 volts AC on VBS-PNL01A. A report from the field confirms 123.2 volts AC at the output of ENB-INV01A. (1) What is the impact of this condition, and (2) what actions should the CRS direct?

- A. (1) ENB-INV01A is inoperable and the associated vital bus distribution system is inoperable;
(2) Direct performance of SOP-0048 Section 5.5, Transfer from ENB-INV01A to ENB-INV01A1 supplying VBS-PNL01A.
- B. (1) Only ENB-INV01A is inoperable.
(2) Direct performance of SOP-0048 Section 5.5, Transfer from ENB-INV01A to ENB-INV01A1 supplying VBS-PNL01A.
- C. (1) Only ENB-INV01A is inoperable
(2) Direct performance of SOP-0048 Section 5.2 Transferring an ENB Inverter from Normal Operation to Maintenance Bypass.
- D. (1) ENB-INV01A is inoperable and the associated vital bus distribution system is inoperable.
(2) Direct performance of SOP-0048 Section 5.2 Transferring an ENB Inverter from Normal Operation to Maintenance Bypass.

Proposed Answer: A.

Explanation:

- A. Correct - The surveillance contains acceptance criteria for both the inverter and the vital bus. Both are outside of the given acceptance criteria therefore the surveillance requirement is not met so both components must be declared inop (TS 3.8.7 & 3.8.9). Transferring to ENB-INV01A will satisfy TS 3.8.7 and will correct the voltage condition which will also allow exit of TS 3.8.9).
- B. The vital bus distribution system is also inop due to voltage being outside the acceptance criteria. Part 2 is correct.
- C. Part 1 is incorrect, see "B". Operating an inverter in Manual Bypass does not satisfy TS 3.8.7. The correct action is to transfer to the alternate inverter.
- D. Part 1 is correct, but operating an inverter in Manual Bypass does not satisfy TS 3.8.7. The correct action is to transfer to the alternate inverter.

Technical Reference(s): STP-302-0102, TS 3.8.7, TS 3.8.9

Proposed references to be provided to applicants during examination: **STP-302-0102, TS 3.8.7, TS 3.8.9 and Bases**

Learning Objective: RLP-STM-0300 Obj 10

Question Source:	New	Question History:	Last NRC Exam	NA
Cognitive Level:	Memory or Fundamental Knowledge <input type="checkbox"/>	Comprehension or Analysis	<input checked="" type="checkbox"/>	4
10 CFR Part 55 Content:	55.43.b.5	Comments:		

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QUESTION 90 Rev 0

Examination Outline Cross-Reference: Level RO ☐ SRO ☒
Tier # 2 Group # 1
K/A # 400000 G.2.4.11 IR 4.2

Knowledge of abnormal condition procedures. (Component Cooling Water System)

Proposed Question:

A total loss of Reactor Plant Component Cooling Water has occurred. In order to accomplish the direction of the associated Abnormal Operating Procedure regarding operation of the Reactor Recirculation System, the CRS should direct performance of which of the following procedures?

- A. OSP-0053 Attachment 18, Securing Reactor Recirculation Pumps, only
- B. OSP-0053 Attachment 19, Isolation of Reactor Recirculation Pumps, only
- C. OSP-0053 Attachment 18 AND OSP-0053 Attachment 19
- D. SOP-0003 Section 5.8. Restoring Seal Purge to an Operating Recirculation Pump

Proposed Answer: C.

Explanation:

- A. This Attachment is performed, but Attachment 19 is also required.
- B. This Attachment is performed, but Attachment 18 is also required.
- C. Correct – Both Attachments 18 and 19 are required. With a loss of cooling only tripping of the pumps is required. With loss of cooling and seal purge both tripping and isolating the pumps is required per AOP-0011. Attachments 18 and 19 are both required to accomplish this action.
- D. With a loss of CCP, seal purge can not be restored due to CRD tripping on low CCP pressure. In addition, the recirc pumps will not remain in service due to the loss of cooling, so there would be no “Operating Recirculation Pump”.

Technical Reference(s): AOP-00011 Rev 19 Pg 5 of 8

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj 9a,11a; RLP-OPS-AOP0011 Obj 4,6b

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 91 Rev 0

Examination Outline Cross-Reference:

Level	RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	2	Group # 2
K/A #	201003 A2.09	IR3.4

Ability to (a) predict the impacts of Low reactor pressure on the Control Rod and Drive Mechanism; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations:

Proposed Question:

During a plant startup an accumulator fault is received on control rod 28-29 due to low accumulator pressure.

Reactor Pressure	530 psig
Charging water header pressure	1750 psig

What is the impact of this condition and what action should the CRS direct to mitigate the condition?

- A. (1) The control rod may not fully insert on a scram signal due to insufficient pressure from the accumulator and/or the reactor via the ball check valve.
(2) SOP-0071, ROD CONTROL AND INFORMATION SYSTEM, Section 5.3 Continuous insert using In Timer Skip to fully insert the control rod.
- B. (1) The control rod may not fully insert on a scram signal due to insufficient pressure from the accumulator and/or the reactor via the ball check valve.
(2) SOP-0002, CONTROL ROD DRIVE HYDRAULICS, Section 4.3 Charging HCU Accumulators with Nitrogen, to restore accumulator pressure.
- C. (1) The control rod may drift due to higher differential pressure across the drive piston.
(2) SOP-0071, ROD CONTROL AND INFORMATION SYSTEM, Section 5.3 Continuous insert using In Timer Skip to fully insert the control rod.
- D. (1) The control rod may drift due to higher differential pressure across the drive piston.
(2) SOP-0002, CONTROL ROD DRIVE HYDRAULICS, Section 4.3 Charging HCU Accumulators with Nitrogen, to restore accumulator pressure.

Proposed Answer: B.

Explanation:

- A. Part 1 is correct, but rod insertion is not required since charging water header pressure is >1540 psig.
- B. Correct – The control rod is capable of scrambling by either accumulator or reactor pressure via the ball check valve on the CRD mechanism. With low pressure in the reactor and the accumulator the rod may not fully scram when required. Since charging water header pressure is >1540 psig, the rod doesn't have to be inserted. The rod must, however, be declared inop within 1 hour. It is prudent therefore to charge the associated accumulator to exit the condition.
- C. Both parts incorrect. The differential pressure across the drive piston is maintained by the pressure control valve which adjusts as reactor pressure changes. See "B" for explanation of Part 2.
- D. Part 1 is incorrect. Part 2 is correct.

Technical Reference(s): Tech Spec 3.1.5 and bases; ARP-680-07-C03 Rev 33 Pg 25 of 51

Proposed references to be provided to applicants during examination: **TS 3.1.5 without Bases**

Learning Objective: RLP-STM-0052 Obj 8q, 11b, 12

Question Source:	New	Question History:	Last NRC Exam	None
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Cognitive Level:	Memory or Fundamental Knowledge <input type="checkbox"/>	Comprehension or Analysis <input checked="" type="checkbox"/>
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10 CFR Part 55 Content:	55.43.b.5	Comments:
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QUESTION 92 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	2	Group # 2	
K/A #	223001 A2.06	IR 4.1	

Ability to (a) predict the impacts of the following on the Primary Containment System and Auxiliaries; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.06 High containment pressure: Mark-III

Proposed Question:

A steam leak has occurred resulting in the following plant conditions:

- Drywell Pressure 1.4 psid
- Containment Pressure 2.1 psig
- Normal containment vent and purge is in service

(1) Which of the following describes the impact of the above conditions; and (2) what procedure should the CRS direct to mitigate the consequences of this condition?

- A. (1) Damage to the HVAC ductwork;
(2) SOP-0059 to secure normal containment vent and purge.
- B. (1) Damage to the HVAC ductwork;
(2) EOP Enclosure 21 for Emergency Containment Venting.
- C. (1) Containment vent valves cannot be opened;
(2) SOP-0059 to secure normal containment vent and purge.
- D. (1) Containment vent valves cannot be opened;
(2) EOP Enclosure 21 for Emergency Containment Venting.

Proposed Answer: A.

Explanation:

- A. Correct – Above 2 psig, normal containment vent and purge must be secured in order to prevent damage to the Aux Bldg HVAC ductwork which is used during this evolution.
- B. Part 1 is correct, but Enclosure 21 is used to maintain pressure below 30 psig. It would be inappropriate to override interlocks and vent containment at 2.1 psig.
- C. The pressure which corresponds to the inability to operate containment vent valves for venting purposes is 15 psig. Part 2 is correct.
- D. See "C". Enclosure 21 is used to maintain pressure below 30 psig. It would be inappropriate to override interlocks and vent containment at 2.1 psig.

Technical Reference(s): EOP-0002, EPSTG*0002 B-8-11

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0514 Obj 5 & 6

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5

Comments:

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QUESTION 93 Rev 0

Examination Outline Cross-Reference:	Level	RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
	Tier #	2	Group # 2
	K/A #	259001 G.2.4.46	IR 4.2

Ability to verify that the alarms are consistent with the plant conditions. (Reactor Feedwater System)

Proposed Question:

Power ascension is in progress per GOP-0001, PLANT STARTUP when the following annunciator is received:

- REACTOR HIGH/LOW WATER LEVEL

The ATC operator reports the following:

- | | |
|--------------------------------------|----------------|
| • RPV Level | 30 inches |
| • Startup Feedwater Regulating Valve | Manual, closed |
| • "A" Feedwater Regulating Valve | Auto, 95% open |
| • "B" Feedwater Regulating Valve | Manual, closed |
| • "C" Feedwater Regulating Valve | Manual, closed |

Which of the following SOP-0009, FEEDWATER procedure sections should the CRS direct?

- A. 4.6 Establishing Flow to the Vessel Through the Startup Level Control Valve
- B. 4.7 Placing Startup Level Control in Automatic
- C. 4.11 Placing the Second or Third FWREG Valve in Service
- D. 5.3 Augmenting FWREGs While at High Power with the Startup FWREG

Proposed Answer: C.

Explanation:

- A. This section is utilized during initial startup of the feed system due to its smaller line size and finer control of level. Based on the above conditions, it is prudent to place available normal feedwater regulation valves in service per Section 4.11.
- B. It is prudent to place available normal feedwater regulation valves in service per Section 4.11. versus placing the Startup in Auto. If it was desired to use the Startup and Normal regulating valves both in Auto, it would be performed with Section 5.3.
- C. Correct – SOP-0009 P&L 2.22 states that FWREG valve position should be limited to 92% to allow for adequate margin for valve modulation. With 2 valves available, the correct action is to place one of them in service per Section 4.11.
- D. Augmentation is not required due to the availability of 2 FWREG valves

Technical Reference(s): SOP-0009, Rev 62 Pg 8, 32 of 99

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107A Obj 6, 7, 9; RLP-STM-0107B Obj 5, 9, 11,

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.5 Comments:

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QUESTION 94 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	3	Group #	Cond. of Ops
K/A #	G.2.1.4		IR 3.8

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

Proposed Question:

The reactor is operating in Mode 1. Current Control Room shift staffing includes:

- 1 Operations Shift Manager (OSM)
- 1 Control Room Supervisor (CRS)
- 1 Shift Technical Advisor (STA)/(CRS)
- 3 Reactor Operators (RO)

Four hours after shift turnover, 2 of the ROs and the OSM fall ill with food poisoning. First Responders have recommended the personnel be sent to the hospital.

In accordance with the Technical Specification manning requirement, _____.

- A. none of the personnel may leave the site until relieved.
- B. two ROs may leave but the OSM may not leave.
- C. sick personnel may leave only after the unit has been placed in Mode 3 when staffing requirements are less.
- D. sick personnel may leave if replacement personnel are immediately called in to ensure that minimum staffing requirements are restored within 2 hours.

Proposed Answer: D.

Explanation:

A. See "D"

B. See "D"

C. See "D"

D. Tech Spec 5.2.2.c. states the minimum staffing requirements may be one less than minimum requirements (2 ROs, 2 SROs, 1 STA) for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

Technical Reference(s): Technical Specification 5.2.2.c.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0416 Obj T2

Question Source: Bank # RBS-NRC-641

Question History: Last NRC Exam RBS Feb 1999

Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒

10 CFR Part 55 Content: 55.43.b.1

Comments:

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QUESTION 95 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	3	Group #	Cond. of Ops
K/A #	G.2.1.41		IR 3.7

Knowledge of the refueling process.

Proposed Question:

Prior to entering Mode 5, the list of applicable Technical Specifications in _____ must be verified met.

- A. REP-0029, Fuel Movement Plan
- B. FHP-0001, Control of Fuel Handling and Refuel Operations
- C. GOP-0002, Plant Shutdown
- D. GMP-0102, Reactor Vessel Disassembly

Proposed Answer: B.

Explanation:

- A. REP-0029 contain guidance regarding assembly of the fuel movement plan and other fuel movement rules, but does not contain the list of applicable Mode 5 Tech Specs.
- B. Correct – Attachment 2 of FHP-0001 contains applicable Mode 5 Tech Specs which must be met prior to entry into Mode 5.
- C. GOP-0002 provides guidance for plant shutdown and entry into Mode 4, but does not contain guidance concerning applicable Mode 5 Tech Specs.
- D. GMP-0102 contains the direction to notify the control room prior to detensioning the vessel head which results in entry into Mode 5, but does not provide the list of applicable Tech Specs which must be met prior to entry into Mode 5.

Technical Reference(s): FHP-0001 Rev 34 Pg 34 of 40

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-RF-PROC Obj 28

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.7

Comments:

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QUESTION 96 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	3	Group #	Equip Control
K/A #	G.2.2.7		IR 3.6

Knowledge of the process for conducting special or infrequent tests.

Proposed Question:

In accordance with EN-OP-116, INFREQUENTLY PERFORMED TESTS OR EVOLUTIONS, the individual who has the experience and expertise to exercise “continuous responsibility” for the oversight of a particular test or evolution is the _____ (1) _____ who is designated by the _____ (2) _____.

- A. (1) Senior Line Manager (SLM); (2) General Manager Plant Operations (GMPO)
- B. (1) Control Room Supervisor (CRS); (2) Operations Shift Manager (OSM)
- C. (1) GMPO; (2) Site Vice President
- D. (1) Test Coordinator; (2) OSM

Proposed Answer: A.

Explanation:

- A. The SLM who is senior to the Shift Manager has continuous responsibility for the oversight of the IPTE and is assigned by the GMPO.
- B. See “A”.
- C. See “A”.
- D. The Test Coordinator is responsible for the overall conduct of the test, but not the oversight (see “A”)

Technical Reference(s): EN-OP-116 Rev 12, Pg 7 of 35

Proposed references to be provided to applicants during examination: None

Learning Objective: Training provided by R-QC-HLO-SROU & R-QC-HLO-SROI

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.3

Comments:

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QUESTION 97 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	3	Group #	Equip Control
K/A #	G.2.2.20		IR 3.8

Knowledge of the process for managing troubleshooting activities.
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Proposed Question:

A component failure requires investigation to determine the specific type of repair required to return the component to operation. The investigation requires complex equipment manipulations beyond available procedure guidance.

(1) Which of the following procedures provides process guidance for this situation and (2) the highest level of approval required to commence the activity?

- A. (1) EN-MA-125, TROUBLESHOOTING CONTROL OF MAINTENANCE ACTIVITIES; (2) Operations Manager approval is required.
- B. (1) EN-OP-115, CONDUCT OF OPERATIONS, (2) Level of approval required is based upon risk.
- C. (1) EN-OP-115, CONDUCT OF OPERATIONS, (2) Shift Manager approval is required.
- D. (1) EN-MA-125, TROUBLESHOOTING CONTROL OF MAINTENANCE ACTIVITIES; (2) Level of approval required is based upon risk.

Proposed Answer: D.

Explanation:

A. Part 1 is correct, but the level of approval is based upon risk.

B. Part 2 is correct, but the applicable procedure is EN-MA-125.

C. Both parts are incorrect. See "D".

D. Correct – The situation described in the stem requires troubleshooting to determine the type of repair required. EN-MA-125 describes the process used for troubleshooting activities. The level of approval is based upon risk as determined by the Troubleshooting Determination Level (Att. 9.2)

Technical Reference(s): EN-MA-125 Rev 16 Pg 9-10 of 54

Proposed references to be provided to applicants during examination: None

Learning Objective: Training provided by R-QC-HLO-SROU & R-QC-HLO-SROI

Question Source: Bank # March 2010 Audit # 97

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒4 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.5

Comments: the original distractor "B" offered the OSM as (1), Validation team suggested this was too close to correct; changed (1) to CRS.

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QUESTION 98 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	3	Group #	Rad Control
K/A #	G.2.3.11		IR 4.3

Ability to control radiation releases.

Proposed Question:

Which of the following is required to discharge an LWS tank to the Mississippi River if RMS-RE107 is INOPERABLE?

- A. Two independent samples of the tank are analyzed. One qualified member of the Chemistry staff and one qualified member of the Radwaste staff independently verify the release rate calculations and the discharge valve lineup.
- B. A single sample is analyzed by two qualified members of the Chemistry staff independently. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.
- C. Two independent samples of the tank are analyzed. Two qualified members of the Chemistry staff independently verify the release rate calculations. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.
- D. A single sample of the tank is analyzed. One qualified member of the Chemistry staff verifies the release rate calculation and one qualified member of the Radwaste staff verifies the discharge valve lineup.

Proposed Answer: C.

Explanation:

- A. See C.
- B. See C.
- C. Correct – ADM-0054 Section 5.5 requires 2 independent samples, 2 Chemistry technicians to verify calculations and 2 qualified members of the Radwaste staff to verify the discharge valve lineup. These actions satisfy the requirements of TRM 3.3.11.2.
- D. See C.

Technical Reference(s): TRM 3.3.11.2 & ADM-0054 Rev 6A Section 5.5 Pg 9 of 12.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0603 Obj 8c

Question Source: Bank # 2008 RBS NRC #97

Question History: Last NRC Exam November 2012

Cognitive Level: Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43b.2 & b.4

Comments: **Appeared on one of last 2 NRC exams (1 of 2)**

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QUESTION 99 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	3	Group #	E-Plan
K/A #	G.2.4.22		IR 4.4

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

Proposed Question:

In accordance with 10CFR50.54x, a licensee may take action that departs from a license condition or a technical specification under certain conditions when this action is needed to _____.

- A. protect company assets.
- B. protect public health and safety.
- C. continue electrical generation during high demand periods.
- D. minimize safety related equipment out of service time.

Proposed Answer: B.

Explanation:

A. See "B".

B. Correct – The departure from license condition is only allowed during emergency conditions to protect public health and safety.

C. See "B".

D. See "B".

Technical Reference(s): 10CFR50.54x, EN-OP-115 Rev 14 Pg 5 of 89

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206 Obj 16

Question Source: New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43.b.1

Comments:

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QUESTION 100 Rev 0

Examination Outline Cross-Reference:

Level		RO <input type="checkbox"/>	SRO <input checked="" type="checkbox"/>
Tier #	3	Group #	E-Plan
K/A #	G.2.4.42		IR 3.8

Knowledge of emergency response facilities.

Proposed Question:

After all emergency response facilities are operational during a declared plant emergency, personnel in which of the areas below have the overall responsibility for determining recommended protective measures for offsite persons within the Emergency Planning Zone and for communication of these recommendations?

- A. Main Control Room
- B. Operations Support Center
- C. Technical Support Center
- D. Emergency Operating Facility

Proposed Answer: D.

Explanation:

A. See "D".

B. See "D".

C. See "D".

D. The shift manager in the MCR has this responsibility initially, but after all facilities are manned, this responsibility is transferred to the Emergency Director in the Emergency Operations Facility.

Technical Reference(s): EIP-2-007 Rev 25 Pg 3 of 9

Proposed references to be provided to applicants during examination: None

Learning Objective: RCBT-EP-SRORMED Obj 17

Question Source: New Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☒2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43b.5

Comments: