QUESTION 1 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295001 AK3.04
	Importance Rating	3.4

Knowledge of the reason for a reactor scram as it applies to a partial or complete loss of forced core flow circulation.

Proposed Question:

What is the reason for an automatic scram upon entry into the Exclusion Region of the Power to Flow map?

a. To avoid exceeding the Reactor Pressure Safety Limits during flux oscillations.

b. To avoid exceeding the MCPR Safety Limit during flux oscillations.

c. To avoid exceeding the MAPRAT operating limit due to low coolant flow.

d. To avoid exceeding the LHGR operating limit due to low coolant flow.

Proposed Answer: B

Explanation (Optional):The Exclusion Region scram as stated in the TS bases avoids exceeding MCPR SL during flux oscillations.

Technical Reference(s): AOP-0024 Rev. 22, STM-503 Rev 2, Tech Spec 3.3.1.1. Bases

Proposed references to be provided to applicants during examination: NA

Learning Objective: STM-503 Obj 24, 27a

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 2 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295003 AA2.03
	Importance Rating	3.2

Ability to determine and/or interpret battery status as it applies to a partial or complete loss of AC power.

Proposed Question:

Following a transient, the following plant conditions exist:

RPV water level -47 inches and stable
Drywell pressure 1.2 psid and stable

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

ENB-SWG1B 125 VDC SWG was being supplied by the backup charger (BYS-CHGR1D) prior to the transient.

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

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

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

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

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

Proposed Answer: B.

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

Technical Reference(s): STM-305, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-STM-305 Obj. 9, 11a	
Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒ 4
10 CFR Part 55 Content:	55.41 b.7	
Comments:		

QUESTION 3 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295004 AA1.01
	Importance Rating	3.3

Ability to operate and/or monitor the DC electrical distribution system during a partial or complete loss of DC power.

Proposed Question:

125 VDC distribution panel BYS-PNL02A2 was inadvertently de-energized due to a clearance error.

Which of the following DC electrical loads is affected by this event?

a. RCIC Gland Seal Compressor

b. ARI Valves

c. Division 1 SRV solenoids

d. ENS-SWG1A breaker control power

Proposed Answer: B.

Explanation (Optional): RCIC loads are supplied from ENB-MCC1. Div 1 SRV Solenoids are supplied from ENB-PNL02A. ENS-SWG1A breaker control power is supplied from ENB-PNL04A. ARI valves are supplied from BYS-PNL02A2.

Technical Reference(s): STM-052, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0052 Obj 3e

Question Source: Modified Bank # INPO 21963; Modified to match RBS mark numbers and altered 2 distractors for plausibility.

Question History: Last NRC Exam Perry 2002

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 4 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295005 AA1.02
	Importance Rating	3.6

Ability to operate and/or monitor RPS following a Main Turbine or Generator trip.

Proposed Question:

A plant startup is in progress in accordance with GOP-0001.
Reactor power is 38%.

The main turbine and generator have just tripped due to a drop in Turbine Bearing Oil Header pressure due to a leak in the Turbine Lube Oil System.

Which of the following describes the response of the RPS system to the turbine trip?

a. RPS trip systems are de-energized, Backup scram valves are energized.

b. RPS trip systems are energized, Backup scram valves are de-energized.

c. RPS trip systems are de-energized, Backup scram valves are de-energized.

d. RPS trip systems are energized, Backup scram valves are energized.

Proposed Answer: A

Explanation (Optional): With reactor power being great than 30.4%, RPS trips on a turbine trip are no longer bypassed. RPS which is normally energized will de-energize and the backup scram valves which are normally de-energized, will energize.

Technical Reference(s): STM-508, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-508, Obj. 3e, 7f

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.6, b.7

Comments:

QUESTION 5 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295006 AK2.02
	Importance Rating	3.8

Knowledge of the interrelations between SCRAM and reactor water level control.

Proposed Question:

The plant is at 100% power.

Feedwater level control is in “automatic” with Narrow Range Level Channel “A” selected.

The “A” Narrow Range Channel has just failed DOWNSCALE.

No operator actions are taken.

Select the cause of the subsequent reactor scram.

a. Reactor vessel high water level.

b. Main Steam Isolation Valve closure.

c. Reactor vessel low water level.

d. APRM high thermal power.

Proposed Answer: A.

Explanation (Optional): A downscale failure of the selected channel will cause the feed system to provide an increase amount of flow to the RPV causing actual level to rise. The resultant high level will cause a high level reactor trip. B. MSIVs close on low level, not high level. C. Level will rise not lower. D. Although the addition of more feedwater will cause a slight reduction in FW temperature which will cause a slight power rise, the amount is not enough to reach the high thermal power trip setpoint.

Technical Reference(s): STM-107, Revision 10

Proposed references to be provided to applicants during examination: NA

Learning Objective: Obj. 14.f.

Question Source:	Bank # 495	
Question History:	Last NRC Exam	RBS NRC 1997
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒ 3
10 CFR Part 55 Content:	55.41 b.4	
Comments:		

QUESTION 6 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295016 G2.4.4
	Importance Rating	4.5

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures as they relate to control room abandonment

Proposed Question:

During normal plant operation, smoke is observed coming from a control room back panel. The control room operators have expressed difficulty in breathing.

In addition, several Safety Relief Valves have unexpectedly opened and unexpected alarms are being received.

Which of the following is the expected operator response?

a. Perform the actions of AOP-0001, Reactor Scram and continue to monitor the situation.

b. Enter EOP-1, RPV Control, using an alternate method of pressure control due to SRV failures.

c. Enter AOP-0035, Stuck Open SRV.

d. Enter AOP-0031, Shutdown From Outside the Main Control Room.

Proposed Answer: D

Explanation (Optional): Indications are present that there is a fire in the control room which is affecting the operators and plant equipment.

Technical Reference(s): AOP-0031, Rev 303

Proposed references to be provided to applicants during examination: NA

Learning Objective: HLO-537 Obj. 2

Question Source: Bank # 499

Question History: Last NRC Exam 1/1997

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 7 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295018 AK3.07
	Importance Rating	3.1

Knowledge of the reasons for the cross connecting of backup systems as it applies to the partial or complete loss of component cooling water.

Proposed Question:

AOP-0011, Loss of CCP, provides guidance to supply certain CCP loads with Standby Service Water.

Why is it desirable to do this during a loss of CCP?

a. To provide cooling to the Reactor Recirculation Pumps to avoid seal degradation.

b. To provide cooling to the RWCU Non Regenerative Heat Exchanger to avoid RWCU resin damage.

c. To provide cooling to the Spent Fuel Pool Cooling Heat Exchanger to ensure adequate decay heat removal from the Spent Fuel Pool

d. To provide cooling to the drywell sample cooler to protect chemistry sample probes from high temperature conditions.

Proposed Answer: C.

Explanation (Optional): When SSW is cross connected to CCP loads, SFC HXs, CRD pumps and RHR pump A&B seal cooler receive cooling. Recirc pumps, RWCU non regen HXs and RWCU pumps and the drywell sample cooler do not receive cooling from SSW when it is aligned to the CCP header.

Technical Reference(s): STM-115, Rev 4, AOP-0011 Rev 16

Proposed references to be provided to applicants during examination: NA

Learning Objective: Obj 2b, 3e, 5b, 11a

Question Source: New

Question History: Last NRC Exam NA

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

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

Comments:

QUESTION 8 Rev 0

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

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material during a partial or total loss of instrument air.

Proposed Question:

During operation at 100% power, the following annunciator is received:

INSTRUMENT AIR COMPRESSOR TROUBLE

Indication on H13-P870 show all instrument air compressor have tripped and IAS header pressure is lowering.

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

a. SAS-AOV134 IAS-SAS CROSS TIE VLV opens, then
SAS-AOV133 SERVICE AIR HEADER BLOCK VLV closes, then
MSIVs fail shut, then
Feedwater Regulating Valves Lock-up

b. SAS-AOV133 SERVICE AIR HEADER BLOCK VLV closes, then
SAS-AOV134 IAS-SAS CROSS TIE VLV opens, then
Feedwater Regulating Valves Lock-up, then
MSIVs fail shut

c. SAS-AOV133 SERVICE AIR HEADER BLOCK VLV closes, then
SAS-AOV134 IAS-SAS CROSS TIE VLV opens, then
MSIVs fail shut, then
Feedwater Regulating Valves Lock-up

d. SAS-AOV134 IAS-SAS CROSS TIE VLV opens, then
SAS-AOV133 SERVICE AIR HEADER BLOCK VLV closes, then
Feedwater Regulating Valves Lock-up, then
MSIVs fail shut

Proposed Answer: D.

Explanation (Optional): SAS-AOV134 opens at 113 psig, SAS-AOV133 closes at 110 psig, FW Reg Vlvs lock up at 85 psig, MSIVs shut at ~50 psig.

Technical Reference(s): AOP-0008 Rev 26, STM-0121, Rev 6

Proposed references to provide to applicants during examination: NA

Learning Objective: RLP-STM-0121 Obj 13 & 14

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 9 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295021 AK3.05
	Importance Rating	3.6

Knowledge of the reason for establishing alternate heat removal paths during a loss of shutdown cooling.

Proposed Question:

The plant is in Mode 5.
RPV level is 85 inches

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

Why does the abnormal procedure direct this action?

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

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

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

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

Proposed Answer: D

Explanation (Optional): Thermal shock limitations for Recirc pumps are of concern during pump startup. Radiological concerns during a fuel handling accident are accounted for by maintaining >23 feet of water over the vessel flange during fuel handling. Although high temperatures can damage RWCU resin, other interlocks protect the resin from high temperature. If an alternate decay heat removal method is not placed in service, boiling will eventually occur.

Technical Reference(s): AOP-0051Rev 304

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-HLO-543 Obj 1
Question Source:	New
Question History:	Last NRC Exam NA
Question Cognitive Level:	Memory or Fundamental Knowledge ☒3 Comprehension or Analysis ☐
10 CFR Part 55 Content:	55.41 b.10
Comments:	

QUESTION 10 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295023 AA1.06
	Importance Rating	3.3

Ability to operate or monitor neutron monitoring during a refueling accident.

Proposed Question:

The following plant conditions exist:

Mode 5
Core alterations in progress

Which of the following conditions require entry into AOP-0027 Fuel Handling Mishaps?

a. The refuel SRO reports a malfunction of IFTS with a new fuel bundle loaded in the carrier.

b. The refuel SRO reports air bubbles coming from the main hoist grapple.

c. The ATC operator observes a steadily rising neutron count rate with a measurable period.

d. The refuel SRO reports that a control rod blade was dragged across the portable radiation shield (cattle chute)

Proposed Answer: C.

Explanation (Optional): A malfunction of IFTS with an irradiated bundle in the carrier would require AOP-27 entry. Air bubbles from the fuel would require AOP-0027 entry. Air bubbles from the grapple is indicative of an air hose leak. Significant bumping of irradiated fuel requires entry in AOP-0027. Observation of a steadily rising neutron count rate with a measurable period is indication of inadvertent criticality which requires entry into AOP-0027.

Technical Reference(s): AOP-0027, Rev 23

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-535, Obj 3

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒3
	Comprehension or Analysis	☐
10 CFR Part 55 Content:	55.41 b.10	
Comments:		

QUESTION 11 Rev 0

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

Knowledge of the reasons for Reactor Scram as it applies to High Drywell Pressure.

Proposed Question:

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

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 (Optional): 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 in the RPV. 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): STM-508, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0508 Obj. 2

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.6

Comments:

QUESTION 12 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295026 EA1.01
	Importance Rating	4.1

Ability to operate and/or monitor suppression pool cooling as it applies to a suppression pool high water temperature.

Proposed Question:

Following an ATWS, the following conditions exist:

Reactor power	0%, all rods in
RPV level	-50 inches, slowly raising to normal band
RPV pressure	0 psig

Suppression Pool Level	19 feet 11 inches
Suppression Pool Temp	140°F

RHR A in Sup Pool Cooling @ 5200gpm, SWP flow @ 5900 gpm
RHR B in Sup Pool Cooling @ 5400 gpm, SWP flow @ 6300 gpm
Both Division of Standby Service Water are in service.

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

a. RHR B system flow has exceeded limits.

b. SWP flow has exceeded limits.

c. RHR pump may experience air entrainment due to vortex limit concerns.

d. RHR pumps may experience cavitation due to NPSH concerns.

Proposed Answer: B

Explanation (Optional): RHR system flow limitation is 5550 gpm, SWP flow limits is 5800 gpm per loop with SSW in service, Vortex limit is 10 feet in the Sup Pool, NPSH concerns are at 160°F

Technical Reference(s): SOP-0031, Rev 304; EOP-0001, Rev 21

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-0511 Obj. f, RLP-STM-204 Obj. 8

Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.10	
Comments:		

QUESTION 13 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295027 EA1.03
	Importance Rating	3.5

Ability to operate and/or monitor emergency depressurization as it applies to High Containment Temperature

Proposed Question:

An ATWS has resulted in degraded conditions in containment due to difficulties in restoring Containment Unit Coolers.

In which of the following situations is Emergency Depressurization REQUIRED?

a. 187°F and lowering at 3°F per minute due to Containment Unit Cooler restoration.

b. 184°F and stable, Containment Unit Coolers will be restored in 2 minutes.

c. 180°F and stable, Containment Unit Coolers CANNOT be restored.

d. 180°F and rising at 2°F per minute, Containment Unit Coolers will be restored in 2 minutes.

Proposed Answer: A

Explanation (Optional): 185°F is the Containment Design Temperature. Only answer “A” is above 185°F. Although both B and C contain conditions where the temperature has approached the design temperature limit and no UCs are in service, both distractors state that the temperature is STABLE. Although distractor D contains a condition where temperature is approaching the design limit, it has not yet been reached therefore ED is not REQUIRED as stated in the stem. Additionally, distractor D states that UCs are about to be restored.

Technical Reference(s): EOP-2, CT4,5,6 Rev 14
EPSTG-2 B-8-9 Rev 12

Proposed references to be provided to applicants during examination: NA

Learning Objective: HLO-514 Obj 5

Question Source: Modified Bank # 758 (See Comments.)

Question History: Last NRC Exam RBS 2/2003

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

10 CFR Part 55 Content: 55.41 b.9

Comments: This question was used on the 2/2003 NRC exam at RBS. A second answer was selected based on post-exam comments due to answer D being considered correct. Original wording of answer D was *"180°F and slowly rising, Containment Unit Coolers CANNOT be restored"*. Modified answer D to indicated that the Containment Unit Coolers are about to be restored to have A as the only correct answer.

QUESTION 14 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295028 EK1.02
	Importance Rating	2.9

Knowledge of the operational implications of equipment environmental qualifications as they apply to high drywell temperature.

Proposed Question:

Which of the following lineups could potentially be affected by a high steam environment in the drywell if operator action is not taken early into the event?

a. LPCI Injection lineup due to accelerated corrosion of magnesium alloy rotor on E12-MOVF042A RHR Pump A LPCI Injection Isol Valve

b. Shutdown cooling flowpath due to accelerated corrosion of magnesium alloy rotor on E12-MOVF009 RHR Shutdown Cooling Inbd Isol Valve

c. LPCI Injection lineup due to accelerated corrosion of magnesium alloy rotor on E12-MOVF027A RHR Pump A Outboard Isolation Valve

d. Alternate injection lineup per EOP-0005 Enclosure 32 due to accelerated corrosion of magnesium alloy rotor on E12-F053A RHR Pump A SDC Injection Valve

Proposed Answer: B

Explanation (Optional): Caution 2 of EOP-001 RPV control identifies E12-MOV009 as having a magnesium alloy rotor which is susceptible to accelerated corrosion between the magnesium alloy shorting ring and the rotor conductor bars. The LPCI injection valves are also located in the drywell, but are not susceptible to this failure mechanism. E12-MOV27A(B)(C) are E12-MOVF008 are not located in the drywell.

Technical Reference(s): EOP-0001 Caution 2, Rev 21

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-OPS-HLO-511 Obj. F

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒3
	Comprehension or Analysis	☐
10 CFR Part 55 Content:	55.41 b.7	
Comments:		

QUESTION 15 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295030 EA2.01
	Importance Rating	4.1

Ability to determine and/or interpret suppression pool level as it applies to a low suppression pool water level.

Proposed Question:

What is the minimum allowable Suppression Pool level for opening SRVs during Emergency Depressurization?

a. >13 feet

b. >15 feet 3 inches

c. >16 feet

d. >19 feet 6 inches

Proposed Answer: A.

Explanation (Optional):Suppression pool level must be verified to be above 13 feet prior to opening SRVs during ED.

Technical Reference(s): EOP-0001, Rev 21

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-512 Obj. E

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.9, b.10

Comments:

QUESTION 16 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295031 EA1.06
	Importance Rating	4.4

Ability to operate and/or monitor the automatic depressurization system as it applies to a reactor low water level.

Proposed Question:

A LOCA has occurred and High Pressure Core Spray has failed to initiate.

The following conditions exist.

ADS Inhibit switches are in INHIBIT
Drywell differential pressure is 1.05 psid and rising
RPV pressure is 890 psig and lowering
RPV water level is -155 inches and stable on wide range instrumentation

All other systems are functioning as designed.

Which of the following describes the operation of the Automatic Depressurization System (ADS) valves under the current conditions?

a. ADS valves can be opened by using the ADS Manual Initiation pushbuttons.

b. ADS will automatically initiate to open ADS valves when the 105 second timer times out.

c. ADS will automatically initiate to open ADS valves when the 5 minute and 105 second timers time out.

d. ADS valves can only be opened by their individual handswitches.

Proposed Answer: A.

Explanation (Optional): Placing the ADS Inhibit switches to inhibit prevents automatic operation of ADS. This eliminates choices “B” and “C”. Choice “D” is incorrect because in addition to their individual handswitches, the ADS valves can also be opened by manual initiation provided the associated divisional ECCS pump is running. Based on given plant conditions, ECCS pumps are running due to a Level 1 signal.

Technical Reference(s): STM-202, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective:	STM-202, Obj 7	
Question Source:	Bank 1030	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒ 2
10 CFR Part 55 Content:	55.41 b.7	
Comments:		

QUESTION 17 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	295037 EK1.01
	Importance Rating	4.1

Knowledge of the operational implications of reactor pressure effects on reactor power as they apply to an ATWS.

Proposed Question:

During an ATWS in order to avoiding exceeding the Heat Capacity Temperature Limit (HCTL) curve, the SRO has ordered reactor pressure be lowered to 700 psig using SRVs.

Which of the following describes reactor power response immediately following the opening of the SRVs and why?

a. Reactor power will rise due to the lowering of the reactor coolant temperature along with adding positive reactivity.

b. Reactor power will rise due to the water level inside the core rising causing more moderation of neutrons.

c. Reactor power will drop due to the voiding of the water in the core as it flashes to steam.

d. Reactor power will drop due to the moderator temperature rising caused by low flow through the core.

Proposed Answer: C

Explanation (Optional): The pressure drop that occurs as SRVs are opened will result in an increase in void fraction in the core as saturated moderator flashes to steam. The increase presence of voids in the core will result in a decrease in power due drop in thermal neutrons. Moderator temperature will decrease to saturation temperature for the lower pressure value, but the void coefficient effect is the primary factor.

Technical Reference(s): HLO-161, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: HLO-161, Obj 7

Question Source: Modified Bank # 569 (Modified to removed superfluous information in stem and to provide conditions which would require pressure reduction. Previously listed conditions did not challenge HCTL.)

Question History:	Last NRC Exam	RBS 2/1999
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.1	
Comments:		

QUESTION 18 Rev 0

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

Ability to determine and/or interpret the Radiation Levels during a High Offsite Release Rate.

Proposed Question:

While in Mode 1 at 100% power, a significant leak occurred on the steam supply to MSR#1. The CRS has directed the ATC operator to place the mode switch in SHUTDOWN. Control Rods failed to insert. EOP-1A execution is in progress. The following conditions exist:

Reactor power: 17%
MSIVs open

RMS-RE125 MAIN PLANT EXAUST Green status
RMS-RE110 AUX BLDF VENTILATION Green status
RMS-RE118 TURBINE BLDG VENT Green status

Emergency Response Organization has been activated.
Offsite release teams have reported 850 mR/hour at the site boundary.

Which of the following accurately describes the current condition?

a. An unfiltered, monitored release is in progress.

b. An unfiltered, unmonitored release is in progress.

c. A filtered, monitored release is in progress.

d. A filtered, unmonitored release is in progress.

Proposed Answer: B.

Explanation (Optional): A leak in the MSR area producing 850mR/hr at the site boundary should be observed on RMS-RE118 and RMS-RE125. Since these monitors are not in alarm, the release is unmonitored. A leak outside secondary containment is unfiltered.

Technical Reference(s): PID22-03A, PID-22-01C

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-NEO-050 Obj 2, STM-0409 Obj 2a
Question Source:	New
Question History:	Last NRC Exam NA
Question Cognitive Level:	Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒3
10 CFR Part 55 Content:	55.41 b.11,b.13
Comments:	

QUESTION 19 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	600000 AA1.09
	Importance Rating	2.5

Ability to operate and/or monitor the plant fire zone panel (including detector location) during a Plant Fire On Site.

Proposed Question:

A fire has erupted in the Control Building 70' elevation cable tray area. The associated water spray system has actuated.

Which of the following accurately describes methods by which the control room team may obtain information concerning this fire and its extinguishment?

a. Alarming detector location on H13-P680 Plant Process Computer screen. Fire pump status on FPM-PNL861 Fire Control Console.

b. Alarming detector location and water spray system status from FPM-PNL861 Fire Control Console. Fire pump status from the Plant Process Computer screen on H13-P680.

c. Alarming detector location on H13-P680 Plant Process Computer screen.Fire pump status and water spray system status from FPM-PNL861 Fire Control Console.

d. Alarming detector location on FPM-PNL861 Fire Control Console. Fire pump status and water spray system status from H13-P680 Plant Process Computer screen.

Proposed Answer: B.

Explanation (Optional): Fire pump status is available on the plant process computer. Fire detector status and flow switch status for each water system is available on the Fire Protection Control Console P861.

Technical Reference(s): SOP-0036, Rev 301, STM-0250, Rev

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0250 Obj 4

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒3
	Comprehension or Analysis	☐
10 CFR Part 55 Content:	55.41 b.7	
Comments:		

QUESTION 20 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	1
	K/A #	700000 AA2.06
	Importance Rating	3.4

Ability to determine and/or interpret generator frequency limitations as they apply to Generator Voltage and Electrical Grid Disturbances.

Proposed Question:

During a severe weather event, several generating units in the area have tripped offline. As a result, grid frequency has dropped to 56.8 Hertz.

How will the Main Generator regulator respond. Assume no other conditions.

a. The exciter voltage circuit will control exciter field voltage at a preset level to stabilize the condition.

b. The exciter voltage circuit will raise exciter field voltage to raise grid frequency.

c. The Volts/Hertz circuit will lower generator excitation to protect the regulator.

d. The Volts/Hertz circuit will raise generator excitation to raise grid frequency.

Proposed Answer: C

Explanation (Optional): At 57 Hertz, the Volts/Hertz circuit develops a take over signal to drive excitation down as frequency decreases to protect frequency/voltage sensitive components of the regulator.

Technical Reference(s): STM-310, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-310, Obj. 10

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 21 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	2
	K/A #	295002 AK2.04
	Importance Rating	3.2

Knowledge of the interrelations between loss of main condenser vacuum and the reactor/turbine pressure regulating system.

Proposed Question:

Following an ATWS, the following conditions exist:

Reactor power	3%
Condenser vacuum	18 inches Hg
RPV water level	18 inches
Reactor pressure	960 psig
Pressure setpoint	950 psig

Based on the above conditions which of the following represent the expected positions of the Control Valves (CVs) and Bypass Valves (BPVs)?

a. CVs open, BPVs open

b. CVs closed, BPVs closed

c. CVs open, BPVs closed

d. CVs closed, BPVs open

Proposed Answer: D.

Explanation (Optional): A main turbine trip occurs at 22.3" Hg. This results in the control valves being closed at the current conditions. The bypass valves do not isolate due to low vacuum until8.5" Hg. With reactor pressure being higher than pressure setpoint, the current conditions will cause the BPVs to be open.

Technical Reference(s): STM-509, Rev 6

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0509 Obj 16d

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 22 Rev 0

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

Knowledge of the interrelations between High Reactor Pressure and NSSSS.

Proposed Question:

Following a planned reactor shutdown, a plant cooldown is in progress with RHR A as the inservice 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?

a. E12-F053A CLOSED E12-F027A OPEN

b. E12-F053A CLOSED E12-F027A CLOSED

c. E12-F053A, OPEN E12-F027A OPEN

d. E12-F053A OPEN E12-F027A CLOSED

Proposed Answer: A.

Explanation (Optional):
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.

Technical Reference(s): STM-0058, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-508, Obj. 2f

Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.9	
Comments:		

QUESTION 23 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	2
	K/A #	295011 AK1.01
	Importance Rating	4.0

Knowledge of the operational implications of containment pressure as it applies to a High Containment Temperature.

Proposed Question:

While at 100% power, a failure of the Turbine Building Chilled Water System caused containment temperature to rise to its 90°F Technical Specification limit.

Which of the following parameters could also be expected to exceed its Tech Spec limit based on current conditions assuming no operator actions are taken?

a. Drywell temperature

b. Suppression pool temperature

c. Drywell pressure

d. Containment pressure

Proposed Answer: D

Explanation (Optional): Due to the direct relationship between pressure and temperature, containment pressure would be expected to rise as containment temperature rises.The drywell is cooled by Service Water during normal operation. Cooling of the drywell with HVN is only allowed during plant outages, therefore drywell temperature and pressure will be unaffected by this failure. Due to the large heat capacity of water, suppression pool temperature would not be expect to rise based on this failure.

Technical Reference(s): STM-403, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-403 Obj 11 &16

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.5 & b.14

Comments:

QUESTION 24 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	2
	K/A #	295013 G2.2.12
	Importance Rating	3.7

Knowledge of surveillance procedures associated with High Suppression Pool Temperature

Proposed Question:

Which of the following describes the surveillance requirement associated with suppression pool temperature monitoring?

a. Once per 12 hours when the normal temperature limit has been exceeded.

b. Every 5 minutes during operation of suppression pool cooling.

c. Every 5 minutes while testing RCIC.

d. Once per hour during SRV testing.

Proposed Answer: C.

Explanation (Optional): STP-057-0700 provides guidance for suppression pool average temperature monitoring during testing which adds heat to the suppression pool. This procedure provides direction to monitor temperature every 5 minutes.

Technical Reference(s): STP-057-0700, Rev 300

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0209 Obj. 10 &13c

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 25 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	2
	K/A #	295022 AA2.03
	Importance Rating	3.1

Ability to determine and/or interpret CRD mechanism temperature as it applies to a Loss of CRD Pumps.

Proposed Question:

Following a trip of the running CRD pump, the following annunciator is received on H13-P680:

CONT RD DRIVE HYDRAULIC SYS HIGH TEMP

Which of the following is the appropriate location to determine the current temperature of the alarming control rod(s)?

a. Local temperature indication on each Hydraulic Control Unit (HCU).

b. Temperature Acquisition and Monitoring and Recording Information System (TAMARIS).

c. OD-3 report from Plant Process Computer.

d. CRD Temperature Recorders in the Auxiliary Building.

Proposed Answer: D.

Explanation (Optional): The local temperature recorders in the Auxiliary Building is the initiating device for the alarm provided. None of the other options provide CRDM temperature indication.

Technical Reference(s): STM-0052, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0052 Obj. 10e & 14c

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.6

Comments:

QUESTION 26 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	2
	K/A #	295033 EK2.01
	Importance Rating	3.8

Knowledge of the interrelations between High Secondary Containment Area Radiation Levels and the area radiation monitoring system.

Proposed Question:

LPCS Penetration Area Radiation Monitor, RMS-RE218, has just gone into high alarm and is currently reading 100mr/hr.

This alarm means that the radiation level ___________________referenced in EOP-3 (Secondary Containment Control).

a. Exceeds the maximum safe operating value

b. Is below the maximum normal operating value

c. Is at the maximum normal operating value

d. Is at the maximum safe operating value

Proposed Answer: C.

Explanation (Optional): All DRMS high alarm setpoints are at the maximum normal operating values.

Technical Reference(s): EOP-3, Rev14

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-OPS-HLO511 Obj. E15

Question Source: Bank # NRC2007#26

Question History: Last NRC Exam 2007

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 27 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	1
	Group #	2
	K/A #	295034 EK3.02
	Importance Rating	4.1

Knowledge of the reasons for starting SBGT as it applies to a Secondary Containment Ventilation High Radiation condition.

Proposed Question:

During normal plant operation, RMS-RE110, Auxiliary Building Ventilation went into High Alarm. The Unit Operator has performed the required manual actions.

Which of the following describes the reason for starting Standby Gas Treatment in this condition?

a. To maintain negative pressure in Primary Containment to ensure offsite release rates are not exceeded.

b. To provide a radiologically controlled environment to maintain control room habitability.

c. To maintain negative pressures in the Auxiliary Building and Annulus to ensure offsite release rates are not exceeded.

d. To process all main plant stack exhaust to ensure offsite release rates are not exceeded.

Proposed Answer: C.

Explanation (Optional): With RMS-RE110 in high alarm, the operator is required to manually isolate the auxiliary building and start SGTS. In this lineup, STGS maintains negative pressures in the Aux Bldg and Annulus areas. SGTS does not draw a suction off containment under this condition, nor does it process all main plant stack exhaust; only the auxiliary building and annulus.

Technical Reference(s): STM-0257, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0257, Obj. 1 & 2

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.13

Comments:

QUESTION 28 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	203000 A1.02
	Importance Rating	3.9

Ability to predict and/or monitor changes in reactor pressure associated with the operating of RHR/LPCI injection mode.

Proposed Question:

Following a Loss of Coolant Accident, the following plant parameters exist:

Reactor pressure	450 psig
RPV level	-95 inches
Drywell pressure	1.8 psid
Containment pressure	Normal and steady

Which of the following describes the Low Pressure Coolant Injection mode of the Residual Heat Removal system?

a. Pumps have started, but are not injecting because the injection valves, E12-F042A,B, and C have not opened.

b. Pumps have started, injection valves E12-F042A, B, and C have opened, but reactor pressure is too high for injection.

c. Pumps have not started, but injection valves E12-F042A, B, and C have opened.

d. Pumps have started, injection valves E12-F042A, B, and C have opened and injection has started.

Proposed Answer: B.

Explanation (Optional): Pumps have started on high drywell pressure greater than 1.68 psid. Injection valves have opened at <487 psig. Injection has not commenced because reactor pressure is above the LPCI pump shutoff head of 339 psig.

Technical Reference(s): STM-0204, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: STM-204 Obj. 3b, 3d, 4, 10

Question Source:	Bank # 145	
Question History:	Last NRC Exam 10/2000	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.7	

Comments:

QUESTION 29 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	203000 A3.01
	Importance Rating	3.8

Ability to monitor automatic valve operation of the RHR/LPCI injection mode

Proposed Question:

A LOCA has occurred during normal plant operation.

Within 2 minutes, ECCS systems have recovered reactor water level. The CRS has directed the unit operator to place RHR ‘A’ in the Suppression Pool Cooling mode.

If the LOCA signal was received 7 minutes ago, which of the following represents the current status of the Suppression Pool Cooling flowpath?

a. E12-F048A RHR A HX BYPASS cannot be maintained CLOSED for another 3 minutes. E12-F024A RHR PUMP A TEST RTN TO SUP PL can be manually overridden immediately via handswitch.

b. E12-F048A RHR A HX BYPASS cannot be OPENED for another 3 minutes. E12-F024A RHR PUMP A TEST RTN TO SUP PL can be manually overridden immediately via handswitch.

c. E12-F048A RHR A HX BYPASS can be OPENED immediately. E12-F024A RHR PUMP A TEST RTN TO SUP PL cannot be opened for another 3 minutes.

d. E12-F048A RHR A HX BYPASS can be CLOSED immediately. E12-F024A RHR PUMP A TEST RTN TO SUP PL cannot be opened for another 3 minutes.

Proposed Answer: A

Explanation (Optional): E12-F048A RHR A HX BYPASS receives on open signal for 10 minutes following a LOCA signal to ensure maximum ECCS flow is provided to the RPV. E12-F024A RHR PUMP A TEST RTN TO SUP PL receives a signal to close during a LOCA, but can be manually overridden at any time.

Technical Reference(s): STM-204, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-STM-0204, Obj.6	
Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒ 3
10 CFR Part 55 Content:	55.41 b.7	

Comments:

QUESTION 30 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	205000 K3.03
	Importance Rating	3.8

Knowledge of the effect that a loss or malfunction of the Shutdown Cooling System will have on reactor temperatures (moderator, vessel, flange)

Proposed Question:

During a refueling outage, a loss of power condition has caused the operating Shutdown Cooling Loop to trip. The alternate loop is NOT available.

Which of the following describes the effect of the condition on the reactor vessel and moderator?

a. The heatup will cause the amount of available shutdown margin to increase and the 80°F per hour Tech Spec limit may be exceeded if cooling is not restored.

b. The heatup will cause the amount of available shutdown margin to decrease and the 100°F per hour Tech Spec limit may be exceeded if cooling is not restored.

c. The heatup will cause the amount of available shutdown margin to decrease and the 80°F per hour Tech Spec limit may be exceeded if cooling is not restored.

d. The heatup will cause the amount of available shutdown margin to increase and the 100°F per hour Tech Spec limit may be exceeded if cooling is not restored.

Proposed Answer: D.

Explanation (Optional): Technical Specification limits reactor coolant system heatups and cooldowns to 100°F per hour. Shutdown margin increases with increasing temperature.

Technical Reference(s): TS 3.4.11, HLO-175, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: HLO-175 Obj. 11

Question Source: New

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒4

10 CFR Part 55 Content: 55.41 b.3, b.14

Comments:

QUESTION 31 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	209001 K2.01
	Importance Rating	3.0

Knowledge of the electrical power supply to the LPCS pump.

Proposed Question:

Which of the following is the electrical power supply for E21-PC001, Low Pressure Core Spray Pump?

a. E22-S002

b. ENS-SWG1A

c. ENS-SWG1B

d. EJS-SWG1A

Proposed Answer: B.

Explanation (Optional):

Technical Reference(s): STM-205, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-205 Obj 17a

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 32 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	209001 A3.01
	Importance Rating	3.6

Ability to monitor automatic valve operations of LPCS.

Proposed Question:

Given the following conditions:

The Low Pressure Core Spray (LPCS) system is running in the test return to the suppression pool mode.

A leak has caused drywell pressure to increase to 1.95 psid.
Reactor water level is -62 inches
Reactor pressure is 750 psig

Select the expected AUTOMATIC response of the LPCS system.

a. The LPCS Pump trips, the Test Return Valve to the Suppression Pool (E21-F012) closes, the Pump restarts and Injection Isolation Valve (E21-F005) opens.

b. The Test Return Valve to the Suppression Pool (E21-F012) closes and the LPCS Pump continues to run on minimum flow.

c. The LPCS Pump trips, the Test Return Valve to the Suppression Pool (E21-F012) closes, and the Pump restarts and runs on minimum flow.

d. The LPCS Pump continues to run, the Test Return Valve to the Suppression Pool (E21-F012) closes and the Injection Isolation Valve (E21-F005) opens.

Proposed Answer: B.

Explanation (Optional): Based on the conditions provided, the pump will not trip. Only load shedding and sequencing would cause the pump to trip and restart. The injection valve (E21-F005) will not open with reactor pressure at 750 psig. The Test Return (E21-F012) will shut on the high drywell pressure signal. The low flow condition will cause the minimum flow valve to open.

Technical Reference(s): STM-205, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-STM-0205 Obj 5		
Question Source:	Bank #	468	
Question History:	Last NRC Exam	1/1997	
Question Cognitive Level:	Memory or Fundamental Knowledge		☐
	Comprehension or Analysis		☒3
10 CFR Part 55 Content:	55.41 b.7		

Comments:

QUESTION 33 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	209002 A3.06
	Importance Rating	2.8

Ability to monitor lights and alarms associated automatic operation of HPCS.

Proposed Question:

During the recovery from an ATWS condition, the following light indications are present for the HPCS system on H13-P601:

Directly above E22-ACB02 HPCS PUMP SUPPLY BRKR Control Switch

Green Light	ON
Amber Light	ON
White Light	ON
Red Light	OFF

HPCS MANUAL OVERRIDE

Amber light	OFF

Directly above E22-F004 HPCS INJECTION ISOL VALVE

Green light	ON
Amber light	ON
Red light	OFF

Based on the indications provided, which of the following describes the current status of HPCS?

a. E22-F004 will open if a Level 2 signal is received. The HPCS Pump has been overriden.

b. E22-F004 will NOT open if a Level 2 signal is received. The HPCS pump is overridden.

c. E22-F004 will open if a Level 2 signal is received. The HPCS pump has tripped.

d. E22-F004 will NOT open if a Level 2 signal is received. The HPCS pump has tripped.

Proposed Answer: D.

Explanation (Optional): The amber light above E22-F004 HPCS Injection Isol Valve indicates that the valve has been manually overridden. When in this condition, the valve will not open on Level 2. The amber light above the HPCS pump breaker control switch indicates that the breaker has tripped. The indication that the HPCS pump has not been overridden is indicated by the.HPCS PUMP MANUAL OVERRIDE amber light being OFF.

Technical Reference(s): STM-203, Rev 6

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP- OPS-203, Obj 6 & 7

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 34 Rev 0

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

Ability to predict and/or monitor changes in pump discharge pressure associated with operating the Standby Liquid Control System.

Proposed Question:

While operating at 100% power an ATWS occurred.

The MSIVs are closed.

Reactor pressure has been lowered to 650 psig to maintain in the safe zone of the Heat Capacity Temperature Limit curve.

The CRS has directed injection with Standby Liquid Control.

Which of the following is indicative of proper SLC operation under these conditions?

a. SLC pump discharge pressure 750 psig, SLC squib continuity light OFF

b. SLC pump discharge pressure 1400 psig, SLC squib continuity light OFF

c. SLC pump discharge pressure 1400 psig, SLC squib continuity light ON

d. SLC pump discharge pressure 750 psig, SLC squib continuity light ON

Proposed Answer: A.

Explanation (Optional): Proper SLC operation occurs when the squib valve has been fired. This is indicated by the continuity light being extinguished. Proper SLC discharge pressure is slightly above reactor pressure. SLC discharge line relief valves lift at 1400. A pressure this high is indicative of blockage in the discharge line.

Technical Reference(s): STM-0201, Rev 4

Proposed references to be provided to applicants during examination:NA

Learning Objective: RLP-STM-0201 Obj 2e

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.6

Comments:

QUESTION 35 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	212000 K3.09
	Importance Rating	3.2

Knowledge of the effect that a loss or malfunction of RPS will have on the magnitude of heat energy that must be absorbed by the containment during accident/transient conditions.

Proposed Question:

During an ATWS condition, EOP-1A directs the operators to install the following Enclosures:

ENCLOSURE 16 Defeating Containment Instrument Air Isolation
ENCLOSURE 24 Defeating RPV Low Level 1 MSIV and MSL Drains Isolation Interlocks
ENCLOSURE 34 Defeating Offgas High Radiation Isolation Interlocks

Why does the procedure direct the performance of these three actions?

a. To ensure the availability of the Standby Liquid Control tank level indication.

b. To minimize the amount of heat energy being absorbed by containment.

c. To allow resetting RPS by preventing an MSIV closure signal.

d. To ensure air is available to the scram discharge volume vents and drains.

Proposed Answer: B.

Explanation (Optional):. Installation of these 3 enclosure ensures that the Main Condenser is maintained available as a heat sink. Encl 16 maintains air to the MSIVs, Encl 24 bypasses the MSIV Level 1 isolation and Encl 34 aids in maintaining condenser vaccum by bypassing any Ofg high radiation signal.

Technical Reference(s): EOP-1A Rev 21 EPSTG-0002 Rev 12

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-0513 Obj. 4

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒2
	Comprehension or Analysis	☐
10 CFR Part 55 Content:	55.41 b.8, 10	
Comments:		

QUESTION 36 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	215003 K4.04
	Importance Rating	2.9

Knowledge of the IRM design feature and/or interlocks that provide for varying system sensitivity levels using range switches.

Proposed Question:

Given the following plant conditions:

Reactor startup in progress.
IRM “C” indicating 36/125 on Range 4

Select the statement that best describes the response of the plant if IRM “C” is inadvertently ranged down by the operator depressing the down range button.

a. Control rod movement can continue as normal.

b. Only a rod block will be initiated.

c. Only a half-scram will be initiated.

d. A rod block and a half-scram will be initiated.

Proposed Answer: B.

Explanation (Optional): At the current power level, placing IRM “C” on Range 3 will result in the IRM displaying a value of 36/40 since Range 3 and 4 are of the same decade. Control Rod Block is initiated at 108/125 (equivalent to 34.5/40 on the odd range scale) therefore a rod block would be present. An RPS trip is initiated at 120/125 (equivalent to 38.4/40 on the odd range scale. With a value of 36/40, IRM “C” is between the rod block and half scram setpoints, therefore only a rod block will be initiated.

Technical Reference(s): R-STM-0503, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0503 Obj. 12 & 13

Question Source: Modified Bank # NRC 11 (Changed original value of IRM reading (from 75/125 to 36/125) such that correct answer changed from D to B.

Also original question stated that a shutdown was in progress. This left A as a possibly correct answer since the rod block signal is a withdrawal block not an insertion block).

Question History:	Last NRC Exam 7/1997	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒ 3
10 CFR Part 55 Content:	55.41 b.7	

Comments:

QUESTION 37 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	215004 A2.02
	Importance Rating	3.4

Ability to predict the impact of an SRM inop and based on those predictions, use procedures to correct, control, or mitigate the consequences of that condition.

Proposed Question:

The plant is operating in Mode 2 when the following annunciator is received:

SRM UPSCALE OR INOPERATIVE

All IRMs are on Range 2.

<u>SRM Countrates</u>:

SRM A	$2x10^3$ cps
SRM B	$8x10^2$ cps
SRM C	$3x10^4$ cps
SRM D	$3x10^5$ cps

The shorting links are installed

Reactor period has lengthened to 400 seconds. The reactor engineer has requested that additional control rods be withdrawn.

With present plant conditions, which of the following is correct regarding reactor status?

a. A rod block is present, but may be cleared by withdrawing SRM D to maintain count rates between $1x10^3$ and $1x10^5$ cps.

b. A rod block is present, but SRM D may not be withdrawn until all IRMs are on Range 3.

c. A rod block is present, but may be cleared, by fully withdrawing SRM D since all IRMs are on Range 2.

d. No rod block exists.

Proposed Answer: A

Explanation (Optional): GOP-0001 directs withdrawal of SRM detectors to maintain count rates between $1x10^3$ and $1x10^5$ cps. The short links bypass the SRM RPS trip, but not the control rod withdrawal block. SRMs may not be fully withdrawn until all IRMs are on Range 3 or above. A control rod block is received when SRM count rate exceeds $1x10^5$ cps.

Technical Reference(s): STM-503, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0503 Obj. 1, 4 & 7

Question Source: Modified Bank # INPO 16346
Significantly modified distractors to fit River Bend System

Question History: Last NRC Exam Grand Gulf 4/2000

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

10 CFR Part 55 Content: 55.41 b.2, b.7, b.10

Comments:

QUESTION 38 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	215005 K2.02
	Importance Rating	2.6

Knowledge of electrical power supplies to APRM channels.

Proposed Question:

A loss of RPS bus 'B' will cause a loss of power to which of the following?

a. APRMs A, B, C, D

b. APRMs E, F, G, H

c. APRMs A, C, E, G

d. APRMs B, D, F, H

Proposed Answer: D

Explanation (Optional):

Technical Reference(s): STM-503, Neutron Monitoring Instruments System, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective: Obj 27 b

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 39 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	217000 A3.04
	Importance Rating	3.6

Ability to monitor system flow during automatic operation of RCIC.

Proposed Question:

A small reactor coolant system leak has occurred, along with a loss of offsite power.

Level has decreased to -110 inches, but is now slowly rising using RCIC injection only. RCIC automatically initiated on Level 2.

Drywell pressure	2.6 psid
Sup Pool Level	20 feet 6 inches

Forty minutes have passed and the only operator actions taken have been to lower reactor pressure to 700 psig and to verify proper operation of automatic features.

Which of the following would be the expected RCIC system operation?

a. RCIC will be injecting at approximately 1000 gpm since reactor pressure has been lowered with suction from the suppression pool.

b. RCIC will be tripped due to low suction pressure.

c. RCIC will be injecting 600 gpm with suction from the CST.

d. RCIC will be injecting 600 gpm with suction from the Suppression Pool.

Proposed Answer: D.

Explanation (Optional):

RCIC flow controller is set to 600 gpm. It will inject the flowrate at any pressure. High suppression pool level (20’3.5”) will cause a RCIC suction swap to the suppression pool.

Technical Reference(s): STM-0209, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-209 Obj.5i & 12

Question Source:	Bank #	275	
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		☐
	Comprehension or Analysis		☒3
10 CFR Part 55 Content:	55.41 b.7		
Comments:			

QUESTION 40 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	218000 A4.09
	Importance Rating	3.9

Ability to manually operate and/or monitor suppression pool temperature in the control room.

Proposed Question:

The plant is in a casualty situation and Automatic Depressurization (ADS) has automatically initiated. The following conditions exist:

Reactor Pressure	0 psig
Containment Temperature	165°F
Suppression Pool Level	15 feet 6 inches

RHR ‘A’ is running in Suppression Pool Cooling mode
RHR B is injecting into the RPV

Which of the following Main Control Room indications provides the most accurate suppression pool temperature indication?

a. CMS-TR24A and CMS-TR24B recorders on H13-P808

b. CMS-TR40A and CMS-TR40B recorders on H13-P808

c. E12-R601 RHR Temperature recorder – Point 1 RHR inlet to HX1 A1 (E12-N004A) on H13-P601

d. E12-R601 RHR Temperature recorder – Point 2 RHR inlet to HX1 B1 (E12-N004B) on H13-P601

Proposed Answer: C.

Explanation (Optional): CMS-TR24 recorders are not accurate with SP Level below 19’3”. CMS-TR40 recorders are not accurate with SP Level below 16’. Point 2 is not accurate without flow through the heat exchanger. With RHR ‘A’ in SP Cooling mode, Point 1 on E12-R601 provides accurate SP Temp.

Technical Reference(s): STM-0057 Rev1; EOP-0001 Caution 8

Proposed references to be provided to applicants during examination:EOP Cautions 3-8

Learning Objective: RLP-HLO-0511 Obj. 6

Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.7	
Comments:		

QUESTION 41 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	223002 K4.07
	Importance Rating	2.8

Knowledge of NSSSS design features which provide for physical separation of system components (to prevent localized environmental factors, electrical faults, and physical events from impairing system response).

Proposed Question:

G33-F004 RWCU Outboard Isolation Valve has isolated due to a ground fault in the isolation logic circuit.

Which of the following would also be affected by this fault?

a. B21-F028A MSL A OTBD MSIV will close

b. G33-F001 RWCU Inboard Isolation Valve will close

c. G33-F054 RWCU Outboard Isolation Valve will close

d. B21-F022A MSL A INBD MSIV will close

Proposed Answer: C.

Explanation (Optional): B21-F028A & B21-F022A will not close, due to physical and electrical separation. G33-F001 also will not be affected due to physical and electrical separation. G33-F054 is in the same division and utilizes the same isolation logic. There is no physical, nor electrical separation.

Technical Reference(s): STM-0058 Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0058 Obj. 11c

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 42 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	239002 K3.02
	Importance Rating	4.2

Knowledge of the effect that a loss or malfunction of the Relief/Safety Valves will have on reactor over pressurization.

Proposed Question:

A loss of 125VDC has rendered automatic SRV relief operation unavailable due to de-energization of both solenoids.

How does this condition affect SRV operation and the Reactor Coolant System Pressure Safety Limit?

a. The safety limit will not be exceeded provided the SRVs are opened manually with their handswitches.

b. The safety limit will not be exceeded because the SRVs will still function in ADS (Automatic Depressurization System) mode.

c. The safety limit will not be exceeded because the SRV will lift in Safety mode prior to reaching the limit.

d. The SRVs will lift in Safety mode, but not prior to exceeding the safety limit.

Proposed Answer: C.

Explanation (Optional): SRVs will not open in any mode other than Safety when the solenoids are de-energized. The safety limit is 1325 psig. All 16 SRVs would have opened in Safety Mode by 1210 psig therefore the Safety limit will not be exceeded..

Technical Reference(s): Technical Specification 2.0; STM-109, Rev 1; STM-202, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-109 Obj 3b, 21b, 4a, 24b; RLP-HLO-401 Obj. 2

Question Source: New

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒3

10 CFR Part 55 Content: 55.41 b.3 & b.7

Comments:

QUESTION 43 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	259002 K1.03
	Importance Rating	3.8

Knowledge of the physical connections and/or cause-effect relationships between Reactor Water Level Control System and reactor water level.

Proposed Question:

The plant is at 100% steady state power.

The 'B' Feedwater Regulating Valve has failed closed.

What is the expected plant response to this malfunction with no operator action?

a. 'A' and 'C' Feedwater Regulating will open to stabilize level in the normal range with reactor power remaining at 100%.

b. 'A' and 'C' Feedwater Regulating valves will open but will be unable to maintain vessel level. The reactor will scram on low water level.

c. Reactor Recirculation Flow Control Valves will run back. The 'A' and 'C; Feedwater Regulating Valves will stabilize level with the plant at a lower power level.

d. Reactor Recirculation Flow Control Valves will run back. The 'A' and 'C' Feedwater Regulating Valves will be unable to maintain vessel level. The reactor will scram on low water level.

Proposed Answer: B.

Explanation (Optional): A 100% power, 3 feedwater pumps are in service. The runback signal only occurs if less than 3 FWS pumps are running. Two Feed Reg Valves alone can not maintain water level at 100% power.

Technical Reference(s): STM-0107, Rev 10

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0107 Obj. 16f

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.3	
Comments:		

QUESTION 44 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	261000 K3.01
	Importance Rating	3.3

Knowledge of the effect that a loss or malfunction of the Standby Gas Treatment System will have on secondary containment and environment differential pressure.

Proposed Question:

Following a LOCA condition, the following abnormal parameters were observed:

Auxiliary Building pressure	+0.15 psig
Annulus pressure	+0.10 psig

Which of the following is responsible for BOTH of these abnormal conditions?

a. Failure of the Auxiliary Building Supply Fans to trip when the associated Exhaust Fans tripped.

b. Trip of both Annulus Pressure Control fans.

c. Trip of both Auxiliary Building Exhaust fans.

d. Failure of Standby Gas Treatment to initiate when required.

Proposed Answer: D.

Explanation (Optional): On a LOCA signal, Standby Gas Treatment initiates and aligns to draw negative pressure on the Auxiliary Bldg and the Annulus. During a LOCA, the annulus pressure control fans, auxiliary bldg supply fans and exhaust fans are all isolated and secured.

Technical Reference(s): STM-257, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0257 Obj. 11d

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.13

Comments:

QUESTION 45 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	261000 A4.09
	Importance Rating	2.7

Ability to manually operate and/or monitor ventilation valves and dampers in the control room.

Proposed Question:

Standby Gas Treatment Exhaust Fan GTS-FN1A must be manually started from the control room by _______________.

a. Opening GTS-AOD1A, SGT FILTER A SUCT ISOL valve before depressing the START pushbutton

b. Opening GTS-AOD3A, SGT EXH FAN A DISCH valve before depressing the START pushbutton

c. Opening both GTS-AOD1A, SGT FILTER A SUCT ISOL and GTS-AOD3A,SGT EXH FAN A DISCH valves before depressing the START pushbutton

d. Depressing the START pushbutton until GTS-AOD1A, SGT FILTER A SUCT ISOL valve opens and the fan starts

Proposed Answer: D.

Explanation (Optional): Depressing the start pushbutton sequences opening the suction damper and starting the fan motor once the damper is fully open.

Technical Reference(s): STM-257, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0257, Obj. 3g, 5b

Question Source: Bank # NRC

Question History: Last NRC Exam 2007

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 46 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	262001 K5.02
	Importance Rating	2.6

Knowledge of the operational implications of breaker control as it applies to the AC electrical distribution system.

Proposed Question:

A 4160 volt ITE type breaker was racked in following maintenance on its associated pump. The breaker was then closed. Thirty minutes later, the breaker experienced a loss of DC control power. No operator actions were taken.

Which of the following describes the operational capabilities of this breaker?

a. The breaker will trip open on loss of control power and no further breaker operations are possible.

b. The breaker will trip open on loss of control power and all additional breaker operations must be performed locally.

c. The breaker cannot be remotely operated but can be locally tripped, then closed and tripped open one more time.

d. The breaker cannot be remotely operated but can be locally tripped one time with no further operation possible.

Proposed Answer: C.

Explanation (Optional): Breakers do not trip open on loss of control power. No remote operation is available when control power is loss, but local tripping is always available. When the breaker was closed prior to the loss of control power, the charging motor energized to charge the springs, so even after control power was loss, the springs were charged and available for a subsequent closure.

Technical Reference(s): STM-300, Rev 11

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0300 Obj. 14a

Question Source: Bank # 1105

Question History: Last NRC Exam 1/1997

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 47 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	262001 A4.03
	Importance Rating	3.2

Ability to manually operate and/or monitor local operation of breakers in the control room.

Proposed Question:

The CRS notifies the Unit Operator that the Control Building operator will be performing a breaker test on ENS-ACB03, E12-C002A RHR A PUMP breaker. The breaker will be racked to the TEST position, control power fuses INSTALLED, and the breaker will be CLOSED to support maintenance testing.

Which of the following represents the expected H13-P601 light indications for the RHR A pump breaker when the test conditions mentioned above are established?

a. Red light OFF, Green light OFF, White light OFF

b. Red light OFF, Green light OFF, White light ON

c. Red light ON, Green light OFF, White light OFF

d. Red light ON, Green light OFF, White light ON

Proposed Answer: C

Explanation (Optional): Breaker position indication will be available in the MCR when the control power fuses are installed. The white light however will extinguish if the breaker is not fully racked in to the OPERATE position due to the 52H contact being open.

Technical Reference(s): ESK-05RHS01

Proposed references to be provided to applicants during examination: NA

Learning Objective: HLO-157 Obj 7 & 12; RLP-STM-300 Obj 5

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 48 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	262002 K4.01
	Importance Rating	3.1

Knowledge of the UPS design feature and/or interlocks which provide for the transfer from preferred power to alternate power supplies.

Proposed Question:

Uninterruptible Power Supply ENB-INV01A is in its normal lineup when a malfunction of the inverter section occurs bringing the inverter output to 0 volts.

What is the expected response of the UPS ENB-INV01A?

a. The UPS will transfer to the battery backup and continue carrying the bus load.

b. The supply to the bus will continue as normal as the inverter section only provides power with the UPS in BYPASS mode.

c. The UPS static transfer switch will transfer and provide power to the bus loads via the alternate power supply.

d. The UPS will not maintain bus voltage due to a LOSS OF SYNCH preventing transfer.

Proposed Answer: C.

Explanation (Optional): The battery backup requires the inverter to provide power therefore 'A' is incorrect. 'B' would be true if in BYPASS, but the normal lineup is through the inverter which is not associated with the BYPASS lineup.The static transfer switch will transfer from the inverter output to the alternate source and maintain bus loads energized. No Loss of Synch conditions exists. The bus will transfer and maintain bus loads.

Technical Reference(s): STM-300 Rev 11

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0300 Obj. H15

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b. 5

Comments:

QUESTION 49 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	263000 K1.02
	Importance Rating	3.2

Knowledge of the physical connections and/or cause-effect relationships between DC electrical distribution system and the battery charger and batteries.

Proposed Question:

Which of the following accurately describes the safety related 125 VDC Electrical Distribution System during NORMAL operation?

a. An ENB charger supplies the ENB switchgear which supplies the ENB battery.

b. An ENB charger supplies the ENB batteries which supply the ENB switchgear.

c. An ENB Inverter supplies the ENB switchgear which supplies the ENB battery.

d. An ENB battery supplies the ENB switchgear which supplies an ENB inverter.

Proposed Answer: A.

Explanation (Optional): During normal operation, the switchgear voltage is maintained by the charger. The battery is a load on the switchgear. If the charger is lost, the battery then becomes the supply voltage.

Technical Reference(s): STM-0305, Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0305 Obj. 2, 12b

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 50 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	263000 A2.02
	Importance Rating	2.6

Ability to predict the impact of a loss of ventilation during charging and based on those predictions, use procedures to correct, control, or mitigate the consequence of this condition.

Proposed Question:

With an equalize charge in progress on ENB-BAT1A, the running battery room fan tripped. The associated standby fan failed to automatically start.

Which of the following represents the expected operator response to this condition and reason for the reponse?

a. No action necessary at this time. Ventilation may be secured for up to 96 hours.

b. Attempt to manually start the standby fan, or provide a temporary ventilation system due to explosive concentrations of hydrogen that can build up during charging.

c. Attempt to manually start the standby fan, or provide a temporary ventilation system to avoid excessive room temperatures beyond the EQ limit.

d. Prop open the battery room door to provide cooling from other areas to minimize exceeding EQ temperature limits.

Proposed Answer: B.

Explanation (Optional): Ventilation fans provide no cooling, but do ensure that hydrogen does not build up in the battery rooms. The 96 hour limit provide does not apply when a battery is being charged at a rate higher than float charge.

Technical Reference(s): SOP-0058, Rev 20

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0058 Obj. 12

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 51 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	264000 A1.03
	Importance Rating	2.8

Ability to predict and/or monitor changes in operating voltages, currents, and temperatures associated with the Emergency Diesel Generators.

Proposed Question:

The monthly surveillance run for the Division 1 diesel generator is in progress. The diesel is synchronized to the bus.

If the voltage regulator control switch is taken to the LOWER position, the diesel generator real load (KW) will _____________ and diesel generator reactive load (KVA) will ____________.

a. decrease; be unchanged

b. decrease; decrease

c. be unchanged; decrease

d. be unchanged; be unchanged

Proposed Answer: C.

Explanation (Optional):

Technical Reference(s): HLO-154 Obj. 20

Proposed references to be provided to applicants during examination: NA

Learning Objective: HLO-154 Obj 20

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 52 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	300000 A2.01
	Importance Rating	2.9

Ability to predict the impact of air dryer and filter malfunctions on the Instrument Air System and based on those predictions, use procedures to correct, control, or mitigate the consequences of this condition.

Proposed Question:

The plant is operating at 100% power with IAS-DRY2 in service.

A failure of IAS-DRY2 regeneration solenoids has resulted in IAS being vented to atmosphere via the regeneration purge line.

What impact will the failure of IAS-DRY2 have on the Instrument Air System and how will the system be restored?

IAS air pressure will lower until IAS-AOV300A IAS-DRY2 PURGE ISOLATION VALVE…

a. …isolates. Local operator action will be required to open IAS-AOV300A when the failed solenoid is repaired.

b. …opens. Local operator action will be required to close IAS-AOV300A when the failed solenoid is repaired.

c. …isolates. IAS-AOV300A will automatically open when IAS pressure is restored to normal.

d. …opens. IAS-AOV300A will automatically close when IAS pressure is restored to normal.

Proposed Answer: A

Explanation (Optional): IAS-AOV300A isolates at 113 psig IAS header pressure to prevent a dryer component failure from causing a complete loss of IAS. The isolation will stop the depressurization therefore automatic re-opening of IAS-AOV300A is not desirable when pressure is restored. Manual action is required at IAS-PNL31.

Technical Reference(s): STM-0121, Rev 6

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0121 Obj. 3c

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.4

Comments:

QUESTION 53 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	1
	K/A #	400000 AK4.01
	Importance Rating	3.4

Knowledge of the CCW design feature and/or interlocks which provide for the automatic start of the standby pump.

Proposed Question:

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

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

What automatic feature failed to function as designed?

a. Trip of the running CRD pump.

b. Start of the standby pump, CCP-P1C.

c. Initiation of both Standby Service Water Divisions.

d. Isolation of cooling water to the CCP heat exchangers.

Proposed Answer: B.

Explanation (Optional):Only the standby pump auto start feature utilizes a single transmitter for initiation. The trip of a running CRD pump would occur if either CCP vital loop sensed <56 psig. The initiation of both SSW divisions would require sensing a ,56 psig signal in both CCP vital loops. The isolation of the CCP heat exchangers would occur if <56 psig was sensed in the Div 2 CCP vital loop.

Technical Reference(s): STM-115, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0115 Obj. 4

Question Source: Bank # 405

Question History:	Last NRC Exam 2004	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.4	

Comments:

QUESTION 54 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	201001 G2.2.39
	Importance Rating	3.9

Knowledge of less than or equal to one hour Technical Specification action statements for CRD Hydraulics.

Proposed Question:

While operating at 100% power, the plant experienced a trip of CRD Pump "A" due to a significant water leak. The water from the leak has impinged upon CRD Pump "B" resulting in a failure to start due to grounding of the motor. Consider the following timeline:

CRD Pump A trip	0915
1st Accumulator Fault	0921
2nd Accumulator Fault	0925

Based on the information above, what is the required action for this condition?

a. Restore charging water header pressure to ≥1540 psig by 1015, or declare the associated control rod accumulators SLOW.

b. Restore charging water header pressure to ≥1540 psig by 0945 or place the Mode Switch in Shutdown.

c. Restore charging water header pressure to ≥1540 psig by 0941 or place the Mode Switch in Shutdown.

d. Restore charging water header to ≥1540 by 1025 or declare the associated control accumulators SLOW.

Proposed Answer: B.

Explanation (Optional): Tech Specs require charging water header to be restore within 20 minutes of the second accumulator fault..

Technical Reference(s): LCO 3.1.5 Condition B.

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0052 Obj. 12

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.10	
Comments:		

QUESTION 55 Rev 0

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

Ability to predict and/or monitor changes in first stage shell pressure associated with RCIS

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 Control Plan has the ATC operator select and continuously withdraw control rod 28-49 from position 12 to position 24.

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

Which one of the following describes the response of Control Rod 28-49 when the ATC operator attempts withdrawal under this condition?

a. Control Rod 28-49 will remain at position 12 due to a control block generated from a failure of the First Stage Shell Pressure transmitter.

b. Control Rod 28-49 will withdraw to position 20 and settle due to the withdrawal limitations between the Low Power Setpoint and the High Power Setpoint.

c. Control Rod 28-49 will withdraw to position 16 and settle due to the withdrawal limitations imposed by the High Power Setpoint.

d. Control Rod 28-49 will withdraw to position 14 and settle due to the single notch withdrawal constraints of the Rod Pattern Controller.

Proposed Answer: C.

Explanation (Optional):Rod withdrawal limitations are dependent on reactor power as sensed by First Stage Turbine Pressure. An upscale failure will indicate to the RC&IS system that reactor power is above the HPSP. The RWL will then limit rod withdrawals to 2 notch positions.

Technical Reference(s): STM-0500, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-STM-0500 Obj. 22a	
Question Source:	Bank #	665
Question History:	Last NRC Exam	2/1999
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.6	

Comments: Original question provided a copy of the reactivity plan as a reference to the candidate. Determined it was unnecessary.

QUESTION 56 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	202001 K6.09
	Importance Rating	3.4

Knowledge of the effect that a loss of reactor water level will have on the Recirculation System.

Proposed Question:

Following a transient, the following plant conditions exist.

Reactor power 0%
Reactor pressure 875 psig
Reactor level -20 inches slowly rising
All ECCS systems are in standby
All systems worked as designed and no operator action has been taken.

Which of the following represents the status of the Reactor Recirculation System?

a. Reactor Recirculation pumps are OFF with cooling water AVAILABLE.

b. Reactor Recirculation pumps are in SLOW speed with cooling water UNAVAILABLE

c. Reactor Recirculation pumps are in SLOW speed with cooling water AVAILABLE.

d. Reactor Recirculation pumps are OFF with cooling water UNAVAILABLE.

Proposed Answer: C.

Explanation (Optional): Level is less than Level 3, but above Level 2 therefore, the Recirc pumps are running in slow speed. Cooling water isolates on Level 2 or 1.68 psid. ECCS systems, specifically HPCS being in standby indicates that neither Level 2, nor 1.68 psid signal has been received.

Technical Reference(s): AOP-0003, Rev 26 STM-053, Rev 1

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0503 Obj 20f & 20k

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒2
10 CFR Part 55 Content:	55.41 b.6	
Comments:		

QUESTION 57 Rev 0

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

Knowledge of the physical connections and/or cause effect relationships between Recirculation Flow Control and reactor water level.

Proposed Question:

The plant is operating at 100% power.
“A” narrow range level channel is selected as input to Feedwater Level Control.

A leak in the reference leg of the “A” narrow range level transmitter has altered the level input to the Feedwater Level Control System. The ATC operator promptly placed the Master Feedwater Controller in Manual in accordance with AOP-0006.

As a result of this condition, both Reactor Recirculation Pumps will…

a. remain at present speed, however the Recirc Flow Control Valves will runback to minimum position.

b. transfer to SLOW speed operation, with the Recirc Flow Control Valves remaining at their present position.

c. transfer to SLOW speed operation, with the Recirc Flow Control Valves running back to 60% drive flow position.

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

Proposed Answer: D.

Explanation (Optional): The Recirc Pump transfer and FCV runback logics receive input from the narrow range channel selected for FWLC. A leak on the reference leg will cause transmitter differential pressure to lower and hence indicated level to rise, therefore the Recirc system will not receive a Level 3 or Level 4 input to cause a speed transfer or runback.

Technical Reference(s): STM-107, Rev 10; STM-503, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-107 Obj 10c, 13c; RLP-STM-503, Rev 2 Obj 20i

Question Source:	Modified Bank #	RBS NRC 9 (Changed failure from a flow transmitter to a level transmitter and change distractor C to make it plausible based on transmitter change.)
Question History:	Last NRC Exam	2/2003
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.41 b.6	

Comments:

QUESTION 58 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	216000 A1.03
	Importance Rating	2.9

Ability to predict and/or monitor changes in parameters associated with operating the Nuclear Boiler Instrumentation including surveillance testing.

Proposed Question:

With the plant operating in Mode 2 at 150 psig, Narrow Range Reactor Water Level indication reads 35 inches.

What is the expected indication on Wide Range level instrumentation under these conditions?

a. Wide Range instrumentation is Upscale.

b. Wide Range instrumentation reads 10 inches.

c. Wide Range instrumentation is Downscale.

d. Wide Range instrumentation reads 35 inches.

Proposed Answer: A.

Explanation (Optional): Narrow Range instrumentation is calibrated for 1055 psig and 130°F drywell temperature.. Wide Range instrumentation is also calibrated for 1055 psig and 130°F drywell temperature. Since both instruments are not at calibrated conditions, both will experience a differential between actual and indicated level. The differential is due to the density of the water in the reference leg. Due to the temperature in the vicinity of the reference leg being lower than calibrated conditions, both instruments will display an indicated level that is higher than actual. The effect is more pronounced on the Wide Range instrument since its operating range is 220 inches (-160” to 60”) versus 60 inches (0-60”) for Narrow Range.

Technical Reference(s): STM-0051, Rev 3 Ref Fig 11 & 12

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0051 Obj. 5e

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 59 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	234000 A2.03
	Importance Rating	2.8

Ability to predict the impacts of a loss of electrical power on the Fuel Handling Equipment and based on those predictions, use procedures to correct, control, or mitigate the consequences of this condition.

Proposed Question:

While moving a fuel bundle from the vessel to the spent fuel pool, a loss of offsite power occurred. The fuel bundle is presently hanging one foot above the top of the spent fuel storage rack.

All emergency diesel generators are supplying their respective loads.

Under these conditions…

a. the main hoist motor will be without power. A hand crank is available to lower the bundle.

b. the main hoist motor will automatically swap to an alternate safety related power source.

c. the main hoist motor must be manually swapped to its alternate safety related power source.

d. the main hoist motor will be without power. The bundle must be left in its current location until electrical power is restored.

Proposed Answer: A.

Explanation (Optional): The fuel handling and refueling platforms have a non-safety related power source. There is no alternate power source available for automatic nor manual alignment. A handcrank is provided to operate the main hoist winch to place a bundle in a safe location should a loss of power to the bridge occur.

Technical Reference(s): STM-0055, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0055 Obj 13a

Question Source:	Bank #	INPO#25946	
Question History:	Last NRC Exam	Pilgrim 10/2003	
Question Cognitive Level:	Memory or Fundamental Knowledge		☒3
	Comprehension or Analysis		☐
10 CFR Part 55 Content:	55.41 b.13		
Comments:			

QUESTION 60 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	239001 G2.4.20
	Importance Rating	3.8

Knowledge of the operational implications of EOP warnings, cautions, and notes as related to main and reheat steam..

Proposed Question:

An override in EOP-1, Step RP-1 states:

IF	THEN
Emergency Depressurization is anticipated	Rapidly depressurize the RPV with the Main Turbine Bypass Valves and Main Steam Line drains ➢ OK to exceed 100°F/Hr cooldown rate

The purpose for this override is to conserve margin to the ___________ prior to an emergency depressurization being required.

a. heat capacity temperature limit (HCTL)

b. RPV saturation temperature limit (RPVST)

c. pressure suppression pressure (PSP)

d. primary containment pressure limit (PCPL)

Proposed Answer: A.

Explanation (Optional): Per the EOP bases, anticipating ED is to send steam to the main condenser to conserve suppression poll heat capacity prior to emergency depressurization.

Technical Reference(s): EOP-0001 Step RP-1 and bases.

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-OPS-HLO-512 Obj. 5

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒3
	Comprehension or Analysis	☐
10 CFR Part 55 Content:	55.41 b.10	

QUESTION 61 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	245000 A3.05
	Importance Rating	3.0

Ability to monitor operation of main turbine generator and control valve automatic operation.

Proposed Question:

While operating at 85% power, which of the following represents the expected positions of the Main Turbine Control Valves?

	CV-1	CV-2	CV-3	CV-4
a.	~ full open	~ full open	~ full open	closed
b.	85% open	85% open	85% open	85% open
c.	85% open	85% open	85% open	throttled
d.	full open	full open	full open	full open

Proposed Answer: A.

Explanation (Optional): CV-4 opens at approximately 90% power. CV1-3 are full open at this point.

Technical Reference(s): STM-509, Rev 6; STM-0110 Rev 7

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0110 Obj. 2d

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.7

Comments:

QUESTION 62 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	259001 K1.05
	Importance Rating	3.2

Knowledge of the physical connections and/or cause-effect relationships between Reactor Feedwater System and Condensate.

Proposed Question:

The plant is at 80% power.

Two condensate pumps (A and B) and two feedwater pumps (B and C) are in service.

If a loss of both condensate pumps occurs, FEEDWATER PUMP…

a. “B” trips 15 seconds after suction pressure decreases to 260 psig.

b. “B” trips 10 seconds after suction pressure decreases to 280 psig.

c. “C” trips 10 seconds after suction pressure decreases to 260 psig.

d. “C” trips 20 seconds after suction pressure decreases to 280 psig.

Proposed Answer: A.

Explanation (Optional):

Low suction pressure trips occurs at 260 psig + associated time delay. (280 psig represents the alarm function). The time delay for each pumps is as follows: A=10 seconds, B=15 seconds, C=20 seconds.

Technical Reference(s): STM-107, Rev 10

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0107, Obj. 5, 8a, 16b

Question Source: Bank # 304

Question History: Last NRC Exam 7/1997

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

10 CFR Part 55 Content: 55.41 b.4

Comments:

QUESTION 63 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	271000 K5.11
	Importance Rating	2.6

Knowledge of the operational implications of the necessity of reducing the relative humidity of carbon bed filters in the Offgas System.

Proposed Question:

What is the operational concern of high relative humidity in the offgas charcoal adsorbers?

a. Moisture in the charcoal will significantly increase the radiation levels at the adsorber outlet.

b. Wet charcoal becomes acidic and cause significant system damage.

c. Wet charcoal can freeze and plug the adsorbers.

d. Moisture can cause adsorber vessel corrosion.

Proposed Answer: C.

Explanation (Optional): There is no significant rise in adsorber outlet radiation. Charcoal acidity is not an operational concern. While moisture in could potentially cause corrosion in the adsorber, it is not an operational concern. Freezing and plugging of the adsorbers, however, is a common industry concern.

Technical Reference(s): STM-0606, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0606 Obj. 12

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.4

Comments:

QUESTION 64 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	286000 A4.05
	Importance Rating	3.3

Ability to manually operate and or monitor the fire pump in the main control room.

Proposed Question:

A fire erupts in the Division 1 Diesel Generator room causing the sprinkler system to initiate and fire water header pressure drops to 115 psig.

Which of the following actions would be expected to occur?

a. The Electric Fire Pump will receive an auto start signal and the Diesel Driven Fire Pump "A" and "B" will start immediately if the Electric Fire Pump fails to start.

b. The Electric Fire Pump will receive an auto start signal, but if it fails to start and header pressure is still at 115 psig after 15 seconds, then Diesel Driven Fire Pump "A" will start.

c. The Diesel Driven Fire Pump "A" will auto start, if fire water header pressure remains at 115 psig for 10 seconds, whether the Electric Driven Fire Pump starts or NOT.

d. The Diesel Driven Fire Pump "A" will auto start, if fire water header pressure remains below 140 psig for 10 seconds and the Electric Fire Pump is running.

Proposed Answer: D.

Explanation (Optional): The Electric Fire Pump receives an auto start signal at 120 psig. Diesel Fire Pump "A" receives a start signal at 110 psig with a 10 sec TD. Diesel Fire Pump "B" receives a start signal at 100 psig with a 15 second TD. If the Electric Fire Pump starts and pressure is still <140 psig for 10 seconds the Diesel Fire Pump "A" will start (<140 psig and 15 seconds for Diesel Fire Pump "B").

Technical Reference(s): STM-250, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-250 Obj. 4b, 4c, 5a, 5b

Question Source: Bank # 610

Question History:	Last NRC Exam 10/2000	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒4
10 CFR Part 55 Content:	55.41 b.7	

Comments:

QUESTION 65 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	2
	Group #	2
	K/A #	290002 K6.01
	Importance Rating	2.8

Knowledge of the effect that a loss or malfunction of the CRD Hydraulic system will have on Reactor Vessel Internals.

Proposed Question:

With a loss of Control Rod Drive hydraulic pumps and accumulators, a control rod can still scram using reactor pressure.

What is the minimum reactor pressure required to scram a control rod?

a. 800 psig

b. 600 psig

c. 400 psig

d. 200 psig

Proposed Answer: B.

Explanation (Optional): At less than 600 psig, it can not be assured that rods will insert on a scram without CRD hydraulics.

Technical Reference(s): STM-0052 Rev 3

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0052 Obj. 11

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.2

Comments:

QUESTION 66 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Conduct of Ops
	K/A #	G 2.1.19
	Importance Rating	3.9

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

Proposed Question:

The plant is operating at rated conditions.

The following are parameter values just taken from the core monitoring computer system.

MFLCPR	0.91
MAPRAT	0.82
MFLPD	0.89
FDLRX	0.89
FCBB	1.12

Which of the following identifies the consequences of continued operation with thermal limits at their present value?

a. Fuel cladding could exhibit in excess of 1% plastic strain during a normal operation or during a transient.

b. Possible peak cladding temperature in excess of 2200°F during a LOCA

c. There is a possibility of the onset of transition boiling in greater than 0.1% of the fuel rods during a transient.

d. The core may become unstable during operation in the Restricted Region of the Power to Flow Map.

Proposed Answer: D

Explanation (Optional): All parameters are within limits except for Fraction of Core Boiling Boundary. This limit is only in effect when operating in the Restricted Region of the Power to Flow Map.

Technical Reference(s): HLO-174, Rev 3; HLO-0534, Rev 1

Proposed references to be provided to applicants during examination: NA

Learning Objective:	_HLO-174 Obj 7,11,23 HLO-0534 Obj 4
Question Source:	Bank # 1043
Question History:	Last NRC Exam NA
Question Cognitive Level:	Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒3
10 CFR Part 55 Content:	55.41 b.2
Comments:	

QUESTION 67 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Conduct of Ops
	K/A #	G 2.1.29
	Importance Rating	4.1

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Proposed Question:

The Unit Operator has just completed venting the Primary Containment in accordance with the System Operating Procedure. All equipment operated as expected.

What type of control board lineup is required following this evolution?

a. Documentation of the Unit Operator Rounds (OSP-0028), with the lineup verified by a different operator.

b. Documentation on the Unit Operator Rounds (OSP-0028), however the lineup is not required to be verified by different operator.

c. Documentation in the Main Control Room Log Book, with the lineup verified by a different operator.

d. Documentation in the Main Control Room Log Book, however the lineup is not required to be verified by a different operator.

Proposed Answer: C.

Explanation (Optional):OSP-0022, Operations General Administrative Guidelines states the control board lineups shall be performed after completion of any operation or component manipulation on Safety Related-Tech Spec systems. Lineups performed shall be verified by a different operator and documented in the Control Room Log Book.

Technical Reference(s): OSP-0022, Rev 11 Operations General Administrative Guidelines, Step 5.2.26.

Proposed references to be provided to applicants during examination: NA

Learning Objective: NA

Question Source: Bank # 637

Question History: Last NRC Exam 2/1999

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 68 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Conduct of Ops
	K/A #	G. 2.1.34
	Importance Rating	2.7

Knowledge of primary and secondary plant chemistry limits.

Proposed Question:

Which of the following represents the operating limit for reactor coolant conductivity in Mode 1?

a. ≤ 0.1 μmhos

b. ≤ 1.0 μmhos

c. ≤ 2.0 μmhos

d. ≤ 10.0 μmhos

Proposed Answer: B.

Explanation (Optional):

Operating limit per TRM 3.4.13 is ≤ 1.0 μmhos.

Technical Reference(s): TRM 3.4.13

Proposed references to be provided to applicants during examination: NA

Learning Objective: NA

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.5

Comments:

QUESTION 69 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Equipment Control
	K/A #	G 2.2.23
	Importance Rating	3.1

Ability to track Technical Specification limiting conditions for operations

Proposed Question:

Where would the onshift Reactor Operator find a listing of Actual and Potential Technical Specification Limiting Conditions for Operation that are in effect?

a. ESOMS Main Control Room Narrative Log

b. Surveillance Testing Procedure Log

c. ESOMS LCO Tracking Log

d. Shift Manager Relief Checklist

Proposed Answer: C.

Explanation (Optional):

Technical Reference(s): Actual and Potential LCO are recorded tracked using the ESOMS software program.

Proposed references to be provided to applicants during examination: NA

Learning Objective: NA

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 70 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Equipment Control
	K/A #	G. 2.2.17
	Importance Rating	2.6

Knowledge of the process for managing maintenance activities during plant operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Proposed Question:

The plant is operating at 100% power.

The Unit Operator receives a phone call from the Fancy Point Switchyard gate. Entergy Transmission & Distribution personnel are requesting entry into the switchyard to perform routine switchyard inspections. The River Bend Electrical Maintenance Superintendent has reviewed the work scope and will enter the switchyard with T&D personnel.

How should the Unit Operator proceed?

a. Allow the T&D personnel to enter the switchyard since their maintenance is non-intrusive.

b. Brief the T&D personnel on OSP-0048, SWITCHYARD, TRANSFORMER YARD AND SENSITIVE EQUIPMENT CONTROLS, then allow them to enter and perform their inspections.

c. Direct the phone call to the CRS who will brief the T&D personnel on OSP-0048, SWITCHYARD, TRANSFORMER YARD AND SENSITIVE EQUIPMENT CONTROLS, then allow them to enter and perform their inspections.

d. Direct the phone call to the CRS who will allow them to enter the switchyard since their maintenance is non-intrusive.

Proposed Answer: C.

Explanation (Optional): Only an OSM/CRS is allowed to authorize entry into Fancy Point Switchyard. Prior to entry, a pre-job brief is required.

Technical Reference(s): OSP-0048, Rev 5 Section 7.1

Proposed references to be provided to applicants during examination: NA

Learning Objective:	NA	
Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☒ 3
	Comprehension or Analysis	☐
10 CFR Part 55 Content:	55.41 b.10	
Comments:		

QUESTION 71 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Equipment Control
	K/A #	G. 2.2.18
	Importance Rating	2.6

Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Proposed Question:

During a refueling outage, the Plant Safety Risk color code is YELLOW.

Which of the following defines this condition?

a. Failure to meet both an adequate level of safety and defense in depth.

b. Adequate level of safety and defense in depth exist. Acceptable risk.

c. High level of safety and defense in depth exist.

d. Failure to meet both an adequate level of safety and defense in depth, but specific contingency plans are in place.

Proposed Answer: B.

Explanation (Optional): As defined in OSP-0037.

Technical Reference(s): SOP-0037 Rev 18

Proposed references to be provided to applicants during examination: NA

Learning Objective: NA

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 72 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Radiation Control
	K/A #	G. 2.3.7
	Importance Rating	3.5

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Proposed Question:

What type of information would you expect to find on a General Radiation Work Permit (RWP)?

a. An individual's dose margin.

b. Electronic Alarming Dosimeter (EAD) Settings.

c. Dose rates at Hot Spots.

d. Total department cumulative dose and dose goals.

Proposed Answer: B.

Explanation (Optional):. EAD Settings are provide on each RWP. An individuals dose margin is not available on the RWP, nor is the department dose/dose goal. The dose rate at a Hot Spot is found on survey maps.

Technical Reference(s): EN-RP-105, Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: HLO-209 Obj. 1

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 73 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Radiation Control
	K/A #	G. 2.3.15
	Importance Rating	2.9

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question:

The plant is operating at 100% power.

Both Offgas Post Treatment Radiation monitors have alarmed on a High-High-High Radiation signal.

Which one of the following describes the effect on the Offgas System and the main Condenser?

a. Offgas will shift into the bypass mode of operation causing a Loss of Condenser Vacuum.

b. Offgas will isolate only the charcoal adsorbers inlet and outlet valves causing a Loss of Condenser Vacuum.

c. Offgas will continue to operate allowing Main Condenser Vacuum to remain constant.

d. Offgas System will isolate causing a Loss of Condenser Vacuum.

Proposed Answer: D.

Explanation (Optional): On a triple high radiation signal on both post treatment radiation monitors, N64-F060 will isolate resulting in a shutdown of offgas flow. As a result, air and non-condensibles will not be removed from the condenser and ultimate condenser vacuum will be lost.

Technical Reference(s): STM-606, Rev 2

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0606, Obj. 13a, 14b

Question Source: Bank # 607

Question History: Last NRC Exam 2/1999

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

10 CFR Part 55 Content: 55.41 b.4; b.11

Comments:

QUESTION 74 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Emergency Plan
	K/A #	G. 2.4.1
	Importance Rating	4.6

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question:

The reactor has just scrammed. The following plant conditions exist:

Reactor power	3%
Reactor water level	17 inches (lowest level observed was 15 inches)
Reactor pressure	1047 psig
Suppression Pool Level	20 feet 2 inches
Drywell H2	0.4%
Drywell pressure	0.2 psid

Which of the following represents the required EOP(s) to enter.

a. EOP-1 and EOP-2

b. EOP-2 only

c. EOP-1A and EOP-2

d. EOP-1 only

Proposed Answer: B

Explanation (Optional): No EOP-1 entry conditions exist, therefore EOP-1A is also not applicable. Suppression Pool Level requires entry into EOP-2.

Technical Reference(s): EOP-1 Rev 21; EOP-2 Rev 21

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-512 Obj 3; RLP-HLO-514, Obj 3

Question Source: Bank # 132

Question History: Last NRC Exam 1/1993

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒3

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 75 Rev 0

Examination Outline Cross-Reference:	Level	RO☒ SRO☐
	Tier #	3
	Group #	Emergency Plan
	K/A #	G. 2.4.5
	Importance Rating	3.7

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

Proposed Question:

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

a. Fuel Handling Procedures

b. Emergency Implementing Procedures

c. Emergency Operating Procedures

d. Radiation Section Procedures

Proposed Answer: B.

Explanation (Optional):
An Emergency Implementing Procedure (EIP-2-006) provides guidance in this situation.

Technical Reference(s): EIP-2-006, Rev 33

Proposed references to be provided to applicants during examination: NA

Learning Objective: NA

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.41 b.10

Comments:

QUESTION 76 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	1
	K/A #	295005 G 2.4.6
	Importance Rating	4.7

Knowledge of EOP mitigation strategies as they relate to main turbine and generator trips.

Proposed Question:

While operating at 100% power, the Main Turbine tripped. Several control rods failed to insert as required. The ATWS procedure is being implemented.

The following conditions exist:

Reactor power	12%
Reactor level	-20 inches
MSIV	OPEN

Which of the following describes the preferred EOP pressure control mitigation strategy as described in OSP-0053, Emergency and Transient Response Support Procedure?

a. The operator stabilizes pressure 800-1090 psig then requests an expanded pressure band, opens applicable steam line drains and opens the Bypass Valves by lowering the Main Turbine pressure setpoint.

b. The operator stabilizes pressure 950-1090 psig then requests an expanded pressure band, opens applicable steam line drains and opens the Bypass Valves by using the BPV jack.

c. The operator stabilizes pressure 800-1090 psig then requests an expanded pressure band, opens applicable steam line drains and opens the Bypass Valves by using the BPV jack.

d. The operator stabilizes pressure 950-1090 psig then requests an expanded pressure band, opens applicable steam line drains and opens the Bypass Valves by lowering the Main Turbine pressure setpoint.

Proposed Answer: D.

Explanation (Optional): The stabilization band is 950 psig per EOP-1A Step RPA-3. 800-1090 psig represents the expanded band after stabilization. OSP-0053 Attachment 1B describes ATWS Pressure Control Strategies. Page 6 of 6 lists with Preferred method of pressure control as automatic pressure control (pressure set point reduction). Jacking open the BPVs is utilized when the MSIVs CLOSED.

Technical Reference(s): OSP-0053 Rev 9, EOP-1A Rev 21

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-OPS-HLO512 Obj. 8		
Question Source:	New		
Question History:	Last NRC Exam	NA	
Question Cognitive Level:	Memory or Fundamental Knowledge		☐
	Comprehension or Analysis		☒ 3
10 CFR Part 55 Content:	55.43 b.5		

Comments:

QUESTION 77 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	1
	K/A #	295006 AA2.04
	Importance Rating	4.1

Ability to determine and/or interpret reactor pressure as it applies to a SCRAM condition.

Proposed Question:

During a plant startup, with reactor power at 25% a failure occurred in the Main Turbine pressure regulator causing its output to fail low.

The following conditions exist:

Reactor pressure	1105 psig
Turbine Stop Valves	CLOSED
Turbine Control Valves	CLOSED
Turbine Bypass Valves	CLOSED
SRVs	OPEN
Suppression Pool Temp	105°F

Assuming all other systems worked as designed, what is the status of the reactor and applicable procedures for the above conditions?

a. The reactor has scrammed and EOP-1 and EOP-2 should be entered.

b. The reactor has not scrammed due TSV and TCV closure signal being bypass. Only EOP-2 should be entered.

c. The reactor has scrammed and only EOP-1 should be entered.

d. The reactor has not scrammed due TSV and TCV closure signal being bypassed. No EOP entry is required.

Proposed Answer: A.

Explanation (Optional): The high pressure reactor scram setpoint of 1094.7 psig has been exceeded, therefore the reactor has scrammed. This setpoint is also the entry condition for EOP-1. As a result of the high pressure condition, SRVs have opened and have caused the suppression pool temperature to rise above the 100°F entry condition, therefore EOP-2 entry is also required.

Technical Reference(s): EOP-1, Rev 21; EOP-2, Rev 14; AOP-0001, Rev 24

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-STM-0508 Obj. 2; RLP-HLO-0512 Obj. 3; RLP-HLO-0514 Obj. 3
Question Source:	New
Question History:	Last NRC Exam NA
Question Cognitive Level:	Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒ 4
10 CFR Part 55 Content:	55.43 b.5
Comments:	

QUESTION 78 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	1
	K/A #	295019 AA2.01
	Importance Rating	3.6

Ability to determine and/or interpret the instrument air system pressure as it applies to a partial or total loss of instrument air

Proposed Question:

The unit operator reports that Instrument Air header pressure has lowered to 60 psig and is stable.

Which of the following actions are required to be taken?

a. Enter AOP-0008, Loss of Instrument Air and shut the MSIVs and then scram the reactor.

b. Enter AOP-0008, Loss of Instrument Air, scram the reactor and enter AOP-0001, Reactor Scram.

c. Enter AOP-0008, Loss of Instrument Air, attempt to start/restart at least one air compressor per SOP-0022,Instrument Air System and monitor header pressure to determine if further actions are required.

d. Enter AOP-0008, Loss of Instrument Air, initiate RCIC, scram the reactor and close the MSIVs.

Proposed Answer: B.

Explanation (Optional): At 65 psig IAS header pressure, AOP-0008, Loss of Instrument Air, directs entering AOP-0001, Reactor Scram and inserting a reactor scram. At 50 psig IAS header pressure, AOP-0008 requires that the MSIVs be closed.

Technical Reference(s): AOP-0008, Rev 26; AOP-0001 Reactor Scram, Rev 24

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-527 Obj. 3

Question Source: New

Question History: Last NRC Exam NA

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

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 #	295025 G2.2.44
	Importance Rating	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 while experiencing a high reactor pressure condition..

Proposed Question:

The Main Control Room has been evacuated.

When the Remote Shutdown panel is manned, 8 minutes after the event, SRV B21-F051C is OPEN and reactor pressure is cycling around 1210 psig.

Which of the following is correct?

a. Implement EOP-0001, RPV Control, and maintain reactor pressure less than 1090 psig.

b. Implement EOP-001A, RPV Control ATWS, and stabilize pressure pressure below 1090 psig with SRVs.

c. Implement AOP-0031, Shutdown From Outside The Main Control Room, and fully open SRV B21-F051D.

d. Implement EOP-0001A, RPV Control ATWS and utilize RWCU in the blowdown mode per Enclosure 29, RWCU Blowdown Mode, to control reactor pressure.

Proposed Answer: B.

Explanation (Optional):With the given conditions, the plant is in an ATWS condition and EOP-0001A, RPV Control ATWS, should be entered. Given the relative low power condition SRVs should be manually controlled to maintain pressure below 1090 psig.

Technical Reference(s): AOP-0031, Rev 303; EOP-0001A, Rev 21

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-0537, Obj. 2; RLP-HLO-0513 Obj. 4

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 80 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	1
	K/A #	295026 G2.1.27
	Importance Rating	4.0

Knowledge of system purpose and function regarding systems used to control suppression pool water temperature

Proposed Question:

The plant is 4 hours into a station blackout and suppression pool temperature is currently 175°F.

Which of the following would be most effective in lowering suppression pool temperature?

a. Implementing AOP-0050, Station Blackout and gravity draining the CST into the suppression pool

b. Utilizing the HALE Fire pump and transferring water from the Well Water Storage Tank to the suppression pool.

c. Implementing EOP-0005 Enclosure 21, Emergency Containment Venting and Defeating Containment Vent Path Isolation Interlocks.

d. Implementing AOP-0050, Station Blackout, and gravity draining the fire protection storage tanks into the suppression pool.

Proposed Answer: A.

Explanation (Optional): The CST has the highest capacity (lbm) of water and will gravity drain to the suppression pool. Well water storage tank has a much smaller capacity. Venting containment will lower containment pressure and also containment temperature with some minimal effect of suppression pool temperature. The Fire Protection Storage Tanks can no be effectively gravity drained to the suppression pool.

Technical Reference(s): TSG-0001, AOP-0050 Rev 25

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-OPS-HLO-541 Obj. 4.6

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 81 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	1
	K/A #	295028 G 2.4.50
	Importance Rating	4.0

Ability to verify alarm setpoints and operate controls identified in the alarm response manual regarding High Drywell Temperature.

Proposed Question:

During normal operation at 100% power, the following annunciator is received on H13-P601:

AIR TEMP MON R608 DRYWELL AMBIENT HIGH TEMP

The CRS should direct the operator to verify drywell temperature on ______(1)______ and operate additional drywell coolers as needed from _______(2)_______.

Maintaining drywell temperature below the normal operating limit ensures that the design temperature limit of ______(3)_______ will not be exceeded in the event a LOCA occurs.

a. (1) E31-R608 on H13-P863 (2) H13-P601 (3) 145°F

b. (1) E31-R608 on H13-P632 (2) H13-P863 (3) 330°F

c. (1) E31-R608 on H13-P601 (2) H13-P863 (3) 145°F

d. (1) E31-R608 on H13-P863 (2) H13-P632 (3) 330°F

Proposed Answer: B.

Explanation (Optional):
The temperature recorder which serves as the initiating device for this alarm is located on control backpanel H13-P632. The drywell unit cooler controls are located on H13-P863 and the Drywell design temperature limit is 330°F.

Technical Reference(s): ARP-601-19 H03 Rev 25; STM-403 Rev 4

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0403 Obj. 10; RLP-STM-0057 Obj. 4a

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.2, b.5

Comments:

QUESTION 82 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐ SRO ☒
	Tier #	1
	Group #	1
	K/A #	295037 G2.4.35
	Importance Rating	4.0

Knowledge of local auxiliary operator tasks during an emergency and the resultant operator effects during an ATWS condition

Proposed Question:

The plant has undergone a low reactor water level transient. The scram report for an ATWS condition is given and the following annunciators are reported to be in alarm:

RPS TRIP LOGIC A OR C ACTIVATED
RPS TRIP LOGIC B OR D ACTIVATED
SCRAM PILOT VLV AIR HEADER LOW PRESSURE
CRD SCRAM DISCH VOL HIGH WATER LEVEL

The RPS Solenoid status lights on H13-P680 are extinguished.

Which of the following EOP-00005 Enclosure 26 Control Rod Insertion Method Determination actions should you direct the available operator to complete?

a. Direct the control room operator to install Enclosure 10 to De-energize the Scram Solenoids

b. Direct the reactor building operator to install Enclosure 11 to Vent the Scram Air Header

c. Direct the At the Controls operator to reset the scram and retry the scram per AOP-0001, Reactor Scram.

d. Direct the reactor building operator to install Enclosure 17 to vent the CRD over piston volume.

Proposed Answer: D.

Explanation (Optional): Scram solenoids have de-energized from stem indications; the scram air header is depressurized from stem indications; the scram can not be reset from stem indications; of the choices, Enclosure 17 should be utilized to insert individual control rods.

Technical Reference(s): EOP-0005 Rev 302; Enclosure 26; Enclosure 10, Enclosure 17, Enclosure 11

Proposed references to be provided to applicants during examination: NA

Learning Objective:	RLP-HLO-516 Obj. 1	
Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒2
10 CFR Part 55 Content:	55.43 b.5	
Comments:		

QUESTION 83 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	2
	K/A #	295011 G2.2.38
	Importance Rating	4.5

Knowledge of conditions and limitations in the facility license regarding High Containment Temperature.

Proposed Question:

The Unit Operator reports the following parameters:

Drywell Temperature	131°F
Containment Pressure	0.2 psig
Containment Temperature	91°F
Unidentified Leakage	0.08 gpm
Suppression Pool Water Temperature	88°F
Suppression Pool Level	19’11”

Which of the following statements is correct based upon the information above?

a. A reactor scram is required. Direct the At the Controls operator to enter AOP-0001, Reactor Scram.

b. If a LOCA were to occur containment design parameters could be exceeded.

c. Plant parameters should be controlled by entering EOP-0001 RPV Control as directed by EOP-0002 Primary Containment Control.

d. No specific actions required at this moment. The plant is within operating limits.

Proposed Answer: B.

Explanation (Optional): The normal operating limit for containment temperature is 90°F. This limit is based on the containment temperature assumed in the accident analysis to avoid exceeding containment design temperature limit of 185°F during a LOCA.

Technical Reference(s): Technical Specification 3.6.1.5 Bases

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0057 Obj. 9

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.2

Comments:

QUESTION 84 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	2
	K/A #	295034 G2.4.6
	Importance Rating	4.7

Knowledge of EOP mitigation strategies regarding Secondary Containment Ventilation High Radiation.

Proposed Question:

A plant transient has occurred resulting in an unisolable primary system leak into the Main Steam Tunnel. This event has led to an offsite radioactive release rate exceeding the General Emergency levels.

Which of the following actions is required?

a. Enter GOP-0002 and commence an orderly plant shutdown.

b. Enter EOP-0003 Secondary Containment and Radioactive Release Control and then EOP-0001 RPV Control Emergency Depressurization.

c. Enter AOP-0001 Reactor scram, scram the plant and wait for direction from the Emergency Response Organization.

d. Enter EOP-0003 Secondary Containment and Radioactive Release Control and then EOP-0001 RPV Control and commence an orderly shutdown.

Proposed Answer: B.

Explanation (Optional):
The condition stated requires EOP-0003 entry which will direct entry into EOP-0001 and then Emergency Depressurization. An orderly plant shutdown per GOP-0002 is not directed in this condition; Although AOP-0001 Reactor Scram would eventually entered, waiting for the ERO direct actions is not procedurally directed. EOP-0001 provides guidance for Emergency Depressurization as required by EOP-0003.

Technical Reference(s): EOP-0003 Rev 14

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-515 Obj.3 & 6

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 85 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	1
	Group #	2
	K/A #	295036 G2.4.9
	Importance Rating	4.2

Knowledge of low power/shutdown implications in accident mitigation strategies with respect to secondary containment high sump/area water level.

Proposed Question:

An earthquake has occurred at the site and the following conditions exists:

Reactor power 82%

RHR Equipment Room A Radiation Level 9.3 E+03 mR/hr
RHR A Room Water Level 6 inches above the floor

RHR Equipment Room C Radiation Level 8.0E+01 mR/hr
RHR C Room Water Level 5 inches above the floor

Aux Bldg Ventilation Radiation Level 1.23 E-04 uCi/ml

There is <u>no</u> indication of a primary system leak.

What action should be taken?

a. Enter GOP-0002 Plant Shutdown and commence a plant shutdown.

b. Enter AOP-0028 Seismic Event and insert a manual reactor scram.

c. Enter GOP-0002 Plant Shutdown and commence an Emergency Depressurization of the plant.

d. Maintain reactor power, enter AOP-0028 Seismic Event and perform a plant walkdown to determine the extent of plant damage.

Proposed Answer: A.

Explanation (Optional): EOP-0003 Secondary Containment Parameter control direct an orderly plant shutdown via GOP-0002 due to two area water levels being great than the Maximum Safe Operating Value. A reactor scram is not required by GOP-0002, AOP-0028 or EOP-0003. An Emergency Depressurization is not required. A normal plant shutdown is required.

Technical Reference(s): EOP-0003 Rev 14; AOP-0028 Rev 5

Proposed references to be provided to applicants during examination: EOP-0003 Table H & left side of flow chart

Learning Objective:	RLP-HLO-515 Obj. 6
Question Source:	New
Question History:	Last NRC Exam NA
Question Cognitive Level:	Memory or Fundamental Knowledge ☐ Comprehension or Analysis ☒3
10 CFR Part 55 Content:	55.43 b.5
Comments:	

QUESTION 86 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	2
	Group #	1
	K/A #	209001 A2.10
	Importance Rating	3.4

Ability to predict the impact of high suppression pool temperature on LPCS and based on those predictions, use procedures to correct, control, or mitigate the consequences of this condition.

Proposed Question:

The following conditions exist:

Reactor water level	-20 inches slowly rising with injection from LPCS
Drywell Temperature	257°F
Drywell Pressure	1.4 psid
Containment Pressure	2 psig
Containment Temperature	96°F
Suppression Pool Temperature	165°F
Suppression Pool Level	20 feet 9 inches

Which of the following actions should be directed based upon the above conditions?

a. Emergency Depressurization of the RPV due to containment pressure.

b. Restore Drywell Cooling to lower drywell temperature to preclude level instrument reference leg flashing.

c. Lineup for injection from HPCS taking a suction from the CST in anticipation of LPCS cavitation due to exceeding NPSH limits.

d. Emergency Depressurize the RPV due to containment temperature.

Proposed Answer: C.

Explanation (Optional):
EOP-0001 Caution 5 warns of potential damage to pumps taking a suction from the Suppression Pool when SP Temp is >165°F

Technical Reference(s): EOP-0001 Rev 21

Proposed references to be provided to applicants during examination: EOP Figure 4
PSP Curve

Learning Objective: RLP-OPS-511 Obj. F

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒ 3
10 CFR Part 55 Content:	55.43 5	

Comments:

QUESTION 87 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	2
	Group #	1
	K/A #	209002 G2.4.49
	Importance Rating	4.4

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls related to HPCS.

Proposed Question:

The plant is at 100% reactor power when HPCS receives an inadvertent initiation signal. Level has been verified to be normal by 2 independent instruments.

Which of the following directions should be given by the CRS and what is the justification for the action?

a. The HPCS pump should be tripped because this is the quickest way to stop the injection. OSP-0053, Emergency and Transient Response Procedure allows for immediate operator response without reference to procedures.

b. The ATC operator to take manual control of the feedwater system to ensure a high reactor water level is not received.

c. The ATC operator to enter AOP-0001 Reactor Scram and insert a manual scram to preclude receiving an automatic scram due to cold water addition caused by the HPCS injection.

d. E22-F004, HPCS Injection Isol Valve, should be taken to close because OSP-0053 Emergency and Transient Response Procedure allows for immediate operator response without reference to procedures.

Proposed Answer: D.

Explanation (Optional):
OSP-0053 allows for manipulation of injection valves and controllers to control level without reference to procedures. Procedure must be referenced before tripping the pump. Feedwater should be left in Auto to allow the Feed Reg Valves to close down and maintain level with the additional injection. Cold water injection will not cause a reactor scram.

Technical Reference(s): OSP-0053 Rev 9

Proposed references to be provided to applicants during examination: NA

Learning Objective: NA

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 88 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	2
	Group #	1
	K/A #	223002 A2.10
	Importance Rating	4.2

Ability to predict the impact of loss of coolant accidents on NSSSS and based on those predictions use procedures to correct, control, or mitigate the consequence of this condition.

Proposed Question:

The following plant conditions exist:

Reactor water level	-55 inches (lowest was -60 inches)
Reactor pressure	840 psig
Drywell Maximum Pressure	1.0 psid
Drywell MaximumTemperature	125°F steady
Containment Temperature	92°F
Containment Pressure	0.21 psig
Suppression Pool Temperature	92°F steady
HVR-UC1A & HVR-UC1B are in service	

Identify the required action based on the given condition:

a. Open 7 ADS SRVs per EOP-0002.

b. Operate all available Suppression Pool Cooling per SOP-0031 and EOP-0002, Containment Control.

c. Operate all available containment cooling by opening SWP-MOV502A(B) and SWP-MOV503A(B) per SOP-0059, Containment HVAC & EOP-0002, Containment Control.

d. Operate all available drywell cooling defeating interlocks with EOP-0005, Enclosure 20 as necessary.

Proposed Answer: C.

Explanation (Optional): With a level 2 signal present, HVN to the containment unit coolers has received an isolation signal from NSSSS. Despite the unit coolers being in service, no cooling is being provided to containment. EOP-0002 Step CT-3, directs the operation of all available containment cooling which includes the aligning of Service Water to the unit coolers. These valves normally open on Level 1 or Hi Drywell pressure of 1.68 psid. Neither of these signals is present. There is no parameter which has exceeded Emergency Depressurization criteria. Sup Pool temp is less than 100°F therefore operating all available sup pool cooling is not required. Drywell temp is less than 145°F therefore Enclosure 20 is not authorized.

Technical Reference(s): AOP-0003 Rev 26; EOP-0002 Rev14; SOP-00059 Rev 30

Proposed references to be provided to applicants during examination: _EOP-0002 CT leg with temperatures blacked out.

Learning Objective: RLP-STM-0409 Obj. 4 & 12; RLP-HLO-0514 Obj 6

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 89 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	2
	Group #	1
	K/A #	259002 G2.4.1
	Importance Rating	4.8

Knowledge of EOP entry conditions and immediate action steps regarding reactor water level control

Proposed Question:

The following plant conditions exist:

Reactor power	100%
Reactor water level	35 inches
Reactor pressure	1055 psig

B Level channel is selected for Feedwater Level Control

The B Level signal has just failed upscale.

Select the required action based on the above conditions.

a. Select the Single Element control pushbutton and enter EOP-0001 if reactor water level drops below 9.7 inches.

b. Take manual control of the feedwater system by placing the Master controller in Manual and enter EOP-0001 if reactor water level drops below 9.7 inches.

c. Reduce reactor power to mitigate the transient and enter EOP-0001 if the reactor is scrammed.

d. Lineup the Startup Feedwater Regulating Valve to augment level control and enter EOP-0001 if the reactor is scrammed.

Proposed Answer: B.

Explanation (Optional):

AOP-0006 immediate operator actions state, “Manually control the feedwater level control system and/or reduce reactor power to mitigate any level transient. EOP-0001 water level entry condition is 9.7 inches.

Technical Reference(s): EOP-0001 Rev 21; AOP-0006 Rev 017

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-0525 Obj. 4 & RLP-HLO-0512 Obj. 3

Question Source:	New
Question History:	Last NRC Exam NA
Question 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 #	400000 A2.01
	Importance Rating	3.4

Ability to predict the impact of a loss of CCW pump and based on those predictions, use procedures to correct, control, or mitigate the consequences of this condition.

Proposed Question:

While operating at 100%, CCS-P1C tripped.

The standby pump, CCS-P1B failed to automatically start. CCS-P1A remained in service.

Select the appropriate action for this condition.

a. Attempt a manual start of CCS-P1B per AOP-0011, Loss of CCS.

b. Scram the reactor per AOP-0001, Reactor Scram.

c. Secure the condensate and feedwater pumps and enter AOP-0006, Condensate and Feedwater Failures.

d. Secure the Stator Water Cooling pumps and trip the Main Turbine/Generator per AOP-0002, Turbine/Generator Trips.

Proposed Answer: A.

Explanation (Optional): Initial actions are to attempt a manual start of the standby pump. All other actions in distractors are to be taken after it is determine that CCS can not be reestablished.

Technical Reference(s): AOP-0011, Rev 012

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-0531 Obj. 6

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 91 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	2
	Group #	2
	K/A #	201001 G2.4.11
	Importance Rating	4.2

Knowledge of abnormal condition procedures associated with CRD hydraulics.

Proposed Question:

The following annunciator has just been received on H13-P680:

CONTROL ROD DRIFT

The ATC operator has reported that control rod 28-29 is drifting OUT. No other alarms have been received.

What is a possible cause for this condition and what actions should be directed?

a. Excessive cooling water flow. Direct the operator to individually scram the affected control rod per EOP-0005 Enclosure 13 Operating Individual Scram Test Switches.

b. Failed directional control valve. Direct the operator to individually scram the affected control rod per EOP-0005 Enclosure 13 Operating Individual Scram Test Switches..

c. Stuck collet piston. Direct the operator to drive the control rod in per the associated Alarm Response Procedure.

d. Leaking scram valves. Direct the operator to drive the control rod in per the associated Alarm Response Procedure.

Proposed Answer: C.

Explanation (Optional): Excessive cooling water flow and leaking scram valves would cause the rod to drift in, not out. Directional control valve failures and stuck collet pistons are both failure mechanisms for a rod to drift out, but the appropriate action is to drive the rod in, not scram it in.

Technical Reference(s): ARP-680-07

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0052 Obj. 8n, 14c

Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒3
10 CFR Part 55 Content:	55.43 b.5	

Comments:

QUESTION 92 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	2
	Group #	2
	K/A #	219000 A2.10
	Importance Rating	3.2

Ability to predict the impact of nuclear boiler instrument failures on RHR pool cooling mode and based on those predictions, use procedures to correct, control or mitigate the consequences of this condition

Proposed Question:

While in Mode 1, a Division 1 RHR/LPCI drywell pressure trip unit has failed in the tripped condition.

Which of the following describes the effect of this condition on RHR ‘A’ and actions to be taken?

a. RHR ‘A’ will <u>not</u> initiate on a valid high drywell pressure condition, nor can it be aligned into the suppression pool cooling lineup. Tech Spec 3.5.1. ECCS-Operating and 3.6.2.3 RHR Suppression Pool Cooling must be entered.

b. RHR ‘A’ will <u>not</u> initiate on a valid high drywell pressure condition. Suppression Pool Cooling can still be aligned if necessary. Only Tech Spec 3.5.1. ECCS-Operating must be entered.

c. RHR will initiate if a valid high drywell pressure condition occurs. Suppression Pool Cooling can still be aligned if necessary. Tech Spec 3.3.5.1 ECCS INSTRUMENTATION must be entered.

d. RHR will initiate if a valid high drywell pressure condition occurs. Suppression Pool Cooling can not be aligned into the suppression pool cooling lineup. Tech Spec 3.6.2.3 RHR SUPPRESSION POOL COOLING must be entered.

Proposed Answer: C..

Explanation (Optional):.A single instrument has failed into the tripped condition. If the second instrument senses a high drywell pressure, RHR A will initiate (2 out of 2 logic). Only 1 of 2 signals has been received, so RHR can still be placed in the SPC lineup (2 out of 2 logic for valve isolation also). Even if the E12-F024A were to isolate, it can still be overridden open. The appropriate Tech Specs is 3.3.5.1. for ECCS Instrumentation.

Technical Reference(s): Tech Specs, STM-0204

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0204 Obj. 4, 6g, 12, 17h,

Question Source:	New	
Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒4
10 CFR Part 55 Content:	55.43 b.2	
Comments:		

QUESTION 93 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	2
	Group #	2
	K/A #	256000 G2.4.31
	Importance Rating	4.1

Knowledge of the alarms, indications, or response procedures associated with reactor condensate.

Proposed Question:

During operation at 91% power, the following alarm on H13-P680 is received:

CONDENSATE PUMP ‘A’ OVERLOAD

The following plant conditions exist:

CNM- P1A 155 amps
CNM-P1B 150 amps
CNM-P1C Standby
3 Feedwater Pumps in service

What is the correct action for this condition?

a. Secure CNM-P1A per SOP-0006, Condensate System

b. Start CNM-P1C per SOP-0006, Condensate System

c. Begin a plant shutdown per GOP-0002, Plant Shutdown

d. Secure a feedwater pump per SOP-0007, Feedwater System

Proposed Answer: B.

Explanation (Optional): The maximum motor current for the condensate pumps is 152 amps. CNM-P1C which is available should be started to assist in handling the load on the condensate system.

Technical Reference(s): SOP-0006, Rev 303, ARP-680-02-B03

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0104 Obj. 6 & 9

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 94 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	3
	Group #	Conduct of Ops
	K/A #	G 2.1.23
	Importance Rating	4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question:

During the performance of a Feedwater Level Control evolution, reactor water level rose approximately 2 inches. Subsequently reactor thermal power indication showed a corresponding rise in megawatts thermal.

Which of the below values represents a condition outside the facility license limit?

a. 3125 MWth, instantaneous

b. 3105 MWth, 1 hour average

c. 3093 MWth, 2 hour average

d. 3092 MWth, 8 hour average

Proposed Answer: D.

Explanation (Optional): Average thermal power limits are given in GOP-0005, Power Maneuvering. The 8 hour average limit is 3091.0 MWth.

Technical Reference(s): GOP-0005, Rev 302

Proposed references to be provided to applicants during examination: GOP-0005

Learning Objective: RLP-HLO-500 Obj. 2

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.1

Comments:

QUESTION 95 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	3
	Group #	Conduct of Ops
	K/A #	G 2.1.42
	Importance Rating	3.4

Knowledge of new and spent fuel movement procedures.

Proposed Question:

River Bend Station is currently performing a refueling outage with core reload in progress.

A control rod blade guide must be moved from the core to the wall hangers in the upper pool. Due to the length of the blades, the mast must be raised beyond the HOIST UP position while traversing through the Portable Shielding (Cattle Chute).

Who must provide approval authority to allow the Refuel Bridge Driver to utilize the TRAVEL OVERRIDE and HOIST OVERRIDE interlock bypass features to move control rod blade guides through the Cattle Chute?

a. Control Room Supervisor

b. Spotter

c. Refuel SRO

d. Fuel Movement Supervisor

Proposed Answer: C.

Explanation (Optional):

FHP-0003 Roles and Responsibilities lists the Refuel SRO as the individual who may authorize the bypass of certain interlocks.

Technical Reference(s): FHP-0003, Rev 20, STM-0055, Rev

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0055 Obj. 6

Question Source: New

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ☒2

Comprehension or Analysis ☐

10 CFR Part 55 Content: 55.43 b.1

Comments:

QUESTION 96 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	3
	Group #	Equipment Control
	K/A #	G2.2.43
	Importance Rating	3.3

Knowledge of the process used to track inoperable alarms.

Proposed Question:

Per OSP-0015, Problem Annunciator Resolution Program, an annunciator that frequently alarms and clears and is distracting to the operating crew is termed a _____________.

a. Non-valid annunciator.

b. Anticipated annunciator.

c. Nuisance annunciator.

d. Valid annunciator.

Proposed Answer: C.

Explanation (Optional): OSP-0015 Section 4 defines the above terms. A nuisance annunciator may be valid or invalid. It is not considered a Problem annunciator until it is tracked in the OSP-0015 Log.

Technical Reference(s): OSP-0015 Rev 301

Proposed references to be provided to applicants during examination: NA

Learning Objective: NA

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.3

Comments:

QUESTION 97 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	3
	Group #	Radiation Control
	K/A #	G2.3.6
	Importance Rating	3.8

Ability to approve release permits.

Proposed Question:

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

a. Two independent samples of the tank are analyzed. A qualified member of the chemistry staff and a qualified member of the Radwaste staff independently verify the release rate calculations and the discharge valve lineup.

b. A single sample is analyzed by two qualified members of the chemistry staff independently. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.

c. Two independent samples of the tank are analyzed. Two qualified members of the chemistry staff independently verify the release rate calculations. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.

d. Two independent samples of the tank are analyzed. A qualified member of the chemistry staff and a qualified member of the Radwaste staff independently verify the release rate calculations and the discharge valve lineup. This is subsequently verified by the Control Room Supervisor.

Proposed Answer: C.

Explanation (Optional): ADM-0054 Section 5.5 provides guidance for discharges when RMS-RE107 is INOPERABLE. A second set of samples and analysis is required by two chemistry technicians. Independent discharge valve lineup by radwaste staff is also required.

Technical Reference(s): ADM-0054, Rev 6A

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0603 Obj. 8

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.4

Comments:

QUESTION 98 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	3
	Group #	Radiation Control
	K/A #	G 2.3.15
	Importance Rating	3.1

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question:

RMS-RE11A & RMS-RE11B, Reactor Building Annulus Exhaust Radiation Monitors, have both gone into HIGH ALARM.

Which of the following represents the appropriate response to this condition?

a. Direct the operator to verify the Annulus Pressure Control System has isolated in accordance with AOP-0003 Automatic Isolations.

b. Direct the operator to verify the Auxiliary Building Ventilation supply intake dampers have isolated in accordance with AOP-0003 Automatic Isolations.

c. Direct the operator to manually isolate the Auxiliary Building Ventilation System intake and exhaust dampers and start Standby Gas Treatment in accordance with SOP-0059 Reactor Building HVAC.

d. Direct the operator to verify the Auxiliary Building Ventilation supply intake dampers <u>and</u> the Annulus Pressure Control System have isolated in accordance with AOP-0003, Automatic Isolations.

Proposed Answer: A.

Explanation (Optional): The given radiation monitors cause an isolation of the annulus pressure control system and the auxiliary building exhaust fan intake dampers. Manual isolation of the entire Aux. Bldg Vent system and start of GTS is required when RMS-RE110 goes into High Alarm condition.

Technical Reference(s): AOP-0003, Rev 26

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-STM-0403 Obj. 6.3

Question Source: New

Question History: Last NRC Exam NA

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

10 CFR Part 55 Content: 55.43 b.4, b.5

Comments:

QUESTION 99 Rev 0

Examination Outline Cross-Reference:	Level	RO ☐ SRO ☒
	Tier #	3
	Group #	Emergency Procedures
	K/A #	G 2.4.28
	Importance Rating	4.1

Knowledge of procedures relating to a security event (non-safeguards information).

Proposed Question:

The outside operator has just notified the control room that armed intruders have been sighted at the clarifiers. This report has also been confirmed by the Security Shift Supervisor.

What is the appropriate response to this event?

a. Enter AOP-0001, Reactor Scram and insert a manual reactor scram.

b. Enter EOP-0001, RPV control and emergency depressurize the RPV.

c. Enter AOP-0054, Security Events, and wait for further information.

d. Enter GOP-0002, Plant Shutdown and begin a controlled shutdown of the plant.

Proposed Answer: C.

Explanation (Optional):

A reactor scram is not required until the it is confirmed that the armed intruders are in the Protected Area. The clarifiers are not in the Protected Area. Even when it is required to depressurize the RPV, the cooldown rate is limited to 100°F per hour. Emergency Depressurization is not authorized. When a plant shutdown is required, it is done by reactor scram not GOP-0002. The appropriate action is to enter AOP-0054 which provide guidance to enter AOP-0001 at appropriate trigger points which have not yet been reached in this event.

Technical Reference(s): AOP-0054, Rev 8

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-552 Obj. 2

Question Source: New

Question History: Last NRC Exam NA

Question Cognitive Level: Memory or Fundamental Knowledge ☐

Comprehension or Analysis ☒2

10 CFR Part 55 Content: 55.43 b.5

Comments:

QUESTION 100 Rev 0

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	3
	Group #	Emergency Procedures_
	K/A #	G 2.4.49
	Importance Rating	4.4

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question:

The following plant conditions exist:

Reactor power 83%
Core flow 68 Mlbm/hr

A failure of the heater drain system has resulted in a 20°F drop in feedwater temperature requiring entry into AOP-0007, Loss of Feedwater Heating.

What is the expected response upon entry into the procedure?

a. Direct the operator to reduce reactor recirculation flow until reactor thermal power lowers to 80% or 50.7 Mlbm/hr core flow is reached.

b. Direct the operator to lower power as close to 63% as possible without entering the Monitored Region of the Power to Flow Map.

c. Direct the operator to reduce reactor recirculation flow until reactor thermal power lowers to 63% or 50.7 Mlbm/hr is reached.

d. Direct the operator to lower power as close to 80% as possible without entering the Monitored Region of the Power to Flow Map.

Proposed Answer: C.

Explanation (Optional):
The immediate operator actions of AOP-0007 Loss of Feedwater Heating state: “Reduce recirculation flow until thermal power lowers by 20% (620 MWth) or 60% (50.7 Mlbm/hr) core flow is reached.

Technical Reference(s): AOP-0007, Rev 23

Proposed references to be provided to applicants during examination: NA

Learning Objective: RLP-HLO-0526 Obj. 4

Question Source: New

Question History:	Last NRC Exam NA	
Question Cognitive Level:	Memory or Fundamental Knowledge	☐
	Comprehension or Analysis	☒2
10 CFR Part 55 Content:	55.43 b.5	
Comments:		