QUESTION 1 Rev 2

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295001 AK2.02		IR 3.2

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: Nuclear boiler instrumentation.

Proposed Question:

The plant was operating at 90% power when Recirc Pump B tripped.

Immediately following the pump trip, Core Flow as indicated on B33-R613, TOTAL FLOW/ΔPRESS, will be ___.

A. Accurate because the summing logic is subtracting B loop flow from the indicated total core flow.

B. Accurate because the summing logic is adding the B loop flow to the indicated total core flow.

C. Inaccurate because the summing logic is adding the B loop flow to the indicated total core flow.

D. Inaccurate because the summing logic is subtracting B loop flow from the indicated total core flow.

Proposed Answer: D

Explanation

D. Correct – The Core Flow summing logic is driven by Recirc pump breaker position; for several minutes after a recirc pump trip, the pump is coasting down and still has positive flow; but the summing logic is already subtracting this flow from the indicated total core flow; therefore it is inaccurate for several minutes.

Technical Reference(s): R-STM-0051, Rev 5, p. 18 of 47, Figure 22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0051, Obj 10,13,16

Question Source: Bank # RBS-NRC-06263

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: replaced question to better match KA with 3 plausible distractors

QUESTION 2 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295004 AK1.04		IR 2.8

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER : Effect of battery discharge rate on capacity.

Proposed Question:

A 125 VDC bus has experienced a trip of its battery charger supply breaker. Efforts to restore the battery charger to service have been unsuccessful. Assuming battery loading remains un-changed; predict the change in bus voltage under these conditions.

Voltage will _____.

A. decrease sharply and then level off.

B. decrease in a linear fashion.

C. slowly lower and then drop sharply.

D. stair step down as individual cells are depleted.

Proposed Answer: C

Explanation
A. B. and D. are incorrect but plausible because the applicant is being tested on a phenomenon that has to do with battery discharge rate that was an industry misconception for years. SER 3-99 was sent out and trained on to correct this misconception.
C. Correct – Battery voltage will decrease slowly then drop sharply as battery voltage reversal occurs as seen in SER 3-99.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0305, Obj 8

Question Source: Bank # March 2014 NRC exam #48

Question History: Last NRC Exam March 2014 Exam #48

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

10 CFR Part 55 Content: 55.41.b.8

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

QUESTION 3 Rev 2

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295005 AK2.07	IR 3.6	

Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Reactor pressure control

Proposed Question:

The plant is performing a Startup, operating at 30% power when a break occurs in the EHC Hydraulic Supply line to the #2 Main Stop Valve Actuator. The Standby EHC pump starts on low pressure and the main turbine trips.

In this transient, with no operator action, the Turbine Bypass Valves will __(1)__ and the Reactor will scram due to __(2)__.

A. (1) open ; (2) high RPV pressure.

B. (1) open ; (2) MSIV closure after a low steam line pressure.

C. (1) close ; (2) high RPV pressure.

D. (1) close ; (2) turbine stop valve fast closure.

Proposed Answer: A

Explanation

A. Correct - BPVs remain available for automatic pressure control due to an independent hydraulic system; due to the turbine trip, pressure will rise and the BPVs will try to control pressure by opening. If power were above 30.92%, then the Control Valve Fast closure scram would be enabled to send a trip signal to RPS, but this feature is not enabled so pressure will rise until the high pressure scram setpoint.

B. BPVs remain available for automatic pressure control due to an independent hydraulic system.

C. BPVs remain available for automatic pressure control due to an independent hydraulic system.

D. See A.

Technical Reference(s): R-STM-0509, pp. 15, 54 of 81

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0509, Obj 10d

Question Source: Bank # RBS-OPS-3478

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Replaced question to better match the KA (Rev 1) ; edited part 1 of all distractors (Rev 2)

QUESTION 4 Rev 2

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295006 AK3.03		IR 3.8

Knowledge of the operational implications of the following concepts as they apply to SCRAM: Reactor pressure response

Proposed Question:

The reactor has been operating at 100% power for an extended period when a scram occurs. Before the scram, reactor pressure will be approximately _(1)_ ; 5 minutes after the scram, reactor pressure will be approximately _(2)_.

A. (1) 960 psig ; (2) 960 psig

B. (1) 1050 psig ; (2) 1050 psig

C. (1) 1050 psig ; (2) 960 psig

D. (1) 960 psig ; (2) 870 psig

Proposed Answer: C

Explanation
A. Part 2 is correct; The part one pressure is credible because 960 psig is the post scram steady state pressure.
B. Part 1 correct. Part 2 is credible if the student has a misconception about the head loss quality of steam flow; post scram, the BPV's will control pressure at the same setpoint as pre-scram, but due to the lack of steam flow the pressure at the BPV's will be the same as the pressure at the RPV.
C. Correct – Simulator fidelity to plant data curve for an uncomplicated reactor scram demonstrates the RPV pressure pre-scram is stable at 1050 psig and post-scram is stable at 960 psig roughly 200 seconds later. This time would not have allowed a cooldown to have been started based on stem conditions.
D. Both parts are incorrect: see C

Technical Reference(s): ANSI 3.5 required test for Simulator fidelity

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0543, Obj 2

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.1

Comments: Edited stem to raise LOD (Rev 1) ; Replaced question to not test GFES material (Rev 2)

QUESTION 5 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295016 AA1.06	IR 4.0	

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM ABANDONMENT: Reactor water level

Proposed Question:

The Main Control Room must be abandoned due to Halon actuation; there is **not** a fire in the MCR. In accordance with AOP-0031, Shutdown from Outside the Main Control Room, what systems will be initiated for RPV level control?

A. LPCS, HPCS, and RCIC

B. HPCS, and RCIC only

C. RCIC only

D. All Division 1 ECCS systems

Proposed Answer: A

Explanation

A. Correct – Subsequent actions for abandonment of the control room without a fire direct initiation of HPCS, LPCS, and RCIC.
B. See "A" this answer does not include LPCS
C. See "A" this answer does not include LPCS or RCIC
D. See "A" this answer does not include HPCS, and should not include RHR-A

Technical Reference(s): AOP-0031, Rev 322 p.10 of 122

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP-0031, Obj 4

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 6 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295018 AA2.04	IR	2.9

Ability to determine and/or interpret system flow as it applies to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER.

Proposed Question:

With the plant operating at 100%, the following annunciators are received:

- TURB CMPNT CLG WATER SYSTEM LOW PRESSURE
- TURB CMPNT CLG WTR SYS SURGE TK LOW LEVEL

On H13-P870, the unit operator observes:

- All 3 CCS pumps are running with elevated amps
- MWS-AOV132 TPCCW SURGE TK MAKE-UP VALVE is OPEN

Which of the following is the cause of the above?

A. CCS piping failure has caused a loss of inventory.

B. GSN-PCV1C, MANIFOLD REGULATOR (Nitrogen Pressure Control Valve to TPCCW Surge Tank) has failed closed.

C. CCS-LT113, SURGE TANK LEVEL TRANSMITTER has failed low.

D. CCS-PV111, MINIMUM FLOW AND PRESSURE CONTROL VALVE, has failed OPEN.

Proposed Answer: A

Explanation

A. Correct - A large leak would cause all the indications, low pressure - starts stby pump, max amps as the pumps operate in runout condition, and loss of inventory causes the surge tank low level condition **and** opening of the makeup valve.

B. The TPCCW surge tank is blanketed with 18 psig nitrogen for NPSH and for corrosion control; if this valve fails closed it would lower system pressure, but not surge tank level.

C. The surge tank level transmitter failing low would cause the surge tank low level alarm and the opening of the surge tank makeup valve, but would not cause all pumps to run with maximum amps, nor would it cause a system low pressure alarm.

D. This would lower system pressure, but would not account for a low Surge Tank level.

Technical Reference(s): ARP-P870-55-B01, C02, PID 09-07A

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-116 Obj 2, 5

Question Source: Bank # RBS Audit Exam, March 2010 #6

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.4 Comments:

QUESTION 7 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295019 G2.4.47	IR 4.2	

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question:

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

P870-51A-A01, INSTRUMENT AIR COMPRESSOR TROUBLE

P870-51A-B01, SAS COMPRESSOR ALIGNED TO IA SYSTEM

The Unit Operator notices the following:

- Instrument Air Header Pressure on IAS-PI105 is trending downward at a rate of 1 psig every 10 seconds.
- A and B IAS compressors amber lights lit; C IAS compressor red light is lit

Assuming the situation continues to degrade, which of the following represents the correct sequence of events?

A. 1) FRV's lock up and fail as is
2) Fuel Pool Gate seals swap to nitrogen
3) Service Air Header cross-connect opens

B. 1) FRV's lock up and fail as is
2) Fuel Pool Gate seals swap to nitrogen
3) MSIV's fail closed

C. Service Air Header cross-connect opens
2) FRV's lock up and fail as is
3) MSIV's fail closed

D. 1) MSIV's fail closed
2) FRV's lock up and fail as is
3) Fuel Pool Gate seals swap to nitrogen

Proposed Answer: C

Explanation

C. Correct – In accordance with AOP-8, the automatic actions of a loss of instrument air are (in order): the cross tie valve opens at 113 psig; the low air header pressure alarm and the service air block valve closing both occur at 110 psig ; FRVs lock-up at 85 psig; MSIV's go closed at 65 psig; Fuel Pool Gates switch to nitrogen at 26 psig.

Technical Reference(s): AOP-0008, Loss of Instrument Air, Rev 37, pp. 4-5 of 21

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0527, Obj 4

Question Source: Modified Bank # RBS December 2008 NRC Exam #8

Question History: Last NRC Exam December 2008 NRC Exam #8

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Edited the distractors for clarity and credibility (Rev 1)

QUESTION 8 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 1
	K/A #	295021 AK1.03	IR 3.9

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Adequate core cooling

Proposed Question:

The plant has been in MODE 4 for 25 hours, cooling down for a refueling outage with the following conditions:

- RHR-B is in Shutdown Cooling mode of operation
- Recirc Pump A is tagged out
- Recirc Pump B is running in slow speed
- Reactor Coolant Temperature is 190˚F
- RPV level +38" and steady

RHR Pump B trips on motor overload and RHR Pump A will not start.

Which of the following operator actions would assure adequate decay heat removal?

A. Align RWCU for Alternate Shutdown Cooling.

B. Align for Main Steam Line Flooding

C. Align SPC/ADHR in Configuration 1

D. Raise RPV level to greater than 75 inches

Proposed Answer: C

Explanation
Note: In this mode, adequate core cooling is equivalent to adequate decay heat removal.
A. RWCU is only aligned for ADHR in mode 5 with coolant temperature below 125˚F
B. MSL Flooding can only be used between 120 and 170˚F
C. Correct – SPC/ADHR is available when reactor coolant temperature is below 200˚F
(Configuration 1 takes a suction from SDC and discharges through LPCI-C injection line)
D. Raising RPV level to +75 ensures natural circulation to prevent stratification and inadvertent pressurization, but does not assure adequate decay heat removal

Technical Reference(s): OSP-0041, Alternate Decay Heat Removal, Rev 306 pp 8, 18, 40
AOP-0051, Loss of Decay Heat Removal, Rev 312 pp 5&6 of 30

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-543, Obj 5

Question Source: Bank 2010 Grand Gulf NRC exam Q#40

Question History: Last NRC Exam NA

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

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

QUESTION 9 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 1
	K/A #	295023 AK2.05	IR 3.5

Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Secondary containment ventilation

Proposed Question:

Refueling operations are in progress when an irradiated fuel bundle is dropped in the spent fuel pool. The refueling team on the lower bridge reports bubbles rising from the dropped fuel bundle and the following annunciators are received in the MCR:

- H13-P863-75A-H01, DIV I Fuel Bldg Exh PAM Gaseous Radn Alarm
- RMS-DSPL230-1GE005, Fuel Build Stack/Vent Exhaust A – High
- RMS-DSPL230-2GE005, Fuel Build Stack/Vent Exhaust A – High

Which of the following describes the Fuel Building ventilation lineup after the conditions given above?

A. Fuel Building Ventilation is completely isolated

B. Supply air is via normal supply fans and exhaust is through the Div 1 charcoal filter trains only

C. Supply air is via Fuel Receiving Area and exhaust is through both Div 1 and Div 2 charcoal filter trains

D. Supply air is via Fuel Receiving Area and exhaust is through the Div 1 charcoal filter trains only

Proposed Answer: C

Explanation

A. A high radiation condition in the fuel building does not isolate the ventilation system; it isolates the normal supply and exhaust fans, and starts the charcoal filtration system.

B. A high radiation condition in the fuel building isolates the normal supply and exhaust fans, and starts the charcoal filtration system.

C. Correct - RMS-RE5A is the instrument that drives all three annunciators; this instrument will start the Div 1 filter train only, however a low flow signal will also start the Div 2 train. P&L 2.1 states that 1 train will need to be secured after an auto start signal is received.

D. This instrument will send a start signal to the Div 1 filter train only, but both will start (see C)

Technical Reference(s): R-STM-0406, pp 19 & 41 of 50 ; ARP-P863-75A-H01; SOP-0062, FB HVAC, P&L 2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0406, Obj 11

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Added “only” to distractors B & D; added more to the explanation (Rev 1)

QUESTION 10 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	1	Group # 1
	K/A #	295024 EK3.06	IR 4.0

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Reactor Scram

Proposed Question:

What is the reason for the reactor scram that occurs due to a High Drywell Pressure?

A. To minimize the possibility of fuel damage due to a reactor coolant pressure boundary leak by reducing the amount of energy being added to the coolant.

B. To ensure the Pressure Suppression function of the containment is maintained in the event Emergency Depressurization is required.

C. To ensure that offsite dose limits are not exceeded during a reactor coolant pressure boundary leak.

D. To avoid clearing of the suppression pool vents due to high drywell pressure.

Proposed Answer: A

Explanation
A high drywell pressure condition results due to a leak of the primary system. Due to the loss of coolant, an inability to cool the fuel may result. A reactor scram occurs to minimize the energy being produced. The pressure suppression function of the containment is based on containment pressure not drywell pressure. Offsite dose limits are prevented from being exceeded by the high drywell pressure containment isolation, not the high drywell pressure reactor scram. Although the scram signal will reduce the energy being leaked into the drywell, and may avoid clearing of the suppression pool vents, this is not the reason for the scram.

Technical Reference(s): R-STM-0508, RPS, Rev 6 p. 46 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508 Obj 2

Question Source: Bank # 2008 NRC Exam Q#11

Question History: Last NRC Exam RBS 2008

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

10 CFR Part 55 Content: 55.41.b.6

Comments: Rejected KA: EK3.04 ; Randomly selected new KA statement EK3.06
(RBS has no guidance to Emergency Depressurize for a High DW D/P

QUESTION 11 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295025 EA1.05		IR 3.7

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

Proposed Question:

The plant is operating at 93% power.
A RCIC lube oil change has just been completed and RCIC has been started with the "slow-roll startup" section of SOP-0035 in the RPV Pressure Control lineup (CST to CST). All MSIV's go closed and the reactor scrams on high reactor pressure.

What is the status of RCIC after the scram?

A. RCIC will trip on over speed due to the high steam supply pressure.

B. RCIC will re-align and inject into the RPV from the CST.

C. RCIC will re-align and inject into the RPV from the Suppression Pool.

D. RCIC will remain in a CST to CST lineup.

Proposed Answer: D

Explanation

A. Over speed is a RCIC trip signal, but high reactor / steam supply pressure does not cause this. The pressure spike from the MSIV closure will however cause RCIC speed to increase

B. RCIC initiation occurs at RPV Level 2 (-43"). There is no logic setpoint reached to cause a realignment of RCIC. The stem does not indicate a loss of Feedwater, RBS has motor-driven feed pumps

C. There is no logic setpoint reached to cause a realignment of RCIC or a swap to the Suppression Pool

D. Correct – There is no logic setpoint reached to cause an initiation of RCIC

Technical Reference(s): R-STM-0209, Rev 10, pp. 26,27,43 of 52

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209, Obj 7, 12

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Edited all distractors for credibility; fixed explanation (Rev 1)

QUESTION 12 Rev 2

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:

Following a loss of offsite power. Suppression Pool level is currently 19 feet, 11 inches. An operator is controlling RPV pressure with a given pressure control band of 800 to 1090 psig.

What is the lowest Suppression pool temperature that would require emergency depressurization?

A. 142˚F

B. 138˚F

C. 133˚F

D. 127˚F

Proposed Answer: C

Explanation
A, B and D Incorrect – All distractors are plausible if the student uses the incorrect suppression pool level curve
C. Correct –: Suppression pool level, stated as 19’11” in the stem, requires the examinee to determine that the 19’6” line on the HCTL curve should be used (more conservative and not allowed to interpolate). The top of the band must be used to determine if the HCTL will be reached. The lowest s.p. temperature that would require ED-for 1090 psig is 133˚F.

Technical Reference(s): EOPs, Heat Capacity Temperature Limit Curve

Proposed references to be provided to applicants during exam: Heat Capacity Temperature Limit Curve

Learning Objective: RLP-HLO-517, Obj 2

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Edited stem and all answer choices in such a way as to make the applicant plot points using the HCTL curve, to find only one correct answer. (Rev 1) ; Edited stem to provide the pressure band and make the question one part rather than two. Edited explanation for clarity. (Rev 2)

QUESTION 13 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295027 G2.4.20		IR 3.8

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

Proposed Question:

A suppression pool temperature of 150˚F may result in ___.

A. direct pressurization of containment from RCIC turbine exhaust during operation.

B. damage to the RCIC turbine during operation due to reduced lube oil cooling.

C. exceeding the NPSH limit for RCIC pump suction.

D. inaccurate RPV level instrument indication.

Proposed Answer: B

Explanation

A. Direct pressurization is a concern with a high suppression pool level, not temperature (EOP Caution 4)

B. Correct- the lube oil and control oil for RCIC is cooled by process flow; the max allowable cooling water temperature for RCIC lube oil is 140˚F (EOP Caution 3)

C. EOP Caution #5 states that NPSH limits will be exceeded with SP temperatures above 160°F. RCIC is the most limiting, see EPSTG0002 page B-5-13

D. EOP Caution #1 states that RPV level instruments will be unreliable at elevated temperatures. Elevated containment temperatures, caused by high SP temperatures, must exceed 200°F.

Technical Reference(s): EOP-1 Caution 3, Bases p. B-5-10

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-HLO-511, Obj 6

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.9

Comments: Edited temperature given in stem. Replaced distractors C & D and their explanations for credibility. Rev 1

QUESTION 14 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295028 EK1.01	IR 3.5	

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level measurement

Proposed Question:

EOP-1 Caution 1, part 2 identifies RPV level instruments that can be used when the containment or drywell temperature near the reference legs is below the specified limits.

_(1)_ RPV level instruments are limited by Drywell Temperature _(2)_

A. (1) Narrow, Wide, and Fuel Zone only
(2) less than 200 degrees F

B. (1) Narrow, Wide, and Fuel Zone only
(2) less than 440 degrees F

C. (1) Upset and Shutdown only
(2) less than 440 degrees F

D. (1) Upset and Shutdown only
(2) less than 200 degrees F

Proposed Answer: C

Explanation

A. Incorrect because the temp and RPV instruments are wrong per the table that is used for this caution.

B. Incorrect because although the temp is correct the RPV instruments associated with the drywell are the upset and shutdown instruments.

C. Correct- per the table in EOP-1, caution 1, the upset and shutdown RPV level instruments can be used when above MIL and if DW temp er less than 440 degrees F. An applicant should determine that the longer instrument runs are for these two instruments since they go through the Drywell and this helps eliminate the other choices and they should recognize the 440 deg F value.

D. Incorrect because even though the instruments are correct the temp for DW is incorrect.

Technical Reference(s): EOP-1, Caution 1, revision 26.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-HLO-511, Obj 6

Question Source: Bank # New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.2

Comments: Replaced question to address focus, cueing, and credible distractors (Rev 1)

QUESTION 15 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295030 EK2.08	IR 3.5	

Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: SRV discharge submergence

Proposed Question:

For normal SRV operations, which of the following suppression pool levels is based on direct pressurization of containment air space?

A. 13 feet

B. 15 feet, 5 inches

C. 21 feet, 3 inches

D. 21 feet, 6 inches

Proposed Answer: A

Explanation

A. Correct – This is the elevation of the top of the SRV discharge device below which opening of an SRV may cause pressurization of the containment air space

B. 15’5” is 2 feet above the horizontal vents; this level is associated with a leak from the DW passing through the horizontal vents.

C. 21’3” is the SRV tail pipe level limit; operation of an SRV above this limit may cause damage to the SRV discharge lines.

D. 21’6” is based on the Pressure Suppression Pressure (PSP) Curve; the PSP pressure is a function of suppression pool level.

Technical Reference(s): EOP Bases p. B-6-56

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 5

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.9

Comments:

QUESTION 16 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295031 EK2.03		IR 4.2

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Low pressure core spray

Proposed Question:

An IST STP is being performed. An operator closes E21-MOV F011, LPCS Min Flow Valve; all other valves are in their normal, standby positions. A valid RPV low level, Low Pressure Core Spray (LPCS) initiation signal is received.

Assuming no operator action, how will the LPCS system respond?

A. The LPCS Min Flow Valve will remain closed until the initiation signal is reset.

B. The LPCS Min Flow Valve will open until system flow rises above 875 gpm.

C. The LPCS Pump will not receive a start signal because of the pump protection logic permissive not being met.

D. The LPCS Pump will start and will eventually overheat.

Proposed Answer: B

Explanation

A. The reset pushbutton is in the LPCS initiation sequence logic; the min flow valve is not part of this logic.

B. Correct – the design logic for the min flow valve is to OPEN when the pump breaker is closed and the sensed flow is less than 875 gpm.

C. The pump start logic does not include a permissive from the min flow valve

D. The pump would overheat without a flow path providing at least 500 gpm; the min flow valve will open with the given conditions.

Technical Reference(s): R-STM-0205 pp.13-14,16 of 33 ; STP-205-4201, Rev 301

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205, Obj 4,9

Question Source: Modified Bank # RBS-OPS-5554

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: edited stem for clarity (Rev 1)

QUESTION 17 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	1
	K/A #	295037 EA1.06		IR 4.1

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

Proposed Question:

During ATWS conditions, using EOP-001A, the following minimum conditions would require boron injection regardless of suppression pool temperature:

A. APRM's oscillate above 10% one time

B. APRM's oscillate above 10% repeatedly

C. APRM's oscillate above 25% one time

D. APRM's oscillate above 25% repeatedly

Proposed Answer: D

Explanation

D. Correct – to provide reasonable assurance that any rapidly growing oscillations are mitigated in a timely manner, boron is injected when neutron flux oscillations in excess of the Large Oscillation Threshold (LOT) commence and continue. 10% does not meet the minimum of 25% for step RQA-3 in EOP-001A.

Technical Reference(s): EOP Bases, page B-7-57

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-513, Obj

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.1

Comments: Replaced question to not test GFES material (Rev 1)

QUESTION 18 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 295038 EA2.03	IR 3.5	

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Radiation levels

Proposed Question:

The radioactivity release control leg of EOP-0003, Secondary Containment and Radioactive Release Control is entered when radiation monitors reach the rate levels corresponding to the ___ action level defined in the Site Emergency Plan.

A. NOUE

B. ALERT

C. SITE AREA EMERGENCY

D. GENERAL EMERGENCY

Proposed Answer: B

Explanation
A. NOUE is below the entry level for EOP-0003
B. Correct- Entry into EOP-0003 radioactive release leg corresponds to the ALERT action level; radiation monitors' setpoints are set accordingly.
C. See B
D. See B

Technical Reference(s): EOP-0003

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-515, Obj 2

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 19 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 600000 G2.4.11	IR 4.0	

Knowledge of abnormal condition procedures

Proposed Question:

The plant is operating at 100% power. Standby Gas Treatment Filter Train "A" is being operated for surveillance testing.

The Auxiliary Building Operator reports smoke coming from the "A" Standby Gas Treatment Filter Train and the filter train case is glowing red.

Which one of the following describes the method to combat a fire in the Standby Gas Treatment Filter Train?

A. The Fire Protection System will initiate the automatic deluge system and fill the filter train with water.

B. The Fire Protection System will automatically open a deluge isolation valve, however, valves must be manually opened to admit water to the filter train.

C. The Fire Protection system Deluge Valve will have to be manually initiated via the pull station to admit water to the filter train.

D. The Fire Protection System at the filter train must be manually valved in to admit water to the filter train.

Proposed Answer: D

Explanation
A. There is no automatic fire suppression for the GTS train
B. The isolation valves must be manually opened, however there is no auto fire suppression for GTS
C. Some deluge systems on site have this capability but not for the GTS train
D. Correct - The isolation valves must be manually opened, no deluge valve actuation is needed.

Technical Reference(s): AOP-0052, Fire Outside the MCR, Rev 25, step 5.4 p. 4 of 62
Pre-Fire Strategy AB-141-531 SGTS Filter A Room Fire Area AB-14

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0257, Obj 4

Question Source: Bank # RBS-NRC-654

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Added technical reference (Rev 1)

QUESTION 20 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 1	
	K/A # 700000 AK1.03	IR 3.3	

Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Under-excitation

Proposed Question:

Under-excitation of the Main Generator results in ___.

A. Turbine rotor overheating.

B. Generator field overheating.

C. Voltage Regulator shifting to Manual.

D. Generator armature overheating.

Proposed Answer: D

Explanation
A. A loss of generator field causes “turbine torque oscillation” which leads to the Turbine rotor overheating.
B. Over-excitation causes the Generator field to overheat.
C. An exciter field overcurrent causes the Voltage Regulator to shift to Manual.
D. Correct – Under-excitation causes overheating of the armature.

Technical Reference(s): R-STM-0310, Rev 8 p. 29 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0310, Obj 3,10

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41b.5

Comments:

QUESTION 21 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 1	Group # 2	
	K/A # 295007 AK2.06	IR 3.5	

Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: PCIS/NSSSS: plant specific

Proposed Question:
Following a planned reactor shutdown, a plant cooldown is in progress with RHR A as the in service shutdown cooling system.

RHR Pump A subsequently trips due to an overcurrent condition.

Due to the trip, an uncontrolled heatup and pressurization has occurred. The following conditions exist:

Reactor water level 80 inches
Reactor pressure 150 psig

Assuming the shutdown cooling reliability plan is NOT installed and NO operator actions have been taken, which of the following represents the status of E12-F053A, RHR PUMP A SDC INJECTION VALVE and E12-F027A, RHR PUMP A OUTBD ISOLATION VALVE?

	E12-F053A	E12-F027A
A.	CLOSED	OPEN
B.	CLOSED	CLOSED
C.	OPEN	OPEN
D.	OPEN	CLOSED

Proposed Answer: A

Explanation

A. Correct - E12-F053A receives an isolation signal at 135 psig from NSSSS. E12-F027A is normally opened and does not receive an isolation signal therefore remains open.

B and D are incorrect because the F027 valve does not receive an isolation signal and is normally open

C. is incorrect because the F053 valve receives an isolation signal at >135 psig.

Technical Reference(s): R-STM-0204, Rev 11 pp. 39, 41 of 63

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank# RBS Dec.2008 NRC #22

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.9

Comments: Rejected KA AK2.01– GFES related; replaced question (Rev 1)

QUESTION 22 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295008 AA2.05	IR 2.9	

Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: Swell

Proposed Question:

River Bend was operating at full power with RCIC tagged out. A loss of offsite power occurred and HPCS recovered RPV level. Operators closed the HPCS injection valve with an RPV water level of +20 inches. Current conditions are as follows:

- MSIV's are closed
- Reactor pressure is 700 psig and rising at 10psig per minute
- Time after scram is 5 minutes
- Drywell pressure is 0.3 psid steady

What is the expected RPV water level response over the next 10 minutes and why?

A. RPV level will rise due to swell from decay heat

B. RPV level will rise due to HPCS injection valve leak by

C. RPV level will lower due to rising pressure which will collapse the voids

D. RPV level will lower due to injection being secured

Proposed Answer: A

Explanation

A. Correct- Level rises due to expansion cause by the heat up from decay heat

B. This is plausible because Feed Reg Valves do leak by at RBS; However HPCS inj valve does not

C. This is plausible because the applicant may consider the addition of cold water as a means of collapsing voids, when, in fact, RPV water level will rise not lower. Void production/collapse will be minimal under these conditions.

D. RPV water level will rise not lower. Only a loss of RPV inventory would cause level to lower. The stem gives no indication of an RPV leak with drywell pressure steady.

Technical Reference(s): RLP-STM-0107, Rev 027, p. 66 of 105

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107, Obj 4

Question Source: Modified Bank # RBS-NRC-01108

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.14

Comments: Edited distractors for credibility, fixed explanation to match, corrected tech.reference (Rev 1)

QUESTION 23 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295011	G2.4.04	IR 4.5

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures: High Containment Temperature

Proposed Question:

Based on the following current conditions, which one of the following describes the Emergency Operating Procedure(s) that should be entered?

- Reactor Power 0% (all rods in)
- RPV Pressure 1064.7 psig
- RPV Level 55 inches
- Containment Temperature 92˚F
- Containment Pressure 0.25 psig
- Drywell Temperature 140˚F
- Drywell Pressure 0.75 psid
- Annulus Differ. Pressure -1.3 in WC

A. EOP-1, RPV Control ONLY

B. EOP-1, RPV Control AND EOP-2, Primary Containment Control

C. EOP-2, Primary Containment Control ONLY

D. EOP-2, Primary Containment Control AND EOP-3, Secondary Containment and Radioactivity Release

Proposed Answer: C

Explanation

A. No EOP-1 Entry Condition exists; EOP-1 entry is plausible because RPV level of 55" would be a scram setpoint if reactor were at power.

B. No EOP-1 Entry Condition exists; see A

C. Correct- EOP-0002 Entry Condition is Containment Temperature above 90˚F

D. No EOP-3 Entry Condition exists ; EOP-3 entry is plausible because Annulus Differential pressure is an often misunderstood setpoint (actual setpoint -3" WC)

Technical Reference(s): EOP-1, -2, and -3

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 3

Question Source: Modified Bank # RBS-NRC-444

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: KA rejected because there is no Alarm or ARP for Cont. High Temperature.
Redrew from the 50 possible choices in the G2.4 category to get 2.4.4
edited two parameters in stem to give more credibility to distractors (Rev 1)

QUESTION 24 Rev 2

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295012 AK1.02		IR 3.1

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE : Reactor power level control

Proposed Question:

In the step just prior to Emergency Depressurization on High Drywell Temperature, EOP-2 directs ___.

A. that all Drywell coolers be operated to maintain temperature below 135°F

B. that the reactor is scrammed and shutdown by control rod insertion

C. depressurization of the RPV to prevent an immediate challenge to containment integrity

D. depressurization of the RPV to prevent an immediate challenge to SRV operation

Proposed Answer: B

Explanation
A. incorrect because all six unit coolers are not allowed until 145°F is reached (not 135)
B. Correct- Per EOP Bases the reactor is scrammed and shutdown because entry conditions of EOP-2 do not necessarily require entry to EOP-1
C. incorrect because High drywell temperature is not an immediate challenge to containment integrity,
D. incorrect because High drywell temperature is not an immediate challenge to SRV operation. See EPSTG-0002 page B-8-6

Technical Reference(s): EOP BASES p. B-8-6

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 5

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.9

Comments: Edited stem and distractors to raise LOD (Rev 1) ; Edited stem and Dist A per re-validation comments (Rev 2)

QUESTION 25 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295013 AK2.01		IR 3.6

Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following: Suppression pool cooling

Proposed Question:

A safety relief valve has opened and AOP-0035, SAFETY RELIEF VALVE STUCK OPEN has been entered. Following reduction in reactor power to 89%, the CRS has directed you to place RHR A in suppression pool cooling mode.

Why is suppression pool cooling initiated?

A. to obtain localized suppression pool temperature

B. to minimize heat input to the suppression pool

C. to immediately reject suppression pool level

D. to establish bulk mixing of the suppression pool

Proposed Answer: D

Explanation
A. RHR heat exchanger inlet temperature could be used to determine local temperature, however T.S. and TRM actions are based on average temperature not localized temperature
B. SP cooling does not affect the amount of heat added to the pool from the open SRV. RHR will remove the heat after it has been input to the pool.
C. While reject may become required much later, the reason for establishing SP cooling is to establish bulk mixing and to cool the pool
D. Correct- RHR A is placed in suppression pool cooling mode to establish bulk mixing of the suppression pool such that an average suppression pool temperature may be obtained

Technical Reference(s): AOP-0035, Safety Relief Valve Stuck Open, Rev 19 p. 5 of 10

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP035, Obj 4

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Edited two distractors for credibility, adjusted explanation (Rev 1)

QUESTION 26 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	1	Group #	2
	K/A #	295029 EK3.03		IR 3.04

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Reactor SCRAM

Proposed Question:

EOP-2, Suppression Pool Level Control, requires entry into EOP-1, RPV Control if suppression pool level cannot be maintained below 21 feet.
The reason for entering EOP-1 in this condition is that it will ___.

A. ensure adequate margin for SRV operation to prevent damage to the SRV discharge line.

B. force a scram and shutdown of the reactor before the containment pressure instruments become submerged.

C. allow for a reactor shutdown prior to over flowing the weir wall and flooding the drywell.

D. require emergency depressurization to be accomplished.

Proposed Answer: A

Explanation

A. Correct- EOP bases states that This action reduces core heat and the steam generation rate in the RPV to decay heat levels (assuming the scram is successful), thereby assisting in maintaining plant conditions below the SRV Tail Pipe Level Limit

B. EOP Caution #7 does apply to this leg of EOP-2, however the instruments will not be submerged at 21 feet.

C. EOP-1 entry on SP level is based on SRV component loading and injection source termination. Water addition to the drywell is not a factor in the reason to Scram on high suppression pool water level

D. EOP-1 contains the procedural steps for ED. Entering EOP-1 from EOP-2 is required to enter the procedure with the ED steps, since EOP-2 may not have been entered from a condition that would require a Scram. ED is not required until suppression pool level reaches 21'-3".

Technical Reference(s): EOP Bases p. B-8-21, B-8-22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-514, Obj 5

Question Source: Bank # RBS-NRC-20

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Edited stem and answer choices for better KA match (Rev 1)

QUESTION 27 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐	
	Tier #	1	Group # 2	
	K/A #	295032 EA1.03		IR 3.7

Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : Secondary containment ventilation (41.7)

Proposed Question:

During the performance of STP-000-0001 DAILY OPERATING LOGS, the auxiliary building operator reports a high temperature condition in the High Pressure Core Spray pump room.

The Unit Operator verifies the status of HVR-UC5 on __(1)__ and verifies the alignment of cooling water on __(2)__

A. (1) H13-P863 ; (2) H13-P601 or H13-P863

B. (1) H13-P870 ; (2) H13-P870 or H13-P863

C. (1) H13-P870 ; (2) H13-P601 or H13-P863

D. (1) H13-P863 ; (2) H13-P863 or H13-P870

Proposed Answer: D.

Explanation:
A. Part 1 is correct, but the cooling water indication lights on H13-P601 are for the HPCS Diesel Generator. Cooling water indication is found on H13-P863 and indications and controls are on H13-P870.
B. incorrect because HVR-UC5 control switch is on H13-P863. Part 2 is correct.
C. incorrect because HVR-UC5 control switch is on H13-P863 and the cooling water indication lights on H13-P601 are for the HPCS Diesel Generator. H13-P863 is correct and H13-P870 also contains control switches and indicating lights.
D. Correct – The control switch and status lights for HVR-UC5 are on H13-P863. The control switch and indicating lights are on H13-P870. Additionally a set of valve position indicating lights are located on H13-P863.

Technical Reference(s): R-STM-0409 Rev 6 Page 12 of 40, SOP-0018 Rev 52 Pg 151-154 of 158 Att. 4B

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0409 Obj 4

Question Source: Bank # RBS Nov 2012 NRC exam #26

Question History: Last NRC Exam RBS Nov 2012 NRC exam

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

10 CFR Part 55 Content: 55.41b.7

Comments: Appeared on one of last 2 NRC exams (2 of 3)
Rejected KA EA1.01 due to inability to create 3 plausible distractors. (Rev 1)

QUESTION 28 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	203000 K4.03		IR 3.2

Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Pump minimum flow protection

Proposed Question:

The reactor has been shut down due to a transient.

Current plant conditions are:

- Reactor pressure is 528 psig and slowly decreasing
- Reactor water level is -120 inches and slowly decreasing
- Drywell pressure is 1.78 psid and slowly increasing
- All low pressure ECCS systems are running as designed

What is the status of the RHR C system?

A. Injection valve (F042C) is open and pump minimum flow valve F064C is closed.

B. Injection valve (F042C) is open and pump minimum flow valve F064C is open.

C. Injection valve (F042C) is closed and pump minimum flow valve F064C is closed.

D. Injection valve (F042C) is closed and pump minimum flow valve F064C is open.

Proposed Answer: D

Explanation
A. and C are incorrect because the min flow valve, F064C is interlocked to open when flow is <1100 gpm open.
B. and A are incorrect because F042C, injection valve will not receive an open signal until pressure is below 487 psig.
D. Correct, The injection valve will open when pressure drops below 487 psig, stem conditions are above this setpoint; the min flow valve is normally open in the standby lineup and also receives an open signal with the pump running and flow less than 1100 gpm.

Technical Reference(s): R-STM-0204, RHR, Rev 11 pp. 12,14,15 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204, Obj 6

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Rejected KA K4.09 ; for inability to write psychometrically sound question; Randomly selected K4.03
Wording of KA suggests How do design features/interlocks provide for STPs

QUESTION 29 Rev 1

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 2 Group # 1
K/A # 205000 K5.02 IR 2.8

Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : Valve operation

Proposed Question:

The plant is shut down. RHR A is operating in Shutdown Cooling mode (SDC) with an RPV water level of 75 inches.
Which of the following valves, if operated from the main control room, would lead to an inventory loss?

A. E12-F004A, RHR PUMP A SUP PL SUCTION VALVE

B. E12-F006A, RHR PUMP A SDC SUCTION VALVE

C. E12-F024A, RHR PUMP A TEST RTN TO SUP PL

D. E12-F042A, RHR PUMP A LPCI INJECTION ISOL VALVE

Proposed Answer: C

Explanation
A. Incorrect - Interlocks prevent opening of E12-F004A with E12-F006A already open
B. Incorrect - closure of E12-F006A would cause to the RHR pump to trip but would not cause an OPDRV
C. Correct- There are no interlocks preventing operation and opening of E12-F024 would direct RPV water to the suppression pool
D. Incorrect – Opening E12-F042A would redirect water through the LPCI injection line but would not cause an OPDRV

Technical Reference(s): R-STM-0204, RHR, Rev 11 pp. 16,21 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204, Obj 6

Question Source: Bank # RBS-NRC-182

Question History: Last NRC Exam NRC Exam 7/1997

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Changed stem due to validation comment: the definition of OPDRV not being met for this question. (Rev 1)

Question Deleted from Exam due to R4 Panel Recommendation during post exam comments/analysis by RBS - three potentially correct answers.

QUESTION 30 Rev 1

Examination Outline Cross-Reference: Level RO ☑ SRO ☐
Tier # 2 Group # 1
K/A # 209001 K6.03 IR 3.3

Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM : Torus/suppression pool water level

Proposed Question:

In accordance with SOP-0032, the Low Pressure Core Spray (LPCS) pump should **not** be run with suction from the suppression pool if level is ____.

A. Less than 10 feet because of NPSH limits

B. Less than 13 feet 3 inches because of NPSH limits

C. Less than 10 feet because of vortexing limits

D. Less than 13 feet 3 inches because of vortexing limits

Proposed Answer: B

Explanation
A. Incorrect, SP level of 10 feet is the value in the EOPs for all ECCS equipment but is not the limit specifically for LPCS in the SOP. The second part is correct. An applicant might confuse the caution in the EOP and its values and reasons with this choice.
B. Correct- SOP-0032 Precaution and Limitation 2.10 Lists 13'3" as the necessary level to assure NPSH
C. Incorrect, SP level of 10 feet is the value in the EOPs for all ECCS equipment but is not the limit specifically for LPCS in the SOP. The second part is also incorrect (the temp limit listed in the EOP of 160 degrees is for vortexing). An applicant might confuse the caution in the EOP and its values and reasons with this choice.
D. Incorrect, SP level of 13 feet 3 inches is correct he limit specifically for LPCS in the SOP. The second part is incorrect (the temp limit listed in the EOP of 160 degrees is for vortexing). An applicant might confuse the caution in the EOP and its values and reasons with this choice.

Technical Reference(s): SOP-0032, LPCS, Rev 23 P&L 2.10 (p. 3 of 30)

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205, Obj 8

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Replaced question for better credibility on distractors (Rev 1)

QUESTION 31 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	209001 A3.03		IR 3.5

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: System pressure

Proposed Question:

The Low Pressure Core Spray (LPCS) system is running in the test return mode at 5050 gpm.

Two minutes later, a steam leak caused drywell pressure to increase to 1.95 psid.
Reactor water level is -62 inches
Reactor pressure is 450 psig

Which of the following identifies the expected AUTOMATIC response?

A. The Test Return Valve to the Suppression Pool (E21-F012) closes, Injection Isolation Valve (E21-F005) opens and discharge pressure rises.

B. The Test Return Valve to the Suppression Pool (E21-F012) closes, Injection Isolation Valve (E21-F005) opens and discharge pressure lowers.

C. The Test Return Valve to the Suppression Pool (E21-F012) remains open, Injection Isolation Valve (E21-F005) remains closed and discharge pressure rises.

D. The Test Return Valve to the Suppression Pool (E21-F012) remains open, the Injection Isolation Valve (E21-F005) remains closed and discharge pressure lowers.

Proposed Answer: A

Explanation

A. Correct - Upon a LOCA signal of 1.68 drywell diff pressure the LPCS system automatically aligns to inject into the RPV, the test return valve closes, if open, the injection valve opens, if RPV pressure is below 487 psig and the min flow valve opens if flow drops below 875 gpm. Since the stem RPV pressure is below the injection valve interlock pressure but above the pump shut off head (282 psig) the injection valve will open but the pump will not flow into the RPV. The min flow valve will open, flow will drop from the 5050 gpm and discharge pressure will rise. For these reasons all other choices are incorrect.

Technical Reference(s): R-STM-0205, LPCS, Rev 5 pp. 17-18

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0205, Obj 9

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Edited stem for clarity (Rev 1)

QUESTION 32 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	209002 A1.08		IR 3.1

Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) controls including: System lineup: BWR-5,6

Proposed Question:

The plant is in a reactor startup Mode 2. Due to an I&C error, HPCS initiated. The CRS directed the Control Room Operator to close the HPCS injection Valve E22*F004. The Control Room Operator closed E22*F004. The maximum Reactor Water Level reached was + 56 inches. I&C can NOT reset the Initiation Signal.

Reactor Water Level is now + 36 inches and lowering.

Which one of the following describes the operation of E22*F004 HPCS INJECT ISOL VALVE with water level now in the normal band?

A. The valve will automatically open on receipt of a Reactor Water Level - Low Level 2 signal.

B. The valve can be opened using the valve hand switch in the OPEN position.

C. The valve can only be opened if the HPCS High Reactor Water Level signal is first manually reset.

D. The valve will automatically reopen if the HPCS Manual Initiation Pushbutton is depressed.

Proposed Answer: C

Explanation

A. Incorrect With an initiation signal still present and the hand switch taken to close, the injection valve will not auto open on a low level signal

B. Incorrect – the applicant may choose this if they fail to recognize that a level 8 signal has locked in.

C. Correct the level 8 signal will need to be reset(level above 51”) before the injection valve will respond to a manual open signal and will not auto open.

D. Incorrect The initiation signal that is stated in the stem has not been reset, adding a manual signal will not change the logic state to allow the injection to open.

Technical Reference(s): R-STM-0203, HPCS, Rev 8 pp.12, 15,16,20

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0203, Obj 9

Question Source: Bank RBS-NRC-577a Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.8

Comments: Replced question due to LOD (Rev 1)

QUESTION 33 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	211000 K5.06		IR 3.0

Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Tank level measurement

Proposed Question:

An ATWS has occurred and the CRS has directed SLC injection. The CRS directs you to monitor SLC tank level and to report when Hot Shutdown Boron Weight is achieved. While monitoring SLC tank level you notice that SLC tank level indication has failed low. How can it be determined that Hot Shutdown Boron Weight has been injected?

A. Hot Shutdown Boron Weight can be estimated by obtaining a tank level sounding

B. RPV water level can be restored after the tank is empty

C. Hot Shutdown Boron Weight can be estimated using injection time

D. Boron Concentration can be measured by sampling reactor coolant

Proposed Answer: C

Explanation

Hot Shutdown Boron Weight at RBS is determined by the number of gallons injected.

A. Incorrect The level indication reads gallons for tank level a local sounding would produce a percent or depth of fluid. The procedure has no guidance to convert those measurements to gallons.

B. Incorrect With a loss of the level indication the controlroom operator will not know when the tank is empty

C. Correct The procedure requires a time entry when injection is started and also has guidance for the length of time to inject hot shutdown boron weight

D. Incorrect No guidance is given in the procedure to use boron concentration

Technical Reference(s): OSP-0053, Emergency & Transient Response, Attachment 13, Rev 22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0201, Standby Liquid Control, Obj 1

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Edited the original question to allow removal of the handout for LOD (Rev 1)

QUESTION 34 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 211000 A2.03	IR 3.2	

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

Proposed Question:

The plant has experienced an ATWS with the following conditions:

- Reactor power 23%
- Reactor Pressure 875 psig
- MSIVs ISOLATED
- Suppression Pool Temp 117°F
- SLC Pump 'A' Injecting for 12 minutes

Subsequently, an electrical transient occurs, which causes a trip of the following:

- EJS-ACB27, EHS-MCC2A SUPPLY BRKR
- EJS-ACB69, EHS-MCC2F SUPPLY BRKR

Which of the following describes the impact of this condition on SLC and appropriate actions to be taken?

A. SLC 'A' will continue to inject. Use EOP Enclosure 15 to determine the amount of boron that has been injected has been injected.

B. SLC 'A' is no longer injecting. Use OSP-0053 Hardcard to initiate SLC 'B'

C. No SLC pumps are available. Direct the use of Enclosure 15 to commence alternate SLC injection.

D. SLC injection is no longer required. Continue execution of EOP-0001A to shutdown the reactor.

Proposed Answer: B.

Explanation:

A Incorrect because loss of EHS-MCC2A causes a loss of power to SLC A and SLC A squib valve.

B. Correct – Power is lost to the "A" SLC pump, but power is still available to SLC "B"

C. Incorrect because although EJS-ACB69 is a Div 2 power source, it is not the power source for SLC "B"; An applicant that is confused over power supplies may choose this with the loss of the MCC2F.

D. Incorrect - SLC –A did not run long enough to inject cold shutdown boron weight (minimum of 16 minutes required) and Sup Pool temp above 110°F, SLC injection is still required.

Technical Reference(s): EOP-0001A, EE-001AC

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0201 Obj 3, 6, 10; RLP-OPS-HLO-513 Obj 5, 6

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.6

Comments: Re-wrote question to raise LOD (Rev 1)

QUESTION 35 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	212000 A3.06		IR 4.2

Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: Main turbine trip: Plant-Specific

Proposed Question:

Following a main turbine trip from 100% power, the reactor will scram first on which of the following signals?

A. Low water level due to shrink

B. High pressure due to control valve closure

C. High neutron flux due to pressure rise

D. Turbine stop valve position

Proposed Answer: D

Explanation
A. B. and C are incorrect because of design feature of the RPS scram signal associated with a turbine trip.
D. Correct – Closure of either the Turbine Stop Valves or the Turbine Control valves results in the loss of heat sink. This scram anticipates the rise in reactor pressure, neutron flux and heat flux resulting from the loss of heat sink on a turbine trip. When the Turbine Stop valve reaches <95% open position, RPS will initiate a scram to reduce the amount of energy required to be absorbed as well as to ensure that the MCPR Safety Limit is not exceeded.

Technical Reference(s): R-STM-0508, RPS, Rev 6 p. 47 of 59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0508, Obj 2

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.6

Comments:

QUESTION 36 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	215003 A4.05		IR 3.4

Ability to manually operate and/or monitor in the control room: IRM Trip bypasses

Proposed Question:

As the on-coming ATC operator, you receive shift turnover information that IRM "E" was bypassed due to an instrument failure and the half-scram was reset. You expect to see the __(1)__ joystick on the H13-P680 panel in the bypass position for this IRM. You verify the IRM condition on the associated IRM drawer located on panel __(2)__.

A. (1) Left side
(2) H13-P669

B. (1) Right side
(2) H13-P669

C. (1) Right side
(2) H13-P670

D. (1) Left side
(2) H13-P670

Proposed Answer: A

Explanation

A. Correct - There are two IRM joysticks on the 680 panel; the left joystick is for division 1 and provides bypass capability for IRM A, C, E, and G. The right joystick is for div 2 and provides bypass capability for IRMs B. D, F, and H. Panel H13-P669 is for the Division 1 IRMs and Panel 670 is for division 2.

B. and C are incorrect because the right side joystick is for Div 2 IRM's (B,D,F,H)

D. is incorrect because the backpanel for Div 1 IRM's is P669 and Div 2 is P670.

Technical Reference(s): R- STM-0503 , Neutron Monitoring, Rev 7 p. 30

Proposed references to be provided to applicants during examination: MCR Backpanel Map

Learning Objective: RLP-STM-0503, Obj 10

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.2

Comments: Replaced question to remove cueing and raise LOD (Rev 1)

QUESTION 37 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 215004 K2.01		IR 2.6

Knowledge of electrical power supplies to the following: SRM channels/detectors

Proposed Question:

The power supply to SRM 'C' detector is ___.

A. RPS A

B. RPS B

C. VBS-PNL01A1

D. VBS-PNL01B1

Proposed Answer: A

Explanation
A. Correct - RPS A is the power supply to SRM A & C
B. Incorrect - RPS B is the power supply to SRM B & D
C. Incorrect - VBS-PNL01A1 is the power supply to neutron monitoring recorders
D. Incorrect - VBS-PNL01B1 is the power supply to neutron monitoring recorders

Technical Reference(s): R-STM-0503, Neutron Monitoring, Rev 8 p.84 of 112

Proposed references to be provided to applicants during examination: None

Learning Objective: R-STM-0503, Obj 7

Question Source: Modified Bank # Nov 2010 Audit Q#36

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.2

Comments: Fixed typo for proposed answer to the correct answer. (Rev 1)

QUESTION 38 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	2	Group # 1
	K/A #	215004 G2.4.31	IR 4.2

Knowledge of annunciator alarms, indications, or response procedures. SRMs

Proposed Question:

During a reactor startup, the ATC has been withdrawing SRMs in accordance with GOP-0001, Plant Startup. All SRMs **except** C indicate full out; SRM C is reading off scale high and indicates “driving out.” The following alarm is in:

- SRM UPSCALE OR INOPERATIVE

Which of the following additional conditions will initiate a Control Rod Withdraw Block?

A. Mode switch in RUN and all IRMs on Range 8

B. Mode switch in STARTUP/HOT STBY and all IRMs on Range 7

C. Mode switch in STARTUP/HOT STBY and SRM C Bypassed

D. Mode switch in RUN and SRM C Selector switch in Standby

Proposed Answer: B

Explanation
A. and D. The rod withdrawal block is bypassed when the mode switch in run
B. Correct- If IRMs are not on Range 8 or above in Mode 2, an SRM UPSC or INOP initiates a rod withdrawal block .
C. The rod withdrawal block is bypassed when the affected SRM is bypassed

Technical Reference(s): ARP-P680-05-C05

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503, Obj 5,7

Question Source: Modified Bank # RBS-NRC-53

Question History: Last NRC Exam NA (original question was on the 1997 NRC exam)

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

10 CFR Part 55 Content: 55.41.b.6

Comments: Rejected KA: There is no AOP or EOP entry conditions having to do with SRMs.
The only G2.4 category applicable to SRMs is G2.4.31

QUESTION 39 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	215005 K1.09		IR 3.6

Knowledge of the physical connections and/or cause-effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: Reactor recirculation system: BWR-5,6

Proposed Question:

Core flow rate is provided to the APRMs for use by the flow control trip reference cards to establish control rod withdrawal blocks and scram trip setpoints, this flow rate is obtained from which of the following sources?

A. jet pumps 5, 10, 15, and 20

B. the above and below core plate differential pressure

C. the elbow taps on both recirculation loops

D. jet pumps 1, 6, 11, and 16

Proposed Answer: C

Explanation
A. Incorrect - These are the calibrated jet pumps; used for driven flow and fuel zone level indications
B. Incorrect - the above and below core plate d/p is used for core flow indication but not for APRMs
C. Correct- A d/p signal is provided to the APRMs from elbow taps on both recirculation loops
D. Incorrect - These jet pumps are used for driven flow indication only

Technical Reference(s): R-STM-0503, Neutron Monitoring, Rev 7 p. 60 of 112

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503, Obj 27

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.2

Comments: edited distractors A & D for credibility

QUESTION 40 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	217000 K2.01		IR 2.8

Knowledge of electrical power supplies to the following: Motor operated valves

Proposed Question:

The normal power supply to E51-F063, RCIC STEAM SUPPLY INBD ISOL VALVE is ____.

A. EHS-MCC2L

B. EHS-MCC2D

C. ENB-MCC1

D. BYS-SWG01B

Proposed Answer: B

Explanation
A. Incorrect - EHS-MCC2L is an alternate source of power to E51-F063 utilized for AOP-0031
B. Correct- EHS-MCC2D is the normal power source to E51-F063
C. Incorrect - DC supply ENB-MCC1 is the source of power to many E51 MOVs but NOT to E51-F063
D. Incorrect - DC supply BYS-SWG01B is the source of power to the RCIC gland seal compressor NOT to E51-F063

Technical Reference(s): R-STM-0209, RCIC, Rev 10 p. 42 of 52

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209, Obj 13

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.8

Comments:

QUESTION 41 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	218000 K3.02		IR 4.5

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

Proposed Question:

The plant is operating in the Emergency Operating Procedures following a significant transient. During the transient, a short resulted in the loss of ENB-PNL02A.

Plant conditions require Emergency Depressurization per the Emergency Operating Procedures.

Which of the following represents the method that should be used to accomplish Emergency Depressurization?

A. At H13-P601, manually open 7 ADS/SRVs.

B. At H13-P631, manually open 7 ADS/SRVs.

C. Arm and depress the Division 1 ADS Manual Initiate pushbuttons.

D. Alternate depressurization methods listed in the EOPs should be utilized due to SRV failure.

Proposed Answer: B.

Explanation:
A. Loss of power to ENB-PNL02A prevents use of Div 1 SRV solenoids; the switches on P601 are Div 1.
B. Correct – Div 2 solenoids are still available to open the SRVs.
C. Div 1 solenoids are de-energized.
D. This would only be required if SRVs could not be opened. Div 2 solenoids are still available.

Technical Reference(s): R-STM-0202, ADS, Rev 2 pp. 20,21 of 36
R-STM-0109, Main Steam, Rev 13 p. 12 of 95

Proposed references to be provided to applicants during examination: None

Learning Objective: R-STM-0202 Obj. 6,12

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41b.3

Comments: Edited question choices A and B for clarity (Rev 1)

QUESTION 42 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	218000 A4.02		IR 4.2

Ability to manually operate and/or monitor in the control room: ADS logic initiation

Proposed Question:

The plant has scrammed due to a loss of offsite power.
Neither HPCS nor RCIC will start.
Approximately 5 minutes later, RPV water level decreases below -143 inches, the “DIV 2 ADS LOGIC TIMER INITIATED” annunciator illuminates.
The operator inhibits ADS.
Later the Operator ARMS and Depresses the ADS B MANUAL INITIATION pushbuttons.
What is the response of the ADS System in this situation?

ADS will initiate:

A. immediately, if any Div 2 low pressure ECCS subsystem pressure permissive is satisfied.

B. in 105 seconds, if any Div 2 low pressure subsystem pressure permissive is satisfied.

C. immediately, regardless of low pressure ECCS subsystem status.

D. In 105 seconds, regardless of low pressure ECCS subsystem status.

Proposed Answer: A

Explanation
A. Correct- ADS Logic only requires subsystem pressure permissive prior to manual initiation
B. Incorrect – there is no time delay for the logic in this condition
C. Incorrect – the only logic requirement for this condition is ECCS pressure/running
D. Incorrect – there is no time delay for the logic in this condition

Technical Reference(s): R-STM-0202, ADS, Rev 2 p.38 of 41 (Figure 2)
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0202, Obj 7
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.41.b.8
Comments: Edited stem to clarify command versus action of operator (Rev 1)

QUESTION 43 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 223002 K4.01	IR 3.0	

Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Redundancy

Proposed Question:

During performance of STP-058-4201, Containment and Drywell Manual Isolation Actuation LSFT, the I&C technician has the reactor operator arm and depress the “A” CRVICS pushbutton on H13-P680 and then continues with his verifications per the STP. Prior to the isolation signal being reset, the “D” CRVICS pushbutton is armed and depressed.

Which of the following is the status of the MSIVs and Containment Isolation Valves?

A. Only the Main Steam Isolation Valves (MSIVs) are CLOSED.

B. The MSIVs <u>AND</u> the Outboard Containment Isolation Valves are CLOSED.

C. Only the Inboard Containment Isolation Valves are CLOSED.

D. Only the Outboard Containment Isolation Valves are CLOSED.

Proposed Answer: A

Explanation

A. Correct – The design of the CRVICS system is such that the MSL and MSL drain isolation signals seal in but the BOP isolations do not, therefore pushing the pushbuttons sequentially will only cause the MSIVs to close.

B. Incorrect, The design of the CRVICS system is such that the MSL and MSL drain isolation signals seal in but the BOP isolations do not, therefore pushing the pushbuttons sequentially will only cause the MSIVs to close. Plausible if the candidate misunderstands the system design in that they believe that the BOP isolations are seal in.

C. Incorrect, The design of the CRVICS system is such that the MSL and MSL drain isolation signals seal in but the BOP isolations do not, therefore pushing the pushbuttons sequentially will only cause the MSIVs to close. Plausible if the candidate misunderstands the system design in that they believe that the BOP isolations are seal in, additionally the A and D channels are associated with the outboard BOP isolations not the inboard BOP isolations

D. Incorrect, The design of the CRVICS system is such that the MSL and MSL drain isolation signals seal in but the BOP isolations do not, therefore pushing the pushbuttons sequentially will only cause the MSIVs to close. Plausible if the candidate misunderstands the system design in that they believe that the BOP isolations are seal in.

Technical Reference(s): R-STM-0058, CRVICS, Rev 9 p. 39 of 63

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0058, Obj 11

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.3

Comments: Replaced the question for KA mismatch (Rev 1)

QUESTION 44 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	239002 K5.04		IR 3.3

Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES : Tail pipe temperature monitoring

Proposed Question:

The plant is operating at 100% power when the following annunciator comes in: P601-19A-B09, Safety Relief Valve Leaking annunciates.

Which one of the following corresponding values is used to confirm an actual leaking SRV?

A. 225°F tailpipe temperature on temperature monitoring panel P614

B. 250°F tailpipe temperature on temperature monitoring panel P614

C. 3 LED's are lit on acoustic monitoring panel P953

D. 4 LED's are lit on acoustic monitoring panel P953

Proposed Answer: B

Explanation

B. Correct - A tail pipe temperature of 250ºF causes an alarm to come in on this panel; 250F would be indicative of leakage. In accordance with the ARP, tailpipe temperature is used to confirm SRV leaking.

C and D These are plausible because there are 10 LEDs per SRV that are used to determine if an SRV is OPEN; 5 out of 10 LEDs must be lit to bring in the alarm/give confirmation of an open SRV.

Technical Reference(s): ARP-601-19A-B09,
R-STM-0109, Main Steam Rev 13 p. 14 of 95

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109, Obj 26

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.14

Comments: Replaced question to not test GFES knowledge (Rev 1)

QUESTION 45 Rev 0

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 following will have on the REACTOR WATER LEVEL CONTROL SYSTEM : Main steam flow input

Proposed Question:

While operating at 100% power, the 'C' Main Steam Line flow transmitter fails low. What is the expected response of the Feedwater level control system with no operator actions taken?

A. RPV level will lower until it reaches Level 3 and the reactor scrams.

B. RPV level will lower and stabilize at a new lower level above Level 3.

C. RPV level will rise and stabilize at a new higher level below Level 8.

D. RPV level will rise until it reaches Level 8 and the reactor scrams.

Proposed Answer: B

Explanation
A. Only first part is correct; see B
B. Correct – A loss of 1 out of 4 steam flow transmitters results in a sensed reduction in steam flow of 25%; this sensed reduction will cause the FWLC system to close the Feed Reg Valves and consequently RPV level will lower. Level will stabilize below its normal level (but above the Level 3 scram setpoint) because the FWLC system is level dominant.
C. This is the expected response for a single steam line transmitter failing high
D. This is the expected response for a single feed flow transmitter failing low

Technical Reference(s): R-STM-0107, Feedwater, Rev 27 p. 71 of 105

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107, Obj B14

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.4

Comments:

QUESTION 46 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	261000 A1.01		IR 2.9

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY GAS TREATMENT SYSTEM controls including: System flow

Proposed Question:

A leak has occurred in the Drywell and Drywell pressure is 2.14 psid and stable. Both trains of Standby Gas Treatment (GTS) started and consequently OSP-0053 Hard Card Attachment 21, Operating Auxiliary Building Ventilation was used to reduce to one train of SBGT running; GTS-A is running. Subsequently, a loss of RSS#1 occurred and the Div 1 Diesel Generator failed to start.

What is the status of GTS-B following the loss of RSS#1?

A. GTS-B may only be started in High Volume Purge mode due to power failure.

B. GTS-B must be manually initiated due to manually securing.

C. GTS-B will restart automatically due to low flow in GTS-A.

D. GTS-B will restart automatically due to undervoltage trip of GTS-A.

Proposed Answer: C

Explanation
A. Incorrect - Power is only lost to Div 1; Div 2 is still available
B. Incorrect - Securing per the hard card places GTS-B in standby;
C. Correct - DW 1.68 is still locked in, so when low flow occurs in Div 1 due to the loss of power, then the Div 2 will automatically restart.
D. Incorrect - The interlock that will start GTS-B is low flow in the running GTS; not undervoltage.

Technical Reference(s): R-STM-0257, Standby Gas Treatment, Rev 5 p. 15 of 28

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0257, Obj 5,12

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments:

QUESTION 47 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	262001 A2.11		IR 3.2

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Degraded system voltages

Proposed Question:

The plant is operating at 100% power when alarm P808-86A, Grid Trouble is received. SPI-REC102 on P808 indicates 230 KV system voltage at Fancy Point to be 223.3 KV. The SOC contacts the control room and indicates that a transient has occurred and Fancy Point is <u>NOT</u> sufficiently stable at this time.

(1) What is the impact to plant equipment?
(2) What actions should be taken to mitigate this event?

A. (1) Amps will increase on the running equipment.
(2) Perform a normal start of the D/G, parallel it to offsite power and disconnect the bus from the grid.

B. (1) Amps will increase on the running equipment.
(2) Perform an emergency start of the D/G and open the safety related bus supply breaker.

C. (1) Amps will decrease on the running equipment.
(2) Perform an emergency start of the D/G and open the safety related bus supply breaker.

D. (1) Amps will decrease on the running equipment.
(2) Perform a normal start of the D/G, parallel it to offsite power and disconnect the bus from the grid.

Proposed Answer: B

Explanation

Plant indications show that Fancy Point is NOT stable(second note on page 64 of AOP-64)

A. Would be correct if Fancy Point was stable

B. Correct per steps:5.5.3.1 and 5.5.3.4

C. and D. Running equipment amps will rise as voltage lowers. Normal voltage is 230KV the stem gives 223.3KV

Technical Reference(s): AOP-0064, Degraded Grid, Rev 6 p 8 of 11

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-AOP064, Obj 3

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.5

Comments: Edited stem for clarity of report from the SOC and removal of the synchroscope reference. Also removed handout (Rev 1)

QUESTION 48 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 262002 A3.01	IR 2.8	

Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source

Proposed Question:

BYS-INV01A displays the following information:

- Rectifier Output 0 amps
- Battery Output 135 VDC
- Inverter Output 122 VAC

What is the status of the loads normally supplied by this inverter?

A. Loads are currently de-energized

B. Loads are currently being supplied by the battery via the inverter

C. Loads are currently being supplied by the normal AC source via the inverter

D. Loads have automatically swapped to the alternate AC source

Proposed Answer: B

Explanation

A., Incorrect - The applicant may misunderstand the construction of the UPS and choose A because the stem shows 0 amps rectifier output The battery output does have power making A wrong.

B. Correct – The normal AC source is unavailable based on the rectifier output; output of the battery and inverter reveal that the battery is supplying power to the inverter and the inverter is powering the loads

C. Incorrect - Rectifier output reveals that the normal AC power source is de-energized

D. Incorrect - Alternate AC power source is provided through a Manual Bypass Switch, which, according to the given conditions has not been turned Applicant may not assume any actions not given in stem.

Technical Reference(s): R-STM-0300, AC Electrical Distr., Rev 26 pp. 28-29 of 105

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0300, Obj 12

Question Source: Bank # RBS Nov 2008 Audit Q#49

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Edited setm to remove possible cueing with the word “normal” (Rev 1)

QUESTION 49 Rev 2

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	263000 A4.01		IR 3.3

Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses: Plant- Specific

Proposed Question:

The white light on the E12-PC002A, RHR Pump "A" Control Switch is lit.
What does this indicate?

A. An initiation signal is present.

B. The pump has been overridden.

C. A Shutdown Cooling isolation signal is present.

D. The control power fuses are installed.

Proposed Answer: D

Explanation
A. This is plausible because there is a white system initiation light located on another part of the panel.
B. This is plausible because pump override is indicated with an annunciator for this pump; the applicant could confuse this with the HPCS which uses a light to indicate override.
C This is plausible because there is a white light for a SDC isolation on another part of the same panel.
D. Correct -- The white light indicates that control power is available.

Technical Reference(s):

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 ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.7

Comments: Edited the stem for plausibility of distractors (Rev 1) ; Replaced question after validation due to difficulty level being to great (Rev 2)

QUESTION 50 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 264000 K1.05		IR 3.2

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

Proposed Question:

The correct fuel oil supply flowpath for operation of the Div 1 Standby Emergency Diesel Generator is from its associated storage tank, in sequential order, through the _____, to the injectors.

A. fuel oil transfer pump, strainer, duplex filter, fuel oil day tank

B. strainer, fuel oil transfer pump, fuel oil day tank, duplex filters

C. strainer, fuel oil transfer pump, fuel oil day tank, booster pump

D. fuel oil transfer pump, strainer, fuel oil day tank, booster pump

Proposed Answer: D

Explanation
A. the filter is located after the day tank
B. the strainer is after the transfer pump
C. the strainer is after the transfer pump
D. Correct – The full flowpath of fuel oil is storage tank, transfer pump, day tank, duplex strainer, booster pump, duplex filter, injector pump, injectors

Technical Reference(s): R-STM-0309S, Rev 14, pp. 17 & 20 of 117

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0309S, Obj 2

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.8

Comments: Edited stem and distractors for grammar (Rev 1)

QUESTION 51 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	264000 G2.1.31		IR 4.6

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

Proposed Question:

The Div 2 standby diesel generator is loaded and in parallel with bus ENS-SWG1B.

A LOCA signal occurs.

Which of the following describes the effect on the standby diesel generator and bus ENS-SWG1B?

A. The normal bus supply breaker will open and the diesel generator will supply bus loads. Voltage and frequency will be adjusted from H13-P877

B. The normal bus supply breaker and diesel generator output breaker will open, then after loads are shed, the diesel generator output breaker will reclose. Voltage and frequency will be adjusted from EGS-PNL4A

C. The diesel generator output breaker will open and cannot be closed if bus voltage is supplied by the normal feed. Voltage and frequency will be adjusted from H13-P877

D. The diesel generator output breaker will remain closed in parallel operation with the bus. Voltage and frequency will be adjusted from EGS-PNL4A

Proposed Answer: C

Explanation
A, B & D All are contrary to the P&L noted in C.
C. Correct – The DG output breaker will open and wait for a low voltage signal on the emergency bus.(SOP-53 P&L 2.15) The SOP gives guidance for use of controls from H13-P877 (SOP-53 p43)

Technical Reference(s): R-STM-0309S, Rev 13 pp. 43-44 of 117
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0309S, Obj 8
Question Source: Bank # RBS-NRC-199
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.41.b.7
Comments: Replaced question due to KA mismatch (Rev 1)

QUESTION 52 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 1	
	K/A # 300000 K2.01	IR 2.8	

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

Proposed Question:

The Instrument air compressors A, B, and C are powered from ___, ___, and ___ respectively.

A. NJS-SWG 1G, 1H, and 1F

B. NJS-SWG 1F, 1G, and 1H

C. NJS-SWG 1E, 1J, and 1K

D. NJS-SWG 1J, 1E, and 1K

Proposed Answer: A

Explanation
A. Correct- these are the power supplies to the IAS compressors
B. Incorrect - these are the power supplies to the Service Air compressors
C. Incorrect -these are the power supplies to the IAS trim coolers
D. Incorrect -these are the power supplies to the SAS trim coolers

Technical Reference(s): R-STM-0121, Rev 16 p. 9 of 69

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0121, Obj 10

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.4

Comments: rejected KA K2.02; re-selected K2.01 – RBS has diesel operated emergency air compressor
Edited distractors for credibility (Rev 1)

QUESTION 53 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	1
	K/A #	400000 K3.01		IR 2.9

Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS

Proposed Question:

The Division 2 isolation valves in the Reactor Plant Component Cooling Water System (CCP) have closed on a 56 psig isolation signal.

What equipment is still receiving cooling water from the CCP system?

A. Recirc pumps, RWCU NRHXs, RWCU pumps

B. RWCU pumps, RHR Pump A Seal Coolers, RWCU NRHXs

C. Recirc pumps, RHR Pump A Seal Coolers, Fuel Pool Coolers

D. Fuel Pool Coolers, RWCU pumps, RWCU NRHXs

Proposed Answer: A

Explanation

A. Correct – The Safety loop isolates which secures cooling to the RHR Pump Seal Coolers, Fuel pool coolers, and CRD pumps. Cooling is not isolated to the three loads listed.

B. RHR Pump Seal Coolers (both divisions) are isolated as part of the safety loop

C. and D are incorrect because Fuel Pool Coolers (both divisions) are isolated as part of the safety loop

Technical Reference(s): R-STM-0115, Rev 6 pp. 12-14 of 35 and Figure 1

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0115, Obj 4,8,11

Question Source: Bank # RBS-OPS-1789

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Edited distractors for credibility (Rev 1)

QUESTION 54 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	201003 A4.02		IR 3.5

Ability to manually operate and/or monitor in the control room: CRD mechanism position: Plant-Specific

Proposed Question:

The ATC operator just selected a control rod to move in accordance with an approved reactivity maneuver plan when the Control Rod Withdrawal Block Alarm annunciates.

The RC & IS Operator Control Module (OCM) also displays the following:

- DATA FAULT pushbutton is backlit
- CH DATA 1 / CH DATA 2 are consecutively lighting and extinguishing
- CHANNEL DISAGREE is backlit

The rod position displayed on the full core display for the selected rod is flashing alternately between 08 and - - .

What operator action would allow this rod to be moved?

A. Depressing the POSITION BYPASSED pushbutton on the OCM

B. Depressing the DATA MODE pushbutton on the OCM

C. Depressing the DATA SOURCE pushbutton on the OCM

D. Depressing the ENTER SUBST pushbutton on the OCM

Proposed Answer: D

Explanation
A. This button will not bypass a control rod display; it will display rods that are already bypassed
B. This pushbutton works in conjunction with the Data Source pushbutton to allow you to choose either single channel or alternating channel(s) to display on the full core display; this would make the alternating rod position indication to stop, but not clear the rod block.
C. This pushbutton allows you to choose which of the the 2 channels to display on the full core display; pushing this button would show first one channel then pushing it again would show the other.
D. Correct – This pushbutton forces the single channel of bad data to be replaced with the other channels good data; thereby allowing rod movement.

Technical Reference(s): R-STM-0500, Rev 3 pp. 19-25 of 46; SOP-0071, Rev 29, Section 5.12
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0500, Obj 7,9
Question Source: New Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.41.b.6
Comments: Edited distractor A for credibility (Rev 1)

QUESTION 55 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	201005 A1.01		IR 3.2

Ability to predict and/or monitor changes in parameters associated with operating the ROD CONTROL AND INFORMATION SYSTEM (RCIS) controls including: First stage shell pressure/turbine load: BWR-6

Proposed Question:

Given the following plant conditions:

- Reactor power 45%
- Generator load 480 MWe

Power ascension is in progress. The next step of the Reactivity Maneuvering Plan is to select and continuously withdraw control rod 28-49 from position 12 to position 24.

Just prior to withdrawing the rod, the Main Turbine First Stage Shell Pressure transmitter output signal fails upscale.

When the withdraw button is pushed, control rod 28-49 will ___.

A. remain at position 12

B. withdraw to position 16 and settle

C. withdraw to position 20 and settle

D. withdraw to position 24 and settle

Proposed Answer: B

Explanation

A. If turbine first stage shell pressure was indicating failed low, the Rod Pattern Controller logic would be enabled and could cause a rod withdraw block (depending on the current rod pattern)

B. Correct- the rod withdrawal limitations are dependent on reactor power as sensed by First Stage Shell Pressure. An upscale failure would indicate reactor power above the high power setpoint to RC&IS. The Rod Withdrawal Limiter will then limit rod withdrawals to 2 notches (12-16).

C. and D are incorrect because the RWL limits rod motion to 2 notches

Technical Reference(s): R-STM-0500, Rev 3 p.16 of 46

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0500, Obj 22

Question Source: Bank # RBS-NRC-665

Question History: Last NRC Exam Dec 2008

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

10 CFR Part 55 Content: 55.41.b.6

Comments:

QUESTION 56 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑ SRO ☐
	Tier #	2	Group # 2
	K/A #	202001 K4.02	IR 3.1

Knowledge of RECIRCULATION System design feature(s) and/or interlocks which provide for the following: Adequate recirculation pump NPSH

Proposed Question:

When total Feedwater flow drops below the Reactor Recirc system interlock level, the Reactor Recirc Pumps downshift from fast speed to slow speed. This interlock ___.

A. prevents flow velocity effects on the wide range level indication.

B. prevents thermal stress on the Recirc pump

C. prevents cavitation in the Recirc pumps

D. adds negative reactivity in anticipation of a reactor scram

Proposed Answer: C

Explanation
A. the interlock associated with wide range flow velocity effects is the low reactor level interlock
B. the interlock associated with thermal stress is the Steam Dome to Vessel Bottom Head delta T
C. Correct – Cavitation is prevented by inhibiting high speed operation of pumps with feed flow < 19.9%
D. the EOC-RPT interlock anticipates a reactor scram based on a turbine trip causing a pressure transient, which will rapidly add positive reactivity.

Technical Reference(s): R-STM-0053, Rev 13 p. 27 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0053, Obj 17

Question Source: Modified Bank # RBS-NRC-168

Question History: Last NRC Exam 1995

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

10 CFR Part 55 Content: 55.41.b.2

Comments: Edited distractor D for clarity (Rev 1)

QUESTION 57 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	202002 K1.09		IR 3.1

Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION FLOW CONTROL SYSTEM and the following: Reactor water level

Proposed Question:

The plant is operating at 100% power. The "A" narrow range level channel is selected as the input to the Feedwater Level Control System. A leak has developed in the reference leg of the "A" narrow range level transmitter.

The ATC operator promptly placed the Master Feedwater Level Controller in MANUAL. As a result of this condition, both Recirc pumps will ___.

A. remain at their present speed, and the Recirc Flow Control Valves will runback to 60% drive flow position.

B. transfer to SLOW speed operation, and the Recirc Flow Control Valves will remain at their present position.

C. transfer to SLOW speed operation, and the Recirc Flow Control Valves will runback to 60% drive flow position.

D. remain at their present speed, and the Recirc Flow Control Valves will remain at their present position.

Proposed Answer: D

Explanation
A. first part is correct, but the reference leg leak will cause a false high level indication; FCV's will not runback due to a high level
B. and C. The reference leg leak will cause a false high level indication, Recirc pumps will not downshift due to a high RPV level.
D. Correct – Recirc Pumps and FCV's logic receive level signal from the narrow range selected (A); the reference leg leak will cause transmitter differential pressure to rise, therefore the Recirc system will not receive a low RPV level (Recirc pumps downshift at level 3 and FCVs runback at level 4)

Technical Reference(s): R-STM-0107B, Rev 27 pp. 58-59

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0107B, Obj 10

Question Source: Bank # Dec 2008 NRC Exam Q#57

Question History: Last NRC Exam Dec 2008

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

10 CFR Part 55 Content: 55.41.b.2

Comments:

QUESTION 58 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	216000 K6.01		IR 3.1

Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: A.C. electrical distribution

Proposed Question:

A loss of RPS Bus A causes which RPV level indication on the H13-P680 panel to fail downscale?

A. B21-R604, Wide Range meter

B. C33-R606A, Narrow Range Channel A meter

C. B21-R615, Fuel Zone recorder

D. C33-R608R, Upset Range recorder

Proposed Answer: A

Explanation
A. Correct – power supply to the wide range meter comes from RPS –A and fails downscale
B. power supplied by inverter 1VBN-PNL01B1
C. The fuel zone recorder is powered by inverter 1VBN-PNL01B1 and it is not located on the 680 panel
D. The narrow range/upset range recorder has a single power supply (1VBN-PNL01B1)

Technical Reference(s): R-STM-0051, Rev 5 p. 23 of 47

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0051, Obj 13; RLP-STM-0508, Obj 7

Question Source: Bank # RBS-NRC-808

Question History: Last NRC Exam Feb 2003

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Replaced Distractor C for credibility (Rev 1)

QUESTION 59 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 219000 K2.02		IR 3.1

Knowledge of electrical power supplies to the following: Pumps (RHR-Supp Pool Cooling)

Proposed Question:

The electrical power supply to RHR Pump C is__(1)__.
The electrical power supply to its' associated line fill pump is __(2)__.

A. (1) ENS-SWG1B ; (2) ENS-SWG2B

B. (1) ENS-SWG1A ; (2) EJS-SWG2A

C. (1) ENS-SWG1B ; (2) EJS-SWG1B

D. (1) ENS-SWG1A ; (2) ENS-SWG1A

Proposed Answer: C

Explanation:
C. Correct – RHR pumps B & C power supply is ENS-SWG1B; the line fill pump for Div 2 is EJS-SWG1B. The distractors are plausible because the power supplies are all associated with ECCS systems.

Technical Reference(s): R-STM-0204, p. 25 of 63 ; EE-001AC

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0204, Obj 11

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.8

Comments: Edited stem and distractors for awkwardness and credibility (Rev 1)

QUESTION 60 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 223001 G2.1.28	IR 4.1	

Knowledge of the purpose and function of major system components and controls. Primary Containment Systems

Proposed Question:

The function of the ___ System is to control hydrogen concentration that would be generated during a Design Basis Loss of Coolant Accident. If hydrogen is detected in the drywell or containment, continuous operation of this system alone is sufficient to limit the hydrogen concentration to levels below combustible regimes that could result in containment damage.

A. Containment Hydrogen Purge

B. Hydrogen Igniter

C. Hydrogen Mixing

D. Hydrogen Recombiner

Proposed Answer: B

Explanation

A. Incorrect- This system designed as a backup to the H_2 Recombiners; there is no EOP/SAP guidance for its use

B. Correct – The H_2 igniters are designed to handle 75% of the Metal-Water Reaction rate; they are designed to mitigate the consequences of a generation event more severe than a design basis LOCA. The Igniter system bounds the DBA scenario used in sizing the H_2 Mixing and H_2 Recombiner Systems.

C. Incorrect - Primary function is to mix containment and drywell atmospheres thereby temporarily diluting the $H_{2,\ \text{and}}$ together with the Recombiner system are bounded by the Igniter system alone, therefore by itself is not correct.

D. Recombiners are started 14 days after a DBA and together with the Mixing system are bounded by the Igniter system alone, therefore by itself is not correct.

Technical Reference(s): EOP Bases, Rev 16, p. B-8-30, 31

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0057, Obj 20

Question Source: Modified Bank # RBS-OPS-06288

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.9

Comments: Edited stem for cueing (Rev 1)

QUESTION 61 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 239001 K3.15		IR 3.5

Knowledge of the effect that a loss or malfunction of the MAIN AND REHEAT STEAM SYSTEM will have on following: Reactor water level control

Proposed Question:

The plant is operating at 70% reactor power when one inboard MSIV fails closed.

RPV level will ___.

A. increase and stabilize at a higher level.

B. decrease and stabilize at a lower level.

C. decrease and then return to normal level.

D. increase and then return to normal level.

Proposed Answer: C

Explanation

A. RPV level indication measures water level in the downcomer region and due to steam flow across the dryers, level is approximately 7 inches higher in the downcomer than the core. When an MSIV closes, pressure will go up momentarily and concurrently collapse some of the steam void in the core. Water levels equalize and then FWLC will compensate and get level back to the the setpoint on the tape set

B. First part is correct, but RPV level is being controlled by FWLC system and will return to whatever the tape set is dialed to (setpoint setdown is not activated)

C. Correct – RPV level initially drops due to a pressure transient causing voids to collapse; water from downcomer flows into the core and then FWLC compensates and level returns to normal.

D. See A

Technical Reference(s): RPPT-HLO-0316, Slide 46 of 72

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109, Obj 2, 19; RLP-HLO-0316, Obj 1, 2

Question Source: Bank # RBS-NRC-308

Question History: Last NRC Exam June 1995

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

10 CFR Part 55 Content: 55.41.b.5

Comments:

QUESTION 62 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	241000 A2.01		IR 3.5

Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of turbine inlet pressure signal

Proposed Question:

The plant is raising power with control rods with reactor power at 15% and rising. EHC Pressure Regulator Channel A is selected for pressure control. Pressure Regulator Channel B is in TEST.

Pressure Regulator Channel A fails to 0 psig, the plant response is _______ and the actions to mitigate the condition are:

A. (1) Only Turbine control valves and bypass valves fail shut
   (2) Implement Scram actions

B. (1) EHC fault protection circuit overrides TEST on Channel B to stabilize pressure
   (2) Implement ARP actions for failed pressure regulator

C. (1) Only Turbine control valves fail shut
   (2) Control Pressure on bypass valves and drains

D. (1) Pressure regulator swaps to B regulator when preset deviation is reached due to reactor power increase
   (2) Implement ARP actions for failed pressure regulator

Proposed Answer: A

Explanation
A. The pressure regulator sensing a low pressure, will close (not open) the TCVs trying to raise pressure.
B. TCVs will close causing pressure to rise, but the BPVs use the same pressure transmitter and therefore would remain closed (not open).
C. Correct - Turbine Control Valves will close and the BPVs use the same pressure transmitter and therefore would also remain closed
D. The pressure regulator sensing a low pressure, will close (not open) the TCVs trying to raise pressure

Technical Reference(s): R-STM-0509, Rev 14 p.51 of 81
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0509, Obj 10
Question Source: Modified Bank # RBS-LOR-1252
Question History: Last NRC Exam NA
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 2
10 CFR Part 55 Content: 55.41.b.5
Comments: Replaced question due to KA match incomplete (Rev 1)

QUESTION 63 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 2	Group # 2	
	K/A # 272000 K4.02		IR 3.7

Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for the following: Automatic actions to contain the radioactive release in the event that the predetermined release rates are exceeded

Proposed Question:

The plant is operating at 100% power. Annulus Pressure Control system is in operation with HVR-FN16A, Annulus Pressure Control (APC) Fan A running.

RMS-RE11A, Div 1 Annulus Exhaust Radiation Monitor goes into High Alarm (reading greater than 3.89E-5 µci/cc). Which of the following describes the ventilation lineup after this event?

A. HVR-FN16A trips; Both Standby Gas Treatment Trains are running

B. HVR-FN16A stays running; Both Standby Gas Treatment Trains are running

C. HVR-FN16A stays running; ONLY Standby Gas Treatment Train A is running

D. HVR-FN16A trips; ONLY Standby Gas Treatment Train A is running

Proposed Answer: A

Explanation
A. Correct - RMS-RE11A sends a signal to trip FN16A and start GTS-FNA; GTS-FNB will start due to a low APC System flow
B. RMS-RE11A sends a signal to trip FN16A
C. RMS-RE11A sends a signal to trip FN16A
D. Part 1 is correct; Part 2 is wrong because GTS-FNB will start due to a low APC System flow

Technical Reference(s): ARP-DSPL230-1GP011, Rev 9
R-STM-0511, Radiation Monitoring, Rev 15 p. 46 of 48

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0511, Obj 4,6

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.11

Comments:

QUESTION 64 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group #	2
	K/A #	288000 A3.01		IR 3.8

Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: Isolation/initiation signals

Proposed Question:

Containment Unit Coolers 1A and 1B are running.
Which of the following signals will cause (1) chilled water to the Containment Unit Coolers to isolate, and (2) Service Water to the Containment Unit Coolers to align?

A. (1) High Negative D/P between Containment and Annulus (-10") ;
(2) RPV level 1 (-143") after the 60 sec time delay

B. (1) High Drywell D/P (1.68 psid) ;
(2) RPV level 1 (-143") after 60 sec time delay

C. (1) High Negative D/P between Containment and Annulus (-10") ;
(2) RPV level 1 (-143") with no time delay

D. (1) High Drywell D/P (1.68 psid) ;
(2) RPV level 1 (-143") with no time delay

Proposed Answer: B

Explanation
A. Part 1 is correct ; SW aligns on RPV level 1 after 60 time delay
B. Correct – chill water isolates on RPV level2, DW D/P, and -12"; Service Water does not align until level 1 is reached and then after a time delay of 60 seconds to allow separation between CW and SW.
C. Part 1 is incorrect – Level 2 signal isolates CW;
D. Part 1 is correct ; SW aligns on RPV level 1 after 60 time delay

Technical Reference(s): R-STM-0403, Reactor Building HVAC, Rev 8 pp. 7-11 of 50

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0403, Obj 6

Question Source: Modified Bank # RBS-OPS-2307

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.7

Comments: Edited distractors for credibility (Rev 1)

QUESTION 65 Rev 1

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:

An un-isolable steam leak has occurred in the main steam tunnel. The CRS has directed you to manually initiate the control room Charcoal Filter Trains per AOP-0003, Automatic Isolations.

Which of the following represents the status of the Control Building HVAC system?

	HVC-MOD7A/B CR Remote Air Intake	HVC-AOD19C/D/E/F HVC Local Air Intake
A.	OPEN	OPEN
B.	OPEN	CLOSE
C.	CLOSE	OPEN
D.	CLOSE	CLOSE

Proposed Answer: C

Explanation
A. It is plausible for the applicant to misunderstand the operation of manual initiation to realign the the remote intake which is located on the other side of the plant from the local intake.
B and D. It is plausible for the applicant to misunderstand the operation of manual initiation to isolate the local air intake dampers making these plausible.
C. Correct - A high-high rad detected by RMS-RE13 will cause the following: HVC-MOV1 to close, HVC-AOD19s to open, and HVC-FN1 to start, Making C correct.

Technical Reference(s): ARP-P863-74A-H03 and –H08, Rev 24

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0402, Obj 7

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.11

Comments: Edited question for LOD and credibility (Rev 1)

QUESTION 66 Rev 0

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.1.7		IR 4.4

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

Proposed Question:

OSP-0053, Emergency and Transient Response Support Procedure, contains Post-Scram Level Control Strategies.

In non-ATWS conditions, when MSIVs are closed, the prescribed level band is _(1)_.

In an ATWS condition, with Reactor power at 30% and the Turbine on-line, the prescribed level band is _(2)_.

A. (1) 10 to 51 inches; (2) -60 to -140 inches

B. (1) -20 to 51 inches; (2) -100 to -140 inches

C. (1) 10 to 51 inches; (2) -100 to -140 inches

D. (1) -20 to 51 inches; (2) -60 to -140 inches

Proposed Answer: D

Explanation
A. Incorrect - Part one would only be correct if MSIVs were open
B. Part 1 is correct, part two would be correct if SRVs were required for pressure control
C. Incorrect - Part one would only be correct if MSIVs were open
D. Correct – The prescribed level band for a non-ATWS condition is normally 10 to 51", except when certain plant conditions described in the EOP bases are present. One such condition is closure of MSIVs where the expanded band is prescribed. In an ATWS when conditions are not present that would challenge containment (SRVs are not required for pressure control), then the level control band should remain -60 to -100 inches.

Technical Reference(s): OSP-0053, Rev 22 pp. 16-19 of 74
EOP Bases pp. B-6-12 & -13; B-7-21 & -22

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-OPS-0512, Obj 5

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 67 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.1.31		IR 4.6

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

Proposed Question:

The main turbine has just tripped while operating at 40% power.

The operator will see ___.

A. all 8 Scram Pilot Solenoid Valve indicating lights out on H13-P691.

B. the Turbine Bypass Valves indicating open on H13-P680.

C. NPS-SWG1A aligned to the station transformers on H13-P808.

D. the Main generator output breakers tripped and the exciter field breaker closed on H13-P680.

Proposed Answer: B.

Explanation:
A. The 8 scram pilot lights are on H13-P680. P691 contains the RPS trip unit that input to the trip logic.
B. Correct - BPV indications are on H13-P680 and they would be open as a result of the turbine trip.
C. NPS-SWG1A is normally aligned to station transformers; after the trip it re-aligns to the preferred.
D. Exciter field breaker would also be tripped for this transient.

Technical Reference(s): AOP-0001, Reactor Scram, Rev 28 p. 4 of 10;
AOP-0002, Turbine Trip, Rev 26 p. 6 of 10

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0310 Obj. 8

Question Source: Bank # RBS-NRC-01183

Question History: Last NRC Exam RBS Dec. 2010 Q# 67

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

10 CFR Part 55 Content: 55.41b.4

Comments: Edited Distractor C for credibility (Rev 1)

QUESTION 68 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 3		
	K/A # G2.1.40		IR 2.8

Knowledge of refueling administrative requirements

Proposed Question:

During a refueling outage, maintenance personnel require access to the annulus fuel transfer tube area.

In addition to operation of the palm handswitch outside the gate, which of the following is also required to allow access to this area?

A. Keylock switch outside the fuel building support room gate

B. Keylock switch on IFTS Operator Panel F42-P001 in the reactor building

C. Keylock switch outside the containment transfer tube support room

D. Keylock switch on IFTS Master Relay Panel F42-P003 in the fuel building

Proposed Answer: D

Explanation:
A. Incorrect-because This switch would be required if the fuel building fuel transfer tube room required access.
B. Incorrect – because the IFTS keylock switch is on the Fuel Bldg IFTS panel, not reactor bldg.
C. Incorrect because this switch would be required if the containment fuel transfer tube room required access.
D. Correct - To access the annulus fuel transfer area requires: 1) H13-P863 keylock switch, 2) Switch outside specific gate requiring entry in ACCESS/OPEN, 3) Switch on IFTS Master Relay panel in the Fuel Bldg, 4) Local palm switch depressed.

Technical Reference(s): R-STM-0055 page 19, rev 9.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0055 Obj 4, 6

Question Source: Modified from April 2010 NRC Exam Q72

Question History: Last NRC Exam RBS Apr. 2010 Q# 72

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Replaced question with modified bank due to cueing (Rev 1)

QUESTION 69 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier # 3		
	K/A # G2.2.6		IR 3.0

Knowledge of the process for making changes to procedures

Proposed Question:

In accordance with RBNP-001, DEVELOPMENT AND CONTROL OF RBS PROCEDURES, a Comment PAR, may be used for which of the following?

A. To make minor changes that result in a change of intent.

B. To correct typographical errors to a procedure that is being implemented.

C. To make suggestions for future procedure improvements.

D. To change acceptance criteria.

Proposed Answer: C

Explanation
A. Incorrect – A procedure “Revision” is required for changes that result in a change of intent.
B. Incorrect –This requires an Editorial change.
C. Correct - This choice describes a “Comment”.
D. Incorrect – A change in acceptance criteria also results in a change of intent;,would require a “Revision”

Technical Reference(s): RBNP-001, Development and Control of Procedures, Rev 35 p. 19 of 43

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0202, Obj 1

Question Source: Bank # RBS-OPS-1667

Question History: Last NRC Exam RBS April 2010

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

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 70 Rev 1

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.2.14	IR 3.9

Knowledge of the process for controlling equipment configuration or status

Proposed Question:

A decon technician has called the work management center requesting permission to open an MWS supply valve to obtain water for deconning purposes. What action is required?

A. The WMC SRO must authorize the manipulation and logging is not required as long as verification of the valve position is made before the end of shift.

B. The OSM/CRS must authorize the manipulation and it must be logged in the control room log or manipulated device log book.

C. The WMC SRO must authorize the manipulation and it must be logged in the control room log or manipulated device log book

D. The OSM/CRS must authorize the manipulation and logging is not required as long as verification of the valve position is made before the end of shift.

Proposed Answer: B

Explanation
B. Correct OSP-0014 requires OSM/CRS approval and logging for configuration control purposes when manipulating plant components outside of procedures; distractors are plausible because there could be a misunderstanding that any SRO could authorize (not just the OSM/CRS) and, years ago, it was acceptable practice to not require logging manipulations but to instead perform a verification before the end of shift.

Technical Reference(s): OSP-0014, Administrative Control of Equipment, Rev 304 p. 6 of 16

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0216, Obj 2, 5

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Edited distractors for credibility (Rev 1)

QUESTION 71 Rev 2

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.2.35		IR 3.6

Ability to determine Technical Specification Mode of Operation

Proposed Question:

The following plant conditions exist:

- Reactor Coolant temperature is 115˚F
- The mode switch position is in Refuel
- All head bolts have been tensioned
- No control rod testing is in progress

What mode of operation is the plant in?

A. Mode 2 Startup

B. Mode 3 Hot Shutdown

C. Mode 4 Cold Shutdown

D. Mode 5 Refueling

Proposed Answer: A

Explanation
A. With switch in refuel and all head bolts tensioned it is Mode 2 Startup
B. Incorrect, with all head bolts tensioned but mode sw not in shutdown this is incorrect
C. Incorrect, with all head bolts tensioned but mode sw not in shutdown this is incorrect
D. Incorrect, with all head bolts tensioned this is incorrect

Technical Reference(s): Tech Spec Definitions Table 1.1-1

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 ☑ 2

10 CFR Part 55 Content: 55.41.b.10

Comments: Replaced question due to credibility (Rev 1); Added 4th bullet per validation remarks (Rev 2)

QUESTION 72 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.3.11	IR 3.8

Ability to control radiation releases

Proposed Question:

Due to a steam leak, the Main Steam Tunnel area temperatures have caused automatic isolations to occur as designed. Because of the location of the leak, no LOCA signals have been generated by RPV level or DW pressure. An ALERT has been declared due to offsite release rate.

Which one of the following will reduce the UNMONITORED release rate?

A. Shutdown the Turbine Building Ventilation System, if operating.

B. Shutdown the Fuel Building Ventilation System, if operating.

C. Start the Turbine Building Ventilation System, if not operating.

D. Start the Fuel Building Charcoal Ventilation System, if not operating.

Proposed Answer: C

Explanation
A. This action would raise the unmonitored release rate
B. The fuel building normal and charcoal filtration systems are both monitored; this would not affect unmonitored release
C. Correct – IAW EOP-0003
D. See B

Technical Reference(s): EOP-0003 p. B-10-3

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-515, Obj 6

Question Source: Bank # RBS-NRC-797

Question History: Last NRC Exam RBS 2003 NRC

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

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 73 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.3.13	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:

For which of the following evolutions is the licensed operator in the control room procedurally required to notify Radiation Protection prior to performance?

A. Suppression Pool reject to Radwaste with Residual Heat Removal (RHR)

B. Placing Heater Drain pumps in the PUMP FORWARD mode

C. Startup of Circulating Water Blowdown

D. Reactor Core Isolation Cooling (RCIC) slow roll startup

Proposed Answer: D

Explanation:
A. See D. (SOP-0031)
B. See D. (SOP-0010)
C. See D. (SOP-0006)
D. Although notifying RP prior to any of the 4 choices demonstrate good teamwork, only RCIC slow roll startup specifically requires notification per SOP-0035.

Technical Reference(s): SOP-0035, RCIC, Rev 47 p. 16 of 76

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0209 Obj 10c

Question Source: Bank # RBS Oct 2012 Audit Q#72

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments:

QUESTION 74 Rev 1

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	3		
	K/A #	G2.4.12		IR 4.0

Knowledge of general operating crew responsibilities during emergency operations

Proposed Question:

River Bend is in a station blackout and is implementing AOP-50, Station Blackout.

- RCIC has failed
- HPCS has failed

You are dispatched to implement Attachment 2, Injection Into RPV With Fire Water System because core damage can occur as early as:

A. 30 minutes

B. 39 minutes

C. 49 minutes

D. 60 minutes

Proposed Answer: C

Explanation
A. Incorrect. although many TCA are at 30 minutes so this time could be confused with the answer of 49 minutes
B. Incorrect, close to 49 minutes
C. Correct, per AOP-50, Attachment 2, page 1.
D. Incorrect, although there are TCAs at 60 minutes

Technical Reference(s): AOP-50, ATT 2, page 1, rev 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 ☑ 2 Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.41.b.10

Comments: Replaced question due to credibility (Rev 1)

QUESTION 75 Rev 0

Examination Outline Cross-Reference:	Level	RO ☑	SRO ☐
	Tier #	3	
	K/A #	G2.4.26	IR 3.1

Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage

Proposed Question:

According to EN-OP-115, Conduct of Operations, a Fire Brigade of at least __(1)__ members shall be maintained on site at all times; the fire brigade members shall NOT include the __(2)__.

A. (1) Four ; (2) Shift Technical Advisor (STA)

B. (1) Five ; (2) Duty Manager

C. (1) Five ; (2) Shift Technical Advisor (STA)

D. (1) Four ; (2) Duty Manager

Proposed Answer: C

Explanation
A. EN-OP-115 requires a minimum of 5 Fire Brigade Members
B. Part 1 is correct; The Duty Manager is not designated as being precluded by procedure
C. Correct – The fire brigade shall not include the OSM, CRS, STA, ATC, and 1 NEO
D. EN-OP-115 requires a minimum of 5 Fire Brigade Members

Technical Reference(s): EN-OP-115, Conduct of Operations, Rev 15 p. 78 of 89

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-HLO-0206, Obj 7

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41.b.10

Comments: Similar to Bank Question #RBS-OPS-3382

QUESTION 76 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295004 G2.2.44		IR 4.4

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions (Partial or Complete Loss of DC)

Proposed Question:

During normal plant operations with BYS-TRS4, XFR SW FOR SBO POWER CONNECTION is out of service for repairs, the following alarms are received on H13-P808 Insert 87, including:

- 125VDC BAT CHGR ENB-CHGR1B TROUBLE
- DIV II 125VDC CHGR BRKR ENB-ACB580- OPEN

Based on these indications and in order to assess operability of the associated system(s), which of the following conditions should the CRS enter?

A. LCO 3.8.4 Condition A, One required battery charger on Division I or II inoperable

B. LCO 3.8.4 Condition B, Division I or II DC electrical power subsystem inoperable for reasons other than Condition A

C. LCO 3.8.9 Condition C, One or more Division I or II DC electrical power distribution subsystems inoperable

D. No LCO entry required because the backup battery charger is available.

Proposed Answer: A.

Explanation:

A., Correct T.S. 3.8.4 Condition A basis requires the Non safety related Backup charger and SBO Diesel Generator to be available and with BYS-TRS4 out of service the SBO Diesel Generator is NOT available to supply the back up charger, Condition "A" is applicable.

B. Incorrect because this LCO is for conditions other than Condition A which is not applicable due to the SBO diesel generator not being available.

C. Incorrect LCO 3.8.9 is for DC distribution system during normal operations this distractor is plausible because the candidate may decide the opening of the breaker renders the DC distribution system inoperable which it does not because the system is still capable of supply battery power to the respective loads.

D. Incorrect LCO 3.8.4 Condition B is applicable as described above This distractor is plausible because the student may recall technical specification bases that allow the backup battery charger to be utilized to supply the DC bus,, but the charger is only allowed to be utilized to extend the LCO time (T.S. 3.8.4 Condition A vice Condition B).

Technical Reference(s): TS 3.8.4, RLP-STM-0305

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0305 Obj 7

Question Source: New Question History: NA

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

10 CFR Part 55 Content: 55.43.b.2

Comments:

QUESTION 77 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295006 G2.4.34		IR 4.2

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (SCRAM)

Proposed Question:

The plant has experienced an ATWS. The following conditions exist at the 680 panel:

- The Mode Switch is in Shutdown
- ARI has been initiated
- Approximately 50% of the withdrawn rods fully inserted
- All eight white scram solenoid lights are extinguished
- The SDV Vent and Drain Valve position green lights are on and red lights are off
- Annunciator 680-05A-C08, SCRAM Pilot Valve Air Header Low Pressure <u>NOT</u> lit

The CRS is using EOP Enclosure 26, Control Rod Insertion Method Determination. Based on the flow chart from this enclosure, which procedure should the CRS direct next to attempt to get the remaining control rods fully inserted?

A. EOP Enclosure 14, Drive Control Rods

B. EOP Enclosure 11, Venting SCRAM Air Header

C. EOP Enclosure 13, Opening Individual SCRAM Test Switches

D. EOP Enclosure 17, Venting CRD Over Piston Volumes

Proposed Answer: B

A. The stem conditions are indicative the failure of the scram air header to vent; scram solenoids are already de-energized.
B. Correct – the flow chart requires venting the scram air header next based on given stem conditions, which is contained in EOP Enclosure 11.
C. and D. These two enclosures are used <u>after</u> Enclosure 11 has been tried and therefore are incorrect.

Technical Reference(s): EOP-0001A RPV Control - ATWS;
EOP-0005 Enclosure 26, Control Rod Insertion Method Determination

Proposed references to be provided to applicants during examination: None.

Learning Objective: RLP-HLO-0513, Obj 5

Question Source: Modified Bank# RBS-OPS-06265

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43.b.5

Comments: Original KA was G2.4.2, re-rolled (drew G2.4.34) ; The original KA was not conducive to writing an SRO level question.

QUESTION 78 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295021 AA2.04		IR 3.6

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

Proposed Question:

The plant is Shutdown in Mode 4 following 400 days of continuous operation when a spurious Group 3 isolation occurs and the logic cannot be reset, which causes a loss of shutdown cooling.

As reactor water temperature rises, what is the minimum value of reactor water temperature that requires an immediate notification (within 1 hour) to the NRC based on NOUE criteria being reached?

A. 201˚F

B. 204˚F

C. 208˚F

D. 210˚F

Proposed Answer: A

Explanation

A Correct – per the EPIP for NOUE, 200 degrees F is the minimum value for immediate notification to the NRC

B 204 degrees is not correct

C. 208 degrees F is not correct

D. 210 degrees F is not correct

Technical Reference(s): EIP-2-001, Classification of Emergencies, Rev 24, Attach 10
EIP-2-002, Classification Actions, Rev 31, p. 7 of 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 ☑ 2
10 CFR Part 55 Content: 55.43.b.5
Comments:

QUESTION 79 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 1	Group # 1	
	K/A # 295028 G2.4.21	IR 4.6	

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

Proposed Question:

While the CRS is executing the Alternate Level Control Leg of EOP-1, a large break LOCA occurs and now ALL RPV level instruments indicate off-scale low.

Five seconds later, the Fuel Zone Level instruments are back on scale.
The current conditions are:

- Containment temperature 91°F (at EL 119 ft)
- Drywell temperature 285°F (at EL 145 ft)
- RPV Pressure 10 psig
- Fuel Zone Level indication -290 inches, rising and falling rapidly

Based on these conditions, Fuel Zone Level indication ____ be used to determine RPV level and the CRS should direct ____.

A. CANNOT ; Emergency Depressurizing per EOP-1

B. CANNOT ; exiting EOP-1 and entering EOP-4

C. CAN ; restoring and maintaining RPV level above -162 inches per EOP-1

D. CAN ; exiting the ALC leg and entering the Steam cooling leg of EOP-1

Proposed Answer: B

Explanation

A. Incorrect per override RC-2 of EOP-1

B Correct per override RC-2 EOP-4 must be entered

C, D - Incorrect, FZ level indication is unreliable due to boiling in the instrument line run

Technical Reference(s): EOP-1, RPV Control, Rev 26

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: 55.43.b.5

Comments:

QUESTION 80 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	295031 EA2.04		IR 4.8

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Adequate core cooling

Proposed Question:

The following conditions exist:
- Reactor Pressure is 943 psig
- RHR pumps A and C are lined up for injection
- HPCS injection has commenced
- Reactor water level is -190” and slowly lowering

Based on these conditions, the CRS should now transition to:

A. The Steam Cooling flow path.

B. The Spray Cooling flow path.

C. The Emergency Depressurization flow path.

D. EOP-4, RPV Flooding.

Proposed Answer: C

Explanation
A. -200 is the water level for steam cooling without injection
B. -211 is the water level for spray cooling, at 943 psig HPCS will not develop 5000 gpm
C. Correct - adequate core cooling does not exist, ED is necessary to allow low pressure ECCS to inject
D. RPV flooding is not required since RPV level can be determined.

Technical Reference(s): EOP Bases, pp. B-6-29,30

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: 55.43.b.5

Comments: Revised question to include procedural choice to make this SRO level

QUESTION 81 Rev 3

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	600000 AA2.16		IR 3.5

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

Proposed Question:

The control room has received alarms on the Fire Control Console for Zone 30 and 96 (RHR C & RCIC Pump Rooms), a fire brigade member has been dispatched and has reported that upon opening the door to RHR C he was confronted with thick black smoke. Based on the location of the fire what plant equipment do you expect to have available to safely shut down the reactor?

A. Only Division II equipment

B. Only Division III equipment

C. Only Division I and II equipment

D. Only Division I and III equipment

Proposed Answer: D

Explanation
A. Incorrect, AOP-0052 shows that Division 2 is not available

B. Incorrect AOP-0052 shows that both Division I and 3 are available

C. Incorrect AOP-0052 shows that Division I is available but division 2 is not available

D. Correct AOP-052 Shows that Division I and Division III are available.

Technical Reference(s): AOP-0052, Fire Outside the MCR, Rev 25 Attachment 1,
SOP-0036 Fire Detection Supervisory System, Rev 305

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 ☑ 2

10 CFR Part 55 Content: 55.43.b.5

Comments: Re-ordered distractors A & B; changed distractor B for credibility. (Rev 3)

QUESTION 82 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	1
	K/A #	700000 G2.2.37		IR 4.6

Ability to determine operability and/or availability of safety related equipment (Generator Voltage and Electric Grid Disturbances)

Proposed Question:

The plant is operating at rated power.

The system dispatcher reports that there is a grid disturbance and the switchyard voltage has lowered to 223 kV. River Bend Station is currently 100% power and is receiving offsite power from the switchyard.

Based on the given conditions what is the status of River Bends offsite AC sources and what if any technical specifications are required to be entered?

A. INOPERABLE and T.S. 3.8.1. condition D

B. INOPERABLE and T.S. 3.8.9. condition D

C. INOPERABLE and T.S. 3.0.3

D. OPERABLE and no T.S. entry

Proposed Answer: A

Explanation
A Correct – T.S. 3.8.1 Condition D is applicable for two required offsite circuits inoperable
B Incorrect - T.S.3.8.9 is applicable to the AC Distribution System and NOT to AC sources
C Incorrect -T.S. 3.0.3 is only applicable if three or more required AC source are inoperable.
D Incorrect - With switch yard voltage less than 224.25. KV both of the offsite circuits are inoperable per AOP-0064

Technical Reference(s): AOP-0064, Degraded Grid, Rev 6
ENS-DC-199, Offsite Power Supply Design Requirements, Attach 9.3
ITS 3.8
USFAR Chapter 8

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 ☐

10 CFR Part 55 Content: 55.43.b.2

Comments:

QUESTION 83 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	2
	K/A #	295015 AA2.02		IR 4.2

Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Control rod position

Proposed Question:

Following a high drywell differential pressure scram signal, the following conditions exist:

- 142 control rods indicate Full-In
- 3 control rods are not full in and are at various positions
- Reactor power indicates 0%
- SRMs are inserted and count rates are lowering

What are the procedures used for this event, in the order used, by the CRS?

A. Only EOP-1, RPV Control

B. Only EOP-1, RPV Control and EOP-1A, RPV Control, ATWS

C. Only EOP-1A, RPV Control-ATWS

D. Only EOP-1, RPV Control and EOP-2, Primary Containment Control

Proposed Answer: B

Explanation

A. Incorrect-Even though power is at 0% and SRM count rates lowering, with 3 rods out you must know that reactor will not be shutdown under all conditions without boron, and therefore a transition to EOP-1A is required..

B. Correct - With more than one rod out, it can not be determined that the reactor will remain shutdown under all conditions without boron; A high drywell differential pressure scram signal is an EOP-0001 entry condition, transition to EOP-1A is required due to override step RC-2 in EOP-1.

C. Incorrect-if someone does not remember that you can’t start in EOP-1A

Technical Reference(s): EOP Bases p. B-6-5

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 ☑ 2

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 84 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 1	Group # 2	
	K/A # 295029 G2.2.25		IR 4.2

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (High Suppression Pool Wtr Lvl)

Proposed Question:

Technical Specifications (TS) require suppression pool water level to be $\leq$ 20 ft 0 inches. What is the TS basis for this upper limit?

A. Above 20 ft 0 inches, excessive pressure could build up in the drywell before the drywell horizontal vents are cleared during a DBA LOCA.

B. Water levels above 20 ft 0 inches would cause low pressure ECCS systems to operate outside of their design NPSH values.

C. Water levels above 20 ft 0 inches would overflow the drywell weir wall and cause inaccurate drywell leakage indications due to excessive pedestal sump discharge flow.

D. Above 20 ft 0 inches, SRV tail pipe pressure could build up too high during SRV operation and damage the tail pipes.

Proposed Answer: D

Explanation

A. A higher DW pressure buildup would occur as suppression pool level rises but this is within design analysis and "excessive" pressure buildup would not occur. This condition is not mentioned in the bases.
B. Levels above 20 ft would raise NPSH of the low pressure ECCS pumps but would not cause them to operate outside of design parameters.
C. The weir wall would overflow at 21 ft 3 inches and cause the pedestal sump to cycle but this is not mentioned as a basis for the upper limit
D. Correct - TS 3.6.2.2 Bases Statement; If the suppression pool water level is too high, it could result in excessive clearing loads from S/RV discharges".

Technical Reference(s): Tech Spec 3.6.2.2 and Bases

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: 55.43.b.2

Comments:

QUESTION 85 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	1	Group #	2
	K/A #	500000 EA2.03		IR 3.8

Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for drywell

Proposed Question:

The plant has experienced a LOCA. The following conditions exist:

- RPV pressure is 450 psig
- RPV level is +10 inches and slowly rising
- Containment pressure is 1.5 psig
- Drywell temperature is 141°F
- Drywell pressure is 1.7 psid
- Drywell hydrogen concentration is 1.9 percent
- Containment hydrogen concentration is 1.8 percent

Based on these indications, the SRO should ________.

A. Enter EOP-2 and implement Enclosure 20 to operate all available drywell cooling, defeating interlocks as necessary

B. Enter EOP-2 and implement Enclosure 25 to initiate normal containment vent and purge, defeating interlocks as necessary

C. Enter EOP-2 and implement Enclosure 31 to operate all hydrogen igniters

D. Enter EOP-2 and implement Enclosure 31 to operate the hydrogen mixing system

Proposed Answer: C

Explanation
With hydrogen concentration above 1.5%, all igniters should be operated
A – Incorrect. Defeating interlocks and operating all available drywell cooling is not required until 145 degrees. Plausible if applicant believes 140 degrees is the requirement
B – Incorrect. Initiating normal containment vent and purge is not directed by EOP-2. Plausible if applicant believes that containment venting is required for hydrogen or CTMT pressure.
C. Correct. With hydrogen concentration above 1.5%, all igniters should be operated per enclosure 31
D – Incorrect. Hydrogen mixing system operation is not directed by EOP-2. Plausible since it is included as one of the hydrogen control methods outlined by enclosure 31.

Technical Reference(s): EOP-Bases, pp. B-8-26,27
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 ☑ 2
10 CFR Part 55 Content: 55.43.b.2
Comments:

QUESTION 86 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	1
	K/A #	205000 A2.03		IR 3.2

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

Proposed Question:

The plant is shutdown, in Mode 4 with RHR Pump A in SDC mode and no Recirc Pumps running.

A reactor operator is in the process of completing a surveillance test on RHR B. When attempting to secure RHR Pump B, the operator erroneously stops RHR Pump A instead and it cannot be restarted. The operator attempts to close F004B, RHR Pump B suction from suppression pool; however power is lost to the valve before it is fully closed and attempts to manually close F004B have failed.

RHR B ________ and the CRS should _________.

A. Is NOT available for shutdown cooling; refer to OSP-0041, Alternate Decay Heat Removal for determining method of alternate decay heat removal

B. Is available for shutdown cooling; perform STP-050-0700, RCS Pressure/Temperature Limits Verification

C. Is NOT available for shutdown cooling; implement AOP-0051, Loss of Decay Heat Removal and raise reactor water level to 70 inches as indicated on H13-P601, B21-R605, RX WTR LEVEL SHUTDOWN RANGE

D. Is available for shutdown cooling; initiate actions to set Primary Containment Operability in accordance with OSP-0034, Control of Obstructions for Primary Containment/Fuel Building Operability

Proposed Answer: A

Explanation

A. Correct. F006B cannot be opened with F004B not fully closed – therefore RHR B is not available. Alternate DHR is required.
B. Incorrect. RHR B is not available. Plausible if applicant does not know that F006B cannot be opened. An RCS pressure/temp limit verification is required on a loss of SDC.
C. Incorrect. Reactor water level must be raised to greater than 75 inches on a loss of forced circulation. Plausible because RHR is not available and water level must be raised per AOP-0051.
D. Incorrect. RHR B is not available. Plausible if applicant does not know that F006B cannot be opened. Establishing containment operability is required on a loss of SDC.

Technical Reference(s): AOP-0051, Loss of Decay Heat Removal, Rev 312

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 ☑ 2	
10 CFR Part 55 Content:	55.43.b.2	Comments:	

QUESTION 87 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	1
	K/A #	212000 G2.2.12		IR 4.1

Knowledge of surveillance procedures (RPS)

Proposed Question:

The Shift Manager has directed you to perform surveillance testing for RPS instrument channels in accordance with SR 3.3.1.1. When the I and C technician puts the channel in the test position for performance of the surveillance, entry into associated Conditions and Required Actions may be delayed for:

A. Up to four hours provided the associated function maintains RPS trip capability

B. Up to four hours provided LCO 3.04b is not applicable

C. Up to six hours provided LCO 3.04b is not applicable

D. Up to six hours provided the associated function maintains RPS trip capability

Proposed Answer: D

Explanation

A. Incorrect., six hours is correct time

B. Incorrect, six hours is the correct time

C. Incorrect, the second part is incorrect

D. Correct-TS SR3.3.1.1 has two notes at the beginning of it. Note 2 states that "When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability."

Technical Reference(s): TS 3.3.1.1, page 3.3-3, amendment 114.
Proposed references to be provided to applicants during examination: None
Learning Objective: RLP-STM-0508
Question Source: New
Question History: N/A
Cognitive Level: Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☑ 3
10 CFR Part 55 Content: 55.43.b.5
Comments:

QUESTION 88 Rev 1

Examination Outline Cross-Reference: Level RO ☐ SRO ☑
Tier # 2 Group # 1
K/A # 215005 A2.02 IR 3.7

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips

Proposed Question:

GOP-0001 Plant Startup requires verification that all APRMs are >5% prior to taking the mode switch to RUN. This step was signed and then a shift turnover occurred.

The mode switch is taken to run. APRM A has several LPRMs that fail downscale which results in only two operable axial LPRMs and so the CRS directs APRM A to be bypassed. Once APRM A is bypassed, APRM C fails upscale.

All other APRMs are operable and are reading >5%.

These conditions will require the CRS to direct ___ and was bypassing APRM A correct?

A. Actions according to EOP-0001, RPV Control; Yes

B. Actions according to GOP-0001, Plant Startup; No

C. Actions according to GOP-0001, Plant Startup; Yes

D. Actions according to EOP-0001, RPV Control; No

Proposed Answer: B

Explanation

A. Incorrect, with 1APRM channel in bypass {A} and one failed downscale (C} the TS minimum of 3 per division is not met however an automatic reactor trip does not occur so EOP-0001 would not be entered. Also the TS admin limit for axial LPRMs per channel is two therefore with two still operable the APRM is still operable and the answer should be No-this was not correct to bypass channel A.

B. Correct - with 1APRM channel in bypass {A} and one failed downscale (C} the TS minimum of 3 per division is not met however an automatic reactor trip does not occur so EOP-0001 would not be entered. However, because Tech Specs allow bypassing only one channel from each trip system, there are not enough channels left to stay at power so a plant shutdown would be required if the APRM could not be fixed. Also the TS admin limit for axial LPRMs per channel is two therefore with two still operable the APRM is still operable and the answer should be No-this was not correct to bypass channel A..

C. First Part is correct, but second part is wrong.

D. First part is wrong, second part is correct.

Technical Reference(s): SOP-0074, Neutron Monitoring, Rev 306, TS amendment 114, and R-STM-0503, RPS, Rev 7,

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0503, Obj 26,28, 38

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 89 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	1
	K/A #	239002 A2.06		IR 4.3

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

Proposed Question:

During a plant startup with reactor power at 25%, a failure occurred in both Main Turbine pressure regulator channels causing the outputs to fail low. During the Scram **BOTH** RPS MG sets trip.

What procedure(S) is/are applicable for the above condition?

A. Only AOP-0001, Reactor Scram

B. Only AOP-0001, Reactor Scram and EOP-0002, Containment Control

C. Only AOP-0001, Reactor Scram, AOP-0003, Automatic Isolations, EOP-0001, RPV Control, and EOP-0002, Containment Control

D. Only AOP-0001 Reactor Scram, AOP-0003, Automatic Isolations, and EOP-0002 Containment Control

Proposed Answer: C

Explanation

A. Incorrect – AOP-0001 is applicable but not the only procedure applicable in this condition, this is credible because has the reactor tripped due to the turbine tripping only this would have been the only procedure in play.

B. Incorrect – EOP-0001 and EOP-0002 would be entered but AOP-0003 and AOP-0001 are also applicable.

C. Correct – the reactor will scram due to the failure of the pressure regulator channels causing the turbine control valves to close raising reactor pressure to greater than the 1097.4 psig set point EOP-0001 entry conditions are met at this point due to the reactor pressure greater than 1097.4 psig, EOP-0002 entry conditions are met due to the SRV's being the only method of pressure control at this point as the RPS MG sets have tripped and this will close the MSIV's. AOP-0003 is also met as far as entry conditions due to the MSIV isolation.

D. Incorrect – The entry conditions for EOP-001 would be met due to high RPV pressure > 1097.4 psig.

Technical Reference(s): R-STM-0109, Main Steam, Rev 13 pp. 36,37,60 of 95
EOP-1, RPV Control, EOP-0002, AOP-0003

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0109, Obj 22

Question Source: New Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 90 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	1
	K/A #	262001 G2.2.25		IR 4.2

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits (AC Electrical Distribution)

Proposed Question:

Consider the following two separate conditions for the AC distribution system (consider all other equipment not mentioned in the condition to be operable):

- Condition 1: Both qualified offsite power sources have been lost
- Condition 2: One qualified offsite power source has been lost and one Emergency Diesel Generator is out of service

Per the Technical Specification Basis, Condition 1 is considered _________ severe than Condition 2, because __________.

A. less; the configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure

B. more; the configuration of the redundant AC electrical power system that remains available is susceptible to a single bus or switching failure

C. less; qualified offsite power sources are considered less reliable than onsite AC sources

D. more; qualified offsite power sources are considered more reliable than onsite AC sources

Proposed Answer: A

Explanation

a. Correct. TS 3.8.1 bases for actions D.1 and D.2 states, "Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this degradation level: a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching"

b. Incorrect. Condition 1 is less severe according to the basis. Plausible if applicant believes that Condition 2 is more severe due to condition 2 having two diverse sources of AC power.

c. Incorrect. Offsite power is not determined in the basis to be more or less reliable than EDGs. Plausible if applicant determines that EDGs are more reliable than offsite power.

d. Incorrect. Condition 1 is less severe according to the basis. Offsite power is not determined in the basis to be more or less reliable than EDGs. Plausible if applicant determines that offsite power is a more reliable source than EDGs.

Technical Reference(s): Tech Spec Bases 3.8.1 p. B 3.8-10

Proposed references to be provided to applicants during examination: TS LCO 3.8.1

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: 55.43.b.2

Comments:

QUESTION 91 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	2	Group #	2
	K/A #	201005 G2.1.23		IR 4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation (RCIS)

Proposed Question:

A reactor startup is in progress. The reactor operator attempts to withdraw control rod 20-29 from position 42 to position 48. Once control rod 20-29 reaches position 48, a data fault occurs. The reactor operator determines the data fault is associated with control rod 20-29.

The data fault could be caused by __________ and the CRS should ____________.

A. More than one reed switch being closed on one channel; enter AOP-0061, Section 5.7, for control rod movement difficulty and then transition to SOP-0071 to attempt to insert control rod 20-29 one step

B. More than one reed switch being closed on one channel; enter SOP-0071 and direct the reactor operator to enter substitute data for control rod position per section 5.12, Entering Substitute Data for Control Rod Position

C. No reed switches being closed on one channel; enter AOP-0061, Section 5.7, for control rod movement difficulty and then transition to SOP-0071 to attempt to insert control rod 20-29 one step

D. No reed switches being closed on one channel; enter SOP-0071 and direct the reactor operator to enter substitute data for control rod position per section 5.12, Entering Substitute Data for Control Rod Position

Proposed Answer: D

Explanation

A. Incorrect. More than one reed switch being closed gives a data fault only if the rod is not at the full-out or full-in positions. CRS should enter SOP-0071 and enter substitute data from the other channel. Plausible if applicant does not realize that two reed switches being closed does not give a data fault with the rod full out and believes that rod must be returned to a position with operable position indication.
B. Incorrect. More than one reed switches being closed gives a data fault only if the rod is not at the full-out or full-in positions. Plausible if applicant does not realize that two reed switches being closed does not give a data fault with the rod full out.
C. Incorrect. CRS should enter SOP-0071 and enter substitute data from the other channel. Plausible if applicant believes that rod must be returned to a position with operable position indication.
D. Correct. No reed position switches being closed on one channel will give a data fault at any rod position. CRS should enter SOP-0071 and enter substitute data from the other channel.

Technical Reference(s): SOP-0071, Step 5.12
AOP-0061, Mispositioned Control Rod, Rev 7

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 ☑ 2
10 CFR Part 55 Content:	55.43.b.5
Comments:	

QUESTION 92 Rev 1

Examination Outline Cross-Reference:	Level	RO ☐	SRO ☑
	Tier # 2	Group # 2	
	K/A # 204000 A2.14		IR 3.2

Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System High Temperature

Proposed Question:

The plant is operating at normal power when the CRS discovers a common cause failure has resulted in both Leakage Detection System temperature instruments (E31-N620A and E31-N620B) for the RWCU heat exchanger room being declared inoperable.

Based on this condition, TRM 3.3.6.1 allows the CRS ______ hour(s) to ________.

A. 1; isolate the RWCU system

B. 1; initiate action to be in Mode 2 within 7 hours

C. 12; place the affected channels in trip

D. 24; place the affected channels in trip

Proposed Answer: D

Explanation

A. The TRM allows 1 hour to isolate RWCU (Condition F) if the affected channels are not placed in trip within 24 hours (or 25 hours total).

B. This is correct if LCO 3.0.3 is entered. However, LCO 3.0.3 is not applicable in this case since the TRM Condition A allows for one or more affected channels.

C. Twelve hours is not correct since it does not apply to function 4.c (RWCU Heat Exchanger Room).

D. Correct – Condition A allows up to 24 hours to trip the affected channels for function 4.c.

Technical Reference(s): TRM 3.3.6.1; R-STM-0601, RWCU, Rev 8, R-STM-0207, LDS, Rev 6

Proposed references to be provided to applicants during examination: TRM Specification 3.3.6.1

Learning Objective: RLP-STM-0601, Obj 7

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43.b.5

Comments: *Original KA was A2.02, re-rolled (drew 2.14). RBS does not have Low Press RWCU.*

QUESTION 93 Rev 1

Examination Outline Cross-Reference:

Level		RO ☐	SRO ☑
Tier #	2	Group #	2
K/A #	290001 A2.04		IR 3.7

Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High airborne radiation

Proposed Question:

Following a plant transient involving an unisolable leak in the RCIC room, the CRS has entered EOP-3, Secondary Containment and Radioactive Release Control.
Current plant conditions are:

Parameter	Current Value	Max Normal	Max Safe Value
RCIC Equip Area Temp	218ºF	144ºF	200ºF
RCIC Equip Rm Rad Lvl	147 mr/hr	1.20E+02 mr/hr	9.5 E+03 mr/hr
RCIC Rm Water Level	2 in	32 1/8 in	4 in above floor
RHR Equip Rm C Rad Lvl	23 mr/hr	8.20E+01 mr/hr	9.5 E+03 mr/hr
RHR C Rm Water Lvl	1 in	32 1/8 in	4 in. above floor

Based on these indications, the CRS should transition to:

A. Emergency Depressurization in order to preferentially depressurize the reactor using the suppression pool instead of secondary containment.

B. EOP-1, RPV Control in order to reduce the leak rate.

C. Emergency Depressurization in order to reduce the leak rate.

D. EOP-1, RPV Control in order to preferentially depressurize the reactor using the suppression pool instead of secondary containment.

Proposed Answer: B

Explanation

A. An emergency depressurization is not required because no parameters have exceeded their max safe value in more than one area. An emergency depressurization is only required if any parameter exceeds its max safe value in 2 or more areas.

B. Correct – The area temperature has exceeded the Max Normal level requiring a reactor scram via EOP-1. The reactor scram will reduce the leak rate resulting in lowering the temperature and radiation levels in the secondary containment thereby enhancing personnel access, equipment reliability, and lowering risk to the general public.

C. An emergency depressurization is not required because no parameters have exceeded their max safe value in more than one area. An emergency depressurization is only required if any parameter exceeds its max safe value in 2 or more areas.

D. The area temperature has exceeded the Max Normal level requiring a reactor scram via EOP-1. Preferentially depressurizing to the suppression pool would be the reason for an emergency depressurization after exceeding the max safe value in two areas.

Technical Reference(s): EOP BASES, page B-9-12, 13

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: 55.43.b.5

Comments:

QUESTION 94 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.1.20		IR 4.6

Ability to interpret and execute procedure steps

Proposed Question:

The plant is operating at 100% power when an ATWS occurs. Reactor Water Level is lowered.

Subsequently the following conditions exist:

- Hot Shutdown Boron Weight injection is complete.
- HPCS is the only available injection system.
- HPCS is restoring RPV level when reactor Power rises to 5% and continues to rise.

What EOP Mitigating Strategy is implemented and why is it chosen?

A. Transition to the main flow path destination point 7 (Level Reduction Required) to re-perform terminate and prevent injection because Cold Shutdown Boron Weight has NOT been injected.

B. Transition to the main flow path destination point 7 (Level Reduction Required) to re-perform terminate and prevent injection because the amount of boron required to shut down the reactor has NOT reached the core.

C. Continue steps for injection at a reduced rate because Cold Shutdown Boron Weight has NOT been injected.

D. Continue steps for injection at a reduced rate because the amount of boron required to shut down the reactor has NOT reached the core.

Answer: B

b. Transition to the main flow path to re-perform stop and prevent injection because the amount of boron required to shut down the reactor has NOT reached the core.

Explanation (Optional):

EOP-1A, RPV CONTROL, ATWS - Step RLA-12 directs restoration of RPV water level to the normal range after having injected sufficient boron to shut down the reactor. If reactor power commences and continues to increase as RPV water level is raised, the amount of boron required to shut down the reactor has not reached the core. Returning to Steps RLA-11 and RLA-13 under these conditions will again require that RPV water level be lowered to prevent flux oscillations and reduce reactor power while additional actions to shut down the reactor proceed.

As injection into the RPV is initially increased to raise RPV water level, a small transient increase in reactor power is expected as natural circulation core flow is re-established. This power increase will be reversed after several seconds as boron is mixed and carried from the lower plenum up into the core region. The wording in the above step, "*commences and continues to increase*," has been specifically chosen to denote only a sustained increase in reactor power indicative of insufficient boron in the core.

Distracters:

a. Stopping and preventing injection helps lower reactor power but Cold Shutdown Boron weight is not the reason this must be done. EOPs wait for Cold Shutdown Boron weight to be injected before reactor depressurization is begun. The candidate that does not recall the bases for cold shutdown boron weight may choose this answer. This answer is plausible because cold shutdown boron weight is an action point in EOPs and stopping and preventing injection is an accurate response for a reactor power rise that is sustained.

c. Reducing the injection rate would be an acceptable response to the reactor power rise but EOPs require stop and prevent because power continues to rise. Injection is not lowered because of Cold shutdown boron weight. EOPs wait for Cold Shutdown Boron weight to be injected before reactor depressurization is begun. The candidate that does not recall the bases for cold shutdown boron weight may choose this answer. This answer is plausible because cold shutdown boron weight is an action point in EOPs and lowering injection of cold water would help lower reactor power.

d. Reducing the injection rate would be an acceptable response to the reactor power rise but not a power rise that continues. EOPs require terminate and prevent in this instance. The candidate that does not recall this reasoning may select this answer. This answer is plausible because cold water injection can cause reactor power to rise.

Technical Reference(s):
R-LPOPS-HLO-513 Rev 4
EOP BASES EPSTG-00002 Rev 16

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank: CNS 2014 NRC Exam

Question History: NA

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

10 CFR Part 55 Content: 55.43.b.5

Comments:

QUESTION 95 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.1.36		IR 4.1

Knowledge of procedures and limitations involved in core alterations

Proposed Question:

During a refuel outage, the CRS enters AOP-0027 because the refueling cavity water level is dropping and the leakage rate exceeds the sump capacity with a fuel bundle not in a safe, conservative storage location. The order of preference from first to fourth for the first four emergency makeup systems is:

A. 1. HPCS injection using CST suction
 2. Condensate injection via feedwater lines
 3. Emergency water addition
 4. RPV Injection with Service Water

B. 1. HPCS injection using CST suction
 2. Condensate injection via feedwater lines
 3. RPV Injection with Service Water
 4. Emergency water addition

C. 1. Condensate injection via feedwater lines
 2. HPCS injection using CST suction
 3. RPV Injection with Service Water
 4. Emergency water addition

D. 1. Condensate injection via feedwater lines
 2. HPCS injection using CST suction
 3. Emergency water addition
 4. RPV Injection with Service Water

Proposed Answer: C.

Explanation:
A. B and D are incorrect but plausible because they are arll acceptable sources of makeup water, but in the wrong order for the direction of the procedure..
C. Correct – per AOP-0027 the correct order is 1. Condensate injection, 2. HPCS injection, 3. RPV injection w/SW, and 4. Emergency water addition (with Fire water)

Technical Reference(s): AOP-0027, page 8, revision 28.

Proposed references to be provided to applicants during examination: None

Learning Objective: RLP-STM-0055

Question Source: New Question History: NA

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

10 CFR Part 55 Content: 55.43.b.7
Comments:

QUESTION 96 Rev 0

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.2.11		IR 3.3

Knowledge of the process for controlling temporary design changes

Proposed Question:

An event has occurred which presents an imminent threat to the safety of the plant.
In order to mitigate the event a temporary modification must be installed as directed by the Shift Manager. Whose concurrence is required to implement the emergency temporary modification?

A. Another SRO

B. The RBS Vice President (VP)

C. The General Manager Plant Operations (GMPO)

D. The Engineering Director or his designee

Proposed Answer: D

Explanation
A. B. and C. are incorrect because of direct statement made by procedure but are plausible because all of these individuals are required for authorizing various activities in the plant.
D. Correct – Per EN-DC-136, the concurrence of the Engineering Director, or designee is required

Technical Reference(s): EN-DC-136, Temporary Modifications, Rev 10 Section 5.3

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: 55.43.b.3

Comments:

QUESTION 97 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.2.38		IR 4.5

Knowledge of conditions and limitations in the facility license

Proposed Question:

The plant is in MODE 1. The following sequence of events occurs to the Standby Liquid Control (SLC) system.

- At 0700 on 12/7/14, an RO reports that the Division II SLC squib valve has no power available due to a blown fuse.
- At 0900 on 12/9/14, B-10 Solution Storage Tank concentration times enrichment is determined to be 564.
- At 0600 on 12/10/14, the blown fuse is replaced, power is restored to the Division II SLC squib valve, and the valve is declared operable.
- At 1400 on 12/11/14, an RO reports that the fuse has blown again and the Division II SLC squib valve once again has no power available.
- At 0500 on 12/12/14, the B-10 Solution Storage Tank concentration is increased and a sample determines concentration times enrichment to be 585.

Using the reference provided, what is the latest time and date when the reactor must be placed in MODE 3?

A. 0500 on 12/10/14

B. 1900 on 12/17/14

C. 0200 on 12/15/14

D. 2100 on 12/19/14

Proposed Answer: B

Explanation:

A. Incorrect. This is the correct time if Condition C is entered. Plausible if applicant believes that both trains of SLC are out of service when B-10 tank is out of spec.

B. Correct. This is the time corresponding to 10 days from the original LCO entry, based on time line where the LCO was never exited.

C. Incorrect. This is the correct time if you apply the 72 hour time limit to the second failure of the squib valve (1400 on 12/11/14 plus 72 hours, plus an additional 12 hours). Plausible if applicant does not recognize that the LCO was never exited.

D. Incorrect. This is 10 days from the second condition requiring LCO entry. Plausible if applicant does not recognize the restart of the 72 hour clock on the squib valve and believes 10 day clock starts once a second condition is entered.

Technical Reference(s): TS LCO 3.1.7

Proposed references to be provided to applicants during examination: TS LCO 3.1.7

Learning Objective:

Question Source: New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.43.b.1

Comments:

QUESTION 98 Rev 2

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.3.14		IR 3.8

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

Proposed Question:

Which of the following statements explains the significance of the Maximum Safe Area Radiation Level being reached in the areas identified in EOP-0003, Secondary Containment and Radioactive Release Control?

A. This is the value that requires verification of Aux building HVAC isolation.

B. This is the value that requires isolating all systems discharging into the area except systems (1) required for damage control, and (2) required to be operated by EOPs.

C. This value in any one area requires the plant to be shutdown per GOP-0002 Plant Shutdown IAW EOP-0003.

D. This value along with a primary system discharging into secondary containment requires entry into EOP-0001 RPV Control IAW EOP-0003.

Proposed Answer: D

Explanation

A. The maximum Safe value is the max level that would be used to determine other actions there is no Max Safe value for isolations.
B. The max <u>normal</u> operating value is used as the stop sign before this action in EOP-3.
C. EOP-0003 requires two areas to be at the Maximum Safe value prior to shutting down per GOP-0002 Plant Shutdown
D. Correct -

Technical Reference(s): EOP Bases

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: 55.43.b.5

Comments: Distractor B replaced due to plausibility

QUESTION 99 Rev 1

Examination Outline Cross-Reference:	Level		RO ☐	SRO ☑
	Tier #	3		
	K/A #	G2.4.16		IR 4.4

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

Proposed Question:

The plant is responding to an accident. The following plant parameters are observed in the control room:

- CTMT Temperature: 176 °F
- PSP is SAFE
- SP Temperature: 103 °F
- Primary containment Hydrogen: 3.3%
- DW Hydrogen: 3.9%
- SP level: 19 ft

The CRS should:

A. Exit all EOPs and enter all SAPs

B. Remain in EOP-1, Exit EOP-2 and enter SAP-002

C. Remain in all EOPs and enter SAP-002 concurrently

D. Remain in all EOPs only

Proposed Answer: A.

Explanation:

a. Correct. EOP-2, CTMT/DW Hydrogen, directs a transition out of all EOPs into all SAPs when Containment or DW hydrogen concentration reaches 3.5%.
b. Incorrect: EOP-2 directs a transition out of all EOPs into all SAPs when Containment or DW hydrogen concentration reaches 3.5%. Plausible if applicant believes that a transition is only required for primary containment parameters.
c. Incorrect: EOP-2 directs a transition out of all EOPs into all SAPs when Containment or DW hydrogen concentration reaches 3.5%. Plausible if applicant believes that H2 levels are not exceeded or that >3.5% is required in both DW and containment.
d. Incorrect: EOP-2 directs a transition out of all EOPs into all SAPs when Containment or DW hydrogen concentration reaches 3.5%. Plausible if applicant believes that H2 levels are not exceeded or that >3.5% is required in both DW and containment.

Technical Reference(s): EOP-1, EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source:	New	Question History:	NA
Cognitive Level:	Memory or Fundamental Knowledge ☐	Comprehension or Analysis ☑ 3	
10 CFR Part 55 Content:	55.43.b.5	Comments:	

QUESTION 100 Rev 1

Examination Outline Cross-Reference: Level RO ☐ SRO ☑
Tier # 3
K/A # G2.4.44 IR 4.4

Knowledge of emergency plan protective action recommendations

Proposed Question:

The plant is responding to an accident. A General Emergency has been declared and off-site dose calculations are in progress. The following plant parameters are observed in the control room:

- Containment PAM: 10,300 R/hr
- Fuel Damage estimate: 48% per COP-1050
- Containment parameters have been exceeded and cannot be restored

It has been determined that containment failure is imminent.

What Protective Action Recommendation should the Emergency Director initiate?

A. Evacuate 2 mile radius and 5 miles downwind, shelter remaining 10 mile radius and evacuate schools, institutions and recreation areas within 5 miles

B. Shelter within a 10 mile radius only

C. Evacuate 5 mile radius and 10 miles downwind, shelter remaining 10 mile radius and evacuate schools, institutions and recreation areas within 10 miles

D. Evacuate 5 mile radius and shelter remainder of 10 mile radius

Proposed Answer: C

Explanation:

A. Incorrect: Minimum PAR for GE. Plausible if applicant believes the core melt parameters (> 10,000 R/hr OR 100% core damage) have not been exceeded.
B. Incorrect: Shelter PAR for when area near the plant cannot be evacuated before plume arrives. Plausible if applicant believes that 2 hours is insufficient evacuation time.
C. Correct: PAR to be used for an imminent failure of containment with conditions given in stem.
D. Incorrect: PAR for a GE with PAM > 10,000 R/hr and a likely failure of containment (> 2 hours to failure). Plausible if applicant has a misconception about the difference between imminent and likely..

Technical Reference(s): EIP-2-007, Protective Action Recommendations, Rev 25, Attachment 2, p. 7 of 9
Proposed references to be provided to applicants during examination: None
Learning Objective: None identified.
Question Source: New
Question History: NA
Cognitive Level: Memory or Fundamental Knowledge ☑ 4 Comprehension or Analysis ☐
10 CFR Part 55 Content: 55.43.b.5
Comments:

RBS December 2014 NRC Initial Written Exam

Modified Question – Parent Questions

Q#	Parent Question
7	RBS December 2008 NRC Exam #8
16	RBS-OPS-5554
22	RBS-NRC-01108
23	RBS-NRC-444
37	Nov 2010 Audit Q#36
38	RBS-NRC-53
56	RBS-NRC-168
60	RBS-OPS-06288
62	RBS-LOR-1252
64	RBS-OPS-2307
68	RBS April 2010 NRC Exam #72
77	RBS-OPS-06265

Exam Question # 7

Modified from Parent Q# RBS December 2008 NRC Exam #8

QUESTION 8 Rev 0

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

Proposed Question:

Due to severe weather in the area around River Bend Station, a loss of offsite power has occurred.

All three diesel generators have started and are supplying safety related loads.

What is the status of the Plant Air Systems?

a. The Service Air compressors will be de-energized. Instrument Air compressors will be supplied by a safety related power source.

b. The Instrument Air compressors will be de-energized. SAS-AOV134 cross-connect valve will open to maintain IAS header pressure with the SAS compressors.

c. Instrument Air and Service Air compressors will both be de-energized. The diesel air compressor will automatically start to supply plant air needs.

d. Instrument Air and Service Air compressors will both be de-energized. The diesel air compressor must be manually started to supply plant air needs.

Proposed Answer: D.

Exam Question # 16

Modified from Parent Q# RBS-OPS-5554

As part of a routine surveillance, valve timing is being performed on the Low Pressure Core Spray (LPCS) system valves. Currently all valves are in their normal standby position except for E21-MOVF001, Suppression Pool Suction Valve, which is closed. A valid high drywell pressure, Low Pressure Core Spray (LPCS) initiation signal is received.

Assuming no operator action, which of the following statements identifies the expected response of LPCS?

A. LPCS Pump will start and run with pump suction valve E21-MOVF001 closed.
B. LPCS Pump will start and then immediately trip because it has **no** suction path.
C. LPCS Pump will **not** receive a start signal because the suction path permissive is not met.
D. E21-MOVF001 receives an automatic open signal and the LPCS Pump starts when the valve is full open.

Answer: A

Exam Question # 22

Modified from Parent Q# RBS-NRC-01108

River Bend was operating at full power with RCIC tagged out. A trip of all Reactor Feedwater Pumps caused RPV level to drop and a Reactor scram to occur on low RPV water level. HPCS recovered RPV level and E22-F004, HPCS INJECT ISOL VALVE was closed when RPV water level reached +35 inches. Conditions are currently:

- Vessel is isolated from the condenser
- Reactor pressure 700 psig (rising at 10 psig per minute)
- Time after scram 5 minutes
- CRD pumps both tripped
- Drywell pressure 0.3 psid

If no additional operator actions are taken other than the ones listed, what is the expected RPV water level response over the next 10 minutes and why?

A. rise due to swell from decay heat.

B. rise due to Feed Reg. Valve leakage exceeding heat requiements.

C. lower due to cooldown.

D. lower due to steam loads reducing RPV water inventory.

Answer: A

Exam Question # 23

Modified from Parent Q# RBS-NRC-444

Given the following plant conditions:

- Reactor Power 0% (all rods in)
- Reactor Level +33 inches
- Reactor Pressure 890 psig
- Drywell Pressure 1.8 psid
- Drywell Temperature 138°F
- Containment Temperature 88°F
- Containment Pressure 0.35 psig
- Annulus Differential Pressure -4.5 in.WC

Based on the above conditions, which one of the following describes the Emergency Operating Procedures that should be entered?

A. EOP-1 ONLY

B. EOP-1 and 2

C. EOP-2 ONLY

D. EOP-2 and 3

Answer: B

Exam Question # 37

Modified from Parent Q# Nov 2010 Audit Q# 36

Examination Outline Cross-Reference:	Level		RO ☑	SRO ☐
	Tier #	2	Group # 1	
	K/A #	215004	IR 2.6	

Knowledge of the electrical power supplies to SRM channels/detectors.

Proposed Question:

The power supply to SRM 'A' detector is _______________.

A. VBS-PNL01A

B. VBS-PNL01B

C. RPS A

D. RPS B

Proposed Answer: C.

Exam Question # 38

Modified from Parent Q# RBS-NRC-53

During a reactor startup the ATC has been withdrawing SRM detectors per GOP-0001. All SRMs except A indicate full out. SRM A has an upscale high and an upscale High-high trip indicated and is reading off-scale high. The P680 indications show the detector is "driving out."

The ATC should . . .

A. Immediately insert control rods to return SRM A readings on-scale.
B. Insert a Div I half scram and continue with the plant startup.
C. Check the SRM A drive power fuses, if the problem is not corrected, obtain reactor engineering assistance..
D. Since the other drive OUT lights are on, SRM A drive has power therefore contact I & C for assistance.

Answer: C

Exam Question # 56

Modified from Parent Q# RBS-NRC-168

If total feedwater flow drops below the reactor recirculation system interlock level, the Reactor Recirculation pumps will downshift to slow speed.

What is the PRIMARY reason for this interlock?

A. pump cavitation.
B. flow control valve cavitation.
C. excessive axial thrust on the pump.
D. inaccurate wide range level indication.

Answer: B

Exam Question # 60

Modified from Parent Q# RBS-OPS-06288

The primary means of hydrogen control following a LOCA utilizes:

A. Hydrogen Igniters
B. Hydrogen Recombiners
C. Containment Purge subsystem
D. Drywell Purge subsystem

Answer: A

Exam Question # 62

Modified from Parent Q# RBS-LOR-1252

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 to 0 psig?

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.

Answer: D

Exam Question # 64

Modified from Parent Q# RBS-OPS-2307

The HVN Chilled Water system Supply and Return valves to Containment (HVN-MOV127, 128, 129, 130, 102) will AUTO-CLOSE upon __________.

A. RPV Level 1 (-143")
B. RPV Level 2 (-43")
C. Drywell to Annulus d/p greater than 12 inches
D. Opening of Service Water supply and Return valves to Containment

Answer: B

Exam Question #68

Modified from Parent Q# April 2010 NRC Exam Q#72

QUESTION 72 Rev 0

During a refueling outage maintenance personnel require access to the annulus fuel transfer tube area.

In addition to operation of the palm handswitch outside the gate, which of the following are also required to allow access to this area?

A. Keylock switch on H13-P863 in ACCESS

B. Keylock switch outside the fuel building support room gate in ACCESS

C. Keylock switch on IFTS in the reactor building in ACCESS

D. Keylock switch outside the containment transfer tube support room in ACCESS

Proposed Answer: A.

Exam Question # 77

Modified from Parent Q# RBS-OPS-06265

The plant has experienced an ATWS. The following conditions exist at P680:

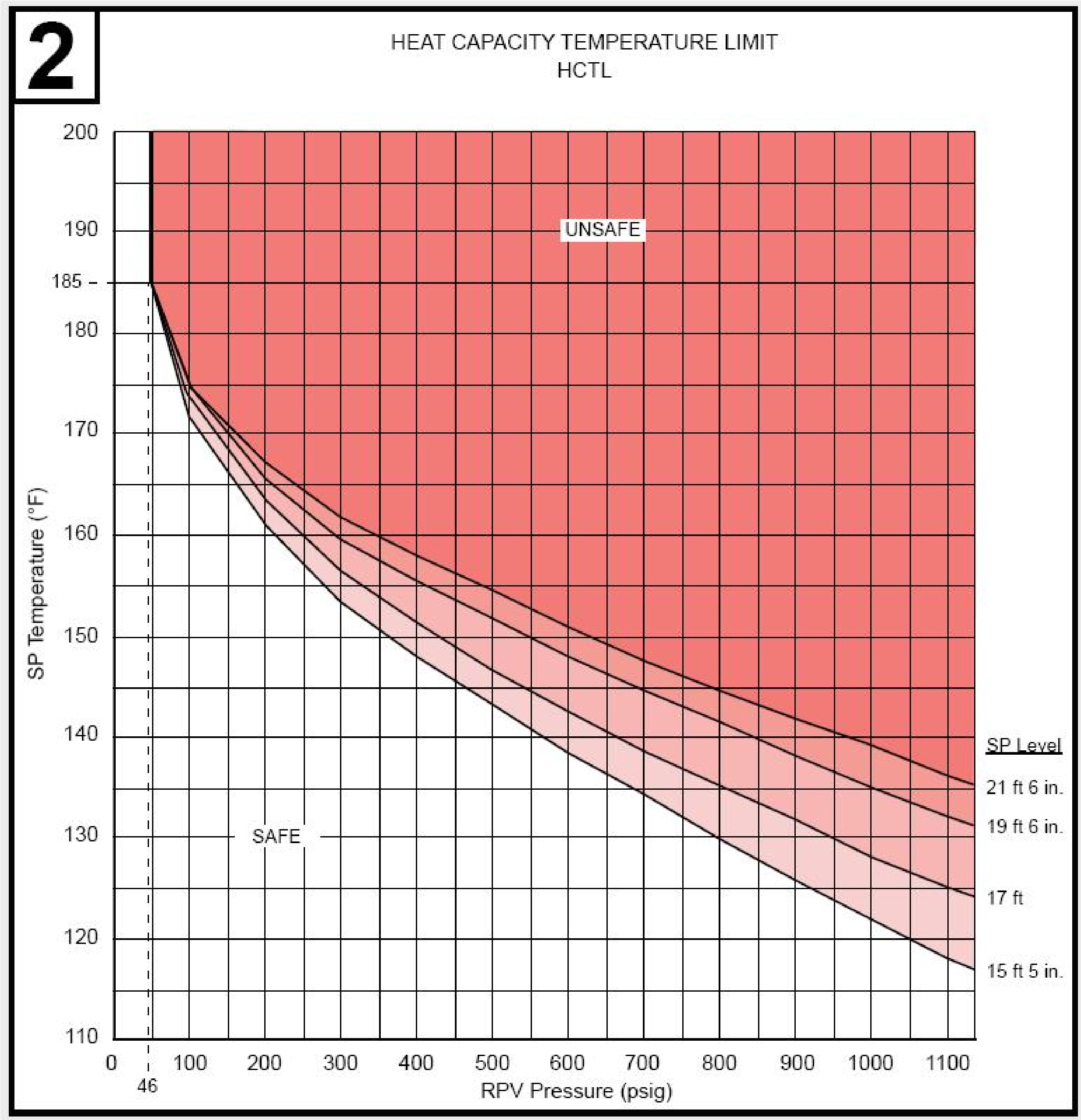
- All eight white scram solenoid lights are extinguished
- Annunciator P680-05-C08 SCRAM PILOT VLV AIR HEADER LOW PRESSURE is alarming
- SDV Vent and Drain valve position lights indicate all four valves are closed
- Approximately 20% of the withdrawn control rods fully inserted
- CRD cooling water differential pressure has been maximized
- ARI has been inititated

Which of the following methods for alternate control rod insertion should be attempted next?

A. venting the scram air header.
B. resetting and reinitiating ARI.
C. removing the scram solenoid power fuses.
D. resetting the scram and intiating a manual scram.

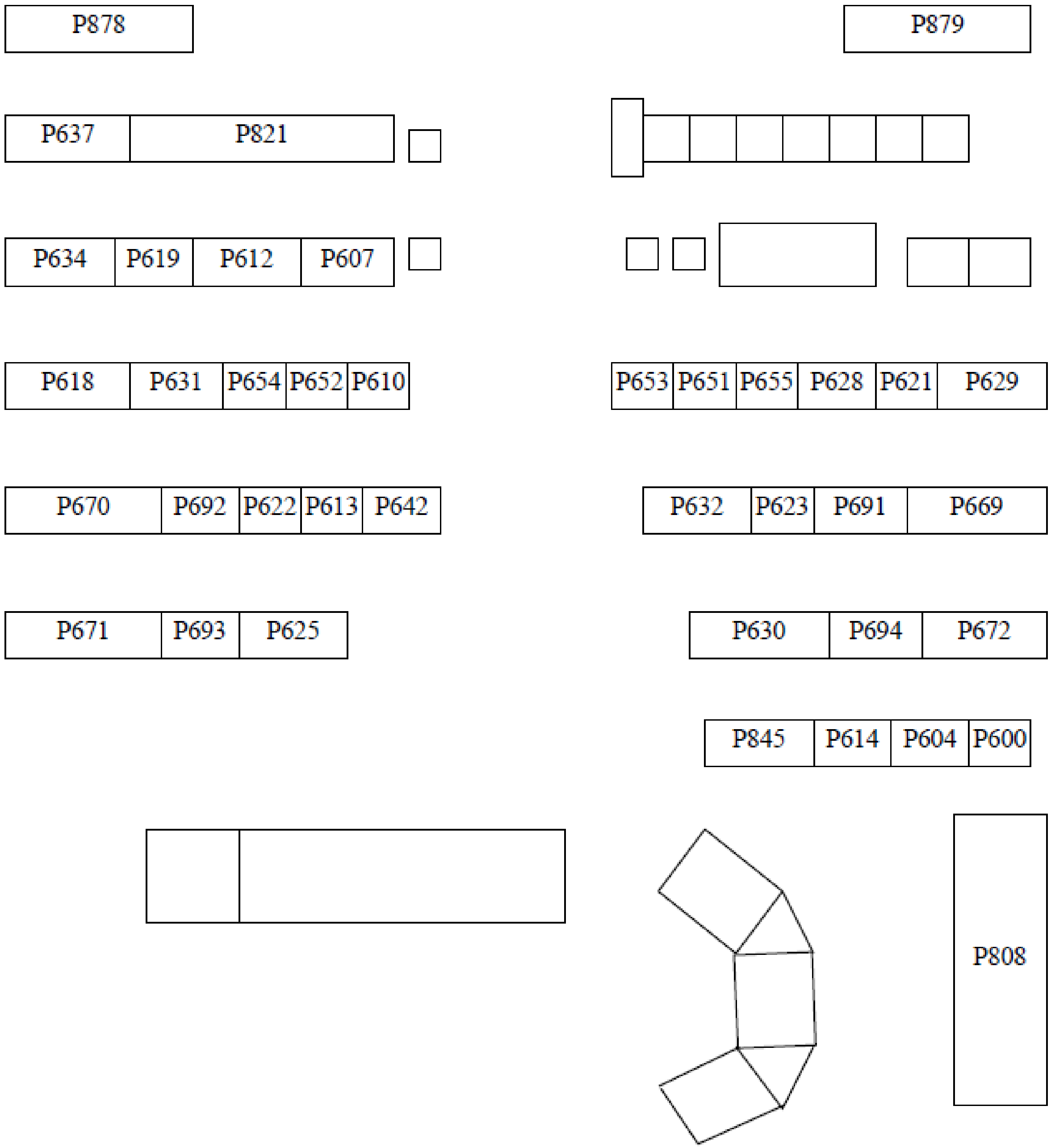
Answer: D

RB-2014-12 Initial Exam
Handouts for final written exam for questions
Q12, Q36
Q90, Q92, and Q97


2
HEAT CAPACITY TEMPERATURE LIMIT
HCTL
UNSAFE
SAFE
SP Temperature (°F)
200
190
185 –
180
170
160
150
140
130
120
110
0
46
100
200
300
400
500
600
700
800
900
1000
1100
RPV Pressure (psig)
SP Level
21 ft 6 in.
19 ft 6 in.
17 ft
15 ft 5 in.

MAIN CONTROL ROOM NSSSS BACKPANELS

All panel numbers are prefixed with H13-



3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources–Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System; and

b. Three diesel generators (DGs).

APPLICABILITY: MODES 1, 2, and 3.

---NOTES---

1. Division III AC electrical power sources are not required to be OPERABLE when High Pressure Core Spray System and Standby Service Water System pump 2C are inoperable.

2. The automatic transfer function for the Division III 4.16 kV system buses shall be OPERABLE whenever the 22 kV onsite circuit is supplying Division III safety related bus E22-S004 from normal power transformer STX-XNS1C.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit. AND	1 hour AND Once per 8 hours thereafter
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	-------------------NOTE------------------- Verification is only required if 22 kV onsite circuit is supplying Division III safety related bus E22-S004 from normal power transformer STX-XNS1C. ---	
	A.2 Verify E22-S004 is aligned to transfer to the preferred station transformer powered by the OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND	
	A.3 Restore required offsite circuit to OPERABLE status.	72 hours AND 24 hours from discovery of two divisions with no offsite power AND 17 days from discovery of failure to meet LCO
B. Automatic transfer function not OPERABLE	B.1 Restore Division III power source to the preferred station service transformers	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One required DG inoperable.	C.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		
	C.2	Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure. OR C.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).	24 hours 24 hours
	AND C.4 Restore required DG to OPERABLE status.	72 hours from discovery of an inoperable Division III DG AND 14 days AND 17 days from discovery of failure to meet LCO
D. Two required offsite circuits inoperable.	D.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable. AND D.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition D concurrent with inoperability of redundant required feature(s) 24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One required offsite circuit inoperable. AND One required DG inoperable.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems–Operating," when any division is de-energized as a result of Condition E. ---	
	E.1 Restore required offsite circuit to OPERABLE status.	12 hours
	OR	
	E.2 Restore required DG to OPERABLE status.	12 hours
F. Two required DGs inoperable.	F.1 Restore one required DG to OPERABLE status.	2 hours OR 24 hours if Division III DG is inoperable
G. Required Action and Associated Completion Time of Condition A, B, C, D, E or F not met.	G.1 Be in MODE 3.	12 hours
	AND	
	G.2 Be in MODE 4.	36 hours
H. Three or more required AC sources inoperable.	H.1 Enter LCO 3.0.3.	Immediately

TR 3.3.6.1 Primary Containment and Drywell Isolation Instrumentation

TLCO 3.3.6.1 The primary containment and drywell isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 1.l, 2.b, 5.b, 5.d, and 5.e AND 24 hours for Functions other than Functions 1.l, 2.b, 5.b, 5.d, and 5.e
	OR	
	A.2 Enter Condition L for function 1.l in MODES 1, 2 and 3.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more automatic Functions with isolation capability not maintained	B.1 Restore isolation capability	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL). <u>OR</u> D.2.1 Be in MODE 3. <u>AND</u> D.2.2 Be in MODE 4.	12 hours 12 hours 36 hours
E. (Not Used)		
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. (Not Used)		

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. <u>OR</u> Required Action and associated Completion Time of Condition F not met.	H.1 Be in MODE 3. <u>AND</u> H.2 Be in MODE 4.	12 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. (not Used)		
J. (not Used)		
K. (not Used)		
L. Entry from Condition A for inoperable Main Steam Line Radiation Monitors.	L.1 Enter TLCO 3.0.3.	Immediately

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (C)(E) < 570.	A.1 Restore (C)(E) ≥ 570.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours