QUESTION 1 Rev 3

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	203000	K3.02	IR 3.5

Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on Suppression pool level ….

Proposed Question:

The plant was operating at 100% rated power when a small leak developed in the drywell resulting in a high drywell pressure signal of 1.68 psid. The plant was subsequently scrammed with plant conditions as follows:

- The reactor is shutdown with all control rods fully inserted
- MSIVs are closed
- RPV Level is +28 inches being controlled by RCIC with suction aligned to the CST
- RPV Pressure is being maintained between 500 to 600 psig with the SRVs
- Drywell pressure is currently 1.70 psid and slowly lowering
- RHR A and C, LPCS, and HPCS pumps and injection valves have been manually overridden
- RHR B is operating in Suppression Pool Cooling
- Div I, Div II, and Div III EDGs are running unloaded with their respective output breakers open

A loss of offsite power to ENS-SWG1B occurs. Which of the following describes the effect on suppression pool level and why?

A. Suppression Pool water level will continue to rise due to the addition of steam from RCIC exhaust and SRVs.

B. Suppression Pool water level will lower as RHR Pump C starts and injects into the RPV, RHR B will restart in the Suppression Pool Cooling lineup.

C. Suppression Pool water level will lower as RHR Pump B realigns and injects into the RPV.

D. Suppression Pool water level will remain constant with RHR B running in Suppression Pool Cooling mode of operation and all other ECCS pumps manually overridden off.

Proposed Answer: A

Explanation

A. Correct- Once ENS-SWG1B is deenergized and then subsequently reenergized by it's respective EDG, the manual override for RHR B/C will clear and with a high DW pressure signal sealed in, RHR B/C will start and realign to inject to the RPV, however RHR B and C **WILL NOT** inject to the RPV since RPV pressure is well above pump shutoff head of 324 psig. Suppression pool level will rise as the SRVs and RCIC exhaust continue to add volume to the suppression pool.

B. Not Correct. The manual override for RHR C pump and injection valve **WILL** reset on a loss of power and subsequent reenergization of ENS-SWG-1B and E12_FO24B (RHR B test return valve) will close, however due to the fact that RPV pressure is at a higher value than pump shutoff head of 225 psig, no injection from RHR will occur. This distractor is plausible if the candidate does not understand the operating characteristics of the RHR

system and keys in on the fact that RPV pressure is below the 487 psig permissive for the associated injection valves. This also requires the candidate to have knowledge of the automatic signals for the RHR test return valve and power supplies for the valve logic

C. Not Correct. The manual override for RHR B injection valve and test return valve **WILL** reset on a loss of power and subsequent re-energization of ENS-SWG-1B. The RHR B test return valve will close aligning RHR B out of the Suppression Pool Cooling mode, however due to the fact that RPV pressure is at a higher value than pump shutoff head of 324 psig, no injection from RHR will occur. This distractor is plausible if the candidate does not understand the operating characteristics of the RHR system and keys in on the fact that RPV pressure is below the 487 psig permissive for the associated injection valves. This also requires the candidate to have knowledge of the automatic signals for the RHR test return valve and power supplies for the valve logic.

D. Not Correct. Once power is lost to the ENS bus, the valve logic for the RHR systems valves is reset (powered from SCM-PNL01B). With an initiation signal still present, the RHR pumps and system valves respond to the initiation signal and align to inject into the RPV. This distractor is plausible is the student does not understand that the valve logic, while safety related, does not come from an inverter supply, and is therefore not maintained on a loss of power to the ENS bus.

Technical Reference(s): Technical Support Guidelines rev01 Pump Data Table;
GE 828E534AA, RHR Electrical Logic drawings; SOP-0031, Residual Heat Removal;
R-STM-0204, Rev 12 pp. 8, 12, 16, 20, 22, 40, 41, and Figure 12

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204, Rev 12, HLO Objectives 4, 6, 9, 10

Question Source: **New**

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 41.7 / 45.4

Comments:MC

QUESTION 2 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	205000	K4.05	IR 3.6

Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for Reactor cooldown rate ….

Proposed Question:

The plant is shutdown in MODE 4 with RHR "B" operating in the Shutdown Cooling mode of operation with the following conditions:

- Reactor Water Cleanup system (RWCU) is in service
- E12-F048B, RHR B Heat Exchanger Bypass valve is OPEN
- E12-F003B, RHR B Heat Exchanger Outlet valve is CLOSED

Which of the following provides an accurate indication of Reactor Coolant temperature?

A. RWCU Non-Regen heat exchanger inlet temperature G33-RTD-N006

B. RHR B heat exchanger (E12-EB001B) inlet temperature E12-T/C-N004B

C. RHR B heat exchanger (E12-EB001B) outlet temperature E12-T/C-N002B

D. RHR B heat exchanger discharge temperature E12-T/C-N027B

Proposed Answer: D

Explanation

A. Not Correct. With RWCU in service G33-RTD-N006 does not provide an accurate indication of Reactor coolant temperature since this water has already been cooled by RWCU return flow through the Regenerative heat exchanger. This answer is plausible if the candidate is unable to accurately understand the flowpath of the RWCU system

B. Not Correct. E12-T/C-N004B does not provide accurate indication of reactor coolant temperature since the heat exchanger outlet valve is closed (E12-F003B). This answer is plausible if the candidate is not able to understand flow in the RHR system, given plant conditions and associated piping diagrams.

C. Not Correct. E12-T/C-N002B does not provide accurate indication of reactor coolant temperature since the heat exchanger outlet valve is closed (E12-F003B) and even if it were open, the coolant flow exiting B.RHR B heat exchanger (E12-EB001B) would have already been cooled by service water. This answer is plausible if the candidate is not able to understand flow in the RHR system, given plant conditions and associated piping diagrams.

D. Correct. With reactor coolant flow bypassing the RHR heat exchanger and the heat exchanger outlet valve closed (E12-F003B), E12-T/C-N027B provides the only accurate indication of reactor coolant temperature.

Technical Reference(s): SOP-0031, Residual Heat Removal System, PID-27-07B, RHR B Piping Diagram, PID-26-03A, RWCU Piping Diagram

Proposed references to be provided to applicants during examination: PID-27-07B, RHR B Piping Diagram, PID-26-03A, RWCU Piping Diagram

Learning Objective: RLP-STM-0204, Rev 2, HLO Objectives 2, 8, 10
Question Source: **New**
Question History: Last NRC Exam N/A
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 41.7
Comments: MC

QUESTION 3 Rev 2

Examination Outline Cross-Reference:

Level	RO ☑	SRO ☐
Tier # 2	Group # 1	
K/A # 209001	A1.02	IR 3.2

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

Proposed Question:

The plant was operating at 100% rated power, when a loss of offsite power occurred followed by a failure of the HPCS EDG and a trip of the RCIC turbine. RPV level is being maintained by LPCS with the following plant conditions:

- RPV level is +25 inches and stable
- RPV pressure is 250 psig and stable.

E21-MOVF021, LPCS MIN FLOW Valve fails open. Which one of the following describes the effect on LPCS Pump Speed and LPCS Pump Discharge Pressure?

A. LPCS Pump speed will LOWER, and LPCS Pump Discharge Pressure will LOWER.

B. LPCS Pump speed will RISE, and LPCS Pump Discharge Pressure will LOWER.

C. LPCS Pump speed will RISE, and LPCS Pump Discharge Pressure will REMAIN CONSTANT.

D. LPCS Pump speed will REMAIN CONSTANT, and LPCS Pump Discharge Pressure will LOWER.

Proposed Answer: D

Explanation

A. Not correct. When the minimum flow valve opens, the system head will decrease resulting in an INCREASE in pump flow, however pump speed is unaffected and is dictated by power to the pump motor. Subsequently LPCS Pump Discharge Pressure will LOWER. This answer is plausible if the candidate inappropriately makes a correlation with the pump laws and assumes that pump speed will follow pump head.

B. Not Correct- Again, pump speed is not dependent on system or discharge head. While pump discharge head will lower as total system head falls due to opening of the minimum flow valve, this answer is also plausible if the candidate misinterprets the application of the pump laws and system operating characteristics and concludes that a rise in system flow correlates in a rise in pump speed.

C. NOT correct. Pump speed is not determined by system operating characteristics, but rather the speed of the pump motor. Additionally, opening of the pump minimum flow valve will result in a lower overall system head and subsequent lowering of pump discharge pressure. This answer is plausible if the candidate does not understand the operating characteristics of the pump and motor and the response on the system and pump discharge pressure of opening the minimum flow valve.

D. Correct. The speed of the motor is independent of system operating characteristics, however pump discharge will lower in response to lowering of overall system head when the minimum flow valve is opened.

Technical Reference(s): RLP-HLO-0153, Pumps, R-STM-0205, Rev 6 Figure 1

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205, Obj. 10 (As available)

Question Source: NEW

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 41.5 / 45.5

Comments:MC

QUESTION 4 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 209002	A2.01	IR 3.8

Ability to (a) predict the impacts of System initiation on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations….

Proposed Question:

The plant was operating at 100% rated power when a total loss of feedwater occurred. The plant subsequently scrammed and RCIC and HPCS automatically initiated on low RPV level.

E22-F004, HPCS injection valve was manually overridden CLOSED and RCIC is currently maintaining RPV level at +36”

The following indications were the highest values recorded for RPV Level:

- Narrow Range Level: +55 inches
- Wide Range Level: +48 inches

The Unit Operator has reported that pressure in the drywell is currently 1.34 psid and slowly rising

Which of the following will allow the HPCS to automatically respond to the rising pressure in the drywell?

A. Depressing E22A-S7, HPCS INITIATION RESET Pushbutton will allow the HPCS system to respond to a high drywell pressure signal.

B. Depressing both the E22A-S7 HPCS INITIATION RESET Pushbutton and the E22A-S6 HPCS HIGH WATER LEVEL 8 RESET Pushbutton will allow HPCS system to respond to a high drywell pressure signal.

C. Depressing the E22A-S6 HPCS HIGH WATER LEVEL 8 RESET Pushbutton will allow HPCS system to respond to a high drywell pressure signal.

D. No action is required to allow HPCS to respond to a high drywell pressure, since RPV level has been restored above the initiation setpoint.

Proposed Answer: A

Explanation

A. Correct- Given the events and the parameters the operator must reset HPCS initiation signal (E22A-S7) since the initiating signal of low level is above the initiation setpoint. Depressing the initiation reset pushbutton will clear the initiation signal from the logic and remove the manual override signal from the valve logic. This action is specifically referenced in the precautions and limitations of the system operating procedure. (SOP-0030)

B. Not Correct. Based on the parameters given, there is currently not a Level 8 signal sealed in for the HPCS injection valve, since the signal is driven from the Wide Range instruments. This answer is plausible if the student incorrectly believes that the Level 8 signal is driven from the Narrow Range instruments, as all other Level 8 signals are.

C. Not Correct. The HPCS Level 8 Reset pushbutton will have no effect since the signal is driven from the Wide Range level instrumentation. This answer is plausible if the student incorrectly assumes that the Level 8 signal is from the Narrow Range instrumentation and also does not understand that the Level 2 initiation signal for HPCS seals in until manually reset and that the initiation signal must be reset to reset the manual override logic for the injection valve.

D. Not Correct. The HPCS injection valve will not respond to any initiation signals until the initiation signal is reset. This answer is plausible if the student does not understand the requirements to reset the manual override logic for the HPCS injection valve.

Technical Reference(s): SOP-0030, Precautions and Limitations, ARP-P601-16A-B04 page 13, ARP-P601-16A-F02 page 33, R-STM-0203 rev 8 page 8

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-203, OBJ. 9

Question Source: **New**

Question History: N/A

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

10 CFR Part 55 Content: 41.5 / 45.6

Comments:MC

QUESTION 5 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 211000	A3.03	IR 3.8

Ability to monitor automatic operations of the STANDBY LIQUID CONTROL SYSTEM including Explosive valves indicating lights….

Proposed Question:

The plant has experienced an ATWS and the CRS has directed the UO to initiate Standby Liquid Control System (SLC) injection.

The following occurred:

- The UO took the P601 SLC PUMP A control switch to RUN and noted that C41-F001A (Suction Valve) failed to open.
- He then took the P601 SLC PUMP B control switch to RUN and observed C41-F001B beginning to open.

The following additional indications now exist:

- SLC "A" SQUIB CONTINUITY light is extinguished
- SLC "B" SQUIB CONTINUITY light is lit

Based on the above, when C41-F001B is full open, which one of the following describes the status of the SLC System?

A. SLC Pump "B" is injecting into the RPV through the "B" Squib valve.

B. SLC Pump "B" is injecting into the RPV through the "A" Squib valve.

C. SLC Pump "A" and "B" are injecting into the RPV through the "A" Squib valve.

D. SLC Pump "A" is NOT running and SLC Pump B is NOT injecting because the "B" Squib valve failed to open.

Proposed Answer: B

Explanation

A. NOT correct. Placing the SLC pump control switch in the RUN position causes the associated squib valve to fire and the associated suction valve to open. Based on indications given, after both pump switches were taken to RUN, the suction valve on SLC “B” was open, however the SLC “B” squib valve failed to fire and open. The resulting configuration is SLC “B” pump running injecting via the “A” SLC squib valve. This answer is plausible if the student does not understand the meaning of the squib valve indications and does not realize that either SLC pump is capable of injecting through either squib valve due to a normally cross-connected lineup on the discharge.

B. Correct: Placing the SLC pump control switch in the RUN position causes the associated squib valve to fire and the associated suction valve to open. Based on indications given, after both pump switches were taken to RUN, the suction valve on SLC “B” was open, however the SLC “B” squib valve failed to fire and open. The resulting configuration is SLC “B” pump running injecting via the “A” SLC squib valve.

C. NOT correct. Placing the SLC pump control switch in the RUN position causes the associated squib valve to fire and the associated suction valve to open. Based on indications given, after both pump switches were taken to

RUN, the suction valve on SLC “B” was open, however the SLC “B” squib valve failed to fire and open. The resulting configuration is SLC “B” pump running injecting via the “A” SLC squib valve. This answer is plausible if the student does not understand the pump logic that requires the associated pump suction valve to be open in order to start.

D. NOT correct. Placing the SLC pump control switch in the RUN position causes the associated squib valve to fire and the associated suction valve to open. Based on indications given, after both pump switches were taken to RUN, the suction valve on SLC “B” was open, however the SLC “B” squib valve failed to fire and open. The resulting configuration is SLC “B” pump running injecting via the “A” SLC squib valve. This answer is plausible if the student does not understand the normal system configuration that allows either SLC pump to inject via either squib valve.

Technical Reference(s): R-STM-201, Rev 8 Page 22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-201, OBJ. 4

Question Source: Bank # RBS-NRC-760

Question History: Last NRC Exam FEBRUARY 2003

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

10 CFR Part 55 Content: 41.7 / 45.7

Comments: MC

QUESTION 6 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	212000	A3.01	IR 4.4

Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including Reactor Power….

Proposed Question:

The plant is operating at 20% power when the following MSIVs go closed:

B21-F022A, MSL A Inboard Isolation valve
B21-F028A, MSL A Outboard Isolation valve
B21-F022C, MSL C Outboard Isolation valve

With NO operator action, which one (1) of the following describes the resulting reactor power and the impact on the RPS?

A. Plant remains operating at 20% power; there is a half-scram on RPS 'A'.

B. Plant remains operating at 20% power; there is a half-scram on RPS 'B'.

C. Plant remains operating at 20% power; there is no RPS actuation.

D. Plant automatically scrams (full scram).

Proposed Answer: C

Explanation

A. NOT correct for the reasons stated in the explanation for answer C. The plausibility is based on the Applicant having the knowledge of the logic arrangement and the ability to analyze the given plant conditions.

B. NOT correct for the reasons stated in the explanation for answer C. The plausibility is based on the Applicant having the knowledge of the logic arrangement and the ability to analyze the given plant conditions.

C. Correct. See RLP-STM-508, Figure 5 logic arrangement (or controlled drawings GE-828E531AA Sheets 5,6,7, and 8). With the stem conditions, a closure of MSL C is needed to cause a half-scram on RPS 'A', while a closure of MSL A would cause a half-scram on RPS 'B'…neither of these conditions occur in the stem conditions. With the stem conditions, a closure of MSL C is needed to cause a half-scram on RPS 'A', while a closure of MSL C would cause a half-scram on RPS 'B'…neither of these conditions occur in the stem conditions. Without there being a closure of three MSLs (only two are closed in the stem), there can be no full scram from MSIV closures. Clearly, the RPS coincident logic is not satisfied to cause any RPS actuation. Placing the plant at only 20% power in the stem conditions assures that there can be no indirect RPS actuation due to high reactor pressure or high steam flow.

D. NOT correct for the reasons stated in the explanation for answer C. The plausibility is based on the Applicant having the knowledge of the logic arrangement and the ability to analyze the given plant conditions.

Technical Reference(s): GE-828E531AA, RPS Elementary Logic Diagram, Sheets 5, 6, 7, and 8
R-STM-508, Rev 6 Figure 5

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-508, OBJ 14

Question Source: New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 41.7 / 45.7

Comments: MC

QUESTION 7 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	215003	G2.4.21	IR 4.0

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. (IRM)

Proposed Question:

During a plant startup at ~ 6% power the Reactor Mode Switch is in START/HOT STBY. The ATC switched all IRM/APRM recorders to APRM and selected all IRM detector drives for withdrawal.

APRM power indications:

A = 4%	B = 7%
C = 4%	D = 5%
E = 6%	F = 6%
G = 7%	H = 5%

The ATC depresses the DRIVE OUT pushbutton for the SRM/IRM detector drives to withdraw the IRM detectors.

Which one of the following describes the expected response?

A. The Retract Permit interlock will prevent IRM detector withdrawal.

B. All IRM detectors will fully withdraw, except IRMs A and C.

C. All IRM detectors will fully withdraw and NO further automatic actions.

D. All IRM detectors will fully withdraw and a control rod withdrawal block initiated.

Proposed Answer: D

Explanation

A. NOT correct. There is no interlock to prevent withdrawal of the IRM detectors from the core. The “IRM Detector Wrong Position” interlock will only generate a rod block IF the Reactor Mode Switch is not in RUN and the IRM detector is not fully inserted into the core. The candidate may select this answer if they have a misconception of the purpose and operation of the IRM Detector Wrong Position interlock.

B. NOT correct. As explained above, power level has no bearing on the ability to withdraw IRM detectors from the core. This answer is plausible if the candidate incorrectly assumes that the IRM detector drive motors are interlocked to prevent withdrawal under certain conditions.

C. NOT correct. With the Reactor Mode Switch in the START/HOT STBY position and the IRM detectors not fully inserted into the core, a control rod withdrawal block will be generated. This answer is plausible if the candidate does not understand the conditions that will generate control rod blocks.

D. Correct. With the Reactor Mode Switch in the START/HOT STBY position and the IRM detectors not fully inserted into the core, a control rod withdrawal block will be generated.

Technical Reference(s): R-STM-503, Table 6

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-503 OBJ H13

Question Source:	Bank #	RBS-NRC-860	
Question History:	Last NRC Exam	FEBRUARY 2003	
Cognitive Level:	Memory or Fundamental Knowledge ☑ 3		Comprehension or Analysis ☐
10 CFR Part 55 Content:	41.7 / 43.5 / 45.12		

Comments:MC

QUESTION 8 Rev 3

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	215004	A4.07	IR 3.4

Ability to manually operate and/or monitor in the control room Verification of proper functioning/ operability SRMs

Proposed Question:

A reactor startup is in progress with the reactor having just gone critical.

- Operators have verified SRM/IRM overlap
- Operators are continuing to withdraw control rods
- SRMs are being withdrawn as necessary to maintain SRM count rate between 1×10^3 and 1×10^5 cps.

The SRM 'B' detector fails and indicates 0 cps output.

Which of the following describes the status of the SRM system?

A. Generates a Div 1 half scram

B. Generates a Control Rod Withdrawal Block

C. Generates a Control Rod Insert Block

D. No action is generated.

Proposed Answer: B

Explanation

A. NOT correct since a) with the shorting links installed (normal configuration the SRMs will not generate scram signals, and b) a downscale on an SRM will only generate a control rod withdrawal block. This answer is plausible if the candidate does not know the normal configuration of the SRM scram logic and shorting links.

B. Correct. With IRM/SRM overlap verified and direction to maintain SRM count rate between 10E3 and 10E5 cps, it is procedurally determined that the IRMs are between Range 1 and Range 3. With the IRMs on Range 3 or below and SRMs counts less than 100 cps (SRM "B") with the SRMs not fully inserted, a control rod withdrawal block is generated.

C. NOT correct since the neutron monitoring systems will not directly generate any control rod insertion blocks. This is a plausible answer is the candidate does not understand that the neutron monitoring system will not generate control rod insertion blocks.

D. NOT correct for the reasons stated in the explanation of the correct answer B. The candidate may select this answer if they have a misconception of the operation of the SRM system or fail to properly analyze the given plant conditions.

Technical Reference(s): SOP-0074, Neutron Monitoring System, Rev 306 p. 13
GOP-0001, Plant Startup, Rev 85 p.36

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-503 OBJ 34

Question Source: New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 41.7 / 45.5 to 45.8

Comments: MC

QUESTION 9 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	215005 A3.08	IR 3.7	

Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including Control rod block status….

Proposed Question:

The plant is operating in Mode 2 with reactor power at 8%.

The following annunciators were received:

- CONTROL ROD WITHDRAWAL BLOCK
- APRM A OR E UPSCALE TRIP or INOP
- Div 1 Half SCRAM

Which of the following caused the given indications?

A. Reactor Mode Switch was taken to RUN

B. APRM Channel E indicates <5%

C. APRM Channel A indicates >15%

D. Less than 2 valid LPRM detectors per level are feeding APRM Channel A

Proposed Answer: C

Explanation

A. NOT correct since taking the Mode switch to RUN would result in no signals being generated based on the given conditions. This answer is plausible since repositioning the Reactor Mode switch will affect the signals that generate rod blocks and upscale trips.

B. NOT correct since a downscale of less than 5% would result in only a control rod withdrawal block and not an INOP signal. This answer is plausible if the candidate does understand that a downscale will not result in an INOP signal.

C. Correct since a reading of greater than 15% with the Reactor Mode Switch in START/HOT STBY would result in both an UPSCALE trip in Div 1 RPS and a control rod withdrawal block.

D. NOT correct. This answer is plausible since the APRM has less than 2 valid LPRM inputs per axial level and this would render the APRM administratively INOP, but would not result in an INOP trip signal.

Technical Reference(s): SOP-0074, Neutron Monitoring System, Rev 306 p. 20
RBS Technical Specifications B 3.3.1.1 RPS Instrumentation Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-503, OBJ 21

Question Source: New

Question History: Last NRC Exam N/A

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

10 CFR Part 55 Content: 41.7 / 45.7

Comments:

QUESTION 10 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	217000	A2.02	IR 3.8

Ability to (a) predict the impacts of Turbine trips on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; 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 transient the following conditions exist:

- RPV Level -50 inches
- RCIC Suction pressure 22" Hg vac. suction pressure
- RPV pressure band 800-1090psig with SRVs

RCIC suction has been aligned to the suppression pool

(1) What is the status of E51-F013 RCIC Injection Valve and E51-F045 RCIC Steam Supply Stop Valve
(2) What procedure actions are needed to utilize RCIC as an injection source?

		1	2
A.	F013	open	Slow Roll RCIC
	F045	open	Place RCIC in RPV Level Control Mode
B.	F013	closed	Reset RCIC trip
	F045	open	Place RCIC in RPV Level Control Mode
C.	F013	open	Slow Roll RCIC
	F045	closed	Place RCIC in RPV Pressure Control Mode
D.	F013	closed	Reset RCIC trip
	F045	closed	Place RCIC in RPV Pressure Control Mode

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible since F045 will be open with an initiation signal present and the procedure to place RCIC in the RPV level control mode requires a slow roll, however,F013 will close when the turbine trips on low suction pressure. And the turbine has not been reset in this distractor.

B. Correct - F045 will be open with an initiation signal present and F013 will close when the turbine trips. The turbine will trip at 20"Hg vac and will need to be reset and then placed in the RPV level control mode until RPV level is >10".

C. NOT Correct. This answer is plausible since the RPV pressure control mode would reduce the amount of heat added to the suppression pool due to SRV operation, however the RCIC turbine has tripped causing the F013 to close and F045 is open due to the initiation signal

D. NOT Correct . This answer is plausible since the RCIC turbine has been tripped and must be reset and RPV pressure control mode would reduce the amount of heat added to the suppression pool due to SRV operation. However the F045 will be open with an initiation signal present

Technical Reference(s): SOP-035 rev 49 p. 19 ; R-STM-0209 rev12 pages 24, 26, 44, 45

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 45.6

Comments:JH

QUESTION 11 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	218000	A1.02	IR 3.7

Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including ADS valve acoustical monitor noise….

Proposed Question:

Due to plant conditions the Automatic Depressurization System (ADS) has initiated.

The SRV tail pipe temperature recorder is out of service for repairs (B21-R614)

What indication is available for the operator to monitor and determine if **ALL** ADS SRVs have opened?

A. An increase in Main Steam Line flows on H13-P680-7B

B. Seven red indicating lights lit on the SRV Status – Acoustic Monitor on, H13-P601-19B

C. Seven SRV red open indicating lights lit on H13-P601-19C

D. Alarm H13-P601-19A-A09, Main Steam Safety Relief Valve Open, sealed in

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible since main steam line flows will increase when the ADS system is initiated,, however the operator will not be able to determine if all seven SRVs have opened, only that at least one per steam line have opened

B. Correct – the acoustic monitor looks at flow noise for each of the 16 SRV tailpipes. When sufficient noise is sensed, the module for that SRV initiates an alarm in the MCR and energizes the red status light on P601-19B. When ADS is initiated all 7SRVs associated with the ADS system will open and all seven of the SRV status lights will energize.

C. Not Correct. This answer is plausible since the red open indicating lights will be lit, however this is only an indication that the solenoid for that SRV has been energized, it is not a true indication that the SRV has opened

D. Not Correct. This answer is plausible since the alarm driven by the acoustic monitor will be sealed in when a SRV is open, however it only indicates that at least one SRV is open and does not “reflash” as more SRVs open.

Technical Reference(s): ARP-601-19A-A09 rev 34 page 17, STM-109 rev14 page 8, 16

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 45.5

Comments:

QUESTION 12 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	223002	K1.01	IR 3.8

Knowledge of the physical connections and/or cause/effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and Main steam system….

Proposed Question:

While operating at 100% power, a loss of RPS 'B' has occurred.

What is the effect on the Main Steam System?

A. Inboard Main Steam Line Drains close. All MSIV remain open with a ½ isolation signal present.

B. Inboard Main Steam Line Drains AND Inboard MSIVs close.

C. Outboard Main Steam Line Drains close. All MSIV remain open with a ½ isolation signal present.

D. Outboard Main Steam Line Drains AND Outboard MSIVs close.

Proposed Answer: A.

Explanation:

A. Correct. RPS B supplies power to the NS^4 inboard logic Channels B & C. The 2 out of 2 logic channels for the inboard MSL drains will be tripped resulting in valve closure. The MSIVs however will remain open with a ½ isolation signal present.

B. Not Correct. This answer is plausible since the isolation logic for the inboiard MSL drains and MSIV is powered from RPS, however, the MSIVs do not close on loss of one RPS bus.

C. Not Correct. This answer is plausible since the isolation logic for the MSL Drains and MSIVs are supplied from RPS, however RPS B supplies Inboard logic channels. Outboard valves are not affected.

D. Not Correct. This answer is plausible since the isolation logic for the MSL Drains and MSIVs are powered from RPS, however RPS B supplied Inboard logic channels. Outboard valves are not affected. Additionally, MSIVs do not close on loss of one RPS bus.

Technical Reference(s): R-STM-0058 rev 10 pp 5-7, 10-16 figures 11 and 12

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0058 Obj. 5, 9, 10

Question Source: Bank # RBS Audit 2010

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.2 to 41.9 / 45.7 to 45.8

Comments: JH

QUESTION 13 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	239002	K5.02	IR 3.7

Knowledge of the operational implications of Safety function of SRV operation concepts as they apply to RELIEF/SAFETY VALVES ….

Proposed Question:

The plant is in a Station Blackout.

All Safety Relief Valve (SRV) accumulator air pressures are zero psig.

How will SRVs respond to rising reactor pressure under these conditions?

A. At the relief setpoint, a solenoid valve will open to admit steam pressure to a spring-loaded disk to pop open the respective SRVs.

B. At the safety setpoint, the actuator spring force begins to be overcome by steam pressure. The valves gradually open and are full open at 110% of the setpoint.

C. Seven Safety Relief Valves will open in the safety mode at 1195 psig. Five more Safety Relief Valves will open in the safety mode at 1205 psig.

D. Ten Safety Relief Valves will open in the safety mode at 1195 psig. The remaining Safety Relief Valves will open in the safety mode at 1210 psig.

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible since during a station blackout DC power will be available for the operation of the SRVs in the relief mode, however for the given conditions, no air pressure is available which is the motive force to operate the SRVs, not steam pressure

B. NOT Correct. This answer is plausible - since it involves the safety mode, but it incorrectly states SRV opening would be gradual when it would actually pop open prior to 110% of setpoint

C. Correct – this explains the operation of the SRVs in the safety mode, 7 open at 1195#, 5 open at 1205# and 4 open at 1210#

D. NOT Correct. This answer is plausible – the pressures given are correct for the safety mode of operation, however the number of SRVs associated with the pressures is not

Technical Reference(s): R-STM-0109 rev 14 pp 10, 61

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 45.3

Comments:JH

QUESTION 14 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
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 Three element control (main steam flow, reactor feedwater flow and reactor water level provide input….

Proposed Question:

The feedwater flow input that is designed to be used by the three element feedwater level control system is generated by:

A. summation of the feedwater suctions flow venturi

B. summation of the feedwater headers flow venturi

C. summation of the feedwater headers ultrasonic flow meters

D. summation of the feedwater headers flow venturi with a correction factor developed by the feedwater header ultrasonic flow meters

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible since these flow elements are used in the recirc flow control valve runback logic and the operation of the feedwater pump min flow valve, they are not in the feedwater level control design

B. Correct – the two flow elements feed two independent flow signals into the feedwater level control system which sums the signals and compares the feed flow and steam flow signals to create a delta signal to anticipate level changes.

C. NOT Correct. This answer is plausible since the flow signal from the ultrasonic feedwater flow meter is used by the plant heat balance calculation not in the feedwater level control design

D. NOT Correct. This answer is plausible since the ultrasonic feedwater flow meter compares its flow signal to the flow signal of the feedwater header venture signal to develop a correction factor if it is needed to be used for the plant heat balance calculation.

Technical Reference(s): R-STM-0107 rev 27 pp 32, 57-58

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7

Comments: JH

QUESTION 15 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	261000	K3.02	IR 3.6

Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on Off-site release rate….

Proposed Question:

A primary system is leaking into the Reactor Water Cleanup Pump Room.

Standby Gas Treatment (GTS) has been manually initiated. All expected isolations have occurred, but both trains of GTS have failed to start.

Which of the following indications would be expected under these conditions?

A. Low containment pressure

B. Negative auxiliary building pressure

C. Elevated annulus radiation levels

D. Elevated off-site radiation levels

Proposed Answer: D.

Explanation

A. NOT Correct. This answer is plausible since on an Initiation signal, GTS takes suction on the Aux Bldg and Annulus, however with containment pressure being referenced to the auxiliary building, a positive pressure in the auxiliary bldg due to the steam leak and no GTS train running will indicate a higher containment pressure.

B. NOT Correct. This answer is plausible - this is the expected indication for GTS in operation or normal Aux. building ventilation, however with both trains of GTS off and the initiation signal present the normal aux building ventilation has shut down and the steam leak would raise the building pressure

C. NOT Correct. This answer is plausible – the annulus and aux building are connected, however the annulus pressure control system will be shutdown due to the GTS initiation signal and the annulus will no longer maintained at a negative pressure. Any radiation due to the steam leak would not be pulled into the annulus .

D. Correct – with no GTS train running the aux building pressure will rise due to the steam leak and exit the building without being treated by the GTS system.

Technical Reference(s): R-STM-0257 rev 5 pp 3-4

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS NRC2010

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.6

Comments: JH

QUESTION 16 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	262001	K2.01	IR 3.3

Knowledge of electrical power supplies to the Following Off-site sources of power

Proposed Question:

Which of the following describes the electrical power supplies to River Bend's offsite power sources?

A. RSS 1 and RSS 2 can both be supplied by the North and South 230 kv buses at Fancy Point Switchyard.

B. RSS 1 can only be supplied by the North 230 kv bus at Fancy Point Switchyard.

C. RSS 2 can only be supplied by the South 230 kv bus at Fancy Point Switchyard.

D. RSS 1 and RSS 2 can both be supplied by 500 kv yard at Fancy Point Switchyard if the 230 kv yard is de-energized.

Proposed Answer: A.

Explanation

A. Correct- Fancy Point Switchyard has multiple possible breaker alignments such that either bus can be removed from service (north or south) and still supply both offsite power sources (RSS 1 & 2).

B. NOT Correct. This answer is plausible since the equipment identification scheme at RBS is a north to south and east to west arrangement. Meaning the north bus would be labeled as A or 1. This is also the convention used during outages. If there is to be an 'A'/Div1 bus outage the north bus at Fancy Point would have maintenance activities conducted at the same time.

C. NOT Correct. This answer is plausible since the equipment identification scheme at RBS is a north to and east to west arrangement. Meaning the south bus would be B or 2. This is also the convention used during outages. If there is to be an 'B'/Div2 bus outage the south bus at Fancy Point would have maintenance activities conducted at the same time.

D. NOT Correct. This answer is plausible – the 500kV section of Fancy Point can supply power to the 230kV busses, however there is no path from the 500kv yard to RSS 1/2 without use of the 230kv switchyard

Technical Reference(s): EE-1AC

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS 2010 Audit

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7

Comments:JH

QUESTION 17 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	262002	K1.10	IR 2.6

Knowledge of the physical connections and/or cause/effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the Fire Protection System.

Proposed Question:

During a loss of all AC power, how are the Fire Detection Remote Data Acquisition (RDAC) Panels affected?

A. All RDAC panels will be de-energized.

B. All RDAC panels will be energized by the station batteries through Uninterruptible Power Supplies (UPS).

C. All RDAC panels will be energized by the station batteries through Uninterruptible Power Supplies (UPS), EXCEPT for FPM-RDAC PNL 26 (Fire Pump House) & FPM-RDAC PNL 29 (Makeup Water Intake Structure) which have local battery backup power source.

D. All RDAC panels will be energized by a local battery backup power source EXCEPT for FPM-RDAC PNL 26 (Fire Pump House) & FPM-RDAC PNL 29 (Makeup Water Intake Structure) which will be powered by the station batteries through Uninterruptible Power Supplies (UPS).

Proposed Answer: C.

Explanation:

A. NOT Correct. This answer is plausible since some RDAC panels are protected with a battery backup (RDAC Panels 26 and 29)

B. NOT Correct. This answer is plausible since most, but not all RDAC panels are protected via uninterruptible power supplies.

C. Correct. All RDAC panels, except RDAC 26 & 29 are powered through a UPS. When the AC source is lost, the station batteries continue to power the loads through the UPS. RDAC 26 & 29 due to their locations outside the protected area have a local battery backup,

D. NOT Correct. This answer is plausible since the power scheme for RDAC Panels 26 and 29 is different, however RDAC Panels 26 and 29 are the panels that are backed up by batteries only and not a UPS.

Technical Reference(s): R-STM-0250, Rev 6 pp. 41-43 ; SOP-0036, Rev 305 p.3

Proposed references to be provided to applicants during examination: none

Learning Objective: R-STM-0250 Obj. 7

Question Source: Bank # RBS Audit 2010

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.2 to 41.9 / 45.7 to 45.8

Comments: JH

May 2015 River Bend Station
NRC Initial License Retake Examination
Reactor Operator

QUESTION 18 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	263000	K2.01	IR 3.1

Knowledge of electrical power supplies to the Following Major D.C. loads

Proposed Question:

ENB-MCC1 is powered from _____.

A. ENB-PNL02A

B. ENB-PNL02B

C. ENB-SWG1A

D. ENB-SWG1B

Proposed Answer: C.

Explanation

A. NOT Correct. This answer is plausible since this is a safety related 125VDC supply panel, however it does not power ENB-MCC1

B. NOT Correct. This answer is plausible since this is a safety related 125VDC supply panel, however it does not power ENB-MCC1

C. Correct - ENB-MCC1 is powered from ENB-SWG1A.

D. NOT Correct. This answer is plausible since this is a safety related 125VDC supply switch gear, however it does not power ENB-MCC1

Technical Reference(s): EE-001AC(J-3)

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS Audit 2010

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7

Comments: JH

QUESTION 19 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	264000	K3.01	IR 4.2

Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on Emergency core cooling systems

Proposed Question:

In response to a Loss of Offsite Power, the plant responded as follows:

EGS-EG1A experienced an Overspeed condition
EGS-EG1B experienced a Jacket Water High Temperature condition
(Both overspeed and Jacket Water Temp values are in excess of the normal trip values)

Current conditions are:

- RPV Level -50 inches
- RPV Pressure 850 psig
- Drywell Pressure 1.75 psid

Regarding ECCS pumps, which of the following is correct?

A. Div 1 OFF, Div 2 Running on Min Flow, Div 3 Injecting

B. Div 1 & Div 2 OFF, Div 3 Injecting

C. Div 1, Div 2, & Div 3 Running on Min Flow

D. Div 1 & Div 2 Running on Min Flow, Div 3 Injecting

Proposed Answer: A.

Explanation

A. Correct - All diesels have received a start signal (LOP and DW pressure) The Div 1 DG has tripped since the overspeed trip is never bypassed. The Div 2 DG will continue to run with the high jacket water temperature present due to the E-start condition. With no power, Div 1 ECCS pumps are OFF, Div 2 ECCS pumps are running, but not injecting due to reactor pressure above the injection valve open permissive of 487 psig. Div 3 DG is running and HPCS is injecting due to being below Level 2(-43").

B. NOT Correct. This answer is plausible since the status of the Div 1&3 ECCS pumps are correct, however Div 2 pumps will be running due to the bypass of the high jacket water temperature with an E start signal

C. NOT Correct. This answer is plausible since the status of Div 2 is correct and Div 3 pumps are running and injecting so the min flow valve will be closed

D. NOT Correct. This answer is plausible since the status of Div 2 & 3 is correct, however Div 1 pumps will not be running due to the loose of power.

Technical Reference(s): R-STM-309S rev 14 p. 9, 55 ; R-STM-0309H Rev 13 p. 4;
R-STM-0204 rev 12 p. 12 ; R-STM-0205 rev 6 p. 6

Proposed references to be provided to applicants during examination: none

Learning Objective: RLP-STM-0204 Obj 4, 17C, RLP-STM-0309 Obj 5, 14

Question Source: Bank # RBS Audit 2010

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.4

Comments: JH

QUESTION 20 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	300000	K4.02	IR 3.0

Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for Cross-over to other air systems

Proposed Question:

The design feature which allows the Instrument Air system to be supplied by the Service Air system causes _____(1)_____ to open at __(2)___ IAS header pressure.

A. (1) SAS-AOV134, SAS/IAS CROSS TIE VALVE; (2) 110 psig

B. (1) SAS-AOV134, SAS/IAS CROSS TIE VALVE; (2) 113 psig

C. (1) SAS-AOV133, SERVICE AIR HEADER BLOCK VLV; (2) 113 psig

D. (1) SAS-AOV133, SERVICE AIR HEADER BLOCK VLV; (2) 110 psig

Proposed Answer: B.

Explanation:

A. NOT Correct. This answer is plausible because SAS-AOV134 will open to cross tie Service Air to IAS however, this occurs at 113 psig.

B. Correct. SAS-AOV134 opens at 113 psig to cross tie SAS to IAS

C. NOT Correct. This answer is plausible - SAS-AOV133 operates at 110 psig. It does not cross tie the systems however, it blocks the SAS system from the SAS compressors to allow all of the SAS air to be supply the IAS system

D. NOT Correct. This answer is plausible - SAS-AOV133 operates at 110 psig. It does not cross tie the systems however, it blocks the SAS system from the SAS compressors to allow all of the SAS air to be supply the IAS system

Technical Reference(s): R-STM-0121 rev16 page 26&27

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0121 Obj. 3

Question Source: Bank # RBS Audit 2010

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7

Comments: JH

QUESTION 21 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	400000	K6.05	IR 3.0

Knowledge of the effect that a loss or malfunction of the Pumps will have on the CCWS

Proposed Question:

A failed motor bearing in CCP-P1A, RPCCW PUMP 1A, has caused the pump to trip.

The standby CCP pump will automatically start when _____(1)_____.
The CCP-P1A anti-pump circuitry _____(2)_____.

A. (1) the running pump trips; (2) must be reset manually.

B. (1) the running pump trips; (2) will automatically reset.

C. (1) system pressure drops to 100 psig; (2) will automatically reset.

D. (1) system pressure drops to 100 psig; (2) must be reset manually.

Proposed Answer: A.

Explanation:

A. Correct – The trip of a running CCP pump automatically starts the standby CCP pump. The anti-pump circuitry is reset by locking out and resetting the tripped pump.

B. NOT Correct. This answer is plausible - Part 1 is correct, but the anti pump circuitry does not automatically reset when the trip condition is cleared. It must be manually reset by locking out and resetting the tripped pump.

C.. NOT Correct. This answer is plausible - The low pressure auto start occurs at 95 psig. Additionally, the anti pump circuitry does not automatically reset when the trip condition is cleared. It must be manually reset by locking out and resetting the tripped pump.

D. NOT Correct. This answer is plausible - The low pressure auto start occurs at 95 psig. Part 2 is correct.

Technical Reference(s): R-STM-0115 Rev 6 p. 7.
SOP-0016 Rev 43 P&L 2.9;

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj 3a, 4b, 6

Question Source: Bank # RBS NRC 2012

Question History: Last NRC Exam 2012 NRC Q#48

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

10 CFR Part 55 Content: 41.7 / 45.7

Comments: JH

QUESTION 22 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	218000	K5.01	IR 3.8

Knowledge of the operational implications of the ADS logic operation concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM

Proposed Question:

Following a plant transient, the following conditions exist:

- RPV Level -155 inches, slowly rising
- RPV Pressure 220 psig, lowering
- ADS actuation has occurred

If the operator places the control switch on H13-P601 for B21-F051C ADS SRV to OFF, how will the valve respond?

The SRV will ____.

A. close and remain closed.

B. remain open until RPV Level rises above Level 1.

C. remain open until the ADS logic is reset.

D. close and will only operate in the Safety mode.

Proposed Answer: C.

Explanation:

A. NOT Correct. This answer is plausible since placing the control switch in the OFF position defeats the capability of the SRV to actuate in Relief mode, however, actuation on ADS is independent of control switch position since the ADS contacts bypass the control switch.

B. NOT Correct. Actuation on ADS is independent of control switch position since the ADS contacts bypass the control switch. This is a plausible distractor because ADS logic level 1 signal does not seal in to bypass the high drywell signal.

C. Correct The valve will remain open because the ADS contacts bypass the control switch in the circuitry. The only method of closing the ADS valves once ADS actuation has occurred is to depress the Level 3 Seal-In Reset Pushbutton..

D. NOT Correct. This answer is plausible since placing the control switch in the OFF position defeats the capability of the SRV to actuate in the Relief mode. Safety Actuation is driven against spring pressure. Actuation on ADS is independent of control switch position since the ADS contacts bypass the control switch.

Technical Reference(s): R-STM-0202 Rev 2 pp. 13, 34, fig 2 and 3

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0202 Obj 7

Question Source: Bank # RBS NRC 2010

Question History: Last NRC Exam April 2010 NRC Q#40

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

10 CFR Part 55 Content: 41.5 / 45.3

Comments: JH

QUESTION 23 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	259002	K6.03	IR 3.1

Knowledge of the effect that a loss or malfunction of the Main steam flow input will have on REACTOR WATER LEVEL CONTROL SYSTEM

Proposed Question:

The plant is operating at 85% power. The Master Level Control System is in Three Element control.

The diaphragm in one of the main steam flow instruments ruptures.

As a result of the failed instrument, the Feedwater Level Control System response will be to ____.

A. close the Feed Reg Valves to a lower position causing actual RPV water level to lower and stabilize at a level above Level 3.

B. open the Feed Reg Valves to full open causing actual RPV water level to rise above Level 8.

C. close the Feed Reg Valves to a lower position causing actual RPV water level to lower below Level 3.

D. open the Feed Reg Valves to a higher position causing actual RPV Water level to rise and stabilize at a level below Level 8.

Proposed Answer: A.

Explanation

A. Correct - With a ruptured diaphragm, zero differential pressure will be detected across the instrument resulting in zero indicated steam flow from the instrument. This causes total steam flow to be lower. As a result, the FWLC will try to match Feed flow to Steam flow by closing the FW Reg Vlvs. The FWLC system will sense the lowering level and readjust Feedwater input to the vessel.

B. NOT Correct. This is a plausible answer if the candidate does not understand a ruptured diaphragm will cause overall steam flow signal to lower and a resultant lower feed flow signal from the FWLC system and that the system is level dominate.

C. NOT Correct. This answer is plausible since a ruptured diaphragm on a steam flow instrument will cause a corresponding lowering of FW demand, however the candidate may choose this answer if they do not understand that the FWLC system is level dominate

D. NOT Correct. This is a plausible answer if the candidate does not understand a ruptured diaphragm will cause overall steam flow signal to lower and a resultant lower feed flow signal from the FWLC system

Technical Reference(s): R-STM-0107 Rev 28 pp 57, 59 and 69

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0107B Obj H14d

Question Source: Bank # RBS March 2010 AUDIT Q#45

Question History: Last NRC Exam NA

Cognitive Level:	Memory or Fundamental Knowledge ☐	Comprehension or Analysis ☑3
10 CFR Part 55 Content:	41.5 / 45.3	

Comments: JH

QUESTION 24 Rev 3

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	223002	A1.02	IR 3.7

Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including Valve closures

Proposed Question:

The plant is operating at rated power when all Reactor Feedwater Pumps trip.

Reactor water level drops to -50" before operators restore it to normal with HPCS and RCIC.

Without any further operator action, which of the following identifies an expected change to plant systems as a direct result of the low reactor water level isolation?

A. Drywell temperature rises.

B. Drywell pressure lowers.

B. Containment Radiation levels rises.

C. Containment pressure lowers.

Proposed Answer: A

Explanation

A. Correct – on a RPV water level of -43" the primary containment isolation system will isolate service water to primary containment and the drywell unit coolers will no longer have a cooling media

B. NOT Correct. This answer is plausible – the drywell pressure would normally lower due to the loss of heat input when the reactor SCRAMed, however the cooling media was isolated at -43". The resulting temperature rise will raise drywell pressure

C. NOT Correct. This answer is plausible – if the candidate considers the CRUD burst that occurs when the reactor is scramed from a high power level.

D. NOT Correct. This answer is plausible – containment pressure would normally lower due to the loss of heat input when the reactor SCRAMed, however the cooling media was isolated at -43". The resulting temperature rise will raise containment pressure

Technical Reference(s): AOP-003 rev33 pp 10, 15 ; R-STM-0118, Rev 25 fig 4 & 5

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # GGNS 2011 NRC

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 45.5

Comments: JH

QUESTION 25 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	1
K/A #	264000	K6.01	IR 3.8

Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS Starting air

Proposed Question:

The plant is operating at 100% power.

The HPCS diesel generator left air start system is tagged out to replace the air receiver tank relief valve.

A loss of power deenergizes the HPCS diesel generator right air start compressor.

How many normal start attempts are available for the HPCS diesel generator?

A. 5

B. 8

C. 10

D. 16

Proposed Answer: A

Explanation
A. Correct – per STM-309H, each receiver tank has the capacity to complete 5 normal start attempts
B. NOT Correct. This answer is plausible – this is the number of normal start attempts for the Div 1&2 diesel generators for each air receiver
C. NOT Correct. This answer is plausible – the HPCS diesel air receiver tanks can be cross tied to have 10 starts available, however the stem indicates that the left receiver tank is tagged out
D. NOT Correct. This answer is plausible – Div 1&2 diesels have two independent air start system that have a combined air start capacity of 16 start attempts

Technical Reference(s): R-STM-309S Rev14 p. 29 ; R-STM-309H rev13 p. 35-36

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.7

Comments: JH

QUESTION 26 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group # 1	
K/A #	400000	G2.1.27	IR 3.9

Knowledge of system purpose and/or function of the CCWS

Proposed Question:

Which of the following describes the function of the Component Cooling Primary (CCP) system?

A. CCP is an intermediate heat transfer system for Reactor auxiliary systems and serves as a backup to the SWP system for safety related equipment.

B. CCP cools Reactor auxiliary systems during normal operation and is a buffer between potentially contaminated systems and the environment.

C. CCP provides a safety related source of cooling to equipment essential for safe reactor shutdown and is a buffer between potentially contaminated systems and the environment.

D. CCP serves as a backup to the SWP system for safety related equipment and provides a reliable source of cooling to equipment essential for safe reactor shutdown

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible – the first half of the distracter is a function of the CCP system, however the SWP system is a backup to the CCP system

B. Correct – both parts of the statement are true concerning the function of the CCP system as described in STM-115

C. NOT Correct. This answer is plausible – the second half of the distracter is true for the CCP system, however the first half is true for the SWP system as described in STM-118

D. NOT Correct. This answer is plausible – CCP does provide a reliable source of cooling to equipment for safe reactor shutdown, however it serves as the normal source of cooling with SWP being the safety related backup.

Technical Reference(s): R-STM-0118 rev25 p.5 ; R-STM-0115 rev 6 p.4

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7

Comments: JH

QUESTION 27 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	2
K/A #	201001	K2.03	IR 3.5

Knowledge of electrical power supplies to the following, Backup SCRAM valve solenoids

Proposed Question:

What is the power supply to the Backup SCRAM valve solenoids?

A. RPS bus A & B

B. EHS-MCC8A

C. ENB-PNL02A & 02B

D. BYS-PNL02A2

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible since this is the power supply to the normal SCRAM valves

B. NOT Correct. This answer is plausible – this is the MC located in the Fuel Building that supplies the CRD isolation valve, C11-F083

C. Correct – the backup SCRAM valves are powered from a safety related DC power supply

D. NOT Correct. This answer is plausible since this is the power supply to the ARI solenoid valves

Technical Reference(s): STM-0052 rev9 page 22,26,28,65

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS-OPS-3838

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7

Comments: JH

QUESTION 28 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	2
K/A #	201003	K1.01	IR 3.2

Knowledge of the physical connections and/or cause/effect relationships between Control Rod and Drive Mechanism and the following, control rod drive hydraulic system

Proposed Question:

Control rod withdrawal speed would increase if the control rod drive hydraulic ____.

A. Flow Control Valve fails open

B. Pressure Control Valve fails open

A. associated stabilizer valve fails closed

B. cooling/exhaust water pressure equalizing valve fails closed

Proposed Answer: A

Explanation

A. Correct. – because it would provide for increased drive water pressure to the CRD mechanisms, causing control rod speed to increase..

B. NOT Correct. Plausible – because CRD drive pressure would decrease, causing control rod speed to decrease

C. NOT Correct. Plausible – because the associated stabilizer valve does go closed on a withdrawal. After withdraw, a failed-closed stab valve becomes meaningless.

D. NOT Correct – because these valves are normally closed at power operations

Technical Reference(s): R-STM-0052 rev9 p. 14-17 ;
OSP-53 rev 22 page 11, 48 and 49

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.2 to 41.9 / 45.7 to 45.8

Comments: JH

QUESTION 29 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	2
K/A #	201005	A4.02	IR 3.6

Ability to manually operate and/or monitor in the control room Rod display module (lights and push buttons): BWR-6

Proposed Question:

The plant was operating at 100% power.

The At-The-Controls Operator observes the following indications.

- The “ACCUM FAULT” pushbutton on Operator Control Module is back-lit

When the “ACCUM FAULT” push button is depressed what will be indicated on the full core display?

A. A solid red LED for all but one control rod

B. A solid green LED for all but one control rod

C. A solid red LED for the effected control rod

D. A solid green LED for the effected control rod

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible - the red status LED will be lit, however only the control rod with the accumulator fault will be lit

B. NOT Correct. This answer is plausible - the status LED will be lit, however the LED is red and only the control rod with open the accumulator fault will be lit

C. Correct – the red status LED on the full core rod display module will light and stay lit while the push button is depressed indicating which control rod has the accumulator fault.

D. NOT Correct. This answer is plausible – the status LED will be lit, however the LED is red

Technical Reference(s): R-STM-0500 rev4 page 20

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Modified Bank # RBS-OPS-1838

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.5 to 45.8

Comments: JH

QUESTION 30 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑ SRO ☐
Tier #	2	Group # 2
K/A #	204000	G 2.4.31 IR 4.2

Knowledge of annunciator alarms, indications, or response procedures. RWCU System

Proposed Question:

A plant startup is in progress. RWCU is in its normal lineup rejecting to the Main Condenser with RWCU Reject Flow Control Valve, G33-AOVF033.

The following conditions exist:

- Recirc Loop Suction temperature is 325°F
- A heatup rate of ~75°F/hr is being maintained
- P680 RWCU Reject Flow Controller G33-R606 is set for a 100 gpm reject flow to the condenser
- Non-Regenerative Heat Exchanger (NRHX) Outlet temperature is 128°F

As the startup continues alarm P680-01A-C01 F/D Inlet High Temp 130 DEG F is received.

The operator's actions are to . . .

A. Reduce the flow rate to the condenser

B. Immediately trip the operating RWCU pump(s)

C. Isolate RWCU system before non-regenerative heat exchanger outlet temperature reaches 140˚ F.

D. Reduce the CCP system temperature by lowering CCP-H/A128 RX PLT CLG WTR SUPPLY HEADER TEMP controller set point

Proposed Answer: A.

Explanation

A. Correct - ARP-P680-01A-C01 lists three action for the operator to take. Reducing the flow rate to the condenser is the first in the list and no other actions are given in this question.

B. NOT Correct. This answer is plausible since RWCU pumps will trip when the isolation set point of 140˚ F is reached

C. NOT Correct. This answer is plausible since this is the set point that will cause the RWCU system to isolate for protection of the F/D

D. NOT Correct. The answer is plausible since the RWCU non-regenerative HX is cooled by CCP, this action would have an effect on the F/D inlet temperature however this is not prescribed by procedure and would have a negative impact on the running equipment cooled by CCP

Technical Reference(s): ARP-P680-01A-C01 rev 20

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source:	New	
Question History:	Last NRC Exam NA	
Cognitive Level:	Memory or Fundamental Knowledge ☐	Comprehension or Analysis ☑ 3
10 CFR Part 55 Content:	41.10 / 43.2 / 45.11	

Comments: JH

QUESTION 31 Rev 1

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: Valve operation

Proposed Question:

RHR Pump 'A' is in standby lineup.

Which of the following describes (1) what will occur if E12-MOVF024A RHR A TEST RETURN TO SUPP POOL is opened and (2) the required actions based on this condition, if any?

A. (1) Line fill pump maintains system full;
(2) Close E12-MOVF024A prior to attempting pump start to minimize starting current.

B. (1) RHR 'A' DISCHARGE PRESSURE LOW alarm is received;
(2) Close E12-MOVF024A, pull control power fuses for RHR 'A' pump breaker and perform fill and vent of RHR 'A.

C. (1) Line fill pump maintains system full;
(2) No further action is required.

D. (1) RHR 'A' DISCHARGE PRESSURE LOW alarm is received;
(2) Perform fill and vent of RHR 'A' and LPCS.

Proposed Answer: B.

Explanation: With E12-MOVF024A open a low pressure condition will occur as system inventory drains to the Suppression Pool. SOP-0031 P&L 2.3.2 requires declaring the loop inop, removing the control power fuses and performing a fill and vent. The line fill pump is not sized to have enough capacity for this flow path. The removal of the control power fuse will prevent manual or auto start of the RHR pump. This will protect the pump from over current if started and system piping from water hammer.

A. NOT Correct – This answer is plausible in that the E12-MOVF024A, test return valve will have to be closed, however the line fill pump is not sized to maintain pressure in the system under these conditions.

B. Correct. With E12-MOVF024A open a low pressure condition will occur as system inventory drains to the Suppression Pool. SOP-0031 P&L 2.3.2 requires declaring the loop inop, removing the control power fuses and performing a fill and vent.

C. NOT Correct. This answer is plausible if the candidate assumes that line fill pump is capable of maintaining the system full and pressurized with E12-MOVF024A open

D. NOT Correct. The answer is plausible since with E12-MOVF024A open a low pressure condition will occur and a fill and vent will have to be performed on the system, however, SOP-0031 P&L 2.3.2 requires declaring the loop inop, removing the control power fuses.

Technical Reference(s): ARP-P601-20A/C04, SOP-0031 P&L 2.3.2.

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0204 Obj 8, 10

Question Source:	Bank #	RBS NRC 2010	
Question History:	Last NRC Exam	NA	
Cognitive Level:	Memory or Fundamental Knowledge ☐		Comprehension or Analysis ☑ 4
10 CFR Part 55 Content:	41.7 / 45.7		

Comments: JH

QUESTION 32 Rev 3

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	2
K/A #	234000	A2.03	IR 2.8

Ability to (a) predict the impacts of the following on the FUEL HANDLING EQUIPMENT ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of Electrical Power

Proposed Question:

Given the following plant conditions:

The plant is shutdown with fuel movement in progress

- The fuel transfer carriage is loaded with 1 spent fuel assembly and is in route to the Fuel Bldg.
- The fuel transfer tube bottom and flap valves are closed
- Due to a mechanical problem and power failure, the Refueling bridge has lost power

What is the impact on fuel transfer to the spent fuel pool and what action is required?

A. Fuel transfer can continue in manual to the spent fuel pool and the spent fuel assembly must be removed from the tube within 5 minutes.

B. Fuel transfer can continue in manual to the spent fuel pool and the spent fuel assembly must be removed from the tube within 30 minutes.

C. Fuel transfer cannot continue and a source of make-up water must be provided to the fuel transfer tube within 5 minutes.

D. Fuel transfer cannot continue and a source of make-up water may need to be provided to the fuel transfer tube within 30 minutes.

Proposed Answer: D

Explanation

A NOT Correct. This answer is plausible if there is a transfer tube position sensor fault. The fuel transfer system would be placed in manual and the carriage lowered to the spent fuel pool for removal, however due to the power loss, this is not possible,

B. NOT Correct. This answer is plausible if there is a transfer tube position sensor fault. The fuel transfer system would be placed in manual and the carriage lowered to the spent fuel pool for removal, however due to the power loss,

C. NOT Correct. This answer is plausible – this is the correct impact for the given conditions however the need for cooling/make up water is 30 minutes

D. Correct – when power is lost to the refuel bridge, the fuel transfer system will not operate. The carriage containing the spent fuel will stop and depending on its location will determine if cooling/make up water will be needed. Per AOP-0027 this time frame is 30 minutes

Technical Reference(s): ARP-F42-P001/A3 rev4 ;
FHP-005 IFTS Operations, rev30 page 3 ;
AOP-027 Fuel Handling Mishaps, rev28 page 5

Proposed references to be provided to applicants during examination: None

Learning Objective: ___ (As available)

May 2015 River Bend Station
NRC Initial License Retake Examination
Reactor Operator

Question Source:	Modified Bank #	RBS-OPS 2648	
Question History:	Last NRC Exam	NA	
Cognitive Level:	Memory or Fundamental Knowledge ☐		Comprehension or Analysis ☑ 4
10 CFR Part 55 Content:	41.5 / 45.6		

Comments: JH

QUESTION 33 Rev 2

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 239003	A1.01	IR 3.1

Ability to predict and/or monitor changes in parameters associated with operating the MSIV LEAKAGE CONTROL SYSTEM controls including: Main steam line pressure

Proposed Question:

The plant has experienced a LOCA. RPV water level fell below level 1 (-143").

MS-PLCS System has been manually initiated for 10 minutes, post LOCA.

The operators have taken successful actions to isolate the leak on the vessel. Vessel level is recovering and expected to settle into the normal operating level band.

Decay heat is causing reactor pressure to rise.

How will the MS-PLCS system respond to the following?

A. When Reactor water level recovers to above Level 1, MS-PLCS will trip

B. As Reactor Pressure rises to 100 psig, MS-PLCS will trip

C. As Reactor Pressure rises to 100 psig, MS-PLCS will not trip

D. When Reactor water level recovers to above Level 1 MS-PLCS can be manually secured by operators

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible – this is the isolation set point for the MSIVs once level rises above -143" the isolation can be reset and the MSIVs opened.

B. Correct – after 5 minutes of operation the MS-PLCS system will trip if the MSL pressure is less than 8.5 psig above RPV pressure. As RPV pressure rises above the design pressure of 25 psig for the MS-PLCS the system will not maintain adequate MSL pressure and trip

C. NOT Correct. This answer is plausible – this system is manually initiated by the control room operators. The key lock switch must be placed in the OFF position to secure the system per the system operating procedure.

D. NOT Correct. This answer is plausible – this system is manually initiated by the control room operators. The key lock switch must be placed in the OFF position to secure the system per the system operating procedure.

Technical Reference(s): SOP-0034 rev 13 p. 14,
R-STM-0208 rev7 pp. 10-12

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS-OPS-06060

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 45.5

May 2015 River Bend Station
NRC Initial License Retake Examination
Reactor Operator

Comments: JH

QUESTION 34 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	2
K/A #	245000	K6.02	IR 3.5

Knowledge of the effect that a loss or malfunction of the following will have on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS : Reactor/turbine pressure control system: Plant-Specific

Proposed Question:

The plant is operating at 80% power with EHC Pressure Regulator Channel A selected for pressure control. Pressure Regulator Channel B is in TEST.

Which one of the following describes the plant response if the Pressure Regulator Channel A Averaging Manifold pressure transmitter fails downscale?

A. Turbine control valves fully open, steam bypass valves fully open, reactor pressure lowers until MSIVs close.

B. Turbine control valves fully close, steam bypass valves fully open, reactor pressure rises until a reactor scram occurs.

C. Turbine control valves fully open, steam bypass valves remain closed, reactor pressure lowers until MSIVs close.

D. Turbine control valves fully close, steam bypass valves remain closed, reactor pressure rises until a reactor scram occurs.

Proposed Answer: D

Explanation

A. NOT Correct. This answer is plausible since this would be the expected response if the pressure transmitter failed upscale

B. NOT Correct. This answer is plausible since this would be the response if the bypass valves had an independent pressure transmitter

C. NOT Correct. This answer is plausible based on the response of the steam bypass valves, however the turbine control valves would only respond in this manner if pressure transmitter failed upscale.

D. Correct – with pressure regulator channel B in “TEST” if a mismatch between the two channels occurs, the auto swap function will not occur. Since channel A is in control a downscale failure will cause the pressure control system to react as if there is a low pressure condition in the RPV. The turbine control valves will close in an attempt to raise RPV sensed pressure and the bypass valves will remain closed for the same reason.

Technical Reference(s): R-STM-0509 rev14 pp. 5 & 51

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS-OPS-7379

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.7

Comments: JH

QUESTION 35 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group # 2	
K/A #	271000	K4.04	IR 3.3

Knowledge of OFFGAS SYSTEM design feature(s) and/or interlocks which provide for the following:
The prevention of hydrogen explosions and/or fires

Proposed Question:

With the "A" train of the Steam Jet Air Ejectors in service and condenser vacuum lowering, the following plant conditions exist:

- Reactor Power is 40%.
- ARC-AOV1A, SJAE Suction Valve was observed CLOSING.

The following annunciators are alarming:

- OFFGAS SYS AFTER FILTER FLOW HI/LO (NORM RNG)
- OFF-GAS POST-TRT HIGH-HIGH RADIATION
- STEAM TO AIR EJECTOR 1A EXTREME LOW FLOW

Which one of the following caused the closure of ARC-AOV1A?

A. High Offgas flow

B. High-High Offgas post treatment radiation

C. Low steam supply flow to the 1A First Stage SJAEs

D. Low steam supply flow to the 1A Second Stage SJAE

Proposed Answer: D

Explanation

A. NOT Correct. This answer is plausible – the off gas flow alarm will be received due to a low flow condition not a high flow.

B. NOT Correct. This answer is plausible since this will cause an isolation of the off gas system, however not ARC-AOV1A

C. NOT Correct. This answer is plausible – ARC-AOV1A does close on steam supply low flow, however steam flow is only sensed on the second stage air ejector steam supply

D. Correct – per ARP-870-53 rev15 page 16 with a low flow condition ARC-AOV1A will close. The steam flow is only sensed on the second stage air ejector steam supply. This isolation occurs because the steam to the second stage air ejector is also used to dilute the H2 within the off gas system piping

Technical Reference(s): ARP-870-52/G03 rev15 ; R-STM-0125, rev3 figure 1.

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS-NRC-872

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7

Comments: JH

QUESTION 36 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group #	2
K/A #	272000	A4.02	IR 3.0

Ability to manually operate and/or monitor in the control room: Radiation Monitoring Meter indications

Proposed Question:

The plant was operating at 100% power when the Control Room Operators rapidly reduced power due to loss of feedwater heating.

Subsequently, alarms H13-P601-19A-D02, Main Steam Line Hi Radiation and H13-P601-19A-C01, MSL Div I Hi-Hi RAD or INOP were received.

Where will the control room operator monitor this parameter for value and trend?

A. DRMS computer, Grid 6

B. H13-P669, Neutron / process RAD Monitor Panel

C. H13- P600, Process Recording Panel

D. H13- P879, Radiation Monitoring Panel

Proposed Answer: B

Explanation

A. NOT Correct. Plausible – this is the ATC area location for all of the safety related radiation monitoring equipment for status, value and trend however the MSL radiation is not displayed here.

B. Correct – this is the correct location for the MSL radiation monitor and it is located next to the neutron monitoring instruments

C. NOT Correct. Plausible - this is the location for the off gas process flow radiation monitoring equipment for status, value and trend

D. NOT Correct. Plausible - this is the MCR back panel location for all of the safety related radiation monitoring equipment for status, value and trend (RM-23)

Technical Reference(s): STM-511 rev15, figures 20, 33, 36, & 20 ;
ARP-601-19A/C01 rev34

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.5 to 45.8

Comments: JH

QUESTION 37 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group # 2	
K/A #	286000	K3.03	IR 3.6

Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on following: Plant protection

Proposed Question:

The plant is in a refueling outage with fuel movement in progress, when failed relays cause a full Balance-Of-Plant (BOP) isolation to occur.

Welding near the RWCU Heat Exchanger Room causes a fire requiring Fire Brigade response.

Which of the following describes the ability of the Fire Brigade to fight the fire?

A. Normally locked closed manual isolation valves may be opened to supply water to the fire suppression system.

B. Normal firefighting equipment and fire hoses are available for use at the RWCU Heat Exchanger room.

C. Firefighting is limited to the use of portable fire extinguishers.

D. Fire suppression systems for the area would have extinguished the fire.

Proposed Answer: C

Explanation

A. NOT Correct. Plausible – the inboard isolation valve for the fire water supply is a locked open valve which could be confused for a locked closed valve

B. NOT Correct. Plausible – there are no automatic fire suppression systems inside containment only fire hose stations, however the BOP isolation will isolate fire water supply header to containment

C. Correct – the RWCU heat exchanger room is located inside containment and the BOP isolation will isolate fire water supply header to containment

D. NOT Correct. Plausible – most areas of the plant are covered by an automatic fire suppression system, however there are no automatic fire suppression systems inside containment

Technical Reference(s): P&ID 15-01C (N-15) rev 13,
AOP-003 automatic isolations rev33 page 15

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS-NRC-1096

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.4

Comments: JH

QUESTION 38 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	2	Group # 2	
K/A #	290003	K5.01	IR 3.2

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC Airborne contamination (e.g., radiological, toxic gas, smoke) control

Proposed Question:

Control Building filter train B was manually started and aligned to supply the MCR for a retest of HVC-FN1B CB Charcoal Filter Fan B

A Hi-Hi radiation signal is received on RMS-RE13A, Local CB Air Intake Monitor.

Which of the following describes the operation of the CB Ventilation System?

A. Filter Train B remains running, HVC-AOD19D and F Local Air Intake Dampers open and Filter Train A starts on the HI-HI Radiation signal

B. Filter Train B stops, HVC-AOD19C, E, D and F Local Air Intake Dampers open and one Filter Train must be manually started

C. Filter Train B stops, HVC-AOD19C, E, D and F Local Air Intake Dampers open and Filter Train A starts on the HI-HI Radiation signal

D. Filter Train B remains running, HVC-AOD19C and E Local Air Intake Dampers open and Filter Train A remains in standby

Proposed Answer: D

Explanation

A. NOT Correct. Plausible – the first part of this statement is true, however the D&F local air dampers will already be open to place the B filter in service and A filter train will not start

B. NOT Correct. Plausible – due to the stated conditions all of the air intake dampers will be/come open, however the B filter train does not stop for the opening of the air intake dampers

C. NOT Correct. Plausible - due to the stated conditions all of the air intake dampers will be open, however the B filter train does not stop for the opening of the air intake dampers, however the A filter train will not start

D. Correct – the local air intake dampers are divisionally oriented, AOD19C/E(Div I) will open on a hi-hi rad signal from RE 13A, AOD19D/F will open on a hi-hi rad signal from RE13B. The manual start for the filter train B will have AOD19D&F open. The hi-hi rad signal from RE13A will cause AOD19C&E to open and because the outlet flow switch for the B train will be satisfied the A train fan will not start

Technical Reference(s): STM-0402 rev6 pages 11&12

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Modified Bank # RBS-NRC-1041

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 45.3

Comments: JH

QUESTION 39 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑ SRO ☐
Tier #	1	Group # 1
K/A #	295001	G2.1.20 IR 4.6

Ability to interpret and execute procedure steps: Partial or Complete Loss of Forced Core Flow Circulation

Proposed Question:

After the trip to OFF of one reactor recirc pump, the following conditions exist:

Reactor power 1720 MW_{th}
Total core flow 37 mlbm/hr

Using the provided power to flow maps, determine which of the following actions must be initiated per AOP-024 Thermal Hydraulic Stability Controls?

A. Raise total core flow to 45 mlbm/hr with the operating loop flow controller to exit the Restricted Region

B. Raise total core flow to 55 mlbm/hr with the operating loop flow controller to exit the Monitored Region

C. Reduce reactor power to 755 mwth with control rod insertion to exit the Monitored Region

D. Verify at least one channel of PBDS is operable and once every 12 hours

Proposed Answer: A

Explanation

A. Correct – the core conditions given indicate that unexpected entry into the restricted region has occurred. Per AOP-0024 the proper steps to execute are step 5.4 with attachment 2 page 1 and determine unexpected entry into the restricted region of the single loop / normal feedwater temperature power to flow map. Step 5.5.2 directs actions to exit the region by raising core flow or control rod insertion

B. NOT Correct. This answer is plausible – but raising core flow to this value will not produce core conditions that are outside of the monitored region

C. NOT Correct. This answer is plausible – inserting control rods to this value will produce core conditions that are outside of the monitored region

D. NOT Correct. This answer is plausible since this would be the proper procedure step to execute if the conditions given indicated entry into the restricted region

Technical Reference(s): AOP-024 rev25 pp. 8,9, attachment 2

Proposed references to be provided to applicants during examination: AOP-24 attachment 1&2 with the region labels removed

Learning Objective: ___ (As available)

Question Source: Modified Bank # RBS-NRC-116

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 43.5 / 45.12

Comments: JH

QUESTION 40 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295003	AA2.03	IR 3.2

Ability to determine and/or interpret Battery status: Plant-Specific as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER

Proposed Question:

The plant has experienced a LOCA and the following plant conditions exist:

RPV water level -47 inches and stable
Drywell pressure 1.7 psid and stable

ENS-SWG1A 4160 VAC SWG is locked out due to a bus fault

ENB-SWG1B 125 VDC SWG was being supplied by the backup charger (BYSCHGR1D) prior to the transient.

Which of the following represents the current status of the 125VDC systems?

A. ENB-SWG1A is being supplied by its charger (ENB-CHGR1A) and ENB-SWG1B is being supplied by the backup charger (BYS-CHGR1D).

B. Both ENB-SWG1A and ENB-SWG1B are being supplied by their respective batteries.

C. ENB-SWG1A is being supplied by its battery and ENB-SWG1B is being supplied by the backup charger (BYS-CHGR1D).

D. ENB-SWG1A is being supplied by its charger (ENB-CHGR1A) and ENB-SWG1B is being supplied by its battery.

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible – this plant configuration is normal for ENB-SWG1A and given for ENB-SWG1B, and if the candidate does not have a full comprehension and understanding of the electrical distribution system could be selected.

B. Correct. With the loss of ENS-SWG1A, the charger has no power to supply the bus. The charger receives 480VAC from EJS-SWG1A which is supplied from ENS-SWG1A, therefore ENB-SWG1A will be supplied by its battery.
At -47 inches, a Level 2 signal has been received resulting in a trip of the backup charger supply breaker (BYS-ACB583), to ENB-SWG1B, therefore ENB-SWG1B will also be supplied from its battery.

C. NOT Correct. This answer is plausible since the first part of the answer is correct and could be selected if the candidate does not have a full comprehension and understanding of the electrical distribution system and all associated interlocks.

D. NOT Correct. This answer is plausible since the second part of the answer correctly describes the state of ENB-SWG1B and could be selected if the candidate does not have a full comprehension and understanding of the electrical distribution system and the effect of the power loss on ENB-SWG1B

Technical Reference(s): R-STM-0305 rev7, page 9,21

Proposed references to be provided to applicants during examination: None

Question Source: Bank # RBS-2008 NRC

Question History: Last NRC Exam RBS 2008 NRC Q#2

Cognitive Level:	Memory or Fundamental Knowledge ☐	Comprehension or Analysis ☑ 4
10 CFR Part 55 Content:	41.10 / 43.5 / 45.13	

Comments: JH

QUESTION 41 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295005	AA1.01	IR 3.1

Ability to operate and/or monitor the Recirculation system: Plant-Specific as they apply to MAIN TURBINE GENERATOR TRIP

Proposed Question:

The plant was operating at 45% power when the main turbine tripped.

All systems responded as designed with reactor water level steady at 18 inches on set point set down.

What is the status of the reactor recirc. pumps and flow control valves?

A. Recirc Pumps are in slow speed with the Flow Control Valves at the 60% drive flow position.

B. Recirc Pumps are tripped with the Flow Control Valves at the 60% drive flow position

C. Recirc Pumps are in slow speed with the Flow Control Valves in their pre-transient positions.

D. Recirc Pumps are tripped with the Flow Control Valves in their pre-transient positions.

Proposed Answer: A

Explanation

A. Correct. With the reactor at 45% power only two feedwater pumps will be operating, and the recirc pumps will be in fast speed. For set point set down to occur the reactor level lowered to below 9.7" this is below level 4 which will cause FCV runback to 60% drive flow position with less than three feedwater pumps in operation. 9.7" is also the set point that will shift the recirc pumps to slow speed

B. NOT Correct. This answer is plausible since this would be the status or recirc system if RPV level lowered to -43" however all systems responded as designed which would not permit RPV level to reach -43"

C. NOT Correct. This answer is plausible since this would be the recirc system status if RPV level remained above level 4, if the candidate does not recognize that actuation of set-point set down would have required level to be below level 4, they may select this answer.

D. NOT Correct. This answer is plausible if the candidate does not recognize that the EOC-RPT is a downshift of the reactor recirc pumps to slow speed and not an actual trip of the pumps to OFF, or if they fail to recognize that water level would not go below -43".

Technical Reference(s): R-STM-0053 rev13 pp 28, 37 ; R-STM-0107 rev28 page 58 ;
GOP-001 rev85 page 58&59 steps 21&23

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Modified Bank # RBS-NRC-1001

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.6

Comments: JH

QUESTION 42 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295006	G2.4.4	IR 4.5

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

Proposed Question:

A plant startup is in progress.

The following sequence of events has occurred:

- "A" Reactor Recirc Pump is in fast speed with its FCV at minimum position
- When the "B" Reactor Recirc Pump is shifted to fast speed, it trips OFF
- Reactor Engineer determines that the Restricted region of the power to flow map has been entered
- Alarm P680-07A-A06 Div I Period Based Detection System Hi Hi Decay Ratio is received
- 2 seconds later alarm P680-07A-B06 Div II Period Based Detection System Hi Hi Decay Ratio is received

What operator action should be taken for these conditions?

A. Insert control rods as specified by RE OR raise recirculation flow by opening the FCVs to exit the region.

B. Enter GOP-0004 for Single Loop Operations.

C. Verify FCBB ≤ 1.0 within 15 minutes and once per 24 hours there after

D. Immediately place the reactor Mode switch in the Shutdown position

Proposed Answer: D

Explanation

A. NOT Correct. This answer is plausible since this action is directed if entry into the restricted region is unexpected per step 5.5 however with both divisions of PBDS in alarm state overriding guidance is given to scram the reactor.

B. NOT Correct. This answer is plausible since this action is directed for a single recirc pump trip per Step 5.9, however with both divisions of PBDS in alarm state overriding guidance is given to scram the reactor.

C. NOT Correct. This answer is plausible this action is directed if entry into the restricted region is unexpected per step 5.5 however with both divisions of PBDS in alarm state overriding guidance is given to scram the reactor.

D. Correct – per AOP-024 Thermal Hydraulic Stability Controls step 5.3 page 8, with both channels of the period based detection system operable and in alarm the alarms are validated and a reactor Scram must be initiated by placing the mode switch to shutdown and enter AOP-001 reactor scram and EOP-1 RPV control

Technical Reference(s): AOP_024 rev 25 page 8&9

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New

Question History:	Last NRC Exam NA	
Cognitive Level:	Memory or Fundamental Knowledge ☐	Comprehension or Analysis ☑ 4
10 CFR Part 55 Content:	41.10 / 43.2 / 45.6	

Comments: JH

QUESTION 43 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295016	AK2.01	IR 4.4

Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Remote shutdown panel: Plant-Specific

Proposed Question:

During control room abandonment, ____(1)_____ Remote Shutdown Panel is preferred because ____(2)_____

A. (1) Division 1;
(2) these circuits are equipped with separate fuses and are less likely to be rendered inoperable from a control room fire.

B. (1) Division 2;
(2) the majority of system interlocks are maintained operable after transfer is complete.

C. (1) Division 1;
(2) the majority of system interlocks are maintained operable after transfer is complete.

D. (1) Division 2;
(2) these circuits are equipped with separate fuses and are less likely to be rendered inoperable from a control room fire.

Proposed Answer: A.

Explanation:

A. Correct - Division 1 and 3 are equipped with separate fuses are less likely to be rendered inop from a control room fire. Div 1 has much more equipment available than Div 3.

B. NOT Correct. This answer is plausible if the student does not understand the effect of placing the remote shutdown panel transfer switches to emergency and that most major interlocks are bypassed in both divisions when transfer is complete.

C. NOT Correct. This answer is plausible since the Div. 1 Remote shutdown panel is the preferred division, however most major interlocks are bypassed in both divisions when transfer is complete.

D. NOT Correct. This answer is plausible due to the fact that one division (Div 1) is equipped with separate fuses. Div 2 does not have standby fuses.

Technical Reference(s): R-STM-0200, rev3 p.4 ; AOP-0031, rev322 p.6

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0200, Obj. 1, 8, 12

Question Source: Bank # RBS 2010 Audit

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.8

Comments: JH

QUESTION 44 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295018	AK1.01	IR 3.5

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

Proposed Question:

A leak in the Reactor Plant Component Cooling Water System (CCP) has resulted in system pressure dropping to 52 psig.

Which of the following components will continue to receive cooling from CCP?

A. RWCU Pump, RWCU Non Regen Heat Exchangers, CRD Pumps

B. RWCU Heat Non Regen Exchangers, SFC Heat Exchangers, Reactor Recirculation Pumps

C. RWCU Pump, Reactor Recirculation Pumps, RWCU Non Regen Heat Exchangers

D. RHR Pump Seal Coolers, Drywell Equipment Drain Sump Cooler, SFC Heat Exchangers

Proposed Answer: C.

Explanation:

A. NOT Correct. This answer is plausible since CRD pumps are considered non-safety related loads, however CRD pump cooling taps off inside the 'B' safety loop and will be isolated from cooling.

B. NOT Correct. This answer is plausible since the RWCU Non-Regen HXs and Reactor Recirc pumps receive cooling from the non-safety related portion of the system, however SFC heat exchangers cooling taps off inside the safety loops and will be isolated from cooling.

C. Correct - None of the listed loads are in the safety loops which isolate at 56 psig.

D. NOT Correct. This answer is plausible since the Drywell Equipment Drain Sump Cooler is non-safety related, however the RHR Pump Seal Coolers and SFC heat exchanger cooling taps off inside the safety loops and will be isolated from cooling.

Technical Reference(s): R-STM-0115, rev6 pp 19-21, fig 1 ; AOP-0011, rev19 pp 3-4

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115 Obj. 2

Question Source: Bank RBS 2010 Audit

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41b.4

Comments: JH

QUESTION 45 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑ SRO ☐
Tier #	1	Group # 1
K/A #	295019	AK2.05 IR 3.4

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Main steam system

Proposed Question:

The plant is experiencing a lowering Instrument Air System pressure and has entered AOP-0008, Loss of Instrument Air.

What component will be affected at 50 psig and how?

A. The Feedwater Level Control valves will drift closed

B. The feedwater heater drain dump valves will fail open

C. The MSIVs will drift closed

D. The MCR HVAC dampers will fail closed

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible – the feedwater level control valves fail as is, and this occurs at 85 psig.

B. NOT Correct. This answer is plausible – the FW heater drain dump valves do fail open, however this will not occur until IAS pressure lowers to less than the control air pressure for these valves of 30 psig.

C. Correct – the MSIVs will start to drift closed at 50 psig.

D. NOT Correct. This answer is plausible – the MCR dampers will fail closed and cause a loss of MCR HVAC, however the MCR damper supply air has a backup supply through compressed air bottles. The backup supply is designed to supply air for 30 days.

Technical Reference(s): AOP-008 rev 37 page 4&6 ;
STM-0121 rev 16 page 34

Proposed references to be provided to applicants during examination: None

Learning Objective: ___ (As available)

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.8

Comments: JH

QUESTION 46 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
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 : Establishing alternate heat removal flow paths

Proposed Question:

The plant is in Mode 5.

RPV level is >23 feet above the RPV flange

A trip of RHR 'A' from the Shutdown Cooling mode has resulted in entry into AOP-0051 Loss of Decay Heat Removal. Steps in this procedure direct the operators to place an alternate decay heat removal system in service.

Why does the abnormal operating procedure direct this action?

A. To ensure adequate mixing of the bulk coolant to avoid exceeding Recirculation Loop to Steam Dome differential temperature limits to protect primary system piping from thermal stresses.

B. To ensure that the radiological consequences of a potential fuel handling accident are within acceptable limits.

C. Because excessive coolant temperature will result in damage to RWCU demineralizer resin.

D. Because decay heat removal must be maintained in order to prevent boiling in the reactor vessel.

Proposed Answer: D

Explanation

A. NOT Correct. This answer is plausible - Thermal shock limitations for Recirc pumps are of concern during pump startup

B. NOT Correct. This answer is plausible - Radiological concerns during a fuel handling accident are accounted for by maintaining >23 feet of water over the vessel flange during fuel handling.

C. NOT Correct. This answer is plausible - Although high temperatures can damage RWCU resin, other interlocks protect the resin from high temperature

D. Correct - If an alternate decay heat removal method is not placed in service, boiling will eventually occur.

Technical Reference(s): AOP-0051, Rev 313 p.5

Proposed references to be provided to applicants during examination: None

Learning Objective: ___ (As available)

Question Source: Bank # RBS NRC 2008

Question History: Last NRC Exam RBS 2008 NRC Q#9

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

10 CFR Part 55 Content: 41.5 / 45.6

Comments: JH

QUESTION 47 Rev 3

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295023	AA1.04	IR 3.4

Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS : Radiation monitoring equipment

Proposed Question:

Core Alterations are in progress when a dropped fuel bundle results in the following DRMS and P863 alarms:

RMS-DSPL230-GE005 Fuel Build Stack / Vent Exhaust A
H13-P863-75A-H01 Div I Fuel Bldg Exh PAM Gaseous Radn Alarm

How would the control room operators validate the alarm?

A. Perform a filter advance from the RM-23 Panel

B. Perform a filter advance from the RM-80 local panel

C. Perform a check source test from the RM-23 Panel

D. Perform a check source test from the RM-80 local panel

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible – filter advance would be appropriate if this was a particulate alarm

B. NOT Correct. This answer is plausible – filter advance would be appropriate if this was a particulate alarm and the RM-80 does not have this function availble.

C. Correct – a check source will allow the control room operators to determine if the radiation detector is responding to the change in radiation level during the check source and the RM-23 panel is located in the MCR

D. NOT Correct. This answer is plausible – the check source is an appropriate action however this function is not available at the RM-80

Technical Reference(s): ARP-P863-75-H01, rev 29 p.65 ; ARP-RMS-DSPL-230, rev 9 p. 33, SOP-0086 rev 16 page 16

Proposed references to be provided to applicants during examination: None

Learning Objective: ___ (As available)

Question Source: Modified Bank # GGNS NRC 15

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.6

Comments: JH

QUESTION 48 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295024	EA2.06	IR
4.1			

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

Proposed Question:

The plant has experienced a Loss of Coolant Accident due to a complete break of the Recirculation System piping.

Which one of the following describes the initial response of Drywell and Suppression Pool Temperature?

A. Drywell pressure will rise to a maximum value clearing all Drywell to Containment Vents releasing steam directly into the Containment pressurizing Containment and raising Suppression Pool temperature slightly.

B. Drywell pressure will rise to a maximum value clearing all Drywell to Containment Vents causing a rise in Containment Pressure due to non-condensable release and a significant rise in Suppression Pool temperature

C. Drywell pressure will rise to greater than ADS initiation setpoints causing ADS depressurization of the reactor to the Suppression Pool, resulting in a significant rise of Suppression Pool temperature.

D. Drywell pressure will rise to greater than the ECCS initiation setpoints causing ECCS injection and collapse of the steam bubble, removing the driving head of Reactor pressure, resulting in a turn of Drywell pressure and a slight rise in Suppression Pool temperature

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible considering the first part of the response, however direct pressurization by steam does not occur.

B. Correct. Per USAR table 6.2-11, suppression pool bubble break through will occur at 1.43 sec. into the event and as the non-condensable gasses are released into containment, pressure will rise and as the blow down rate lowers the steam released through the vents will be condensed by the suppression pool water with a significant rise of pool temperature

C. NOT Correct. This answer is plausible – ADS will initiate during this event, however by the time the time delay (105 sec.) has been satisfied for the ADS SRVs to open the RPV blow down will have completed to a point that little to no steam will be released to the suppression pool.

D. NOT Correct. This answer is plausible since the ECCS systems will initiate however the injection of these systems do not halt the RPV blow down as shown in USAR table 6.2-11

Technical Reference(s): USAR table 6.2-11

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Modified Bank # RBS-NRC-663

Question History:	Last NRC Exam NA	
Cognitive Level:	Memory or Fundamental Knowledge ☐	Comprehension or Analysis ☑ 4
10 CFR Part 55 Content:	41.10 / 43.5 / 45.13	

Comments: JH

QUESTION 49 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑ SRO ☐
Tier #	1	Group # 1
K/A #	295025	G2.1.27 IR 3.9

Knowledge of system purpose and/or function. High Reactor Pressure / 3

Proposed Question:

The plant was operating at 100% power when a low vacuum turbine trip caused reactor pressure to rise rapidly.

Reactor pressure rose to maximum pressure of 1160 psig and is currently cycling between 1063 psig and 956 psig.

Reactor water level dropped to a minimum of -20 inches wide range before recovery to +18 inches.

Which one of the following describes the current status of ARI and SRV valves?

ASSUME NO OPERATOR ACTIONS.

A. Energized 1 SRV open

B. De-energized 2 SRVs open

C. De-energized 1 SRV open

D. Energized 2 SRVs open

Proposed Answer: A

Explanation

A. Correct. When pressure rise above the ARI initiation setpoint of 1153 psig, the ARI logic will energize the ARI valves to open and depressurize the scram air header, The ARI logic is sealed in and cannot be reset until at least 32 seconds after initiation, assuming the initiating signal is clear. Additionally, the peak RPV pressure of 1160 psig would cause all 16 SRVs to initially open which activates the Lo-Lo set function which then lowers the closing setpoints of 5 SRVs to 976 psig for 3 SRVs, 966 psig for 1 SRV and 956 psig for the last SRV

B. NOT Correct. This answer is plausible since normal functions to depressurize the scram air header are fail safe systems and de-energize to perform their intended function. Additionally, of the five low-low set valves, 3 will function to close (in Lo-Lo set mode) at 976 psig, while the other two will close at lower pressures, if the student does not recall that setpoints of the two lowest lo-lo set valves, they may choose this answer.

C. NOT Correct. This answer is plausible since normal functions to depressurize the scram air header are fail safe systems and de-energize to perform their intended function and at the given plant conditions there will only be one SRV open.

C. NOT Correct. The answer is plausible since the ARI valves are energize to open and have a setpoint of 1153 psig and will remain energized until manually reset.

Technical Reference(s): AOP-0001 Reactor scram rev 30 p.8 ;
ARP-680-05/C01 and C02 rev 12 ;
R-STM-0109 rev 14 p.61 ; R-STM-0052 rev 9 page 26 &27

Proposed references to be provided to applicants during examination: none

Learning Objective:	___ (As available)		
Question Source:	Modified Bank #	GGNS NRC 2004	
Question History:	Last NRC Exam	NA	
Cognitive Level:	Memory or Fundamental Knowledge ☐		Comprehension or Analysis ☑ 4
10 CFR Part 55 Content:	41.7		

Comments: JH

QUESTION 50 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295026	EA2.03	IR 3.9

Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor pressure

Proposed Question:

The reactor was scrammed due to a stuck open SRV.

The plant experienced an ATWS.

Plant conditions indicate a slow approach to the Heat Capacity Temperature Limit, due to rising suppression pool temperature.

What action should be taken by the operators to address these conditions?

A. Open 7 ADS/SRVs

B. Maintain RPV pressure below the HCTL

C. Rapidly depressurize the RPV with any Alternate Depressurization system

D. Anticipate emergency depressurization and open steam bypass valves and MSL drains.

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible since the direction given in EOP-2, Primary Containment Control, Steps SPT-6 and SPT-7, if SP temperature cannot be maintained below the HCTL is to perform an emergency depressurization. This action would not be appropriate given the slow rise in SP temperature without first performing the actions to control pressure to prevent exceeding HCTL

B. Correct – per EOP-01A step RPA-4 lowering RPV pressure during ATWS conditions is directed in an effort not to meet conditions that would require ED

C. NOT Correct. This answer is plausible since the direction given in EOP-2, Primary Containment Control, Steps SPT-6 and SPT-7, if SP temperature cannot be maintained below the HCTL is to perform an emergency depressurization. This action would not be appropriate given the slow rise in SP temperature without first performing the actions to control pressure to prevent exceeding HCTL. The candidate may choose this answer if they incorrectly apply the direction in Steps SPT-6 and SPT-7 and further wish to limit the addition of heat to the SP.

D. NOT Correct. This answer is plausible since anticipation of emergency depressurization by opening the steam bypass valves and MSL drains is appropriate when plant conditions approach parameters which required emergency depressurization. This action is however not allowed procedurally during ATWS conditions.

Technical Reference(s): EPSTG-002 page B-7-66 rev 16

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # _____
New X

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.8 to 41.10

Comments:JH

QUESTION 51 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	2
K/A #	295011	EA1.02	IR 3.5

. Ability to operate and/or monitor the following as they apply to HIGH CONTAINMENT TEMPERATURE, **Containment ventilation/cooling: Mark-III**

Proposed Question:

The plant was operating at 100% rated power when a steam leak occurred resulting in a low RPV water level of -80 inches. Current RPV water level is -66 inches and stable. Prior to the transient, HVR-UC1A and HVR-UC1C were in service.

EOP-2 has been entered to address a high containment temperature condition. The CRS has given the direction to operate all available containment cooling.

What actions are required to operate all available containment cooling under these conditions?

A. Open HVN-MOV127, 128, 129, 130 Chilled Water Supply and Return Valves.

B. Start HVR-UC1B.

C. Open SWP-MOV502A(B) and SWP-MOV503A(B) Containment Unit Cooler Supply and Return Valves and start HVR-UC1B.

D. Open HVN-MOV127, 128, 129, 130 Chilled Water Supply and Return Valves and start HVR-UC1B.

Proposed Answer: C.

Explanation

A. NOT Correct. This answer is plausible - these are the normal cooling supply valves however their isolation signal has not cleared or bypassed and they cannot be opened

B. NOT Correct. This answer is plausible – the third unit cooler will need to be started however the SWP valves will need to be opened to meet the definition of available in the EOPs

C. Correct - With level between Level 2 and Level 1, HVN will be isolated to the containment UC's, but SWP will not be valved in. HVR-UC1B is not running because it has not received a Level 1 start signal. Manual action will be required to valve in SWP and start the unit cooler

D. NOT Correct. This answer is plausible - these are the normal cooling supply valves however their isolation signal has not cleared (>-43) or bypassed and they cannot be opened

Technical Reference(s): R-STM-0403 Pg 10-11 of 48, EPSTG-0002 B-8-8

Proposed references to be provided to applicants during examination: none

Learning Objective: RLP-STM-0403 Obj 3, 4, 6 (As available)

Question Source: Bank # RBS 2010 Audit Modified Bank # _____

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.6)

Comments:JH

QUESTION 52 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295028	EK3.02	IR 3.5

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

Proposed Question:

Following a large break LOCA, ALL RPV level instruments went off-scale low.

Five minutes later, the Fuel Zone Level instruments returned on scale. The following conditions now exist:

- Containment temperature	191°F (at EL 119 ft)
- Drywell temperature	335°F (at EL 145 ft)
- RPV Pressure	10 psig
- Fuel Zone Level indication	-290 inches and slowly rising and rapid lowering
- LPCS is injecting	5050 gpm

Which of the following is the reason to enter EOP-004 RPV Flooding?

A. Primary containment temperature have exceeded design limits

B. Dry well temperatures have exceeded design limits

C. Assure adequate core cooling under conditions where RPV level cannot be determined

D. Core damage will begin rapidly under conditions where RPV level cannot be determined

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible since the conditions given exceed the containment design limit of 185ºF and require emergency depressurization, however EOP-4 is not entered for this reason

B. NOT Correct. This answer is plausible – the conditions given exceed the drywell design limit of 330ºF and requirement to emergency depressurization, however EOP-4 is not entered for this reason

C. Correct – Per EPSTG-0002 page B-6-6 rev 16 from caution #1 the stem indicates that RPV level indication is unreliable and RPV level cannot be determined

D. NOT Correct. This answer is plausible – if adequate core cooling is no maintained core damage can occur, however if core damage is indicated by high radiation or H2 levels entry into the SAPs would be required.

Technical Reference(s): EPSTG-0002, STM-057 rev 4 page 6&7

Proposed references to be provided to applicants during examination: steam table
Learning Objective: ___ (As available)

Question Source: New X
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 41.5 / 45.6

Comments:JH

QUESTION 53 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295030	EK1.03	IR 3.8

Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Heat capacity

Proposed Question:

Which one of the following is a consequence of suppression pool water level of 15 feet 5 inches?

With a Suppression pool water level of 15 feet 5 inches. . .

A. operation of SRVs would directly pressurize containment and exceed allowable limits

B. steam from a LOCA may not be adequately condensed and containment could exceed allowable limits

C. will uncover the top two Drywell to Containment horizontal vents.

D. reduces the available net positive suction head for the low pressure ECCS pumps below minimum required.

Proposed Answer: B

Explanation

A. NOT Correct. This answer is plausible – this would be a result of SRV operation with a suppression pool level of 13", EPSTG-002 page B-11-6&7

B. Correct – per EPSTG-0002 rev 16 page B-8-24 a level of 15'5" will compromise the heat suppression capacity of the suppression pool water in the event a LOCA were to occur and could result in excessive containment pressure.

C. NOT Correct. This answer is plausible – low level in the suppression pool approaches the elevation of the horizontal vents and could cause direct pressurization of the containment. However the student may choose this answer if they do not the basis of the specified elevation of 15'5", as the horizontal vents between the drywell and containment will not be uncovered until a suppression pool level of 13'-3"

D. NOT Correct. This answer is plausible because this condition would reduce the available NPSH for ECCS pumps, however the limit for this is a suppression pool level of 10", Caution #5 of EOP-1

Technical Reference(s): EPSTG-002, EOP-1 rev 26 Caution #5, EOP-2, rev 15 SPL-12 and EOP-4rev 14, RF-2.

Proposed references to be provided to applicants during examination: ___

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # RBS-NRC-513
New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 43.5 / 45.13

Comments:JH

QUESTION 54 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295031	EK1.03	IR 3.7

Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL : Water level effects on reactor power

Proposed Question:

During an ATWS event, with the MSIVs open and the main turbine on line.

Why is the direction given to lower RPV level to below -56 inches?

A. To ensure that a Level 2 trip of the reactor recirculation pumps will occur, effecting a reduction in reactor power.

B. Prevent closure of the MSIVs on level prior to bypassing Level 1 isolations.

C. Prevent the potential for high core inlet sub-cooling

D. Prevent exceeding HCTL before hot shutdown boron weight is injected

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible since a reduction in level below level 2 will cause a trip of the reactor recirculation pumps and a reduction in reactor power. However, this action will have already been directed by Step RCA-2 and is not the reason for effecting a reduction in level below -56 inches.

B. NOT Correct. This answer is plausible since this is a consideration for the initial control band of -56 to -100 inches, however the specific reason for -56 inches is to lower level 24 inches below the feedwater spargers in order to minimize inlet subcooling.

C. Correct – per the EOP bases this level is given to uncover the feedwater spargers to allow steam heating of the feedwater prior to entering the core the rise in core inlet temperature has a large negative effect on reactor power production

D. NOT Correct. This answer is plausible since it is the bases for effecting a further level reduction based on power level conditions to allow injection of hot shutdown boron weight at conditions which should preclude the necessity to perform an emergency depressurization based on exceeding HCTL.

Technical Reference(s): EPSTG-002 rev 16 pageB-7-15 and 16, EOP-1A rev. 26

Proposed references to be provided to applicants during examination: ___

Learning Objective: ___ (As available)

Question Source: New X

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.8 to 41.10

Comments:JH

QUESTION 55 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295037	EA2.07	IR 4.0

Ability to determine and/or interpret Containment conditions/isolations as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN

Proposed Question:

A hydraulic ATWS is in progress.

- Reactor level band is -60" to -140" per EOP-1A
- RPV level is -75" and stable
- Reactor power is 15%.

Per AOP-0001, Reactor Scram, which of the following identifies the action(s) that need to be taken to allow draining of the Scram Discharge Volume?

A. Reset the Reactor SCRAM

B. Install EOP-5 Enclosure 12 (Defeating RPS and ARI Logic Trips) ONLY and then reset the Reactor SCRAM

C. Install EOP-5 Enclosures 12 (Defeating RPS and ARI Logic Trips) and 16 (Defeating Containment Instrument Air Isolation Interlocks) and then reset the Reactor SCRAM

D. Install EOP-5 Enclosures 12 (Defeating RPS and ARI Logic Trips) and 14 (Defeating RC&IS Interlocks and Emergency Control Rod Insertion)and then reset the Reactor SCRAM

Proposed Answer: _C

Explanation

A. NOT Correct. This answer is plausible – this is the correct action to open the SDV vent and drain valves with no active scram signal present. The stem indicates a low RPV water level scram would be present, additionally air to operate the SDV vent and drain valves is not available due to the isolation of containment air on level 2(-43").

B. NOT Correct. This answer is plausible – the additional EOP-5 enclosure will bypass the scram and ARI trip logics which allow them to be reset. The SDV vent and drain AOVs fail closed on a loss of air and the air supply to containment is isolated due to the low RPV water level

C. Correct – the addition of EOP-5 enclosure16 allows the air supply valve to containment to be opened and supply air to the SDV vent and drain AOVs

D. NOT Correct. This answer is plausible – EOP-5 enclosure 14 is the correct action to bypass the rod pattern controller and would allow the control rods to be inserted normally

Technical Reference(s): AOP-0001page 9-10 rev30 ESPTG-2 rev 16 EOP-1A rev 26
Proposed references to be provided to applicants during examination: ___
Learning Objective: ___ (As available)
Question Source: New X
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑3
10 CFR Part 55 Content: 41.10 / 43.5 / 45.13
Comments:JH

QUESTION 56 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295038	EK1.03	IR 2.8

Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Meteorological effects on off-site release

Proposed Question:

An accident has occurred that caused fuel damage and a loss of primary containment.

Current site boundary conditions are:

- Wind from the east (90 deg.) at 7 mph
- 60 mRem/hr.
- 1.1 E-9 microcuries/cc I-131

During this event due to weather frontal passage the wind direction has changed.

Current site boundary conditions are:

- Wind from west (280 deg.) at 10 mph
- 25 mRem/hr.
- 1.1 E-9 microcuries/cc I-131

What operational impact will the meteorological changes have on the main control room?

A. None, the main control room is well shielded by concrete walls

B. The main control room will need to be evacuated and control transferred to the Remote Shutdown Panel

C. Verify that the main control room charcoal filter train has started and aligned to the local intake

D. Transfer the main control room ventilation intake to the remote intake

Proposed Answer: D.

Explanation:

A. NOT Correct. This answer is plausible since the MCR walls are concrete however the radiological hazard is not from the dose level but from the I-131 which could enter the MCR via the ventilation intake.

B. NOT Correct – This answer is plausible if the MCR were to become inhabitable, however during accident conditions, the MCR charcoal filter system will start and reduce the overall exposure to the MCR operators.

C. NOT Correct. This answer is plausible since this addresses the conditions under which the MCR filter train would operate, however the MCR filter train should not be aligned to the local air intake since this intake is in the release path.

D. Correct. With the given new wind direction along with the loss of primary containment the MCR local intake is in the path of the release.(local intake is located on top of the control building east of containment) The intake should be transfer to the remote intake which is located in the standby cooling tower which is located west of containment.

Technical Reference(s):STM-402 rev 6, page 8&9, SOP-0058 rev 21, page 19 &20, RMS-DSPL230/1GP013 page 42

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.8 to 41.10

Comments:JH

QUESTION 57 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	600000	AA1.05	IR 3.0

Ability to operate and / or monitor Plant and control room ventilation systems as it applies to PLANT FIRE ON SITE:

Proposed Question:

The plant was operating at 100% rated power when a loss of coolant accident occurred, resulting in an automatic reactor scram.

Plant conditions are as follows:

- Reactor water level is -90 inches and lowering
- Drywell pressure is 2.2 psid and rising
- Smoke from an outside fire is entering the Control Room

An operator is attempting to manually place the Control Room ventilation in the Smoke Removal mode.

Under these conditions the HVC-AOD-107 / 108 Smoke Removal Fan Suction dampers will _____(1)____ and the Smoke Removal fan will ___(2)____.

A. open / start.

B. open / start when RPV water level is restored above Level 2.

C. remain closed / start and run on recirc.

D. remain closed / not start.

Proposed Answer: D

Explanation

A. NOT Correct. This answer is plausible since the Smoke Removal portion of the system does require manual operation, however the student may not recall that these dampers receive an isolation signal on a LOCA in addition to Main Control Room ventilation high raditation.

B. NOT Correct. This answer is plausible since the restoration of RPV level above level 2 will clear the level isolation signal, however with drywell pressure above 1.68 psig and absence of actions to reset seal-in isolation signals, dampers will remain closed.

C. NOT Correct. This answer is plausible – the normal main control room ventilation system is operated in a recirc flow path it is reasonable to assume the smoke removal system would follow the same path

D. Correct – these dampers receive an isolation signal on 1.68 drywell ΔP or level 2. From the stem those conditions still exist and will not be reset. The smoke removal fan is interlocked with the suction dampers as a start permissive

Technical Reference(s): AOP-0003 rev 33 page 10 & 20, SOP-0058 rev 21page 14

Proposed references to be provided to applicants during examination: none
Learning Objective: ___ (As available)

Question Source: Bank # RBS-NRC-167 Modified Bank # _____

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41/41.7/45.6_____

Comments:JH

QUESTION 58 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	700000	AK2.07	IR 3.6

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

Proposed Question:

A grid disturbance has occurred in the Entergy service area.

Given the following plant conditions:

- 100% rated power
- 50 MVARS
- Alarm P808-86A-H01, GRID TROUBLE is received in the MCR
- Bus voltage on SPI-REC102 indicates 224 KV

Per AOP-0064 Degraded Grid, what limits should be maintained on the main generator?

A. Raise main generator hydrogen pressure to 75 psig

B. Lower MVARS and Lower MWATTS to maintain a power factor of 0.9

C. Raise bus voltage to ~ 230 KV by adjusting MVARS, maintaining 0 to +230 MVARS

D. Lower reactor power to remove large pumps from service, until bus voltage is above the alarm set point.

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible – raising the H_2 pressure will give more margin to undesirable areas on the generator capability curve however this guidance is not given in AOP-64

B. NOT Correct. This answer is plausible – the power factor rating for RB is .9, this adjustment would not improve the low voltage condition and this guidance is not given in AOP-64

C. Correct – AOP-0064 gives a preferred band of 0-230 megavars to maintain bus voltage above 224.25 Kv

D. NOT Correct. This answer is plausible – removing large equipment from service will remove real load from the grid / generator, however the generator load reduction would lower bus voltage further.

Technical Reference(s): AOP-0064 Degraded Grid rev 6 page 6, SOP-0080 Turbine Generator Operation rev 327 page 6 and 76

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: New X

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.4, 41.5, 41.7, 41.10 / 45.8

Comments:JH

QUESTION 59 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	2
K/A #	295002	G2.4.4	IR 4.5

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. Loss of Main Condenser Vac

Proposed Question:

The following plant conditions exist:

- Reactor power 100% rated
- RPV level 36 inches
- Condenser vacuum 23.8 inches Hg
- IAS header pressure 115 psig
- CCP header pressure 108 psig
- CCS header pressure 113 psig

Which of the following procedures is appropriate for these conditions?

A. AOP-0008, Loss of Instrument Air

B. AOP-0011, Loss of Reactor plant Component Cooling Water

C. AOP-0012, Loss of Turbine Plant Component Cooling Water

D. AOP-0005, Loss of Condenser Vacuum / Trip of Circulating Water Pump

Proposed Answer: D _

Explanation

A. NOT Correct. This answer is plausible – the value given is lower than normal system pressure however a lowering trend is not indicated, AOP entry is not appropriate.

B. NOT Correct. This answer is plausible – the value given is lower than normal system pressure of 115psig however a lowering trend is not indicated, AOP entry is not appropriate.(standby pump auto start at 95 psig)

C. NOT Correct. This answer is plausible – the value given is lower than normal system pressure of 117 psig however a lowering trend is not indicated, AOP entry is not appropriate. .(standby pump auto start at 96 psig)

D. Correct – the given value of condenser vacuum is in the unacceptable region and requires entry into the AOP

Technical Reference(s): AOP-005 rev 22 page 3, AOP-008rev 37 page3,AOP- 0011 rev19 page 3and AOP-0012 rev 12

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS NRC-01262 Modified Bank # _____

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 43.2 / 45.6

Comments:JH

QUESTION 60 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	2
K/A #	295007	AA2.02	IR 4.1

Ability to determine and/or interpret Reactor power as it applies to HIGH REACTOR PRESSURE :

Proposed Question:

The plant is operating at 100 % rated power when B21-AOVF028B, an outboard MSIV fails closed due to a rupture of the valve actuator air supply.

Determine which one of the following describes the response of the reactor?

A. RPV pressure will rise and stabilize at a higher pressure.
Reactor power will rise and stabilize at a higher power.
RPV water level will lower and then return to normal level.

B. RPV pressure will rise and then lower following the scram.
Reactor power will rise and then drop following the scram.
RPV water level will lower and then stabilize at a lower level.

C. RPV pressure will lower and stabilize at a lower pressure.
Reactor power will lower and stabilize at a lower power.
RPV water level will lower and then stabilize at a lower level

D. RPV pressure will rise and stabilize following the scram.
Reactor power will rise and drop following the scram.
RPV water level will rise and then return to normal level.

Proposed Answer: B _

Explanation
A. NOT Correct. This answer is plausible – if this failure occurred at a lower power this would be the expected response
B. Correct – RBS main steam lines do not have enough capacity to operate at 100% with one steam line isolated
C. NOT Correct. This answer is plausible – two of the three parameters indicate a proper response
D. NOT Correct. This answer is plausible – two of the three parameters indicate a proper response

Technical Reference(s): USAR 15.2.1rev 14 page 15.2-15 to 17

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # RBS 2003 NRC exam Modified Bank # _____

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 43.5 / 45.13

Comments:JH

QUESTION 61 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	1
K/A #	295027	AA1.01	IR 3.6

Ability to operate and/or monitor the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) : Containment ventilation/cooling system

Proposed Question:

HVR-UC1A and 1C Containment Unit Coolers are running.

A small break LOCA occurs.

- Containment temperature is 96°F and slowly rising.
- Drywell pressure is 1.7 psig.

Based on this transient, after 5 minutes, the operator would expect to see the containment unit coolers:

A. HVR-UC1A, 1B and 1C running with service water flow.

B. HVR-UC1A, 1B and 1C running with chilled water flow.

C. HVR-UC1A and 1B running with service water flow.

D. HVR-UC1A and 1B running with chilled water flow.

Proposed Answer: C

Explanation

A. NOT Correct. This answer is plausible since UC1A and UC1B will auto start on a high drywell pressure and service water will supply them. UC1C will trip on high drywell pressure.

B. NOT Correct. This answer is plausible since UC1A and UC1B will start on high drywell pressure, however chilled water will isolate and the the UCs will be supplied by service water.

C. Correct- Drywell pressure 1.7 psig will cause isolation of Plant Chilled Water to containment. The location of containment coolers is such that no chilled water flow to containment coolers will exist. However, containment cooler fans do not shed, and will continue to operate. After 1 minute UC1B will start and service water will be supplied to UC1A and 1B.

D. NOT Correct. This answer is plausible since UC1A and UC1B receive an auto start signal on high drywell pressure, however chilled water to the containment is isolated and the UCs will be supplied by service water.

Technical Reference(s): AOP-003 Automatic Isolations rev 33 page 11 & 15, STM-403rev 8 page 10&11.

Proposed references to be provided to applicants during examination: none

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # RBS-NRC-1193
New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.6)
Comments:JH

QUESTION 62 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	2
K/A #	295013	AK3.01	IR 3.6

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE : Suppression pool cooling operation

Proposed Question:

EOP-2 directs the operator to place all available Suppression Pool Cooling in service when Suppression Pool temperature is cannot be maintained below 100ºF.

Why is this direction given in response to high suppression pool temperature?

A. To prevent exceeding the primary containment pressure limit during a Design Bases Accident

B. To prevent exceeding RCIC pump operating limit for lube oil temperature

C. To prevent exceeding NPSH limits for ECCS pumps when taking suction from the suppression pool

D. To prevent exceeding the Heat Capacity Temperature Limit during ATWS conditions

Proposed Answer: A _

Explanation

A. Correct- per ESPTG-0002 containment failure could occur if suppression pool temperature reaches the upper LCO limit with a DBA

B. NOT Correct. This answer is plausible since elevated suppression pool temperature is a concern with respect to continued RCIC operations as referenced in the EOP Cautions, it does not however have direct correlation with respect to the operation of Suppression Pool Cooling.

C. NOT Correct. This answer is plausible since elevated suppression pool temperature is a concern with respect to NPSH head limitations for pumps taking a suction on the suppression pool but does not direct operation of Suppression Pool Coolling.

D. NOT Correct. This answer is plausible since exceeding the HCTL would require and emergency depressurization of the RPV – this is ED criteria given in EOPs based on suppression pool temperature

Technical Reference(s): _EPSTG-0002 page B-8-16 rev 16 and TS 3.6.2.1 bases__

Proposed references to be provided to applicants during examination: _none__

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # _____
New X

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 45.6

Comments:JH

QUESTION 63 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group # 2	
K/A #	295020	AK2.09	IR 3.1

Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following:
RHR/shutdown cooling: Plant-Specific

Proposed Question:

Given the following conditions:

- RBS is currently in a Refuel outage
- The reactor vessel head is removed and water level is >23 feet above the vessel flange
- RHR A is in the Shutdown Cooling Mode

During a maintenance activity an inadvertent isolation of E12-MOVF008 RHR Shutdown Cooling Outboard Isolation valve occurs.

What Shutdown Cooling method is available to be placed in service?

A. ADHR in Configuration #2.

B. RHR-B in the Shutdown Cooling mode

C. ADHR in Configuration #1

D. RHR-A in the Fuel Pool Cooling Assist mode

Proposed Answer: D _

Explanation
A NOT Correct. This answer is plausible – Configuration #2 with SDC suction and discharge to the Upper Pool is a valid method of decay heat removal, however this configuration requires that E12-MOVF008 be open
B. NOT Correct. This answer is plausible – this is a valid SDC configuration however the RHR pump interlocked and cannot be started in the SDC mode without the E12-MOVF008 open
C. NOT Correct. This answer is plausible – ADHR configuration #3 does not require E12-MOVF008 to be open to create a flow path however configuration #1 does
D. Correct – this mode of RHR does not use the normal SDC path

Technical Reference(s): _OSP-0041, Alternate Decay Heat Removal, RSTM-204 fig 6&8 and RSTM-656 fig 3

Proposed references to be provided to applicants during examination: _none__

Learning Objective: ___ (As available)
Question Source: New X
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 41.7 / 45.8
Comments:JH

QUESTION 64 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	2
K/A #	295022	AK1.02 IR 3.6	

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

Proposed Question:

The plant is operating at 100% power.

Why must the Mode switch be placed in Shut Down under certain conditions after a loss of all CRD pumps?

A. Loss of seal purge to the Recirc. Pumps will cause rapid seal degradation and drywell leakage

B. Loss of keep fill system for reactor level indication will cause non-condensable gases to accumulate in the instrumentation and cause inaccurate level indication

C. Loss of CRD charging water that will cause the control rod accumulator to lose the ability to perform their active function

D. Loss of seal purge to the RWCU pumps will cause rapid degradation and high Auxiliary building temperatures / radiation

Proposed Answer: C _

Explanation

A. NOT Correct. This answer is plausible – the loss of seal purge will occur however a rapid loss of recirc seals does not occur, the loss only shortens the life of the seal

B. NOT Correct. This answer is plausible – the loss of the keep fill system will occur however the accumulation of non-condensable to a point that impacts the level indication is a long term condition

C. Correct. Loss of CRD and consequential accumulator faults can result in a failure of control rods to insert with required scram times upon receipt of a valid scram signal.

D. NOT Correct. This answer is plausible – the loss of seal purge to RWCU will occur however a rapid loss of RWCU seals does not occur, the loss only shortens the life of the seal

Technical Reference(s): _RBS TS 3.1.5 bases__

Proposed references to be provided to applicants during examination: ___

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # _RBS NRC 368_____
New

Question History: Last NRC Exam NA

Cognitive Level: Memory or Fundamental Knowledge ☑ 4 Comprehension or Analysis ☐
10 CFR Part 55 Content: 41.8 to 41.10
Comments:JH

QUESTION 65 Rev 1

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	1	Group #	2
K/A #	295035	EK2.03	IR 3.3

Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: Offsite Release Rate

Proposed Question:

Which one of the following is the reason that EOP 3, SECONDARY CONTAINMENT and FUEL BUILDING CONTROL, must be entered if Annulus differential pressure is 0.1 inches WC?

A. A significant steam leak into secondary containment.

B. A significant water leak from primary system may be discharging directly into secondary containment.

C. Potential for the loss of secondary containment that could result in uncontrolled radioactive releases

D. Potential failure of the Shield Building that will also result in failure of the primary containment vessel

Proposed Answer: C _

Explanation

A. NOT Correct. This answer is plausible – this is the reason to enter EOP-3 for high area temperature conditions

B. NOT Correct. This answer is plausible – this is the reason to enter EOP-3 for high radiation conditions

C. Correct – ESPTG-0002 discussion outlines the need to enter due to the potential loss of secondary containment due to high differential pressure conditions.

D. NOT Correct. This answer is plausible since the LCO value for containment external pressure is limited to a value of -.3 psig the rising annulus pressure could threaten this value.

Technical Reference(s): _ ESPTG-0002 pageB-9-2 rev 19__

Proposed references to be provided to applicants during examination: ___

Learning Objective: ___ (As available)

Question Source: Bank # 847 Modified Bank # _____

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.7 / 45.8

Comments:JH

QUESTION 66 Rev 1

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 3
K/A # G 2.1.14 IR 3.1

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

Proposed Question:

Which of the following would require a plant wide announcement per EN-OP-115-7 Component Deviations?

A. Transfer the RWCU backwash receiving tank

B. Alternating running Control Rod Drive pumps

C. Movement of irradiated fuel in the spent fuel pool

D. Start of the condenser air removal iodine filter train

Proposed Answer: B _

Explanation
A. NOT Correct. This answer is plausible – prior to transferring the backwash receiving tank the procedure requires that RP be notified
B. Correct – per EN-OP-115-7 step 5.2[11] major equipment status changes are to be announced plant wide. RBS has determined that 4160V and above as major equipment.
C. NOT Correct. This answer is – prior to movement of fuel in the spent fuel pool the control room supervisor must be notified as well as when the activity has stopped
D. NOT Correct. This answer is plausible – this is a large piece of equipment in the turbine building that takes suction on the condenser air removal pumps. While the equipment is large the fan is a 480V motor and does not affect TB HVAC

Technical Reference(s): __ EN-OP-115-7rev01, page 5, step 5.2[11] _

Proposed references to be provided to applicants during examination: ___

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # _____
New X

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 43.5 / 45.12

Comments:JH

QUESTION 67 Rev 2

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	3		
K/A #	G 2.1.29		IR 4.1

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Proposed Question:

RHR A was operating in the Suppression Pool Cooling mode to support the RCIC Pump quarterly surveillance. RCIC was secured last shift.

Plant conditions are as follows:

- Suppression pool temperature is stable at 81 degrees.
- RHR A has been secured IAW SOP-0031, Residual Heat Removal system operating procedure.

What, if any, control board lineup is required?

A. Perform a Control Board lineup with SOP-0031 Attachment 4A Control Board Lineup RHR A Loop and document the line up in the attachment

B. Performance of a Control Board lineup is **not** required, SOP-0031 step by step guidance placed the system in the proper lineup

C. Perform an administrative Control Board lineup and document in the MCR log book

D. Perform a Control Board lineup with the white plastic "washers" around the position indication and document in the MCR system check list

Proposed Answer: C C

Explanation

A. NOT Correct. This answer is plausible – this is a requirement of OSP-0022 if the system has under gone major maintenance or extended outage

B. NOT Correct. This answer is plausible – a control board lineup is not required if the step by step guidance has initials / signoff for the steps

C. Correct – Per OSP-0022 step 5.2.6, an administrative lineup shall be conducted and documented in the MCR log book

D. NOT Correct. This answer is plausible – the white washers depict the lights that are normally lit during normal operation and are not to be used to position plant equipment.

Technical Reference(s): _OSP-0022 rev79 page 58 section 5.2.5 &5.2.6.

Proposed references to be provided to applicants during examination: _none__

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # _____
New X

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 45.1 / 45.12

Comments:JH

QUESTION 68 Rev 2

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 3
K/A # G 2.1.37 IR 4.3

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

Proposed Question:

In accordance with OSP-0022, OPERATIONS GENERAL ADMINISTRATIVE GUIDELINES, which of the following activities is TYPE 1 REACTIVITY CHANGE?

A. Reactor startup

B. Sequence exchange where power is lowered to <70%

C. Fully withdrawn control rod testing

D. Partially withdrawn control rod testing

Proposed Answer: C _

Explanation:
A. NOT Correct. This answer is plausible since a reactor startup is a reactivity change, however, it is a Type 3 reactivity change.
B. NOT Correct. This answer is plausible since a sequence exchange is a reactivity change, however with power lowered to <80% it is a Type 3 reactivity change.
C. Correct. Fully withdrawn control rod testing is a Type 1 reactivity change.
D. NOT Correct. This is a plausible answer since partially withdrawn control rod testing is a reactivity change, however it is a Type 2 reactivity change.

Technical Reference(s): OSP-0022 rev 79 step 4.10.2 page 28

Proposed references to be provided to applicants during examination: NA

Learning Objective: None identified.

Question Source: Bank # _____ Modified Bank # _Nov 2010 RBS Audit_____

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.1 / 43.6 / 45.6

Comments:JH

QUESTION 69 Rev 1

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 3
K/A # G 2.2.12 IR 3.7

Knowledge of surveillance procedures

Proposed Question:

Who approves the start of surveillance testing and verifies that plant conditions support the test prior to start?

A. Shift Technical Advisor

B. Field Support Supervisor

C. Operation Shift Manager / Control room Supervisor

D. Work Control Supervisor

Proposed Answer: C _

Explanation
A. NOT Correct. This answer is plausible – this is an on shift duty position that can be filled by an SRO that could approve the start of an STP if standing the OSM/CRS position
B. NOT Correct. This answer is plausible – this is an on shift duty position that can be filled by an SRO that could approve the start of an STP if standing the OSM/CRS position
C. Correct per ADM-0015 rev 38 step 4.7.1
D. NOT Correct. This answer is plausible – this is an on shift duty position that can be filled by an SRO that could approve the start of an STP if standing the OSM/CRS position

Technical Reference(s): __ADM-0015 rev38 step 4.7.1_

Proposed references to be provided to applicants during examination: _none__

Learning Objective: ___ (As available)

Question Source: Bank # _RBS-OPS07687_____ Modified Bank # _____
New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 45.13

Comments:JH

QUESTION 70 Rev 3

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	3		
K/A #	G 2.2.14		IR 3.9

Knowledge of the process for controlling equipment configuration or status.

Proposed Question:

A pump in the plant has developed a greater than normal seal leak allowed by operator rounds. The operator secured the pump and started the standby pump as directed by the CRS.

Which of the following describes the appropriate control method for the status of the secured pump?

A. Place a Caution Tag on the switch that documents the condition of the equipment.

B. Place a Danger Tag on the switch to prevent use of this equipment.

C. Place a Test and Maintenance Tag on the switch to allow maintenance activities.

D. Place a Lockout Device on the switch to prevent use of this equipment.

Proposed Answer: A

Explanation

A. Correct- Caution tags provide precautions or special instructions that relate to unusual or out-of-normal conditions.

B. NOT Correct. This answer is plausible because the Danger hold tag would be used to place the equipment in a safe condition if maintenance were to be performed. When maintenance is to be performed, the Caution tag would be replaced by a Danger tag.

C. NOT Correct. This answer is plausible because a T&M tag would be used for a component that would need to be manipulated during maintenance activities, such as determining the leak location, but is incorrect because a T&M tag permits operation of the equipment by only authorized persons signed on to the tagout.

D. NOT Correct. This answer is plausible because during maintenance activities a lockout device would be used however the device would be installed on the pump breaker

Technical Reference(s): EN-OP-102 rev17 page 5 and 60

Proposed references to be provided to applicants during examination: __none_

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified Bank # _RBS NRC #1057_____

New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 43.3 / 45.13

Comments:JH

QUESTION 71 Rev 3

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 3
K/A # G 2.2.22 IR 4.0

Knowledge of limiting conditions for operations and safety limits.

Proposed Question:

Plant start up is in progress at 80% power.

Control rod(s) withdrawal results in a Minimum Critical Power Ratio (MCPR) of 1.06.

Which of the following would satisfy Technical Specification requirements for this condition?

A. Reduce reactor power below 23.8% within 4 hours.

B. Reduce reactor pressure below 685 psig within 2 hours.

C. Insert all insertable control rods within 2 hours.

D. Insert the last withdrawn control rod(s).

Proposed Answer: C _

Explanation
A. NOT Correct. This answer is plausible because it is a required TS action for exceeding the operating limit MCPR, which is less restrictive than the safety limit MCPR.
B. NOT Correct. This answer is plausible because the MCPR safety limit is based on reactor pressure at or above 685 psig and core flow of 10%, only reducing reactor pressure would not restore compliance
C. Correct - At 80% power, the reactor is at rated pressure with two recirc loops operating. MCPR less than 1.08 under these conditions constitutes a safety limit violation. With any safety limit violation the SL must be restored and insert all insertable control rods inserted within 2 hours.
D. NOT Correct. This answer is plausible because inserting the control rod(s) that caused the violation would restore MCPR to within limits; however, this action is incomplete.

Technical Reference(s): _RBS T.S. 2.1 and 2.2 amendment 182__

Proposed references to be provided to applicants during examination: _none__

Learning Objective: RLP-HLO-401 Obj. B, G___ (As available)

Question Source: Bank # _____ Modified Bank # GGNS 2007 NRC_____
New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.5 / 43.2 / 45.2

Comments:JH

QUESTION 72 Rev 2

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 3
K/A # G 2.3.4 IR 3.2

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question:

Per procedure EIP-2-012, Radiation Exposure Controls, the Emergency Director may authorize a worker to receive up to a maximum of ____ Rem (TEDE) for the purpose of preventing damage to plant equipment.

A. 5

B. 10

C. 25

D. 75

Proposed Answer: B _

Explanation
A. NOT Correct. This answer is plausible – this is the allowed dose for preplanned emergency actions
B. Correct - per EIP-2-012
C. NOT Correct. This answer is plausible – this is the allowed dose without consent to save a life
D. NOT Correct. This answer is plausible – this is the allowed dose to voluntarily save a life

Technical Reference(s): _ EIP-2-012, Radiation Exposure Controls rev21 page 12__

Proposed references to be provided to applicants during examination: _none__

Learning Objective: ___ (As available)

Question Source: New X

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.12 / 43.4 / 45.10

Comments:JH

QUESTION 73 Rev 2

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G 2.3.13	IR 3.4

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

Proposed Question:

River Bend Station is currently performing a refueling outage with core reload in progress.

A control rod blade guide must be moved from the core to the wall hangers in the upper pool. Due to the length of the blade guide, the mast must be raised beyond the HOIST UP position while traversing through the Portable Shielding (Cattle Chute).

Per EN-FAP-OU-108, Fuel Handling Process, who must provide approval authority to allow the Refuel Bridge Driver to utilize the TRAVEL OVERRIDE and HOIST OVERRIDE interlock bypass features to move control rod blade guides through the Cattle Chute?

A. Control Room Supervisor

B. Field Support Supervisor (SRO)

C. Fuel Handling Supervisor (SRO)

D. Operations Shift Manager

Proposed Answer: C.

Explanation:
A. NOT Correct. This answer is plausible since the authority to override refuel bridge interlocks must come from a licensed senior reactor operator, however the individual has to be fulfilling the role of Fuel Handling Supervisor.
B. NOT Correct. This answer is plausible since the Field Support Supervisor is responsible for oversight of operational activities occurring outside the Control Room.
C. Correct – EN-FAP-OU-108 Roles and Responsibilities lists the Fuel Handling Supervisor (Licensed SRO) as the individual who may authorize the bypass of certain interlocks.
D. NOT Correct. This answer is plausible since the OSM must provide authority to commence core alterations.

Technical Reference(s): FHP-0003 rev35 page 18, EN-FAP-OU-108 rev5 page 4, EN-OP-115 rev15 page 15

Proposed references to be provided to applicants during examination: NA
Learning Objective: RLP-STM-0055 Obj. 6
Question Source: Bank # RBS NRC #95
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☑2 Comprehension or Analysis ☐
10 CFR Part 55 Content: 41.12 / 43.4 / 45.9 / 45.10
Comments:JH

QUESTION 74 Rev 0

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 3
K/A # G 2.4.1 IR 4.6

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question:

The reactor has just scrammed. The following plant conditions exist:

- Reactor power 0%, all rods in
- Reactor water level 17 inches (lowest level observed was 15 inches)
- Reactor pressure 1105 psig
- Suppression Pool Level 19'10"
- Drywell Temp 150°F
- Drywell Pressure 0.2 psid
- Main Steam Tunnel Temp 142°F

Which of the following represents the required EOP(s) to enter?

A. EOP-1 only

B. EOP-2 only

C. EOP-1A & EOP-2

D. EOP-1 & EOP-2

Proposed Answer: D _

Explanation
A. NOT Correct. EOP-1 is req'd due to 1105 psig, EOP-2 is required due to 150°F in the drywell
B. NOT Correct. EOP-1 is req'd due to 1105 psig, EOP-2 is required due to 150°F in the drywell
C. NOT Correct. EOP-1A is not req'd due to all rods are in
D. Correct EOP-1: 1105 psig RPV pressure, EOP-2: 150°F drywell temperature

Technical Reference(s): __ EOP-1, EOP-2
Proposed references to be provided to applicants during examination: _none__

Learning Objective: RLP-OPS-HLO-512 Obj. 3; RLP-OPS-HLO-0514 Obj. 3

Question Source: Bank # _____ Modified RBS 2010 NRC exam
New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 41.10 / 43.5 / 45.13

Comments:JH

QUESTION 75 Rev 0

Examination Outline Cross-Reference:

Level		RO ☑	SRO ☐
Tier #	3		
K/A #	G 2.4.32		IR 3.6

Knowledge of operator response to loss of all annunciators.

Proposed Question:

During a reactor shutdown, from 75% power, all Control Room annunciators are lost.

In accordance with AOP-55, Loss of Control Room Annunciators, the crew will _____.

A. Scram the reactor to place the reactor in a safe condition and reduce the number of running equipment

B. Lower power quickly by transferring both Recirc. pumps to slow speed per OSP-053, Emergency and Transient Response Support

C. Continue with the reactor shutdown per GOP-002, Plant Shutdown, to establish shutdown cooling

D. Stop all load changes and equipment or system manipulations to place the plant in a stable condition

Proposed Answer: D _

Explanation
A. NOT Correct. – a SCRAM would induce a significant transient on the plant that could jeopardize proper plant control without annunciators
B. NOT Correct. This action is plausible since OSP-053 does have directions for transfer of Recirc. Pumps to slow speed in an emergency however AOP-055, a higher tier document gives direction to stop equipment manipulations
C. NOT Correct. this action would require load and system changes that are not allowed by AOP-055
D. Correct – per the first subsequent action of AOP-055, step 5.1

Technical Reference(s): _AOP-055rev20, OSP-053rev22__

Proposed references to be provided to applicants during examination: __none_

Learning Objective: ___ (As available)

Question Source: Bank # _____ Modified: June 2007 RBS

Question History: Last NRC Exam _ NA

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

10 CFR Part 55 Content: 41.10 / 43.5 / 45.13

Comments:JH