Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K1.04
	Importance Rating	3.2

EDGs

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

Proposed Question: #1

Which one of the following identifies the Emergency Service Water (ESW) pumps that can supply cooling water to Diesel Generator A?

A. A, only

B. A and B, only

C. A and C, only

D. A, B, C, and D

Proposed Answer: D

Explanation: All four ESW pumps can supply cooling water to DG A.

A. Incorrect – All four ESW pumps can supply cooling water to DG A. Plausible because on manual start of DG A, only ESW pump A automatically starts.
B. Incorrect – All four ESW pumps can supply cooling water to DG A. Plausible because on manual start of DG A, only one ESW loop automatically supplies cooling water to DG A and some plants have A and B components in loop A.
C. Incorrect – All four ESW pumps can supply cooling water to DG A. Plausible because on manual start of DG A, only ESW loop A automatically supplies cooling water to DG A. ESW pumps A and C are in ESW loop A.

Technical Reference(s): OP-054 Att. A

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-24 RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K1.09
	Importance Rating	4.0

HPCI

Knowledge of the physical connections and/or cause-effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: ECCS keep fill system: BWR-2,3,4(P-Spec)

Proposed Question: #2

Which one of the following describes the normal source of water to the HPCI keep fill system and where this water source injects?

A. Dedicated keep fill pump; injects upstream of HV-155-F006, HPCI INJECTION

B. Dedicated keep fill pump; injects downstream of HV-155-F006, HPCI INJECTION

C. Condensate Transfer pumps; injects upstream of HV-155-F006, HPCI INJECTION

D. Condensate Transfer pumps; injects downstream of HV-155-F006, HPCI INJECTION

Proposed Answer: C

Explanation: The HPCI keep fill system is normally supplied with water from the Condensate Transfer pumps. Keep fill water is injected upstream of HV-155-F006, HPCI INJECTION.

A. Incorrect – The HPCI keep fill system is normally supplied with water from the Condensate Transfer pumps. Plausible because many plants have dedicated ECCS keep fill pumps and SSES does have a passive backup supply from a dedicated tank.
B. Incorrect – The HPCI keep fill system is normally supplied with water from the Condensate Transfer pumps. Plausible because many plants have dedicated ECCS keep fill pumps and SSES does have a passive backup supply from a dedicated tank. Keep fill water is injected upstream of HV-155-F006, HPCI INJECTION. Plausible if the candidate mixes up the location of keep fill relative to this valve.
D. Incorrect – Keep fill water is injected upstream of HV-155-F006, HPCI INJECTION. Plausible if the candidate mixes up the location of keep fill relative to this valve.

Technical Reference(s): M-155 Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-52 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K2.01
	Importance Rating	2.6

RHR/LPCI: Injection Mode

Knowledge of electrical power supplies to the following: Pumps

Proposed Question: #3

Unit 1 is operating at 100% power with the following:

- Startup Bus 10 (0A103) de-energizes due to a sustained electrical fault.
- NO ESS buses automatically swap to their alternate power source.
- NO Diesel Generators start.

Which one of the following identifies the Unit 1 RHR pumps available for LPCI?

A. 1A and 1B

B. 1A and 1C

C. 1B and 1D

D. 1C and 1D

Proposed Answer: C

Explanation: When Startup Bus 10 de-energizes, Unit 1 ESS Buses 1A (1A201) and 1C (1A203) de-energize. With failure of DGs A and C to start, these ESS Buses remain de-energized. This makes RHR pumps 1A and 1C unavailable for LPCI. Therefore, only RHR pumps 1B and 1D remain available for LPCI.

A. Incorrect – RHR pump 1A is unavailable because ESS Bus 1A is de-energized. Plausible if candidate mixes up which ESS Buses normally receive power from Startup Bus 10.
B. Incorrect – RHR pump 1A is unavailable because ESS Bus 1A is de-energized. RHR pump 1C is unavailable because ESS Bus 1C is de-energized. Plausible if candidate mixes up which ESS Buses normally receive power from Startup Bus 10.
D. Incorrect – RHR pump 1C is unavailable because ESS Bus 1C is de-energized. Plausible if candidate mixes up which ESS Buses normally receive power from Startup Bus 10.

Technical Reference(s): ON-Sub-001 Att. D, ON-4KV-101 Att J & D

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-49 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 K2.01
	Importance Rating	2.6

SRM

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

Proposed Question: #4

Which one of the following identifies the power supplies to the Unit 1 SRM detectors and drive motors?

	SRM Detectors	SRM Drive Motors
A.	1D672 and 1D682	1Y125
B.	1D672 and 1D682	1Y218
C.	1Y201A and 1Y201B	1Y125
D.	1Y201A and 1Y201B	1Y218

Proposed Answer: B

Explanation: The Unit 1 SRM detectors are powered from 24 VDC Buses 1D672 and 1D682. The Unit 1 SRM drive motors are powered from 208/120 VAC Bus 1Y218.

A. Incorrect – The Unit 1 SRM drive motors are powered from 208/120 VAC Bus 1Y218. Plausible because 1Y125 supplies multiple other 120 VAC instrumentation loads.
C. Incorrect – The Unit 1 SRM detectors are powered from 24 VDC Buses 1D672 and 1D682. Plausible because RPS Buses 1Y201A and 1Y201B supply power to the PRNMs. The Unit 1 SRM drive motors are powered from 208/120 VAC Bus 1Y218. Plausible because 1Y125 supplies multiple other 120 VAC instrumentation loads.
D. Incorrect – The Unit 1 SRM detectors are powered from 24 VDC Buses 1D672 and 1D682. Plausible because RPS Buses 1Y201A and 1Y201B supply power to the PRNMs.

Technical Reference(s): E-13 Sheet 1, ON-YPNL-101 Att. H

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-78A RBO-3

Question Source: Bank – NMP1 2010 Audit #39

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K3.03
	Importance Rating	3.6

PCIS/Nuclear Steam Supply Shutoff

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Off-site radioactive release rates

Proposed Question: #5

Unit 1 has experienced a loss of coolant accident with the following:

- The Suppression Pool is being vented using the Standby Gas Treatment system (SGTS) due to the presence of hydrogen.
- SGTS Exhaust Vent Monitor A (RE-D12-0N017A) fails downscale.
- SGTS Exhaust Vent Monitor B (RE-D12-0N017B) indicates 25 mR/hr, up slow.

Which one of the following describes the resulting status of the release from this flow path?

The release from this flow path...

A. continues at the initial rate.

B. continues at half the initial rate.

C. is stopped by closure of two isolation valves.

D. is stopped by closure of just one isolation valve.

Proposed Answer: D

Explanation: RE-D12-0N017A and B each close one isolation valve, HV-15703 and HV-25703 respectively, in this flow path if they exceed 23 mR/hr. With RE-D12-0N017A failed downscale, HV-15703 does not close. With RE-D12-0N017B indicating 25 mR/hr, HV-25703 closes. This valve closure stops the release from this flow path.

A. Incorrect – With RE-D12-0N017B indicating 25 mR/hr, HV-25703 closes. This valve closure stops the release from this flow path. Plausible if isolation logic needed both rad monitors high to cause valve closure, or if 25 mR/hr was not above the isolation setpoint.
B. Incorrect – With RE-D12-0N017B indicating 25 mR/hr, HV-25703 closes. This valve closure stops the release from this flow path. Plausible because only 1 out of 2 isolation valves close with the given malfunction.
C. Incorrect – With RE-D12-0N017A failed downscale, HV-15703 does not close. Plausible if logic was 1 out of 2 to close both isolation valves.

Technical Reference(s): AR-101-001, AR-015-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-70 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K3.02
	Importance Rating	3.2

Shutdown Cooling

Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor water level: Plant-Specific

Proposed Question: #6

Unit 1 is shutdown with the following:

- Reactor coolant temperature is 120°F, down slow.
- RHR pump 1A is operating in the Shutdown Cooling (SDC) lineup.
- A leak develops on the RHR pump 1A suction flange.
- The leak does NOT cause a high SDC flow isolation.
- Reactor water level is 54", down slow.

Which one of the following identifies the Reactor water level at which an isolation signal will first be generated that will stop this leak?

A. 13"

B. -30"

C. -38"

D. -129"

Proposed Answer: A

Explanation: At 13", an isolation signal will be received that will close the following valve to isolate this leak:

- HV-151-F008, SHUTDOWN CLG SUCT OB ISO
- HV-151-F009, SHUTDOWN CLG SUCT IB ISO
- HV-151-F015A, RHR INJ OB ISO
- HV-151-F015B, RHR INJ IB ISO

B. Incorrect – An isolation signal that will isolate this leak is first received at 13". -30" is the Reactor water level at which RCIC gets a start signal.
C. Incorrect – An isolation signal that will isolate this leak is first received at 13". Another isolation signal is received at -38".
D. Incorrect – An isolation signal that will isolate this leak is first received at 13". -129" is the Reactor water level at which RHR gets a LPCI initiation signal.

Technical Reference(s): ON-159-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-49

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	239002 K4.08
Importance Rating	3.6

SRVs

Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Opening of the SRV from either an electrical or mechanical signal

Proposed Question: #7

Unit 1 is operating at 100% power with the following:

- 125 VDC 1D614 is de-energized due to a sustained electrical fault.
- A Main Turbine trip occurs and Turbine Bypass Valves are slow to operate.
- Reactor pressure reaches a peak of 1220 psig during the resulting transient.

Which one of the following describes the operation of the SRVs during this transient?

A. NO SRVs open on an electrical signal. At least one SRV opens by self-actuation (safety function).

B. NO SRVs open on an electrical signal. NO SRVs open by self-actuation (safety function).

C. All 16 SRVs open on an electrical signal. NO SRVs open by self-actuation (safety function).

D. All 16 SRVs open on an electrical signal. At least one SRV opens by self-actuation (safety function).

Proposed Answer: A

Explanation: Loss of 1D614 results in loss of the relief function of all 16 SRVs, meaning they will not open on high pressure due to an electrical signal. The SRVs can also open on high pressure due to self-actuation (mechanical opening based on Reactor pressure overcoming spring pressure). This is referred to as the safety function. The 16 SRVs have safety function set pressure requirements ranging from 1140 psig to 1241 psig. With Reactor pressure reaching a peak of 1220 psig due to slow operation of the Turbine Bypass Valves, multiple SRVs opened by self-actuation.

B. Incorrect – Reactor pressure rose high enough to open multiple SRVs via the safey function. Plausible because normally operation of TBVs and/or opening of SRVs via the relief function would prevent this from happening.
C. Incorrect – No SRVs opened on an electrical signal due to the loss of 1D614. Plausible because other electrical functions of the SRVs remain operable (ADS, opening using key lock switches in lower relay room). Reactor pressure rose high enough to open multiple SRVs via the safey function. Plausible because normally operation of TBVs and/or opening of SRVs via the relief function would prevent this from happening.
D. Incorrect – No SRVs opened on an electrical signal due to the loss of 1D614. Plausible because other electrical functions of the SRVs remain operable (ADS, opening using key lock switches in lower relay room).

Technical Reference(s): ON-125VDC-101 Att C, DBD 16

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-83 RBO-2

Question Source: Modified Bank – Oyster Creek 2009 NRC #7

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K4.14
	Importance Rating	3.4

Reactor Water Level Control

Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Selection of various instruments to provide reactor water level input

Proposed Question: #8

Unit 2 is operating at 100% power with the following Reactor water level inputs to ICS:

- NRLA +28 inches
- NRLBB +50 inches
- NRLC +35 inches
- UPSETLB +33 inches

Which one of the following describes the ICS response to these inputs?

A. The NRLBB input to the average level calculation is replaced by NRLC.

B. The NRLBB input to the average level calculation is replaced by UPSETLB.

C. Level control will switch from using the average level calculation to using NRLC.

D. Level control will switch from using the average level calculation to using UPSETLB.

Proposed Answer: B

Explanation: ICS has a level validation scheme. A low median signal is selected from NRLA, NRLBB, NRLC and UPSETLB. ICS disregards both the high signal and the low signal. Of the two median (middle) signals ICS chooses the low median for level instrument validation. In this case, the low median signal is 33", based on UPSETLB. If an input deviates by more than 10" from the low median signal, it is marked as DEVIANT. In this case, NRLBB is more than 10" above the low median signal, and therefore DEVIANT. This causes NRLBB to be replaced in the average level calculation by the low median signal of 33" from UPSETLB.

A. Incorrect – NRLBB is replaced by the low median signal, which is 33" from UPSETLB. Plausible because NRLC is the high median signal, indicates at the normal Reactor water level, and is higher priority than UPSETLB in the separate auto selected level logic.
C. Incorrect – Since only one signal (NRLBB) is DEVIANT, ICS will continue to use average level calculation, just with the NRLBB signal replaced. Average level calculation is abandoned if a second NRL instrument is lost and NRLC is one of the possible replacements. NRLA is low enough below low median to cause an alarm, but not so low to be marked DEVIANT.
D. Incorrect – Since only one signal (NRLBB) is DEVIANT, ICS will continue to use average level calculation, just with the NRLBB signal replaced. Average level calculation is abandoned if a second NRL instrument is lost and UPSETLB is one of the possible replacements. NRLA is low enough below low median to cause an alarm, but not so low to be marked DEVIANT.

Technical Reference(s): AR-1651-003 Att A

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-31F RBO-4

Question Source: Modified Bank – LOC27 NRC #26

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K5.04
	Importance Rating	2.8

LPCS

Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM: Heat removal (transfer) mechanisms

Proposed Question: #9

Unit 1 has experienced a loss of coolant accident with the following:

- Emergency RPV Depressurization has been performed.
- Reactor pressure is 75 psig, steady.
- Reactor water level is -198", down slow.
- The only Reactor injection sources are Core Spray pumps 1A and 1C.
- Total Core Spray flow is 6700 gpm.

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

Adequate core cooling ***via Spray Cooling*** is currently...

A. being provided.

B. NOT being provided because Core Spray flow is too low.

C. NOT being provided because Reactor water level is too low.

D. NOT being provided because only one loop of Core Spray is injecting.

Proposed Answer: A

Explanation: Adequate core cooling via Spray Cooling is currently being provided because Reactor water level is above -210" and one loop of Core Spray is injecting greater than 6350 gpm to the Reactor.

B. Incorrect – One Core Spray loop is injecting greater than 6350 gpm to the Reactor, therefore that aspect of Spray Cooling is satisfied. Plausible if candidate believes a higher Core Spray flow is required.
C. Incorrect – Reactor water level is above -210", therefore that aspect of Spray Cooling is satisfied. Plausible because Reactor water level is below the levels for adequate core cooling by other means (-161" for submergence, -179" is steam cooling in EO-100-102).
D. Incorrect – One Core Spray loop is injecting greater than 6350 gpm to the Reactor, therefore that aspect of Spray Cooling is satisfied. Plausible if candidate believes both loops of Core Spray must be injecting (Core Spray pumps 1A and 1C are both in Core Spray loop A).

Technical Reference(s): EO-100-102, EO-000-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC23 NRC #51

Question History: LOC23 NRC #51

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

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

AC Electrical Distribution

Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: Breaker control

Proposed Question: #10

A Unit 1 startup is in progress with the following:

- The Main Generator has been synchronized to the grid.
- Aux Bus 11A is being transferred to the Main Generator supply.
- The control switch for AUX XFMR 11 TO BUS 11A BKR 1A10101 sticks in the CLOSE position when released, and cannot be returned to the Normal After Close position.
- AUX XFMR 11 TO BUS 11A BKR 1A10101 closes.

Which of the following describes the Unit 1 response?

A. TIE BUS TO BUS 11A BKR 1A10104 immediately opens.
Aux Bus 11A ends up fed from the Aux Transformer only.

B. TIE BUS TO BUS 11A BKR 1A10104 opens after a 10–second time delay.
Aux Bus 11A ends up fed from the Aux Transformer only.

C. AUX XFMR 11 TO BUS 11A BKR 1A10101 opens after a 10–second time delay.
Aux Bus 11A ends up fed from Startup Bus 10 only.

D. TIE BUS TO BUS 11A BKR 1A10104 and AUX XFMR 11 TO BUS 11A BKR 1A10101 remain closed.
Aux Bus 11A ends up fed from both Startup Bus 10 and the Aux Transformer.

Proposed Answer: D

Explanation: On normal transfer of this bus, TIE BUS TO BUS 11A BKR 1A10104 opens once the control switch for AUX XFMR 11 TO BUS 11A BKR 1A10101 returns to Normal After Close (red flagged and neutral). This breaker opening logic requires the related control switch in NAT/NAC, the related sync switch ON, and the related Aux Incoming breaker closed. Without all three conditions, TIE BUS TO BUS 11A BKR 1A10104 remains closed. There is no automatic re-opening logic that will cause AUX XFMR 11 TO BUS 11A BKR 1A10101 to open. Therefore, both breakers remain closed and Aux Bus 11A ends up fed from both Startup Bus 10 and the Aux Transformer.

A. Incorrect – 1A10104 will NOT immediately open. Plausible because it would immediately open if not for the control switch malfunction.
B. Incorrect – 1A10104 will NOT open after a time delay. Plausible because it would normally open if not for the control switch malfunction and there is a 10 second time delay associated with electrical bus undervoltage logic.
C. Incorrect – 1A10101 will NOT open after a time delay. Plausible because normal interlock prevents the Aux Bus from being supplied by both sources for a prolonged period of time, the control switch has malfunctioned, and there is a 10 second time delay associated with electrical bus undervoltage logic.

Technical Reference(s): OP-103-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-003 RBO-4

Question Source: Bank – LOC25 NRC #3

Question History: LOC25 NRC #3

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K6.03
	Importance Rating	2.7

Instrument Air

Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: Temperature indicators

Proposed Question: #11

Unit 1 is operating at 100% power with the following:

- Instrument Air Compressor (IAC) 1K107A is running in LEAD.
- The sensed aftercooler outlet air temperature drifts upscale over approximately one (1) minute.

Which one of the following describes the response of Turbine Building Closed Cooling Water (TBCCW) flow to the aftercooler and the response of IAC 1K107A?

TBCCW flow to the aftercooler...

A. rises. IAC 1K107A trips.

B. rises. IAC 1K107A does NOT trip.

C. remains the same. IAC 1K107A trips.

D. remains the same. IAC 1K107A does NOT trip.

Proposed Answer: A

Explanation: TBCCW flow to the aftercooler is automatically modulated by temperature control valve TCV12504A1 based on sensed aftercooler outlet air temperature. With temperature drifting high, TCV12504A1 opens to raise TBCCW flow in an attempt to lower temperature. Actual aftercooler outlet air temperature will therefore lower, but sensed temperature continues to drift upscale based on the given failure. IAC 1K107A trips when sensed temperature exceeds 320°F.

B. Incorrect – IAC 1K107A trips based on aftercooler outlet air temperature >320°F. Plausible because not all sensed IAC parameters lead to a trip.
C. Incorrect – TBCCW flow rises due to the automatic response of TCV12504A1. Plausible because only certain TBCCW loads have automatic TCVs such as this.
D. Incorrect – TBCCW flow rises due to the automatic response of TCV12504A1. Plausible because only certain TBCCW loads have automatic TCVs such as this. IAC 1K107A trips based on aftercooler outlet air temperature >320°F. Plausible because not all sensed IAC parameters lead to a trip.

Technical Reference(s): LA-1140-001, M-125 Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-4

Question Source: Modified Bank – LOC23 Cert #18

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K6.03
	Importance Rating	2.7

UPS (AC/DC)

Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.): Static inverter

Proposed Question: #12

Unit 1 is operating at 100% power with the following:

- Non-Class 1E Instrument AC UPS 1D130 is in service.
- Then, the UPS 1D130 inverter fails and the inverter output goes to zero.

Which one of the following describes the effect on Instrument AC Distribution Panel 1Y128?

1Y128 is...

A. automatically powered by UPS 1D130 from 1B246.

B. automatically powered by UPS 1D130 from 1D133.

C. automatically powered by the Static Switch from 1B226.

D. de-energized and must be manually transferred to 1B226.

Proposed Answer: C

Explanation: Panel 1Y128 is normally powered by UPS 1D130 from 1B246 through the inverter. When the inverter fails and its output goes to zero, power from 1B246 is interrupted. The backup power supply from 1D133 also requires the inverter to be functioning, so it does not automatically supply power in this case. UPS 1D130 has a static switch that does not require the inverter to be functioning to supply 1Y128. Therefore, this static switch automatically supplies 1Y128 with power from 1B226 with the given malfunction.

A. Incorrect – The static switch automatically supplies 1Y128 with power from 1B226. Plausible because 1B246 is the normal power source to 1Y128.
B. Incorrect – The static switch automatically supplies 1Y128 with power from 1B226. Plausible because 1D133 is the normal backup power supply, but also requires a functioning inverter to supply 1Y128.
D. Incorrect – The static switch automatically supplies 1Y128 with power from 1B226. Plausible because the maintenance bypass switch is designed for such a manual transfer, but would only be required if the static switch failed or was initially NOT in the normal AUTO position.

Technical Reference(s): TM-OP-017

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-017 RBO-03

Question Source: Bank – NMP1 2010 Audit #7

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 A1.02
	Importance Rating	3.1

SGTS

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

Proposed Question: #13

A Unit 1 startup is in progress with the following:

- Drywell inerting is in progress.
- Nitrogen is being added to the Drywell.
- Standby Gas Treatment (SGTS) fan A is running and aligned to take suction from the Unit 1 Drywell only.
- Drywell pressure is 0.07 psig, down slow.

Then, the control switch for DRWL/WETWELL BURP DMP HD-17508B is taken to CLOSE.

Which one of the following describes the effect of this switch manipulation on Drywell pressure?

Drywell pressure will...

A. rise.

B. remain approximately constant.

C. continue to lower, but at a faster rate.

D. continue to lower, but at a slower rate.

Proposed Answer: A

Explanation: The Drywell is initially being inerted by adding nitrogen and simultaneously venting air through SGTS fan A through open series dampers HD-17508A and HD-17508B. The nitrogen addition rate is initially lower than the SGTS flow rate, therefore Drywell pressure is lowering slowly. When the control switch for DRWL/WETWELL BURP DMP HD-17508B is taken to CLOSE, HD-17508B closes. This completely isolates the SGTS flow path from the Drywell. No interlock exists which isolates the nitrogen flowing into the Drywell, so Drywell pressure will begin to rise.

B. Incorrect – Drywell pressure rises because SGTS flow out is stopped while nitrogen flow in remains steady. Plausible if interlock also isolated the nitrogen flow path on low SGTS flow. There are other interlocks based on low SGTS flow.
C. Incorrect – Drywell pressure rises because SGTS flow out is stopped while nitrogen flow in remains steady. Plausible if this damper isolated the nitrogen flow path instead of the SGTS flow path.
D. Incorrect – Drywell pressure rises because SGTS flow out is stopped while nitrogen flow in remains steady. Plausible if this damper was only associated with the SGTS train B or had an interlock to not close before the fan was stopped.

Technical Reference(s): M-175 Sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-070 EO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A1.09
	Importance Rating	4.0

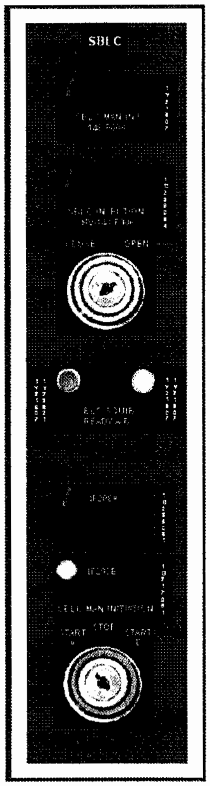
SLC

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: SBLC system lineup

Proposed Question: #14

Unit 1 has experienced a failure to scram with the following:

- An operator has initiated Standby Liquid Control (SLC).
- The following Standby Liquid Control indications are now present:



Which one of the following describes the status of the Standby Liquid Control system?

A. SLC has responded as designed.

B. SLC squib valve A has failed to fire.

C. SLC squib valve B has failed to fire.

D. SLC pump B has failed to start.

Proposed Answer: C

Explanation: The given indications show the SLC manual initiation keylock switch has been taken to A START. This should result in the start of SLC pump A and firing of both SLC squib valves A and B, as evidenced by their white lights extinguishing. The given indications show SLC pump A running as expected, SLC pump B in standby as expected, and SLC squib valve A white light extinguished as expected. However, the SLC squib valve B white light is not extinguished, indicating that it has failed to fire.

A. Incorrect – SLC squib valve B has failed to fire. While SLC is injecting with pump A through squib valve A, squib valve B is also supposed to fire per design.
B. Incorrect – SLC squib valve B has failed to fire. Plausible if candidate mixes up how the squib valve white lights indicate for a fired squib valve.
D. Incorrect – SLC squib valve B has failed to fire. Plausible because SLC pump B is not running, both squib valves are supposed to fire, but only one SLC pump is supposed to run.

Technical Reference(s): OP-153-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-053 RBO-4

Question Source: Modified Bank – LOC24 NRC #34

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

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

DC Electrical Distribution

Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of ventilation during charging

Proposed Question: #15

Unit 1 is operating at 100% power with the following:

- An equalizing charge is in progress on 125 VDC Battery 1D610.
- Approximately 24 hours remain in the equalizing charge.
- All Battery Room Ventilation is lost.
- Maintenance reports that it will take approximately 6 hours to restore Battery Room Ventilation.
- Battery Room temperature is 80°F for Battery 1D610.

Which one of the following describes how to control the equalizing charge and the associated reason or limitation, in accordance with OP-102-001, 125V DC System, and ON-CSHVAC-001, Loss of Control Structure HVAC?

A. Return Battery Charger 1D613 to FLOAT to prevent hydrogen buildup.

B. Return Battery Charger 1D613 to FLOAT to prevent excessive temperatures.

C. Continue the equalizing charge as long as room temperature remains below 102°F.

D. Continue the equalizing charge as long as air samples indicate less than 1% hydrogen.

Proposed Answer: A

Explanation: OP-102-001 Precaution 2.4.2 requires returning the charger to float to prevent hydrogen buildup. ON-CSHVAC-001 also requires ensuring all battery chargers are in float within 3 hours.

B. Incorrect – The equalizing charge must be secured, but the reason is to prevent hydrogen buildup, not prevent excessive temperatures.
C. Incorrect – The equalizing charge must be secured regardless of actual room temperature rise based on hydrogen generation concerns. 102°F is a high temperature limitation associated with Reactor Building Emergency Switchgear Rooms.
D. Incorrect – The equalizing charge must be secured regardless of actual hydrogen concentration. 1% hydrogen concentration is the lowest value used in EOP combustible gas control strategies.

Technical Reference(s): OP-102-001, ON-CSHVAC-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-002 RBO-5

Question Source: Bank – LOC26R NRC #26

Question History: LOC26R NRC #26

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A2.01
	Importance Rating	2.8

IRM

Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded

Proposed Question: #16

Unit 1 is preparing for a startup with the following:

- The Reactor Mode Switch in STARTUP.
- All control rods are fully inserted.

Then, the high voltage power supply to IRM B fails low.

Which one of the following describes (1) the plant response and (2) what action(s), if any, is(are) required to correct conditions?

A. (1) NO control rod block or half scram
(2) NO action is required.

B. (1) Control rod block, but NO half scram
(2) Bypass IRM B.

C. (1) Control rod block and half scram
(2) Bypass IRM B and then rotate REACTOR SCRAM RESET HS to GRP 1/4 position, only.

D. (1) Control rod block and half scram
(2) Bypass IRM B and then rotate REACTOR SCRAM RESET HS to both GRP 1/4 and GRP 2/3 positions.

Proposed Answer: D

Explanation: An IRM INOP condition results from detector power supply high voltage low. A control rod withdrawal block is generated by the INOP condition. An RPS half scram signal is also generated by the INOP condition. To clear these conditions, IRM B must be bypassed, and RPS must be reset by rotating RPS SCRAM RESET HS to both GRP 1/4 and GRP 2/3 positions.

A. Incorrect – A rod block and half scram occur. Plausible because the IRM downscale by itself will not generate a control rod block or half scram, as IRMs are on Range 1 with all control rods inserted and startup about to commence.
B. Incorrect – A half scram occurs. Plausible because, in other conditions, an IRM downscale by itself only causes a rod block.
C. Incorrect – In this situation GRP 2/3 must also be reset. Plausible because only a half scram is received, however all 4 groups on that half of RPS must be reset.

Technical Reference(s): AR-104-A05, OP-158-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078B RBO-4

Question Source: Bank – LOC25 NRC #15

Question History: LOC25 NRC #15

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 A3.03
	Importance Rating	3.7

RCIC

Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: System pressure

Proposed Question: #17

Which one of the following describes a parameter value that would cause an automatic RCIC isolation?

A. Reactor pressure of 150 psig.

B. RCIC steam supply header D/P of 110" H_2O.

C. RCIC equipment room area temperature of 125°F.

D. RCIC turbine exhaust rupture diaphragm pressure of 15 psig.

Proposed Answer: D

Explanation: High RCIC turbine exhaust rupture diaphragm pressure causes an automatic RCIC isolation at 10 psig.

A. Incorrect – RCIC does not isolate on low Reactor pressure until <60 psig. 150 psig is based on the RCIC operability requirement.
B. Incorrect – RCIC does not isolate on high steam supply header D/P until 188" H_2O. 110" H_2O is elevated and based on the approximate RCIC % overspeed trip setpoint.
C. Incorrect – RCIC does not isolate on high equipment room temperature until 167°F. 125°F is based on being higher than the associated Max Normal temperature in EO-100-104

Technical Reference(s): AR-108-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-50 RBO-4

Question Source: Bank – JAF 4/14 NRC #21

Question History: JAF 4/14 NRC #21

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 A3.05
	Importance Rating	3.3

APRM / LPRM

Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Flow converter/comparator alarms

Proposed Question: #18

Unit 1 is operating at 85% power with the following:

- A malfunction has caused the Reactor Recirc loop A flow signal to APRM 1 to drift low.
- NO APRM upscale flow-biased setpoints have been exceeded.
- The Reactor Recirc loop flow signals to APRM 1 stabilize at the following values:
    - Reactor Recirc loop A: 84%
    - Reactor Recirc loop B: 97%

Which one of the following describes the automatic response to this malfunction, if any?

A. NO alarm, rod block, or half scram occurs.

B. Annunciator AR-103-E06, APRM FLOW REFERENCE OFF NORMAL, alarms. NO rod block or half scram occurs.

C. Annunciator AR-103-E06, APRM FLOW REFERENCE OFF NORMAL, alarms and a rod block occurs. NO half scram occurs.

D. Annunciator AR-103-E06, APRM FLOW REFERENCE OFF NORMAL, alarms, a rod block occurs, and a half scram occurs.

Proposed Answer: B

Explanation: The difference in Recirc Loop flows to APRM 1 is 12%. This causes annunciator AR-103-E06 to alarm. This flow comparator function is alarm only. No rod block or half scram occurs.

A. Incorrect – The difference in Recirc Loop flows to APRM 1 is 12%. This causes annunciator AR-103-E06 to alarm. Plausible because if the difference were <7%, no automatic response would occur.
C. Incorrect – No rod block occurs. Plausible because another condition that causes AR-103-E06 to alarm (flow reference >112%) would cause a rod block.
D. Incorrect – No rod block occurs. Plausible because another condition that causes AR-103-E06 to alarm (flow reference >112%) would cause a rod block. No half scram occurs. Plausible because other APRM malfunctions (including a Recirc flow signal failed downscale) would cause a half scram.

Technical Reference(s): AR-103-E06

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-78D RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 A4.11
	Importance Rating	3.7

RPS

Ability to manually operate and/or monitor in the control room: Scram air header pressure

Proposed Question: #19

The plant is operating at 15% power with the following:

- A malfunction of the Feedwater system occurs.
- A manual Scram was inserted at T=0.
- Reactor water level responds per the graph below.
- Reactor pressure remains within a band of 800 to 1000 psig throughout the transient.

Note: Assume the Scram Discharge Volume water level remains below the scram setpoint during this time frame.

Which one of the following describes the approximate **earliest** time the Scram Air Header can be re-pressurized following the Reactor scram?

The Scram Air Header can **first** be re-pressurized at approximately...

A. 12 seconds.

B. 20 seconds.

C. 30 seconds.

D. 55 seconds.

Proposed Answer: B

Explanation: When Reactor water level drops below 13" (at approximately 2 seconds), an automatic Reactor scram signal is generated. The scram results in Scram Air Header pressure dropping to 0 psig. For the scram to be reset, at least 10 seconds must pass after the initial scram. This time delay is designed to ensure all control rods insert. Additionally, the initiating condition must be clear for the scram to be reset. The scram reset time delay times out at approximately 12 seconds. Reactor water level recovers above 13" at approximately 20 seconds. Therefore, the earliest time in this scenario that the Scram Air Header can be re-pressurized is approximately 20 seconds.

A. Incorrect – The Scram Air Header cannot be re-pressurized until approximately 20 seconds. 12 seconds is plausible because this is when the 10 second time delay from the initiating scram expires, however Reactor water level is still too low to reset the scram.
C. Incorrect – The Scram Air Header can be re-pressurized at the earlier time of approximately 20 seconds. 30 seconds is plausible because this is 25 seconds after Reactor water level reached -30". 25 seconds is the time delay for resetting ARI, which actuates at -38", not -30". -30" is the level at which RCIC initiates.
D. Incorrect – The Scram Air Header can be re-pressurized at the earlier time of approximately 20 seconds. 55 seconds is plausible because this is 25 seconds after Reactor water level recovered above 13". 25 seconds is the time delay for resetting ARI, but not an RPS scram signal.

Technical Reference(s): M-1-C72-22 Sheet 10 & 11

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-058 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A4.01
	Importance Rating	3.1

Component Cooling Water

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

Proposed Question: #20

Unit 1 is operating at 100% power with the following:

- RBCCW COOLER TEMP TIC-11028 is in Manual.
- The OPEN pushbutton is depressed.

Which one of the following describes the response of RBCCW temperature?

RBCCW temperature...

A. rises due to less RBCCW flow through the RBCCW heat exchangers.

B. rises due to less Service Water flow through the RBCCW heat exchangers.

C. lowers due to more RBCCW flow through the RBCCW heat exchangers.

D. lowers due to more Service Water flow through the RBCCW heat exchangers.

Proposed Answer: D

Explanation: Depressing the OPEN pushbutton on TIC-11028 when it is in Manual causes Temperature Control Valve TV-11028 to open. This valve is on the combined Service Water outlet from the RBCCW heat exchangers. Opening this valve causes more Service Water flow to pass through the RBCCW heat exchangers, lowering RBCCW temperature.

A. Incorrect – RBCCW temperature lowers due to more Service Water flow through the RBCCW heat exchangers. Plausible if the controller was setup to be reverse acting, as they are at many plants, or if it controlled a bypass valve, which upon opening decreased flow through the heat exchanger.
B. Incorrect – RBCCW temperature lowers due to more Service Water flow through the RBCCW heat exchangers. Plausible if the controller was setup to be reverse acting, as they are at many plants, or if it controlled a bypass valve, which upon opening decreased flow through the heat exchanger.
C. Incorrect – RBCCW temperature lowers due to more Service Water flow through the RBCCW heat exchangers. Plausible because some plants maintain full Service Water flow through RBCCW heat exchangers and modulate the RBCCW flow for temperature control.

Technical Reference(s): M-110 Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-014 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 2.1.27
	Importance Rating	3.9

ADS

Knowledge of system purpose and / or function.

Proposed Question: #21

Unit 1 is operating at 100% power with the following:

- SRV G inadvertently opens.
- SRV G is closed by pulling fuses in Upper Relay Room panel 1C628 in accordance with ON-SRV-101, Stuck Open SRV.

Which one of the following describes the resulting ability of SRV G to perform its automatic pressure relief control function and its ADS function?

A. Both the automatic pressure relief control function and the ADS function will still work.

B. The automatic pressure relief control function will work, but the ADS function will NOT.

C. The ADS function will work, but the automatic pressure relief control function will NOT.

D. NEITHER the automatic pressure relief control function NOR the ADS function will work.

Proposed Answer: C

Explanation:

A. Incorrect – The pulled fuses only affect the electrical opening of the SRV including the relief function.
B. Incorrect –The spring safety function will still work.
D. Incorrect – ADS is a different set of fuses and will work.

Technical Reference(s): ON-SRV-101, M1-B21-129 sheet 4 or E-180 sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083E RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 2.4.35
	Importance Rating	3.8

Instrument Air

Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Proposed Question: #22

Both Units are operating at 100% power with the following:

- Instrument Air Compressors (IACs) 1A and 1B have both tripped.
- Unit 1 Instrument Air header pressure is 78 psig, down slow.
- Service Air has NOT been successful in restoring Instrument Air header pressure.
- The Shift Manager has directed cross-tying Unit 1 and Unit 2 Instrument Air.

Which one of the following describes (1) how this task is performed and (2) the ability of this lineup to automatically isolate if Unit 2 Instrument Air header pressure also degrades?

	(1)	(2)
A.	Performed in Main Control Room	Lineup will automatically isolate
B.	Performed in Main Control Room	Lineup will NOT automatically isolate
C.	Performed by Operator in the field	Lineup will automatically isolate
D.	Performed by Operator in the field	Lineup will NOT automatically isolate

Proposed Answer: D

Explanation: Cross-tie of Unit 1 and Unit 2 Instrument Air requires an Operator in the field to open manual valve 025091, Unit 1 Unit 2 Instr Air Cross Connect Vlv. There is no automatic isolation feature in this lineup to protect against one Unit from negatively affecting the other Unit.

A. Incorrect – Cross-tie of Unit 1 and Unit 2 Instrument Air requires an Operator in the field to open manual valve 025091. Plausible because some Instrument Air controls and indications are in the Main Control Room, just not this control. The lineup will NOT automatically isolate on lowering pressure. Plausible because this is a specific concern with this lineup (including a procedural caution) and it is common for some air cross-connect lineups to automatically isolate given an adverse trend in pressure.
B. Incorrect – Cross-tie of Unit 1 and Unit 2 Instrument Air requires an Operator in the field to open manual valve 025091. Plausible because some Instrument Air controls and indications are in the Main Control Room, just not this control.
C. Incorrect – The lineup will NOT automatically isolate on lowering pressure. Plausible because this is a specific concern with this lineup (including a procedural caution) and it is common for some air cross-connect lineups to automatically isolate given an adverse trend in pressure.

Technical Reference(s): ON-118-001, OP-118-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A1.05
	Importance Rating	2.9

Reactor Water Level Control

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: FWRV/startup level control position: Plant-Specific

Proposed Question: #23

Unit 1 was operating at 100% power when the following occurred:

- The Reactor was manually scrammed.
- HPCI and RCIC were overridden after initiation to prevent Reactor water level from reaching +54 inches.
- Feedwater Level Control (FWLC) Setpoint Setdown initiated.
- Reactor water level is being maintained at +25" by Feedwater Level Control.
- Feedwater pump 1A is injecting.
- Ten minutes have elapsed since Reactor water level was stabilized at +25".

Which one of the following describes the plant response to resetting the FWLC Setpoint Setdown logic with NO additional operator action?

The FWLC system will...

A. automatically restore Reactor water level to +35" by raising RFPT A speed in flow control mode.

B. automatically restore Reactor water level to +35" by opening FW LOW LOAD VALVE LV-10641.

C. continue to maintain Reactor water level at +25" by maintaining RFPT A speed in flow control mode.

D. continue to maintain Reactor water level at +25" by controlling the position of FW LOW LOAD VALVE LV-10641.

Proposed Answer: D

Explanation: The FWLC system will automatically align the Feedwater system to startup level control via LV-10641 with one Feedwater pump in discharge pressure control mode following a scram. The low load controller LIC-C32-1R602 setpoint will remain at 25" following reset of Setpoint Setdown until operators return the setpoint to +35".

A. Incorrect – The level setpoint will remain at +25" until manually adjusted. Plausible because originally level would have been controlled at 35". The FWLC system automatically aligns to startup level control mode, therefore LV-10641 will control flow, not RFPT A in flow control mode. Plausible because RFPT A is initially in flow control mode.
B. Incorrect – The level setpoint will remain at +25" until manually adjusted. Plausible because originally level would have been controlled at 35".
C. Incorrect – The FWLC system automatically aligns to startup level control mode, therefore LV-10641 will control flow, not RFPT A in flow control mode. Plausible because RFPT A is initially in flow control mode.

Technical Reference(s): OP-145-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-045 RBO-4

Question Source: Bank – LOC26R NRC #3

Question History: LOC26R NRC #3

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K1.07
	Importance Rating	3.0

IRM

Knowledge of the physical connections and/or cause-effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: Reactor vessel

Proposed Question: #24

Which one of the following describes the location of the IRMs when (1) fully inserted and (2) fully withdrawn?

	(1)	(2)
A.	Below the Reactor core plate	Inside the Reactor vessel
B.	Above the Reactor core plate	Inside the Reactor vessel
C.	Below the Reactor core plate	Outside the Reactor vessel
D.	Above the Reactor core plate	Outside the Reactor vessel

Proposed Answer: B

Explanation: When fully inserted, the IRM detectors are located above the Reactor core plate approximately 18" above the center of the core. When fully withdrawn, the IRM detectors are located approximately 30" below the Reactor core plate, but still inside the Reactor vessel

A. Incorrect – When fully inserted, the IRM detectors are located above the Reactor core plate approximately 18" above the center of the core. Plausible because some other nuclear power plants locate neutron monitoring detectors outside of the Reactor core, even when in use. Doing so has a benefit of reducing irradiation and prolonging detector life.
C. Incorrect – When fully inserted, the IRM detectors are located above the Reactor core plate approximately 18" above the center of the core. Plausible because some other nuclear power plants locate neutron monitoring detectors outside of the Reactor core, even when in use. Doing so has a benefit of reducing irradiation and prolonging detector life. When fully withdrawn, the IRM detectors are located approximately 30" below the Reactor core plate, but still inside the Reactor vessel. Plausible because TIP detectors are outside the Reactor vessel when fully withdrawn and this would prolong IRM detector life
D. Incorrect – When fully withdrawn, the IRM detectors are located approximately 30" below the Reactor core plate, but still inside the Reactor vessel. Plausible because TIP detectors are outside the Reactor vessel when fully withdrawn and this would prolong IRM detector life.

Technical Reference(s): DBD 055

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078B RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 K3.01
	Importance Rating	3.5

AC Electrical Distribution

Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: Major system loads

Proposed Question: #25

Unit 1 is operating at 100% power with the following:

- Diesel Generator (DG) C is out of service due to catastrophic turbocharger failure.
- DG E has not yet been transferred for DG C.
- At time T=0 seconds, a loss of all offsite power occurs.
- At time T=30 seconds, a DBA LOCA occurs.

Which one of the following describes which Core Spray pumps will start and when?

A. Only pumps 1A, 1B, and 1D will start at T=40.5 seconds

B. Pumps 1A, 1B, 1C, and 1D will start at T=40.5 seconds

C. Only pumps 1A, 1B, and 1D will start at T=45 seconds

D. Pumps 1A, 1B, 1C, and 1D will start at T=45 seconds

Proposed Answer: A

Explanation: At time T=0 seconds, the loss of all offsite power causes DGs A, B, and D to automatically start and close in on ESS buses 1A, 1B, and 1D, respectively. ESS bus 1C will de-energize, preventing the start of Core Spray pump 1C later when the DBA LOCA occurs. Due to the loss of offsite power, Core Spray pumps 1A, 1B, and 1D start with a time delay of 10.5 seconds. Therefore, Core Spray pumps 1A, 1B, and 1D start at time T=40.5 seconds.

B. Incorrect – Core Spray pump 1C does not start. Plausible because some other plants have two DGs load onto each ESS bus, such that loss of just one DG would not prevent any Core Spray pump from starting.
C. Incorrect – Core Spray pumps start at 40.5 seconds, not 45 seconds. Plausible because Core Spray pumps normally have a time delay of 15 seconds when starting on a LOCA signal, which would make them start at 45 seconds.
D. Incorrect – Core Spray pump 1C does not start. Plausible because some other plants have two DGs load onto each ESS bus, such that loss of just one DG would not prevent any Core Spray pump from starting. Core Spray pumps start at 40.5 seconds, not 45 seconds. Plausible because Core Spray pumps normally have a time delay of 15 seconds when starting on a LOCA signal, which would make them start at 45 seconds.

Technical Reference(s): ON-4KV-101 Att J, OP-103-001 Att A, OP-151-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-004 RBO-4

Question Source: Modified Bank – LOC24 NRC #47

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 2.1.28
	Importance Rating	4.1

UPS (AC/DC)

Knowledge of the purpose and function of major system components and controls.

Proposed Question: #26

Unit 1 is operating at 100% power when MCC 1B236 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this loss on Uninterruptible Power Supplies (UPS)?

UPS __(1)__ will be supplied by __(2)__.

A. (1) 1D240
(2) MCC 1B216, only.

B. (1) 1D240
(2) Its dedicated battery for approximately 20 minutes, then by MCC 1B216.

C. (1) 1D666
(2) MCC 1B216, only.

D. (1) 1D666
(2) Its dedicated battery for approximately 20 minutes, then by MCC 1B216.

Proposed Answer: B

Explanation: MCC 1B236 is the preferred source for UPS 1D240. When MCC 1B236 is lost, the UPS initially is powered from its dedicated battery. This lasts approximately 20 minutes. Then the UPS swaps to the alternate AC source, MCC 1B216.

A. Incorrect – The UPS first swaps to the battery source, then to the alternate AC source after the battery discharges for approximately 20 minutes. Plausible because swapping to the alternate AC source first would preserve battery capacity.
C. Incorrect – UPS 1D666 is normally supplied by MCC 1B246, not 1B236. Plausible because these are similar MCCs. The UPS first swaps to the battery source, then to the alternate AC source after the battery discharges for approximately 20 minutes. Plausible because swapping to the alternate AC source first would preserve battery capacity.
D. Incorrect – UPS 1D666 is normally supplied by MCC 1B246, not 1B236. Plausible because these are similar MCCs.

Technical Reference(s): ON-4KV-107, AR-106-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-017 RBO-3

Question Source: Bank – LOC24 Cert #18

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 K1.02
	Importance Rating	2.5

Main Turbine Generator and Auxiliary Systems

Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: Condensate system

Proposed Question: #27

Unit 1 is operating at 60% power with the following:

- Condensate pumps 1A, 1B, and 1D are operating.
- Condensate pump 1C is secured.

Then, a steam leak in the Drywell results in the following:

- The Reactor automatically scrams due to high Drywell pressure.
- Reactor water level reaches a low of -20", then recovers to the normal band.
- Drywell pressure is 4 psig, up slow.

Which one of the following describes the number of operating Condensate pumps one (1) minute later?

A. Zero (0)

B. Two (2)

C. Three (3)

D. Four (4)

Proposed Answer: A

Explanation: The Reactor scram leads to the Main Generator lockouts tripping. With a LOCA signal present due to Drywell pressure above 1.72 psig, the Main Generator lockout leads to Aux Bus load shedding and slow transfer of the Aux buses. This load shedding scheme trips all Condensate pumps to prevent overloading ESS transformers. Therefore, zero Condensate pumps remain operating.

B. Incorrect – Zero Condensate pumps remain operating due to the combination of Main Generator lockout and LOCA signal resulting in Aux Bus load shedding. Plausible because ON-SCRAM-101 PCO Actions allow reducing the number of operating Condensate pumps to two.
C. Incorrect – Zero Condensate pumps remain operating due to the combination of Main Generator lockout and LOCA signal resulting in Aux Bus load shedding. Plausible because three Condensate pumps were initially running.
D. Incorrect – Zero Condensate pumps remain operating due to the combination of Main Generator lockout and LOCA signal resulting in Aux Bus load shedding. Plausible because Reactor water level lowered below +13", which initiates automatic Feedwater response in the form of Setpoint Setdown, just not auto-start of the standby Condensate pump.

Technical Reference(s): EO-000-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-003 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	230000 K4.09
	Importance Rating	3.0

RHR/LPCI: Torus/Pool Spray Mode

Knowledge of RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE design feature(s) and/or interlocks which provide for the following: Spray flow cooling

Proposed Question: #28

Unit 1 is operating at 100% power with the following:

- RHR loop 1A is operating in Suppression Pool cooling.
- HV-151-F048A, HX A SHELL SIDE BYPS, is fully closed.

Then, a steam leak in the Drywell results in the following:

- Drywell pressure is 3 psig, up slow.
- Suppression Chamber spray is desired to be placed in service using RHR loop 1A.

Which one of the following describes the current status of HV-151-F048A?

HV-151-F048A is...

A. open and may be re-positioned without override.

B. open and requires an override to be re-positioned.

C. closed and may be re-positioned without override.

D. closed and requires an override to be re-positioned.

Proposed Answer: B

Explanation: HV-151-F048A automatically opens when Drywell pressure exceeds 1.72 psig to allow a higher flow rate from LPCI (both through and around the heat exchanger. This open signal also seals in. Therefore, in order to close HV-151-F048A (to force more water through the heat exchanger and provide greater cooling of the spray flow), the open signal must be overridden.

A. Incorrect – The open signal seals in, therefore it must be overridden to re-position the valve. Plausible because the open signal could not seal in since this valve is likely be re-positioned in a situation with high Drywell pressure.
C. Incorrect – Although HV-151-F048A was initially closed, it automatically opens when Drywell pressure exceeds 1.72 psig. Plausible for this valve to remain closed to ensure cooling of LPCI flow by directing it all through the heat exchanger. The open signal seals in, therefore it must be overridden to re-position the valve. Plausible because the open signal could not seal in since this valve is likely be re-positioned in a situation with high Drywell pressure.
D. Incorrect – Although HV-151-F048A was initially closed, it automatically opens when Drywell pressure exceeds 1.72 psig. Plausible for this valve to remain closed to ensure cooling of LPCI flow by directing it all through the heat exchanger.

Technical Reference(s): OP-149-004

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201002 K3.02
	Importance Rating	2.9

RMCS

Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: Rod block monitor: Plant-Specific

Proposed Question: #29

Unit 1 is operating at 75% power with the following:

- A control rod pattern adjustment is in progress.
- A malfunction with the RMCS selection circuitry results in two control rods being selected at the same time.

Which one of the following describes the resulting operation of the Rod Block Monitor (RBM)?

The RBM...

A. enforces both a rod withdrawal block and a rod insert block.

B. enforces a rod withdrawal block, but NOT a rod insert block.

C. does NOT enforce a rod block, but does provide protection for both selected control rods.

D. does NOT enforce a rod block, but only provides protection for one of the selected control rods.

Proposed Answer: B

Explanation: When the RBM detects more than one control rod selected at the same time, an INOP trip is generated. The INOP trip results in a rod withdrawal block, but not a rod insert block.

A. Incorrect – The INOP trip results in a rod withdrawal block, but not a rod insert block. Plausible because other system malfunctions (RWM INOP, RDCS failure) result in rod insert blocks also.
C. Incorrect – The RBM has a feature that detects more than one control rod selected at the same time and enforces a rod withdrawal block. Plausible because if this feature did not exist, the RBM circuitry would have to be designed to either protect both selected rods or just one (first selected vs. last selected).
D. Incorrect – The RBM has a feature that detects more than one control rod selected at the same time and enforces a rod withdrawal block. Plausible because if this feature did not exist, the RBM circuitry would have to be designed to either protect both selected rods or just one (first selected vs. last selected).

Technical Reference(s): AR-103-C04

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078K RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	219000 K4.05
	Importance Rating	3.0

RHR/LPCI: Torus/Pool Cooling Mode

Knowledge of RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE design feature(s) and/or interlocks which provide for the following: Pump minimum flow protection

Proposed Question: #30

Unit 1 is operating at 100% power with the following:

- HPCI testing has just been completed.
- RHR loop A is operating in the Suppression Pool Cooling lineup.

Then, a loss of coolant accident results in the following:

- Reactor water level is -140", down slow.
- Reactor pressure is 450 psig, down slow.
- Drywell pressure is 15 psig, up slow.
- RHR loop A valves are aligned as shown in the picture on the next page.

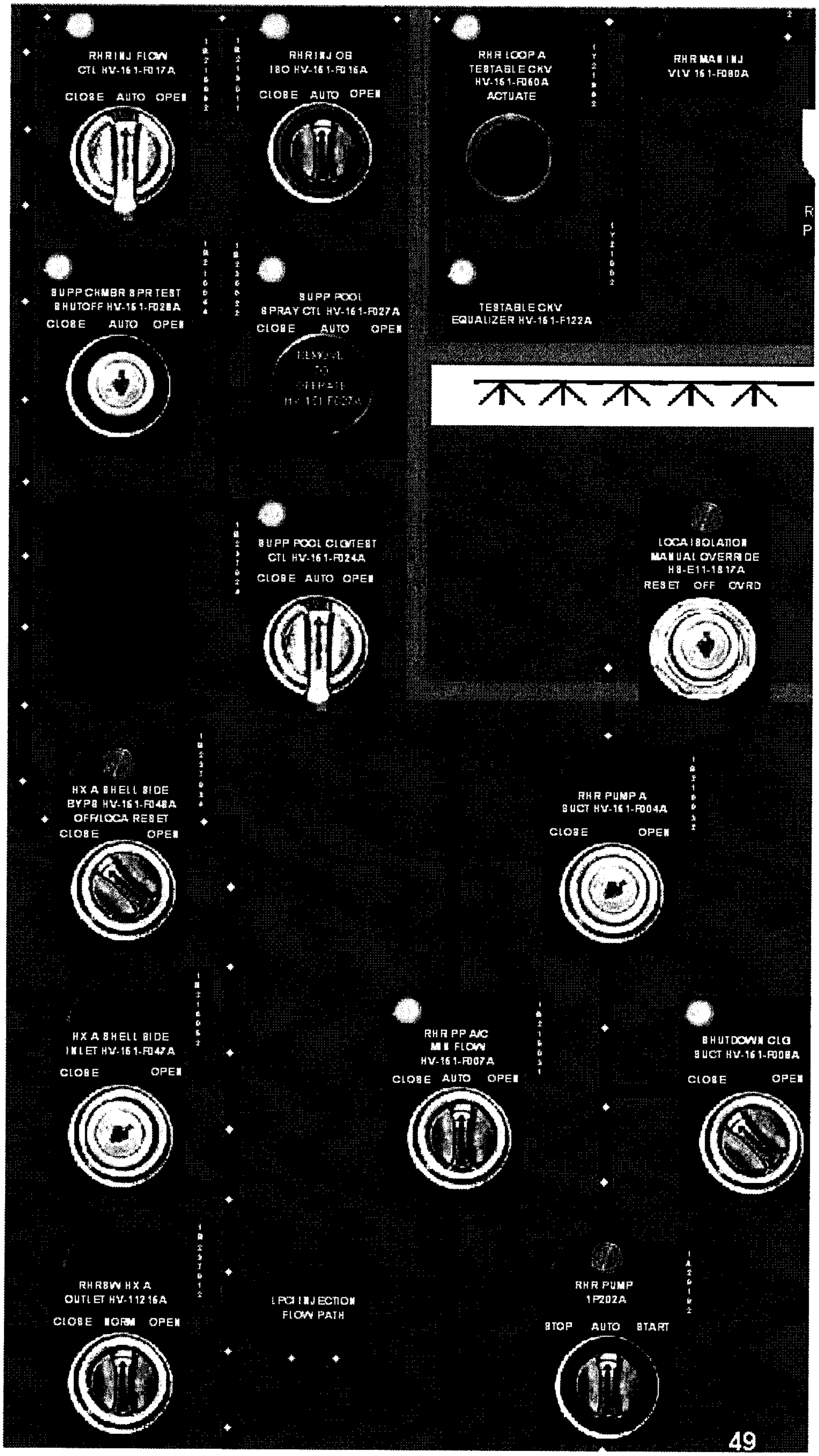
Which one of the following describes the status of RHR loop A valves?

A. RHR loop A valves have operated as designed.

B. HV-151-F007A, RHR PP A/C MIN FLOW, is closed, but should be open.

C. HV-151-F017A, RHR INJ FLOW CTL, and HV-151-F015A, RHR INJ OB ISO, are closed, but should be open.

D. HV-151-F028A, SUPP CHMBR SPR TEST SHUTOFF, and HV-151-F024A, SUPP POOL CLG/TEST CTL, are closed, but should be open.


RHR INJ FLOW CTL HV-151-F017A
CLOSE AUTO OPEN
RHR INJ OB ISO HV-151-F016A
CLOSE AUTO OPEN
RHR LOOP A TESTABLE CKV HV-151-F060A ACTUATE
RHR MAN INJ VLV 151-F080A
SUPP CHMBR SPR TEST SHUTOFF HV-151-F028A
CLOSE AUTO OPEN
SUPP POOL SPRAY CTL HV-151-F027A
CLOSE AUTO OPEN
TESTABLE CKV EQUALIZER HV-151-F122A
SUPP POOL CLG/TEST CTL HV-151-F024A
CLOSE AUTO OPEN
LOCA ISOLATION MANUAL OVERRIDE HS-E11-1817A
RESET OFF OVRD
HX A SHELL SIDE BYPS HV-151-F048A OFF/LOCA RESET
CLOSE OPEN
RHR PUMP A SUCT HV-151-F004A
CLOSE OPEN
HX A SHELL SIDE INLET HV-151-F047A
CLOSE OPEN
RHR PP A/C MIN FLOW HV-151-F007A
CLOSE AUTO OPEN
SHUTDOWN CLG SUCT HV-151-F008A
CLOSE OPEN
RHRSW HX A OUTLET HV-11216A
CLOSE NORM OPEN
LPCI INJECTION FLOW PATH
RHR PUMP 1P202A
STOP AUTO START
49

Proposed Answer: B

Explanation: With Suppression Pool Cooling initially in service, F017A and F015A are closed, F028A and F024A are open, and F07A is normally closed (RHR min flow valve is normally open) but could be open if flow was low enough through the cooling flow path. When Drywell pressure exceeds 1.72 psig and/or Reactor water level lowers below -129", the Suppression Pool Cooling lineup secures (F028A and F024A close). However, since Reactor pressure is above approximately 420 psig (407-433 psig range per TS 3.3.5.1), the LPCI injection valves (F017A and F015A) do not yet receive a signal to open and therefore remain closed. This combination of valve closures results in low flow, therefore the minimum flow valve (F07A) should be open.

A. Incorrect – F07A should be open due to the low flow condition. Plausible because a LOCA signal is present and LPCI injection valves will soon open.
C. Incorrect – F017A and F015A should be closed because Reactor pressure is above 420 psig. Plausible because a LOCA signal is present which has already causes F028A and F024A to re-position.
D. Incorrect – F028A and F024A should be closed because Drywell pressure is above 1.72 psig and Reactor water level is below -129". Plausible because these two valves were initially open for the SPC lineup, and Reactor pressure is still above the value at which LPCI injection actually occurs.

Technical Reference(s): OP-149-001, OP-149-004

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-3

Question Source: Modified Bank – LOC26R NRC #8

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	234000 K5.02
	Importance Rating	3.1

Fuel Handling Equipment

Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT: Fuel handling equipment interlocks

Proposed Question: #31

Unit 1 is in a refueling outage with the following:

- The Reactor Mode Switch is in REFUEL.
- One control rod is withdrawn to position 04.
- A fuel bundle is latched on the Refuel Platform grapple in the Spent Fuel Pool.
- The Refuel Platform grapple is in the Normal Up position.

Which one of the following describes when fuel movement would **first** be interrupted by an interlock if the Refuel Platform Operator attempted to place the fuel bundle in the Unit 1 core?

This fuel movement would **first** be interrupted by an interlock...

A. as soon as any attempt is made to move the Refuel Platform towards the core.

B. when the Refuel Platform reaches a position close to the core.

C. as soon as any attempt is made to lower the fuel bundle once over the core.

D. when the bottom of the fuel bundle is lowered enough to reach the height of the core top guide.

Proposed Answer: B

Explanation: This movement would first be interrupted by the Bridge Reverse #1 interlock. This interlock stops Refuel Platform motion when the Refuel Platform is nearing a location over the core, a control rod is withdrawn, and the grapple is loaded.

A. Incorrect – Motion towards the core is allowed until the Refuel Platform nears a position over the core. Plausible because motion towards the core will eventually be interrupted, just not until the Refuel Platform nears the core, and this also would prevent loading fuel with a rod out.
C. Incorrect – The fuel movement would be interrupted earlier than this before the Refuel platform moved over the core. Plausible because if the Bridge Reverse #1 interlock failed, a separate interlock would prevent lowering the grapple.
D. Incorrect – The fuel movement would be interrupted earlier than this before the Refuel platform moved over the core. Plausible because if the Bridge Reverse #1 interlock failed, a separate interlock would prevent lowering the grapple. Plausible that this interlock would not occur until near the core top guide because this is the latest that an interlock would prevent simultaneous fuel loading and rod withdrawal.

Technical Reference(s): OP-181-001 Att A

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-081A RBO-4

Question Source: Modified Bank – Vision SYSID 34030

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	214000 K6.02
	Importance Rating	2.7

RPIS

Knowledge of the effect that a loss or malfunction of the following will have on the ROD POSITION INFORMATION SYSTEM: Position indication probe

Proposed Question: #32

Unit 1 is operating at 70% power with the following:

- A control rod is notch withdrawn from position 46 to position 48.
- Misalignment of the position indicating probe reed switches causes the following when the control rod settles:
    - o The position 47 reed switch is closed.
    - o The position 48 reed switch is open.

Which one of the following describes the effect of this malfunction?

Annunciator...

A. AR-104-H05, ROD DRIFT, alarms when the control rod timer times out.

B. AR-104-H05, ROD DRIFT, alarms when the next control rod is selected.

C. AR-104-H06, ROD OVERTRAVEL, alarms when the control rod timer times out.

D. AR-104-H06, ROD OVERTRAVEL, alarms when the next control rod is selected.

Proposed Answer: A

Explanation: AR-104-H05, ROD DRIFT, alarms when the control rod timer times out because the control rod fails to indicate at the next even position (position 48 reed switch open) and remains detected at an odd position (position 47 reed switch closed).

B. Incorrect – The annunciator alarms as soon as the timer times out, not later when the next rod is selected. Plausible that the alarm would be blocked while a rod is selected and being moved since odd reed switch positions are expected to be picked up during this evolution.
C. Incorrect – AR-104-H06 does not alarm. Plausible because this alarm comes in if the rod moves past position 48, and in this case position 48 fails to indicate.
D. Incorrect – AR-104-H06 does not alarm. Plausible because this alarm comes in if the rod moves past position 48, and in this case position 48 fails to indicate. The correct annunciator alarms as soon as the timer times out, not later when the next rod is selected. Plausible that the alarm would be blocked while a rod is selected and being moved since odd reed switch positions are expected to be picked up during this evolution.

Technical Reference(s): AR-104-H05, AR-104-H06

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-056 RBO-4

Question Source: Modified Bank – NMP1 2010 Audit #29

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Facility:	Nine Mile Point Unit 1
Vendor:	GE
Exam Date:	2010
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	214000	K3.03	
	Importance Rating	3.1		

Knowledge of the effect that a loss or malfunction of the ROD POSITION INFORMATION SYSTEM will have on following: RMCS: Plant-Specific

Proposed Question: RO Question # 29

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

- Rod select power is turned OFF
- Control rod 02-19 is at position 48
- Then, control rod 02-19 reed switch S47 (for position 47) fails in the closed position
- No Control rod movement occurs

Which one of the following describes the impact of this switch failure?

F3-2-6, CONTROL ROD DRIFT, will annunciate...

A. at this time.

B. only when Rod select power is turned ON.

C. only when Control Rod 02-19 is selected for movement.

D. only when Control Rod 02-19 is selected AND the rod movement timer has started.

Proposed Answer: A

Explanation (Optional):

A. Correct - The rod drift alarm will be received due to an odd reed switch being closed with the rod not selected for movement.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 A1.24
	Importance Rating	2.6

Reactor/Turbine Pressure Regulator

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR/TURBINE PRESSURE REGULATING SYSTEM controls including: Main turbine eccentricity

Proposed Question: #33

Which one of the following describes:

(1) the availability of Main Turbine eccentricity indication in the Main Control Room (MCR), and

(2) whether a high Main Turbine eccentricity signal directly causes a Main Turbine trip?

	(1) Eccentricity Indication	(2) High Eccentricity Signal
A.	Available in MCR	Causes a trip
B.	Available in MCR	Does NOT cause a trip
C.	NOT available in MCR	Causes a trip
D.	NOT available in MCR	Does NOT cause a trip

Proposed Answer: B

Explanation: There is a recorder (XR10116) in the MCR that displays Main Turbine eccentricity, along with TBV position, Turbine speed, and TCV position, on panel 1C652. A high eccentricity signal does not directly cause a Main Turbine trip.

A. Incorrect – A high eccentricity signal does not directly cause a Main Turbine trip. Plausible because high eccentricity is undesirable, may lead to Turbine damage, and may indirectly lead to a Main Turbine trip on high vibration.
C. Incorrect – Main Turbine eccentricity indication is available on a recorder in the MCR. Plausible because this is not a widely used indication and alternate indication is available at the Turbine Supervisory Instrumentation Panel, which is outside the MCR.
D. Incorrect – Main Turbine eccentricity indication is available on a recorder in the MCR. Plausible because this is not a widely used indication and alternate indication is available at the Turbine Supervisory Instrumentation Panel, which is outside the MCR. A high eccentricity signal does not directly cause a Main Turbine trip. Plausible because high eccentricity is undesirable, may lead to Turbine damage, and may indirectly lead to a Main Turbine trip on high vibration.

Technical Reference(s): AR-105-A01

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-093 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	268000 A2.01
	Importance Rating	2.9

Radwaste

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

Proposed Question: #34

Unit 1 is operating at 100% power with the following:

- Radwaste tanks are in a normal alignment.
- An unidentified leak has developed in the Drywell.
- Input to the Drywell Floor Drain sumps has risen from 0.3 gpm to 1.9 gpm in the last four hours.
- No change has been made to Radwaste tank alignment based on this leak.

Which one of the following describes:

(1) the Radwaste tanks that will be receiving this excess leakage, and

(2) whether the leakage exceeds any Technical Specification limit?

	(1) Radwaste Tanks That Receive This Excess Leakage	(2) Leakage Exceeds Any Technical Specification Limit?
A.	Surge Tanks	No
B.	Surge Tanks	Yes
C.	Collection Tanks	No
D.	Collection Tanks	Yes

Proposed Answer: C

Explanation: The Drywell Floor Drain sumps discharge to the Reactor Building sumps, which then are normally aligned to discharge to the Radwaste Collection Tanks. This leakage represents a rise in Unidentified Leakage of 1.6 gpm in four hours. This is within the TS 3.4.4 limit of ≤ 2 gpm increase in Unidentified Leakage in a four hour period while in Mode 1. The TS 3.4.4 limit of ≤ 5 gpm Unidentified Leakage is also not exceeded.

Note: The K/A requires testing the impact of a "system rupture" on Radwaste, including use of procedure(s) to correct, control, or mitigate the impact. The term "system rupture" is interpreted as either a rupture of some component within the Radwaste system or a rupture of an external system that directly impacts Radwaste. Additionally, NUREG 1123 indicates the Radwaste system includes plant sumps and drains. Therefore, the question meets the K/A by:

- giving a system rupture (RCS system inside Drywell) that impacts Radwaste (Drywell floor drain sumps),
- requiring prediction of the associated impact (Radwaste tank that receives leakage), and
- requiring determination of correct procedure usage to deal with the impact (assessing status of Technical Specification leakage limits).

A. Incorrect – The Drywell Floor Drain sumps discharge to the Reactor Building sumps, which then are normally aligned to discharge to the Radwaste Collection Tanks. Plausible because an alternate alignment is available to the Surge Tanks.
B. Incorrect – The Drywell Floor Drain sumps discharge to the Reactor Building sumps, which then are normally aligned to discharge to the Radwaste Collection Tanks. Plausible because an alternate alignment is available to the Surge Tanks. This leakage represents a rise in Unidentified Leakage of 1.6 gpm in four hours. This is within the TS 3.4.4 limit of ≤ 2 gpm increase in Unidentified Leakage in a four hour period while in Mode 1. The TS 3.4.4 limit of ≤ 5 gpm Unidentified Leakage is also not exceeded. Plausible because this is a significant rise in leakage over a short period of time, elevated well above normal rates, and is in excess of 1 gpm.
D. Incorrect – This leakage represents a rise in Unidentified Leakage of 1.6 gpm in four hours. This is within the TS 3.4.4 limit of ≤ 2 gpm increase in Unidentified Leakage in a four hour period while in Mode 1. The TS 3.4.4 limit of ≤ 5 gpm Unidentified Leakage is also not exceeded. Plausible because this is a significant rise in leakage over a short period of time, elevated well above normal rates, and is in excess of 1 gpm.

Technical Reference(s): M-161 Sheet1, M-161 Sheet 3, M-126 Sheet 1, TS 3.4.4

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-069 RBO-3

Question Source: Modified Bank – LOC27 NRC #34

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	288000 A3.01
	Importance Rating	3.8

Plant Ventilation

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

Proposed Question: #35

Unit 2 is in Mode 2 with a heatup in progress and IRMs on Range 8.

Unit 1 is operating at 100% power with the following:

- Unit 1 experiences a loss of Feedwater.
- The Unit 1 Reactor is manually scrammed.
- Unit 1 Reactor water level drops to a low of -45".
- Unit 1 Reactor water level is recovered to +35", stable, using HPCI and RCIC.

Which of the following describes the response of the Reactor Building Ventilation zones (I, II, III)?

	Zone I	Zone II	Zone III
A.	Isolates	Isolates	Isolates
B.	Does NOT isolate	Isolates	Isolates
C.	Isolates	Does NOT isolate	Isolates
D.	Does NOT isolate	Does NOT isolate	Does NOT isolate

Proposed Answer: C

Explanation: Unit 1 Reactor water level below -38" causes both a Zone I and Zone III isolation signal. Zone II does not receive an isolation signal because Unit 2 Reactor water level and Drywell pressure are unaffected.

A. Incorrect – Zone II does not receive an isolation signal because Unit 2 Reactor water level and Drywell pressure are unaffected. Plausible if candidate confuses Zones and because the low Reactor water level does partially affect Unit 2 through the Zone III isolation.
B. Incorrect – Zone I does receive an isolation signal and Zone II does not receive an isolation signal because Unit 2 Reactor water level and Drywell pressure are unaffected. Plausible because this is the correct answer if the transient happened on Unit 2 instead of Unit 1.
D. Incorrect – Unit 1 Reactor water level below -38" causes both a Zone I and Zone III isolation signal. Plausible because this would be the correct answer if Reactor water level lowered slightly less.

Technical Reference(s): ON-159-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-70 RBO-4

Question Source: Modified Bank – LOC24 NRC #46

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	204000 A4.07
	Importance Rating	3.1

RWCU

Ability to manually operate and/or monitor in the control room: System temperature

Proposed Question: #36

A Unit 1 shutdown is in progress when an Operator establishes RWCU letdown flow to the Main Condenser, in accordance with OP-161-001, Reactor Water Cleanup System.

Which one of the following describes the effect of establishing RWCU letdown flow on calculated core thermal power (CTP) and Non-Regenerative Heat Exchanger (NRHX) outlet temperature?

	Calculated CTP	NRHX Outlet Temperature
A.	Lowers	Lowers
B.	Lowers	Rises
C.	Rises	Lowers
D.	Rises	Rises

Proposed Answer: D

Explanation: Calculated CTP rises due to the change in RWCU flow upon establishing letdown flow. NRHX outlet temperature rises due to greater RWCU flow with no additional RBCCW flow.

A. Incorrect – Calculated CTP rises due to the change in RWCU flow upon establishing letdown flow. NRHX outlet temperature rises due to greater RWCU flow with no additional RBCCW flow. Both plausible if candidate does not understand the interaction between RWCU letdown flow and these parameters.
B. Incorrect – Calculated CTP rises due to the change in RWCU flow upon establishing letdown flow. Plausible if candidate does not understand the interaction between RWCU letdown flow and calculated CTP.
C. Incorrect – NRHX outlet temperature rises due to greater RWCU flow with no additional RBCCW flow. Plausible if candidate does not understand the interaction between RWCU letdown flow and NRHX outlet temperature.

Technical Reference(s): OP-161-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-061 RBO-7

Question Source: Bank - NMP1 2009 NRC #62

Question History: NMP1 2009 NRC #62

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201003 2.4.31
	Importance Rating	4.2

Control Rod and Drive Mechanism

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #37

Unit 1 is operating at 100% power when annunciator AR-103-H05, CRD PANEL 1C007 HI TEMP, alarms due to high temperature on one control rod.

Which one of the following describes the ability of this annunciator to re-flash if a second control rod experiences a high temperature and the location of CRD panel 1C007?

	Will Annunciator Re-Flash?	Panel Location
A.	No	Relay Room
B.	No	Reactor Building
C.	Yes	Relay Room
D.	Yes	Reactor Building

Proposed Answer: B

Explanation: Annunciator AR-103-H05 is provided with re-flash capability, such that if a second CRDM alarms on high temperature the alarm will not be masked. This is only available after the local pael has been reset. If no operator action is taken, the annunciator will not re-flash. CRD panel 1C007 is located in the Reactor Building.

A. Incorrect – Annunciator AR-103-H05 is provided with re-flash capability, such that if a second CRDM alarms on high temperature the alarm will not be masked. Plausible because some multi-point alarms do not have such a re-flash capability. CRD panel 1C007 is located in the Reactor Building. Plausible because the Relay Room contains many panels/indications.
C. Incorrect – CRD panel 1C007 is located in the Reactor Building. Plausible because the Relay Room contains many panels/indications.
D. Incorrect - Annunciator AR-103-H05 is provided with re-flash capability, such that if a second CRDM alarms on high temperature the alarm will not be masked. Plausible this alarm does have a re-flash capability but is not active until the local panel is reset..

Technical Reference(s): AR-103-H05

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-055M RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	2
K/A #	290002 K3.06
Importance Rating	3.1

Reactor Vessel Internals

Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following: PCIS/NSSSS

Proposed Question: #38

A Unit 1 startup is in progress with the following:

- Reactor power is approximately 1%.
- The Mechanical Vacuum pump (MVP) is operating.
- The Main Steam Isolation Valves (MSIVs) are open.

Then, major fuel damage occurs as a result of defective fuel bundles:

- AR-111-C03, MN STM LINE RAD MONITOR HI RADIATION, alarms.
- AR-103-D01, MN STM LINE HI HI RADIATION, alarms.
- AR-104-D01, MN STM LINE HI HI RADIATION, alarms.

Which one of the following describes the response of the MVP and the MSIVs?

The MVP...

A. remains in service unless manually secured and the MSIVs remain open unless manually closed.

B. remains in service unless manually secured and the MSIVs automatically isolate.

C. automatically trips and the MSIVs remain open unless manually closed.

D. automatically trips and the MSIVs automatically close.

Proposed Answer: C

Explanation: Major fuel damage will cause the Main Steam Line radiation monitors to reach the hi-hi trip setpoint. This causes the Mechanical Vacuum pump to trip. The MSIV isolation on hi-hi Main Steam Line radiation has been removed, therefore the MSIVs remain open unless manually closed.

A. Incorrect – The MVP trips on hi-hi Main Steam Line radiation. Plausible since the MSIVs do not automatically isolate.
B. Incorrect – The MVP trips on hi-hi Main Steam Line radiation. Plausible since the MSIVs do not automatically isolate. The MSIVs remain open unless manually closed. Plausible because original plant design include automatic MSIV isolation on hi-hi Main Steam Line radiation and manual isolation is still procedurally required.
D. Incorrect – The MSIVs remain open unless manually closed. Plausible because original plant design include automatic MSIV isolation on hi-hi Main Steam Line radiation and manual isolation is still procedurally required.

Technical Reference(s): ON-159-002, ON-179-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059X RBO-2

Question Source: Modified Bank – Vision SYSID 314

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AK1.02
	Importance Rating	3.3

Generator Voltage and Electric Grid Disturbances

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

Proposed Question: #39

Unit 1 is operating at approximately 100% power with the following:

- A grid disturbance has occurred.
- TCC has requested adjusting Generator reactive power to the maximum allowed by the Generator capability curve.
- The following Generator conditions are present and stable:
  - Generator reactive power: 550 MVAR lagging
  - Generator power: 1130 MWe
  - Generator hydrogen pressure: 70 psig
-

Note: The Main Generator capability curve is provided on the following pages.

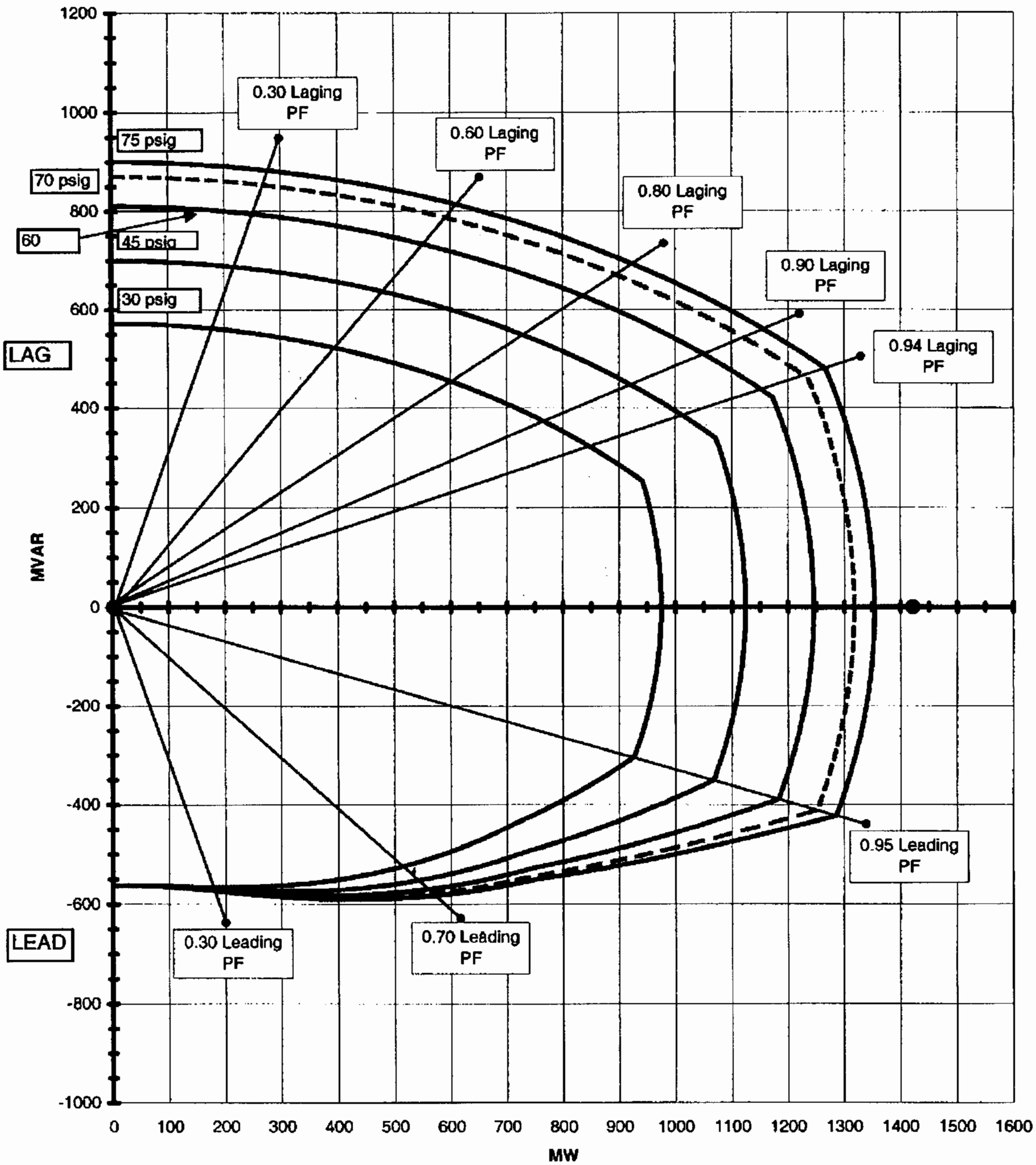
Which one of the following describes the required manipulation to carry out the request from TCC?

A. Depress LOAD SELECTOR INCREASE pushbutton.

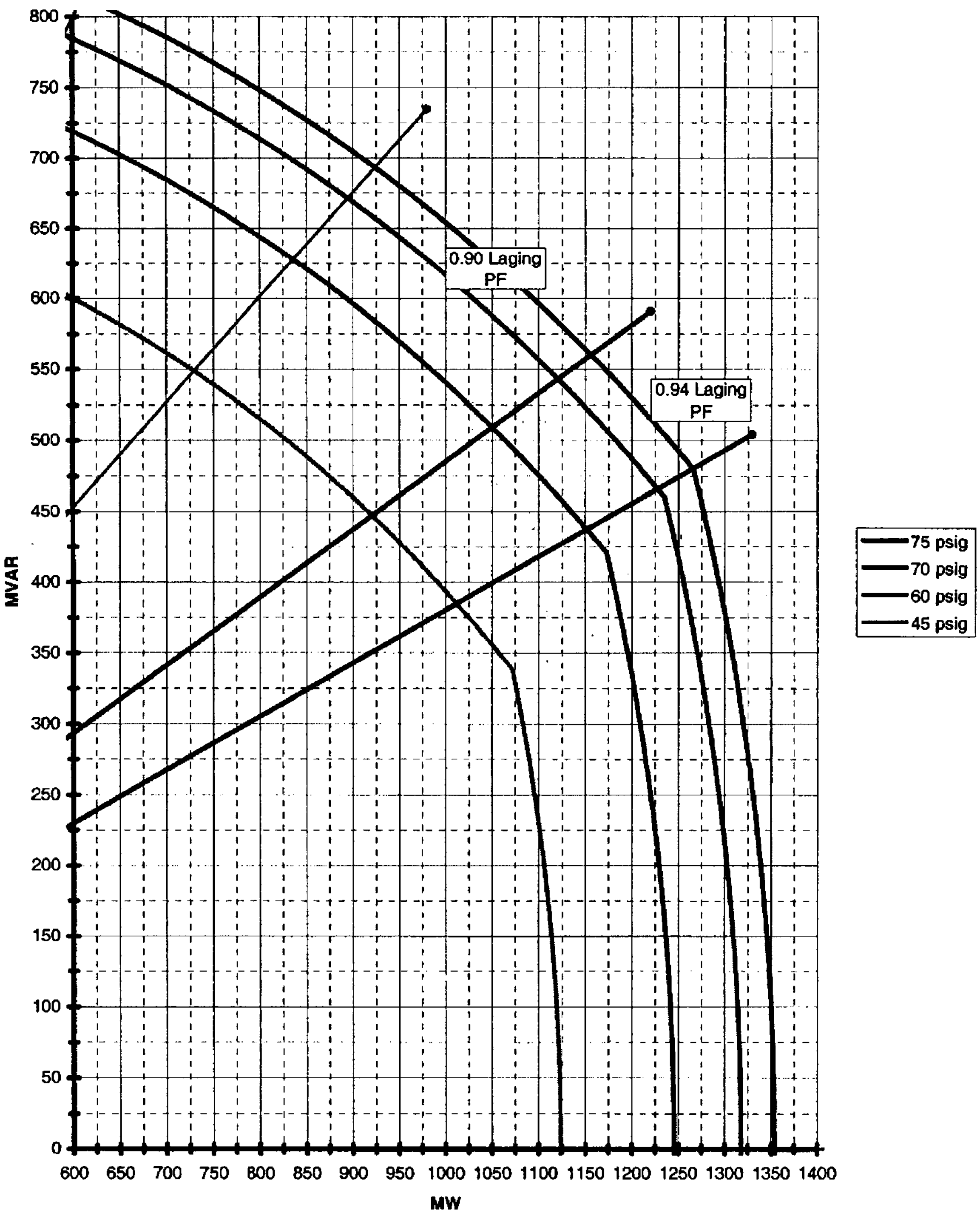
B. Depress LOAD SELECTOR DECREASE pushbutton.

C. Rotate AUTO VOLT REG ADJUST HC-10001 potentiometer clockwise (↻).

D. Rotate AUTO VOLT REG ADJUST HC-10001 potentiometer counterclockwise (↺).



0.30 Laging PF
0.60 Laging PF
0.80 Laging PF
0.90 Laging PF
0.94 Laging PF
75 psig
70 psig
60
45 psig
30 psig
LAG
LEAD
0.95 Leading PF
0.30 Leading PF
0.70 Leading PF
MVAR
MW
1200
1000
800
600
400
200
0
-200
-400
-600
-800
-1000
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1300
1400
1500
1600



0.90 Laging PF
0.94 Laging PF
75 psig
70 psig
60 psig
45 psig
MVAR
MW
800
750
700
650
600
550
500
450
400
350
300
250
200
150
100
50
0
600 650 700 750 800 850 900 950 1000 1050 1100 1150 1200 1250 1300 1350 1400

Proposed Answer: D

Explanation: The given Generator parameters are exceeding the capability curve. MVARs must be lowered. This is accomplished by rotating the AUTO VOLT REG ADJUST HC-10001 potentiometer counterclockwise.

A. Incorrect – The correct control is the AUTO VOLT REG ADJUST HC-10001 potentiometer, not the LOAD SELECTOR pushbuttons. Plausible because these pushbuttons are used to adjust Generator power (MWe) during a startup.
B. Incorrect – The correct control is the AUTO VOLT REG ADJUST HC-10001 potentiometer, not the LOAD SELECTOR pushbuttons. Plausible because these pushbuttons are used to adjust Generator power (MWe) during a startup.
C. Incorrect – The given Generator parameters are exceeding the capability curve. MVARs must be lowered, not raised. This is accomplished by rotating the AUTO VOLT REG ADJUST HC-10001 potentiometer counterclockwise, not clockwise. Plausible if the candidate mis-interprets conditions as requiring more MVARs or if candidate confuses operation of the AUTO VOLT REG ADJUST HC-10001 potentiometer.

Technical Reference(s): ON-198-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-098 RBO-7

Question Source: Modified Bank – LOC23 NRC #58

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK1.02
	Importance Rating	3.8

Reactor Low Water Level

Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Natural circulation: Plant-Specific

Proposed Question: #40

Unit 1 is shutdown with the following:

- A complete loss of Shutdown Cooling has occurred due to manual isolation of a leak.
- Both Reactor Recirculation pumps are unavailable.
- Reactor water level is 20", steady.

Which one of the following identifies the status of Reactor water level to ensure adequate Reactor coolant circulation in accordance with ON-SDC-101, Loss of Shutdown Cooling?

Reactor water level is...

A. adequate at the current value.

B. too low. The minimum required value is 35".

C. too low. The minimum required value is 45".

D. too low. The minimum required value is 54".

Proposed Answer: C

Explanation: With no Reactor Recirculation pumps running and a complete loss of Shutdown Cooling, there is no forced circulation through the Reactor core. ON-SDC-101 requires Reactor water level to be a minimum of 45" in this situation to ensure adequate natural circulation.

A. Incorrect – The current Reactor water level to 20" is below the minimum required value of 45". Plausible because Reactor water level is above 13", which is the lower end of the EO-100-102 control band.
B. Incorrect – The minimum required value is 45". 35" is plausible because it is the normal Feedwater level control value.
D. Incorrect – The minimum required value is 45". 54" is plausible because it is the upper end of the EO-100-102 control band.

Technical Reference(s): ON-SDC-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-7

Question Source: Modified Bank – JAF 9/14 NRC #6

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 2
Group # 1
K/A # 205000 K3.02
Importance Rating 3.2

Shutdown Cooling

Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor water level: Plant-Specific

Proposed Question: #6

The plant is shutdown with the following:

- RHR B is operating in Shutdown Cooling mode.
- Both Recirculation loops are in service.
- Reactor water level is 195 inches and stable.

Then, a spurious signal causes isolation of Shutdown Cooling.

Which one of the following describes the status of Reactor water level, in accordance with AOP-30, Loss of Shutdown Cooling?

Reactor water level...

A. is within the preferred control band of 177 to 270 inches.

B. is within the preferred control band of 177 to 222.5 inches.

C. should be raised to the preferred control band of 200 to 270 inches.

D. should be raised to the preferred control band of 234.5 to 270 inches.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EK1.03
	Importance Rating	2.8

High Off-site Release Rate

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

Proposed Question: #41

Both Units are operating at 100% power with the following:

- I&C reports that the wind speed and direction indications have been lost from all levels of the 200' Meteorological Tower (60 meter tower) located near the Well Water Storage tank.

Which one of the following describes the impact of this loss?

The operational concern with loss of the Meteorological Tower indications is the effect on the ability to ___(1)___.

Alternate wind speed and direction indications are ___(2)___ from installed instrumentation in the Control Room.

	(1)	(2)
A.	assess radiological releases	available
B.	assess radiological releases	NOT available
C.	predict adverse weather conditions	available
D.	predict adverse weather conditions	NOT available

Proposed Answer: A

Explanation: The purpose of the Met Tower is to provide wind speed and direction to assist in assessing the impact of radiological releases. While the 200' tower is the primary tower, alternate indications for wind speed and direction are also available in the Control Room from a 30' backup tower.

B. Incorrect – Alternate indications for wind speed and direction are also available in the Control Room from a 30' backup tower. Plausible because the 200' tower is the primary tower and provides more indications.
C. Incorrect – The operational concern with loss of indications is the ability to obtain wind speed and direction for assessment of radiological releases. Plausible because met tower data could be used to predict weather and ON-NAT-PHENOM uses wind speed indication to monitor current weather conditions.
D. Incorrect – Alternate indications for wind speed and direction are also available in the Control Room from a 30' backup tower. Plausible because the 200' tower is the primary tower and provides more indications. Plausible because met tower data could be used to predict weather and ON-NAT-PHENOM uses wind speed indication to monitor current weather conditions.

Technical Reference(s): EP-PA-001 Att JJ

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2015 Cert #39

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK2.01
	Importance Rating	3.3

Partial or Complete Loss of CCW

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System loads

Proposed Question: #42

Unit 1 is operating at 100% power with the following:

- RBCCW pump 1A trips due to motor overload.
- RBCCW pump 1B fails to automatically start.
- RBCCW discharge header pressure lowers to 0 psig.
- Thirty (30) seconds later, an Operator manually starts RBCCW pump 1B.

Which one of the following loads automatically trips due to this transient?

A. Offgas chiller

B. Low Pressure Air compressor

C. Reactor Water Cleanup pump

D. Containment Instrument Gas compressor

Proposed Answer: D

Explanation: Containment Instrument Gas compressors automatically trip if their cooling water supply from RBCCW drops below 20 psig.

A. Incorrect – RBCCW supplies cooling water for Offgas chillers, but they do not automatically trip on low RBCCW pressure.
B. Incorrect – RBCCW supplies cooling water to the Low Pressure Air compressor, but it does not automatically trip on low RBCCW pressure.
C. Incorrect – RBCCW supplies cooling water to RWCU, but the pumps do not automatically trip on low RBCCW pressure. RWCU pumps would trip on a sustained loss of RBCCW due to high NRHX outlet temperature isolation.

Technical Reference(s): ON-RBCCW-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-25 RBO-4

Question Source: Modified Bank – LOC25 NRC #44

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK2.01
	Importance Rating	4.4

Control Room Abandonment

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

Proposed Question: #43

A Control Room evacuation has been performed due to a fire with the following:

- ON-100-009, Control Room Evacuation is being performed.
- All required actions were completed prior to exiting the Control Room.
- All required Remote Shutdown Panel transfer switches have been placed in the EMERG position.

Which one of the following:

(1) lists the SRVs that can be operated from the Remote Shutdown Panel and

(2) the status of their Control Room controls?

A. (1) A, B, and C
(2) Disabled

B. (1) A, B, and C
(2) Still enabled

C. (1) G, J, K, L, M, and N
(2) Disabled

D. (1) G, J, K, L, M, and N
(2) Still enabled

Proposed Answer: A

Explanation: SRVs A, B, and C are the only SRVs that can be operated from the Remote Shutdown Panel. Placing the associated transfer switches in EMERG both enables the Remote Shutdown Panel control and disables the Control Room control.

B. Incorrect – Placing the associated transfer switches in EMERG both enables the Remote Shutdown Panel control and disables the Control Room control. Plausible that Control Room controls remain enabled such that any actions taken in the Control Room would not be overridden by taking the transfer switches to EMERG.
C. Incorrect – SRVs A, B, and C are the only SRVs that can be operated from the Remote Shutdown Panel. SRVs G, J, K, L, M, and N are plausible because they are the ADS SRVs and ON-100-009 provides direction to operate them from the Upper Relay Room if necessary.
D. Incorrect – SRVs A, B, and C are the only SRVs that can be operated from the Remote Shutdown Panel. SRVs G, J, K, L, M, and N are plausible because they are the ADS SRVs and ON-100-009 provides direction to operate them from the Upper Relay Room if necessary. Placing the associated transfer switches in EMERG both enables the Remote Shutdown Panel control and disables the Control Room control. Plausible that Control Room controls remain enabled such that any actions taken in the Control Room would not be overridden by taking the transfer switches to EMERG.

Technical Reference(s): ON-100-009

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-001 RBO-4

Question Source: Bank – Hope Creek 2005 NRC #6

Question History: Hope Creek 2005 NRC #6

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AK2.04
	Importance Rating	2.5

Plant Fire On-site

Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Breakers, relays, and disconnects

Proposed Question: #44

Which one of the following loads does ON-013-001, Response to Fire, provide specific direction for tripping by local breaker operation?

A. Condensate pumps

B. Main Generator output

C. RHR Service Water pumps

D. Reactor Recirculation pumps

Proposed Answer: D

Explanation: One concern ON-013-001 is losing the ability to trip the Reactor Recirculation pumps due to a fire. The procedure includes direction on tripping Reactor Recirculation pumps using the field, supply, or RPT breakers.

A. Incorrect – ON-013-001 provides specific direction for locally tripping Reactor Recirculation pump breakers, not Condensate pumps. Plausible because the procedure does have other steps for responding to uncontrolled Feedwater injection (closing MSIVs).
B. Incorrect – ON-013-001 provides specific direction for locally tripping Reactor Recirculation pump breakers, not the Main Generator output. Plausible because the procedure does lead to a manual Reactor scram, which requires verifying the Main Generator output breaker open.
C. Incorrect – ON-013-001 provides specific direction for locally tripping Reactor Recirculation pump breakers, not RHR Service Water pumps. Plausible because the procedure does provide specific direction for RHRSW pump control if it trips spuriously.

Technical Reference(s): ON-013-001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295005 AK3.04
Importance Rating	3.2

Main Turbine Generator Trip

Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: Main generator trip

Proposed Question: #45

Unit 1 is operating at 20% power with the following:

- Leakage has developed in the Stator Cooling system.
- Annunciator AR-106-A04, STATOR COOLING WATER GEN INLET/OUTLET HI CONDUCTIVITY, is in alarm.
- Stator Cooling inlet and outlet conductivity are indicating upscale at 10 μmho.
- Annunciator AR-106-B04, STATOR COOLING WATER OUTLET HEADER HI TEMP, is in alarm.
- Stator Cooling outlet header temperature is 168°F, stable.
- Annunciator AR-106-D04, STATOR COOLING WATER LO PRESS, is in alarm.
- Stator Cooling pressure is 46 psig, stable.

Which one of the following describes the need for a Main Turbine / Generator trip?

Main Turbine / Generator trip...

A. is NOT required.

B. is required due to the conductivity values.

C. is required due to the outlet temperature value.

D. is required due to the pressure value.

Proposed Answer: B

Explanation: A Main Turbine / Generator trip is required by AR-106-A04 because Stator Cooling conductivity is >9.9 μmho.

A. Incorrect – A trip is required due to conductivity. Plausible because two of the three given parameter values do not require a trip and specific AR knowledge is required to know the third parameter value requires a trip.
C. Incorrect – A trip is required due to conductivity, not temperature. Plausible because temperature is high and a higher temperature (176°F) would require a trip.
D. Incorrect – A trip is required due to conductivity, not pressure. Plausible because pressure is low and a lower pressure (40.5 psig) would require a trip.

Technical Reference(s): AR-106-A04, AR-106-B04, AR-106-D04

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-097 RBO-7

Question Source: Bank – NMP1 2010 NRC #66

Question History: NMP1 2010 NRC #66

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK3.03
	Importance Rating	4.1

SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Lowering reactor water level

Proposed Question: #46

Which one of the following explains the reason for intentionally lowering Reactor water level during an ATWS?

A. Concentrate boron in the core.

B. Lower void fraction inside the shroud.

C. Raise preheating of the incoming feedwater.

D. Raise natural circulation through the core to mix boron.

Proposed Answer: C

Explanation: EO-100-113 bases state that the reason for lowering level is to uncover the feedwater spargers sufficiently to reduce core inlet subcooling (raise feedwater pre-heating).

A. Incorrect – The reason is to raise feedwater pre-heating by uncovering the spargers, not concentrate boron in the core. Lowering Reactor vessel water inventory would act to concentrate boron, but this is not the reason for lowering level.
B. Incorrect – The reason is to raise feedwater pre-heating by uncovering the spargers, not lowering void fraction inside the shroud. Plausible because another part of the reason for intentionally lowering water level is to stabilize oscillations causes by changes in void fraction inside the shroud.
D. Incorrect – The reason is to raise feedwater pre-heating by uncovering the spargers, not raise natural circulation and mix boron. Plausible because this is the reason for raising Reactor water level in subsequent steps of EO-100-113.

Technical Reference(s): EO-000-113

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2009 NRC #24

Question History: NMP1 2009 NRC #24

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AK3.01
	Importance Rating	3.3

Partial or Complete Loss of Instrument Air

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Backup air system supply: Plant-Specific

Proposed Question: #47

Unit 1 is operating at 100% power when the following events occur:

Time (mm:ss)	Event
00:00	An Instrument Air line break results in Instrument Air header pressure slowly and continuously lowering
01:00	The Lead IA Compressor fully loads
02:30	The Standby IA Compressor starts and loads
03:00	PCV-12560, SERVICE AIR CROSSTIE TO INSTRUMENT AIR, begins to open
04:00	Annunciator AR-124-A01, INSTRUMENT AIR LOOP A LO PRESSURE, alarms

Which one of the following identifies the approximate Instrument Air ***header pressure*** at time 03:00?

A. 95 psig

B. 87 psig

C. 85 psig

D. 80 psig

Proposed Answer: C

Explanation: At time 03:00, Instrument Air header pressure is approximately 85 psig because this is the pressure at which the Service Air system provides automatic backup to the Instrument Air system through opening of PCV-12560.

A. Incorrect – At time 03:00, IA header pressure is approximately 85 psig. 95 psig is plausible because this is the IA header pressure at time 01:00 as evidenced by full loading of the Lead IAC. Also plausible because PCV-12560 starts to open at a locally sensed pressure of approximately 95 psig, which corresponds to a ***header pressure*** of 85 psig.
B. Incorrect – At time 03:00, IA header pressure is approximately 85 psig. 87 psig is plausible because this is the IA header pressure at time 02:30 as evidenced by start of the Standby IAC.
D. Incorrect – At time 03:00, IA header pressure is approximately 85 psig. 80 psig is plausible because this is the IA header pressure at time 04:00 as evidenced by AR-124-A01.

Technical Reference(s): ON-118-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-4

Question Source: Bank – LOC25 Cert #53

Question History:

Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AA1.03
	Importance Rating	4.4

Partial or Complete Loss of AC

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Systems necessary to assure safe plant shutdown

Proposed Question: #48

A Station Blackout is in progress.

Which one of the following describes steps that are taken to limit battery discharge, in accordance with EO-100-030, Unit 1 Response to Station Blackout?

Shedding of non-class 1E 250 VDC loads is required to be completed within __(1)__ minutes from the start of the Station Blackout. HPCI and RCIC are __(2)__.

A. (1) 15
(2) run continuously unless erratic operation is observed.

B. (1) 15
(2) secured as soon as NOT needed for Reactor water level control.

C. (1) 45
(2) run continuously unless erratic operation is observed.

D. (1) 45
(2) secured as soon as NOT needed for Reactor water level control.

Proposed Answer: C

Explanation: The time requirement for load shedding of non-class 1E 250 VDC loads is 45 minutes. HPCI and RCIC are run continuously unless erratic operation is observed. Both of these strategies are meant to reduce battery discharge rate to preserve battery capacity.

A. Incorrect – The time requirement for load reductions is 45 minutes.
B. Incorrect – The time requirement for load reductions is 45 minutes.
D. Incorrect – HPCI and RCIC are run continuously unless erratic operation is observed. Plausible that securing these systems when not needed for Reactor water level control would result in preservation of more battery capacity.

Technical Reference(s): EO-1(2)00-030

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC25 Cert #40

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EA1.07
	Importance Rating	4.1

High Reactor Pressure

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

Proposed Question: #49

Unit 1 is operating at 100% power with the following:

- A malfunction occurs with the Main Turbine pressure regulator.
- Reactor pressure rises.
- An Operator places the Reactor Mode Switch in SHUTDOWN.
- All control rods insert.
- Reactor pressure rises to a high value of 1150 psig and then lowers to 920 psig on Turbine Bypass Valves.
- Reactor pressure was above 1080 psig for two seconds during the transient.

Which one of the following describes the status of the Alternate Rod Insertion (ARI) solenoids and the Reactor Recirc pumps after this transient?

	ARI Solenoids	Reactor Recirc Pumps
A.	Energized	Tripped
B.	Energized	Running
C.	De-energized	Tripped
D.	De-energized	Running

Proposed Answer: A

Explanation: Reactor pressure rising above 1135 psig actuates both the ARI and ATWS-RPT logic. The ARI logic energizes the ARI solenoids to vent the scram air header and cause a alternate Reactor scram method. The ATWS-RPT logic trips the RR pumps. There is no time delay on the high pressure actuation of ARI and ATWS-RPT logic.

B. Incorrect – RR pumps are tripped due to ATWS-RPT logic actuation. Plausible because there is a 10 second time delay on ATWS-RPT logic for low Reactor water level, but not high pressure.
C. Incorrect – ARI solenoids are energized due to ARI logic actuation. Plausible because there is a 10 second time delay on ATWS-RPT logic for low Reactor water level, but not high pressure.
D. Incorrect – RR pumps are tripped due to ATWS-RPT logic actuation. ARI solenoids are energized due to ARI logic actuation. Plausible because there is a 10 second time delay on ATWS-RPT logic for low Reactor water level, but not high pressure.

Technical Reference(s): DBD021, SI-164-304

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-064 RBO-4

Question Source: Bank – JAF 4/14 NRC #50

Question History: JAF 4/14 NRC #50

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AA1.03
	Importance Rating	3.4

Partial or Complete Loss of DC Power

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: A.C. electrical distribution

Proposed Question: #50

Unit 1 is operating at 100% power when a loss of the normal 125 VDC supply occurs to ESS Bus 1A.

Which one of the following describes how ESS Bus 1A will be affected if its normal offsite power source de-energizes?

A. ESS Bus 1A Normal Feeder Breaker would trip open.
Diesel Generator A would NOT start.

B. ESS Bus 1A Normal Feeder Breaker would remain closed.
Diesel Generator A would NOT start.

C. ESS Bus 1A Normal Feeder Breaker would trip open.
Diesel Generator A would start but NOT load onto the bus.

D. ESS Bus 1A Normal Feeder Breaker would remain closed.
Diesel Generator A would start but NOT load onto the bus.

Proposed Answer: B

Explanation: A loss of 125 VDC to ESS Bus 1A results in a loss of breaker control functions. In a normal situation, a loss of the normal offsite power source would result in a fast transfer to the alternate supply and the DG would not start. However, with the loss of DC, the feeder breakers will not operate. The DG would not start, even with a loss of bus voltage, because it needs to see all three bus feeder breakers (Normal, Alternate, DC Output) open before a DG start signal occurs.

A. Incorrect – ESS Bus 1A Normal Feeder Breaker would NOT trip open due to a loss of control power. Plausible that this breaker would have automatic DC backup power or maintain trip capability.
C. Incorrect – ESS Bus 1A Normal Feeder Breaker would NOT trip open due to a loss of DC control power. Plausible that this breaker would have automatic DC backup power or maintain trip capability. The DG would not start – it needs to see all three bus feeder breakers (Normal, Alternate, DC Output) open. Plausible that the DG would start, but just not close in on the bus due to breaker control power loss.
D. Incorrect – The DG would not start – it needs to see all three bus feeder breakers (Normal, Alternate, DC Output) open. Plausible that the DG would start, but just not close in on the bus due to breaker control power loss.

Technical Reference(s): ON-125VDC-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-003 RBO-5

Question Source: Bank – LOC25 NRC #48

Question History: LOC25 NRC #48

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295028 EA2.02
Importance Rating	3.8

High Drywell Temperature

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

Proposed Question: #51

Which one of the following describes the requirement in EO-100-103, Primary Containment Control, for performing an Emergency RPV Depressurization due to high Drywell temperature?

EO-100-103 requires Emergency RPV Depressurization if Drywell temperature cannot be restored and maintained below the threshold value of...

A. 230°F.

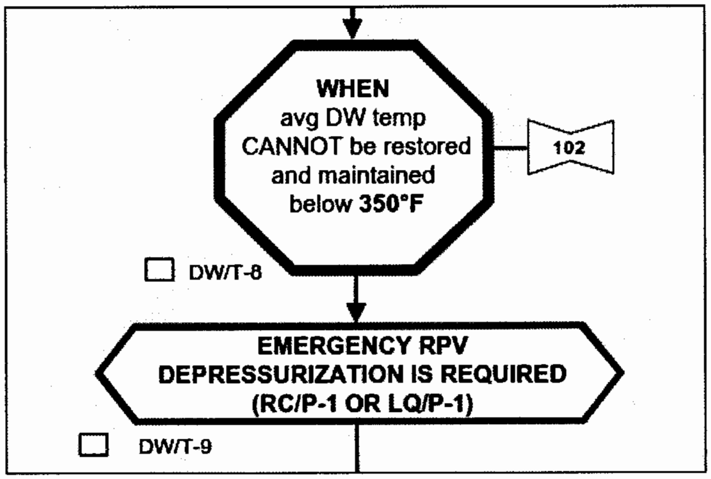
B. 340°F.

C. 350°F.

D. 440°F.

Proposed Answer: C

Explanation: EO-100-103 contains the following steps:



A. Incorrect – The threshold value requiring Emergency RPV Depressurization is 350°F, not 230°F. Plausible because this is well above the 150°F EOP entry condition and 230°F is used in EO-100-103 as the upper limit on average SPOTMOS indication.
B. Incorrect – The threshold value requiring Emergency RPV Depressurization is 350°F, not 340°F. Plausible because 340°F is the threshold value for scramming the Reactor and initiating Drywell spray.
D. Incorrect – The threshold value requiring Emergency RPV Depressurization is 350°F, not 440°F. Plausible because 440°F is used in EO-100-103 as the upper limit on Drywell temperature indication.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: Modified Bank – JAF 4/14 NRC #45

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 1
Group # 1
K/A # 295028 EK3.01
Importance Rating 3.6

High Drywell Temperature

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: Emergency depressurization

Proposed Question: #45

Which one of the following describes the requirement in EOP-4, Primary Containment Control, for performing an emergency depressurization due to high Drywell temperature and the basis of this requirement?

EOP-4 requires emergency depressurization if Drywell temperature cannot be restored and maintained below...

A. 260°F. This requirement is based on applicable component qualification or structural design temperature limits.

B. 260°F. This requirement is based on the upper limit for Primary Containment temperature indication.

C. 309°F. This requirement is based on applicable component qualification or structural design temperature limits.

D. 309°F. This requirement is based on the upper limit for Primary Containment temperature indication.

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

SCRAM

Ability to determine and/or interpret the following as they apply to SCRAM: Reactor pressure

Proposed Question: #52

Unit 1 is operating at 100% power when power is lost to both RPS Bus A and RPS Bus B.

Which one of the following describes the automatic Reactor pressure control response?

Reactor pressure will be controlled...

A. by automatic cycling of the Safety Relief Valves.

B. at approximately 934 psig by the Turbine Bypass Valves.

C. at approximately 1040 psig by the Turbine Bypass Valves.

D. at approximately 1040 psig by the Turbine Control Valves.

Proposed Answer: A

Explanation: Loss of power to both RPS Bus A and B causes a Reactor scram and closure of the MSIVs. With MSIVs closed, Reactor pressure will rise until SRVs automatically cycle.

B. Incorrect – The Reactor does scram on this loss of power. Turbine Bypass Valves normally control Reactor pressure ~934 psig following a scram. However, Turbine Bypass Valves are unavailable in this situation due to automatic closure of the MSIVs due to the power loss.
C. Incorrect – The Reactor does scram on this loss of power. Turbine Bypass Valves normally control Reactor pressure following a scram. 1040 psig is normal Reactor pressure at 100% power. However, Turbine Bypass Valves are unavailable in this situation due to automatic closure of the MSIVs due to the power loss.
D. Incorrect – This loss of power results in an automatic Reactor scram. If no Reactor scram occurred, Reactor pressure would continue to be controlled ~1040 psig by the Turbine Control Valves.

Technical Reference(s): ON-RPS-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-058 RBO-6

Question Source: Modified Bank – LOC26R NRC #44

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.01
	Importance Rating	3.5

Loss of Shutdown Cooling

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water heatup/cooldown rate

Proposed Question: #53

Unit 1 is being shutdown for a refueling outage with the following:

- The Reactor Mode Switch is in SHUTDOWN.
- The Reactor Vessel head closure bolts are still fully tensioned.
- RHR loop A is operating in Shutdown Cooling.

Then, a complete loss of Shutdown Cooling results in the following Reactor Coolant temperature response:

Time (hhmm)	Reactor Coolant Temperature (°F)
0800	140
0802	144
0804	148

Assuming the rate of temperature rise remains constant, which one of the following describes the status of the heatup rate and the approximate time when boiling will **first** occur?

	Status of Heatup Rate	First Boiling Time
A.	Below Technical Specification Limit	0830
B.	Below Technical Specification Limit	0836
C.	Above Technical Specification Limit	0830
D.	Above Technical Specification Limit	0836

Proposed Answer: D

Explanation: The given data shows a heatup rate of 4°F every 2 minutes, or 120°F/hr. This is above the Technical Specification 3.4.10 heatup rate limit of 100°F/hr. A Mode change occurs when temperature reaches 200°F. At the given heatup rate, this will occur at 0830 [0804 + (200°F – 148°F) x (2 min / 4°F) = 0830]. Boiling first occurs when temperature reaches 212°F. At the given heatup rate, this will occur at 0836 [0804 + (212°F – 148°F) x (2 min / 4°F) = 0836].

A. Incorrect – The given data shows a heatup rate of 4°F every 2 minutes, or 120°F/hr, which is above the Technical Specification 3.4.10 heatup rate limit of 100°F/hr. 0830 is the time when a Mode change first occurs, but boiling will not occur until 0836.
B. Incorrect – The given data shows a heatup rate of 4°F every 2 minutes, or 120°F/hr, which is above the Technical Specification 3.4.10 heatup rate limit of 100°F/hr.
C. Incorrect – 0830 is the time when a Mode change first occurs, but boiling will not occur until 0836.

Technical Reference(s): TS Table 1.1-1, TS 3.4.10

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-062 RBO-8

Question Source: Modified Bank – LOC27 NRC #40

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 2.2.22
	Importance Rating	4.0

Refueling Accidents

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #54

Which one of the following describes the requirement for Spent Fuel Pool water level in accordance with Technical Specification 3.7.7, Spent Fuel Storage Pool Water Level?

The minimum required level is...

A. 794', which corresponds to 8' over the top of fuel assemblies in the racks.

B. 794', which corresponds to 22' over the top of fuel assemblies in the racks.

C. 816', which corresponds to 8' over the top of fuel assemblies in the racks.

D. 816', which corresponds to 22' over the top of fuel assemblies in the racks.

Proposed Answer: D

Explanation: TS 3.7.7 requires SFP water level to be a minimum of 22' above the top of fuel assemblies in the racks. This corresponds to an indicated level of 816'.

A. Incorrect – TS 3.7.7 requires SFP water level to be a minimum of 22' above the top of fuel assemblies in the racks. This corresponds to an indicated level of 816'. 794' is plausible because this is the value used in EO-100-104 for initiating SFP spray. 8' is plausible because this is the approximate amount of water above fuel when it is grappled and in the full up position during refueling operations.
B. Incorrect – TS 3.7.7 requires SFP water level to be a minimum of 22' above the top of fuel assemblies in the racks. This corresponds to an indicated level of 816'. 794' is plausible because this is the value used in EO-100-104 for initiating SFP spray.
C. Incorrect – TS 3.7.7 requires SFP water level to be a minimum of 22' above the top of fuel assemblies in the racks. This corresponds to an indicated level of 816'. 8' is plausible because this is the approximate amount of water above fuel when it is grappled and in the full up position during refueling operations.

Technical Reference(s): TS 3.7.7

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-035 RBO-8

Question Source: Modified Bank – Vision SYSID 32112

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 2.1.23
	Importance Rating	4.3

Low Suppression Pool Water Level

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: #55

Unit 1 is operating at 100% power with the following:

- A Suppression Pool leak has developed.
- Suppression Pool water level is 21.5', down slow.
- EO-100-103, Primary Containment Control, is being executed.
- The Unit Supervisor has directed initiation of makeup to the Suppression Pool per EO-100-103.

Which one of the following describes the procedurally allowed method that provides the highest Suppression Pool makeup flow rate?

A. Start HPCI and operate it on minimum flow.

B. Start RCIC and operate it on minimum flow.

C. Gravity drain the Condensate Storage Tank through a Core Spray suction line.

D. Use Condensate Transfer to pump water through a Core Spray minimum flow line.

Proposed Answer: C

Explanation: EO-100-103 and OP-159-001 allow all four of these methods to add water to the Suppression Pool. Out of these methods, gravity draining the CST to the Suppression Pool through a Core Spray suction line provides the highest rate of makeup.

A. Incorrect – Running HPCI on minimum flow is one of the procedurally allowed methods to makeup to the Suppression Pool, but the flow rate is not as great as draining the CST through a Core Spray suction line.
B. Incorrect – Running RCIC on minimum flow is one of the procedurally allowed methods to makeup to the Suppression Pool, but the flow rate is not as great as draining the CST through a Core Spray suction line.
D. Incorrect – Pumping Condensate Transfer through the Core Spray minimum flow line is one of the procedurally allowed methods to makeup to the Suppression Pool (and it is the normal method), but the flow rate is not as great as draining the CST through a Core Spray suction line.

Technical Reference(s): EO-100-103, OP-159-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-5

Question Source: Bank – LOC24 Cert #49

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 2.2.12
	Importance Rating	3.7

Suppression Pool High Water Temperature

Knowledge of surveillance procedures.

Proposed Question: #56

Unit 1 is operating at 80% power and surveillance SO-150-002, Quarterly RCIC Flow Verification, is going to be performed this shift.

Which one of the following describes the required control of Suppression Pool Cooling in accordance with SO-150-002?

Suppression Pool Cooling must...

A. be initiated prior to starting RCIC.

B. be initiated while RCIC is operating when Suppression Pool average water temperature exceeds 90°F.

C. be initiated while RCIC is operating when Suppression Pool average water temperature exceeds 105°F.

D. NOT be initiated until after RCIC is secured.

Proposed Answer: A

Explanation: SO-150-002 requires starting Suppression Pool Cooling prior to starting RCIC to mitigate the expected rise in Suppression Pool water temperature.

B. Incorrect – SO-150-002 requires starting Suppression Pool Cooling prior to starting RCIC. Plausible because 90°F is the normal SP temperature limit in TS 3.6.2.1 and the EO-100-103 entry condition.
C. Incorrect – SO-150-002 requires starting Suppression Pool Cooling prior to starting RCIC. Plausible because 105°F is the applicable SP temperature limit in TS 3.6.2.1 during surveillance testing.
D. Incorrect – SO-150-002 requires starting Suppression Pool Cooling prior to starting RCIC. Plausible if test required measurement of SP temperature rate of rise or if candidate believed there was an operability concern with simultaneous operation of RCIC and RHR.

Technical Reference(s): SO-150-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-050 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK2.11
	Importance Rating	4.2

High Drywell Pressure

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Drywell spray (RHR) logic: Mark-I&II

Proposed Question: #57

Unit 1 is operating at 100% power with the following:

- HV-151-F016A, Drywell Spray Outboard Isolation Valve, is open for testing.

Then, leak in the Drywell occurs:

- Drywell pressure is 5 psig, up slow.
- Reactor water level is -45", down slow.

Which one of the following describes the response of HV-151-F016A?

A. Closes due to a Drywell pressure signal, only.

B. Closes due to a combination of Drywell pressure and Reactor water level signals.

C. Remains open, but will automatically close if Reactor water level lowers below -129".

D. Remains open and will NOT automatically close if Reactor water level lowers below -129".

Proposed Answer: A

Explanation: HV-151-F016A automatically closes given any of the following signals:

- Reactor water level < -129"
- Drywell pressure > 1.72 psig
- Manual Initiation Div 1 or 2 PB Armed and Depressed.

Therefore, HV-151-F016A closes automatically based on a Drywell pressure signal only.

B. Incorrect – The given value of Reactor water level is too high to cause a signal that causes this valve to close. Plausible because Reactor water level <-129" does cause the valve to close.
C. Incorrect – The valve closes on a Drywell pressure signal. Plausible because Reactor water level is above -129", which is a separate closure signal.
D. Incorrect – The valve closes on a Drywell pressure signal. Plausible because the valve is initially open, is opened during LOCA conditions on purpose per the EOPs, and auto closure can be overridden.

Technical Reference(s): ON-159-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-4

Question Source: Modified Bank – Vision SYSID 32768

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AA2.04
	Importance Rating	3.0

Partial or Complete Loss of Forced Core Flow Circulation

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Individual jet pump flows: Not-BWR-1&2

Proposed Question: #58

Unit 1 is operating at 100% power when the jet pump mixer for Jet Pump (JP) 11 becomes displaced.

Which one of the following identifies the resulting change in JP 2 and 12 individual flows, in accordance with AOP-164-001, Reactor Recirculation Abnormal Operating Procedure?

A. Both flows rise.

B. Both flows lower.

C. JP 2 flow rises and JP 12 flow lowers.

D. JP 2 flow lowers and JP 12 flow rises.

Proposed Answer: C

Explanation: JPs 1-10 are in Recirc loop B. JPs 11-20 are in Recirc loop A. JPs 11 and 12 share a riser. With a displaced mixer on JP 11, the individual flow for the JP on the same riser (JP 12) lowers. The individual flows for JPs in the intact loop (loop B – JPs 1-10) rise.

A. Incorrect – JP 12 flow lowers because it shares a riser with the failed JP 11. Plausible if the candidate does not recall which JPs share a riser or understand the dynamics of this failure.
B. Incorrect – JP 2 flow rises because it is in the opposite loop as the failed JP 11. Plausible if the candidate does not recall the JP loop assignments or understand the dynamics of this failure.
D. Incorrect – JP 2 flow rises because it is in the opposite loop as the failed JP 11. Plausible if the candidate does not recall the JP loop assignments or understand the dynamics of this failure. JP 12 flow lowers because it shares a riser with the failed JP 11. Plausible if the candidate does not recall which JPs share a riser or understand the dynamics of this failure.

Technical Reference(s): AOP-164-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-62 RBO-6

Question Source: Modified Bank – Peach Bottom 2013 NRC #52

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

52. Unit 2 was operating at 100% power.

- The jet pump mixer for Jet Pump '11' becomes displaced.

This failure will cause a sudden RISE in:

A. Core Plate Flow

B. DP on Jet Pump '12'

C. 'A' Recirc Loop Drive Flow

D. 'B' Recirc Pump Speed

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295014 AK1.05
	Importance Rating	3.7

Inadvertent Reactivity Addition

Knowledge of the operational implications of the following concepts as they apply to INADVERTENT REACTIVITY ADDITION: Fuel thermal limits

Proposed Question: #59

Unit 1 is operating at 100% power with the following:

- Two control rods inadvertently drift out.
- The control rods CANNOT be inserted.
- Reactor power is lowered 20% using Recirculation flow.
- Thermal limits are currently:
    - FDLRX: 0.89
    - MFLCPR: 0.97
    - MAPRAT: 1.03

Which one of the following describes the significance of these thermal limit values, in accordance with Technical Specifications (TS)?

A. All three of these thermal limits are satisfactory.

B. TS 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR), is NOT met.

C. TS 3.2.2, Minimum Critical Power Ratio (MCPR), is NOT met.

D. TS 3.2.3, Linear Heat Generation Rate (LHGR), is NOT met.

Proposed Answer: B

Explanation: FDLRX, MFLCPR, and MAPRAT are ratios calculated by the core monitoring system to determine if LHGR, MCPR, and APLHGR thermal limits are satisfactory. If any of these ratios is greater than 1, the corresponding thermal limit is unsatisfactory and the associated TS is not met. In this case, since MAPRAT is greater than 1, the APLHGR thermal limit is unsatisfactory and TS 3.2.1 is not met.

A. Incorrect – Since MAPRAT is greater than 1, the APLHGR thermal limit is unsatisfactory and TS 3.2.1 is not met. Plausible if candidate does not understand that the given values are ratios that must be less than 1 for the associated thermal limits to be SAT.
C. Incorrect – Since MFLCPR is less than 1, the associated MCPR thermal limit is SAT. Plausible if candidate does not understand that the given values are ratios that must be less than 1 for the associated thermal limits to be SAT, or does not know which ratio corresponds to which TS.
D. Incorrect – Since FDLRX is less than 1, the associated LHGR thermal limit is SAT. Plausible if candidate does not understand that the given values are ratios that must be less than 1 for the associated thermal limits to be SAT, or does not know which ratio corresponds to which TS.

Technical Reference(s): TS 3.2.1-3, COLR

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-62X RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 AK2.05
	Importance Rating	3.7

High Drywell Pressure

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Drywell cooling and ventilation

Proposed Question: #60

Unit 1 is operating at 100% power with the following:

- A small loss of coolant accident occurs.
- Drywell pressure peaks at 2.0 psig.

Which one of the following describes the resulting status of Drywell Cooling?

Drywell Cooling fans are...

A. running and cooling water is being supplied to the coolers.

B. tripped and cooling water is being supplied to the coolers.

C. running and cooling water is isolated to the coolers.

D. tripped and cooling water is isolated to the coolers.

Proposed Answer: D

Explanation: Drywell pressure > 1.72 psig causes the Drywell Cooling fans to trip and isolates RBCW flow to the Drywell Coolers.

A. Incorrect – Drywell pressure > 1.72 psig causes the Drywell Cooling fans to trip. Plausible because it is desired to have the Drywell Cooling fans running and procedures direct restarting them in this situation. Drywell pressure > 1.72 psig isolates RBCW flow to the Drywell Coolers. Plausible because during a loss of offsite power, RBCCW is automatically aligned to replace flow from RBCW, but not in this situation.
B. Incorrect – Drywell pressure > 1.72 psig isolates RBCW flow to the Drywell Coolers. Plausible because during a loss of offsite power, RBCCW is automatically aligned to replace flow from RBCW, but not in this situation.
C. Incorrect – Drywell pressure > 1.72 psig causes the Drywell Cooling fans to trip. Plausible because it is desired to have the Drywell Cooling fans running and procedures direct restarting them in this situation.

Technical Reference(s): ON-159-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-073 RBO-4

Question Source: Modified Bank – LOC25 NRC #65

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295007 AK3.03
	Importance Rating	3.4

High Reactor Pressure

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: RCIC operation: Plant-Specific

Proposed Question: #61

Which one of the following describes the reason for operating RCIC in the CST to CST mode, in accordance with OP-150-001, RCIC System?

A. To utilize RCIC for Reactor pressure control.

B. To preserve Suppression Pool water for other systems.

C. To inject to the Reactor from a cleaner source of water.

D. To minimize the amount of time RCIC operates on minimum flow.

Proposed Answer: A

Explanation: The reason for placing RCIC in the CST to CST mode of operation is to utilize RCIC for Reactor pressure control (OP-150-001 2.6.1.f).

B. Incorrect – The reason for placing RCIC in the CST to CST mode of operation is to utilize RCIC for Reactor pressure control. Plausible because an alternate suction source for RCIC is the Suppression Pool.
C. Incorrect – The reason for placing RCIC in the CST to CST mode of operation is to utilize RCIC for Reactor pressure control. Plausible because an alternate suction source for RCIC is the Suppression Pool, which is likely to be a less clean source of water than the CST.
D. Incorrect – The reason for placing RCIC in the CST to CST mode of operation is to utilize RCIC for Reactor pressure control. Plausible because this does provide RCIC with an additional flow path to the minimum flow line.

Technical Reference(s): OP-150-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-050 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295033 EA1.03
	Importance Rating	3.8

High Secondary Containment Area Radiation Levels

Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Secondary containment ventilation

Proposed Question: #62

Unit 1 is operating at 100% power with the following:

- An irradiated fuel bundle has been damaged in the Unit 1 Fuel Pool.
- Multiple Refuel Floor area radiation monitors are in high alarm but indicate below the Max Safe radiation level.
- Unit 1 Refuel Floor Wall Exhaust rad monitors indicate 15 mR/hr, up slow.
- Unit 1 Refuel Floor High Exhaust rad monitors indicate 25 mR/hr, up slow.

Which one of the following describes the status of the Zone III HVAC Supply and Exhaust systems and the required control of these systems, in accordance with EO-100-104, Secondary Containment Control?

Zone III HVAC Supply and Exhaust systems...

A. are operating and are required to be maintained operating.

B. are operating, but are required to be secured.

C. are secured, but are required to be restarted.

D. are secured and are required to be maintained secured.

Proposed Answer: D

Explanation: A Zone III isolation signal is received due to Refuel Floor High Exhaust rad monitors indicating above 18 mR/hr. This isolates Zone III HVAC Supply and Exhaust systems and trips the associated fans. With Refuel Floor High Exhaust rad monitors indicating above the Hi-Hi alarm setpoint, EO-100-104 requires maintaining the Zone III HVAC Supply and Exhaust systems secured.

A. Incorrect – Zone III HVAC Supply and Exhaust systems are secured due to Refuel Floor High Exhaust rad monitors indicating above 18 mR/hr. Plausible because the Refuel Floor Wall Exhaust rad monitors are indicating below their Hi-Hi alarm setpoint of 21 mR/hr.
B. Incorrect – Zone III HVAC Supply and Exhaust systems are secured due to Refuel Floor High Exhaust rad monitors indicating above 18 mR/hr. Plausible because the Refuel Floor Wall Exhaust rad monitors are indicating below their Hi-Hi alarm setpoint of 21 mR/hr. Also plausible that manual securing would be required based on fuel damage and high area radiation conditions.
C. Incorrect – EO-100-104 requires maintaining the Zone III HVAC Supply and Exhaust systems secured. Plausible because in some situations, such as Zone III isolation on a LOCA signal, EO-100-104 provides direction to restart RB HVAC following automatic shutdown.

Technical Reference(s): EO-100-104

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-034 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295032 EA2.02
	Importance Rating	3.3

High Secondary Containment Area Temperature

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Equipment operability

Proposed Question: #63

Unit 1 is operating at 100% power with the following:

- A steam leak has developed in the HPCI equipment area.
- Annunciator AR-114-E05, HPCI LEAK DETECTION HI TEMP / HI DIFF TEMP, alarms.
- Annunciator AR-114-F04, HPCI LEAK DETECT LOGIC A HI TEMP, alarms.
- Both annunciators have been confirmed as valid based on high temperature in the HPCI equipment area.
- NO other annunciators are in alarm.

Which one of the following describes the operability / availability of HPCI?

HPCI...

A. remains available because it will NOT isolate unless annunciator AR-114-F05, HPCI LEAK DETECT LOGIC B HI TEMP, also alarms.

B. remains available but will isolate if these conditions are maintained for at least 15 minutes.

C. is inoperable and unavailable. Just one of the two steam supply isolation valves has automatically closed.

D. is inoperable and unavailable. Both steam supply isolation valves have automatically closed.

Proposed Answer: C

Explanation: Multiple temperature detectors input to annunciators AR-114-E05 and AR-114-F04. AR-114-F04 being in alarm based on high temperature in the HPCI equipment area causes the inboard steam supply isolation valve (HV-155-F002) to close. Since annunciator AR-114-F05 is NOT also in alarm, the outboard steam supply isolation valve (HV-155-F003) does NOT close.

A. Incorrect – The HPCI isolation logic is setup such that high temperature logic A is sufficient alone to close one HPCI steam supply isolation valve. Plausible because many logics are setup such that they require both A and B to actuate to cause an automatic action.
B. Incorrect – High temperature in the HPCI equipment area isolates the steam supply without time delay. Plausible because high temperature in the HPCI pipe routing area only causes an isolation after a 15 minute time delay.
D. Incorrect – AR-114-F04 being in alarm based on high temperature in the HPCI equipment area causes the inboard steam supply isolation valve (HV-155-F002) to close. Since annunciator AR-114-F05 is NOT also in alarm, the outboard steam supply isolation valve (HV-155-F003) does NOT close. Plausible that if one side of the logic can cause an isolation, it would close both isolation valves.

Technical Reference(s): AR-114-F04(05)

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-052 RBO-4

Question Source: Bank – JAF 4/14 NRC #64

Question History: JAF 4/14 NRC #64

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295035 2.4.1
	Importance Rating	4.6

Secondary Containment High Differential Pressure

Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: #64

Unit 1 is operating at 100% power with the following:

- A malfunction occurs with the Zone I differential pressure controller.
- Zone I differential pressure drops to 0.10" WG Vacuum, steady.

Which one of the following describes the need for entering EO-100-104, Secondary Containment Control?

EO-100-104 entry is...

A. required now.

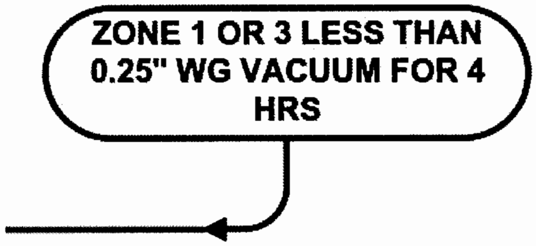
B. NOT required unless differential pressure degrades further.

C. first required if this differential pressure persists for 15 minutes.

D. first required if this differential pressure persists for 4 hours.

Proposed Answer: D

Explanation: EO-100-104 contains the following entry condition:



A. Incorrect – EOP entry is not yet required because the given DP has not persisted for 4 hours. Plausible because many plants do not have such a time requirement on EOP entry for high RB DP.
B. Incorrect – The given DP is high enough to require EOP entry, but must persist for 4 hours. Plausible because many plants do not require EOP entry until RB DP goes positive.
C. Incorrect – EOP entry is not required unless the given DP persists for 4 hours. Plausible because 15 minutes is a common time requirement in emergency response.

Technical Reference(s): EO-100-104

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-034 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 AK2.01
	Importance Rating	3.6

Inadvertent Containment Isolation

Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following: Main steam system

Proposed Question: #65

An inadvertent MSIV isolation has occurred with the following:

- The isolation signal is now clear.
- The MSIV control switches have been left in the AUTO position.

Then, the NSSS Logic reset pushbuttons are depressed and released.

Which one of the following describes the MSIV response?

A. The MSIV logic will NOT reset.

B. The MSIV logic will reset and the MSIVs will automatically open.

C. The MSIV logic will reset and the MSIVs will remain closed until the control switches are taken to OPEN.

D. The MSIV logic will reset and the MSIVs will remain closed until the control switches are cycled to CLOSE and back to AUTO.

Proposed Answer: A

Explanation: The MSIV logic will not reset when the NSSS Logic reset pushbuttons are depressed unless the MSIV control switches are first placed in CLOSE.

B. Incorrect – The logic will not reset. Plausible because the isolation signal is clear and the AUTO position could have been designed as the desired position for reset. Also plausible that with the signal reset, the AUTO position would automatically reopen MSIVs.
C. Incorrect – The logic will not reset. Plausible because the isolation signal is clear and the AUTO position could have been designed as the desired position for reset. Also plausible that with the signal reset, the OPEN position would be required to reopen MSIVs.
D. Incorrect – The logic will not reset. Plausible because the isolation signal is clear and the AUTO position could have been designed as the desired position for reset. Also plausible that with the signal reset, the control switch would need to be taken to CLOSE to reset some further seal-in.

Technical Reference(s): ON-184-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083 RBO-4

Question Source: Bank – Vision System ID 5159

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.41
	Importance Rating	2.8

Knowledge of the refueling process.

Proposed Question: #66

Given the following separate evolutions during a refueling outage:

(1) Removal of an irradiated fuel bundle from the Reactor core.

(2) Removal of a control rod blade from a fuel cell that contains two fuel bundles.

(3) Removal of a control rod blade from a fuel cell that contains NO fuel bundles.

(4) Removal of an LPRM string from the Reactor core.

Which one of the following identifies which of these evolutions is considered a Core Alteration, in accordance with NDAP-QA-0507, Conduct of Refuel Floor?

A. (1) only

B. (1) and (2) only

C. (1), (2), and (3) only

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

Proposed Answer: B

Explanation: (1) and (2) are the only listed core alterations per the following definition in NDAP-QA-0507:

5.2 CORE ALTERATIONS: Movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

5.2.1 Movement of Source Range Monitors (SRMs), Local Power Range Monitors (LPRMs), Intermediate Range Monitors (IRMs), Traversing Incore Probes (TIPs), or special moveable detectors (including undervessel replacement); and

5.2.2 Control rod movement, provided there are no fuel assemblies in the associated core cell.

A. Incorrect – Removal of a control rod blade from a fuel cell that contains two fuel bundles is also considered a core alteration. Plausible because this represents a partially de-fueled cell and a fully de-fueled cell is not considered a core alteration.
C. Incorrect – Removal of a control rod blade from a fuel cell that contains NO fuel bundles not considered a core alteration. Plausible because a control rod blade is a reactivity control component and this specific evolution is an exception to the general rule.
D. Incorrect – Removal of a control rod blade from a fuel cell that contains NO fuel bundles is not considered a core alteration. Plausible because a control rod blade is a reactivity control component and this specific evolution is an exception to the general rule. Removal of an LPRM string from the Reactor core is not considered a core alteration. Plausible because an LPRM string is a core component necessary for monitoring reactivity and this specific evolution is an exception to the general rule.

Technical Reference(s): NDAP-QA-0507

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.26
	Importance Rating	3.4

Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

Proposed Question: #67

Which one of the following is the voltage threshold for "Hazardous Electrical Energy", in accordance with NDAP-QA-0322, Energy Control Process?

A. 5 V

B. 12 V

C. 50 V

D. 100 V

Proposed Answer: C

Explanation: Per NDAP-QA-0322, Hazardous Electrical Energy is defines as a voltage ≥ 50 V.

A. Incorrect – Per NDAP-QA-0322, Hazardous Electrical Energy is defines as a voltage ≥ 50 V. Plausible because 5 V is a common low voltage used in control systems.
B. Incorrect – Per NDAP-QA-0322, Hazardous Electrical Energy is defines as a voltage ≥ 50 V. Plausible because 12 V is a common low voltage, such as in engine starting batteries.
D. Incorrect – Per NDAP-QA-0322, Hazardous Electrical Energy is defines as a voltage ≥ 50 V. Plausible because 100 V is a common low voltage, such as in common outlets.

Technical Reference(s): NDAP-QA-0322

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2013 Cert #68

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.2
	Importance Rating	4.6

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Proposed Question: #68

A Unit 1 startup is in progress with the following:

- The Reactor is NOT yet critical.
- The next control rod to be withdrawn is at position 12 and has a target position of 48.
- The control rod is NOT identified as fast on the list of CRD deficiencies.
- The following table gives SRM count rates before the startup commenced and current SRM count rates:

SRM	Count Rate Prior to Commencing Startup (cps)	Current Count Rate (cps)
A	110	770
B	140	850
C	120	820
D	90	740

Which one of the following describes the ability to withdraw the control rod using continuous rod withdrawal, in accordance with GO-100-002, Plant Startup, Heatup and Power Operation?

A. The control rod may be withdrawn continuously from position 12 to 48.

B. The control rod must be notch withdrawn at all positions from 12 to 48.

C. The control rod must be notch withdrawn from position 12 to 16. The control rod may be withdrawn continuously from position 16 to 48.

D. The control rod must be notch withdrawn from position 12 to 32. The control rod may be withdrawn continuously from position 32 to 48.

Proposed Answer: A

Explanation: All of the SRMs have increased well above their initial count rate, but none have reached 10x their initial count rate. GO-100-002 does not restrict continuous control rod withdrawal until one SRM reaches 10x its initial count rate. Therefore, the control rod may be continuously withdrawn from position 12 to 48.

B. Incorrect – None of the SRMs have reached 10x their initial count rate, therefore continuous control rod withdrawal is not yet restricted. Plausible because all SRMs have reached at least 6x their initial count rate and SRM D has achieved 3 count rate doublings, which is a common threshold used in the industry.
C. Incorrect – None of the SRMs have reached 10x their initial count rate, therefore continuous control rod withdrawal is not yet restricted. Plausible because all SRMs have reached at least 6x their initial count rate and SRM D has achieved 3 count rate doublings, which is a common threshold used in the industry. Position 16 is a plausible threshold because it is a common threshold for when control rod worth begins to become less significant.
D. Incorrect – None of the SRMs have reached 10x their initial count rate, therefore continuous control rod withdrawal is not yet restricted. Plausible because all SRMs have reached at least 6x their initial count rate and SRM D has achieved 3 count rate doublings, which is a common threshold used in the industry. Position 32 is the correct threshold for when continuous withdrawal is restricted.

Technical Reference(s): GO-100-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078A RBO-7

Question Source: Modified Bank – JAF 4/14 NRC #68

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 3
Group #
K/A # 2.2.2
Importance Rating 4.6

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Proposed Question: #68

A plant startup is in progress with the following:

- All SRMs are indicating five (5) count rate doublings over the initial count rate.
- The Reactor is NOT yet critical.
- The next control rod to be withdrawn is at position 12 and has a target position of 48.
- The control rod is NOT identified as fast on the list of CRD deficiencies.

Which one of the following describes the ability to withdraw the control rod using continuous rod withdrawal, in accordance with OP-26, Control Rod Drive Manual Control, and OP-65, Startup and Shutdown?

A. The control rod may be withdrawn continuously from position 12 to 48.

B. The control rod must NOT be withdrawn continuously at any position from 12 to 48.

C. The control rod must be notch withdrawn from position 12 to 30. The control rod may be withdrawn continuously from position 30 to 48.

D. The control rod may be withdrawn continuously from position 12 to 16 and from position 30 to 48. The control rod must be notch withdrawn from position 16 to 30.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.39
	Importance Rating	3.9

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

Proposed Question: #69

Unit 1 is operating at 100% power with the following:

- A Main Turbine pressure regulator malfunction occurs.
- Reactor pressure slowly rises to 1065 psig and stabilizes.

Which one of the following describes the impact of this Reactor pressure on Technical Specifications, if any?

A. NO Technical Specification Condition entry is required.

B. Technical Specifications require lowering Reactor pressure within a maximum of 15 minutes, but do NOT require an immediate Reactor scram.

C. Technical Specifications require lowering Reactor pressure within a maximum of 1 hour, but do NOT require an immediate Reactor scram.

D. Technical Specifications require an immediate Reactor scram.

Proposed Answer: B

Explanation: Reactor pressure above 1050 psig requires entry into Technical Specification 3.4.11 Condition A. This condition requires restoring pressure within a maximum of 15 minutes.

A. Incorrect – TS Condition entry is required because Reactor pressure is above the 1050 psig limit in TS 3.4.11. Plausible because Reactor pressure is still below other limits (1087 psig scram setpoint in TS 3.3.1.1, 1325 psig safety limit of TS 2.1.2).
C. Incorrect – The maximum time allowed by TS 3.4.11 Condition A to restore pressure is 15 minutes, not one hour. Plausible because other TS use one hour completion times and ROs are responsible for knowing completion times up to one hour.
D. Incorrect – An immediate Reactor scram is not required. Plausible because some other TS require immediate scram (TS 3.6.2.1 for example) and higher Reactor pressure would require a scram (TS 3.3.1.1 – 1093 psig).

Technical Reference(s): TS 3.4.11

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-062 RBO-8

Question Source: Modified Bank – NMP2 2009 NRC #100

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Facility:	Nine Mile Point 2
Vendor:	GE
Exam Date:	August 2009
Exam Type:	S

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			2
	K/A #	G2	2.2.39	
	Importance Rating			4.5

Equipment Control: Knowledge of less than one hour technical specification action statements for systems.

Proposed Question:	SRO Question # 100

The plant is operating at 100% power. A Pressure Regulator malfunction results in reactor pressure slowly rising from 1020 psig to 1041 psig.

To comply with Technical Specifications which one of the following is (1) the maximum reactor pressure permitted AND (2) what is the maximum time permitted above this pressure?

A.	(1) 1024 psig (2) 15 minutes.
B.	(1) 1024 psig (2) 1 hour.
C.	(1) 1034 psig (2) 15 minutes.
D.	(1) 1034 psig (2) 1 hour.

Proposed Answer:	C

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.4
	Importance Rating	3.2

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: #70

Unit 1 is shutdown with the following:

- A 31 year old Operator is entering the Drywell for a job.
- Expected dose for transiting to and from the job location is a total of 20 mRem.
- Expected dose rate at the job location is 3.5 Rem/hr.
- It will take 45 minutes to complete the job.
- The Operator's previous TEDE for the year is 1850 mRem.
- No dose extension has been previously obtained for this Operator.
- No emergency is in progress.

Which one of the following describes the Operator's expected dose, in accordance with NDAP-QA-0625, Personnel Radiation Exposure Monitoring Program?

The Operator's expected dose...

A. will stay within the normal annual dose control level *without* an extension.

B. will exceed the normal annual dose control level *without* an extension, but stay within the allowable dose control level *with* an extension.

C. will exceed the allowable dose control level *with* an extension, but stay within the federal dose limit.

D. will exceed the federal dose limits.

Proposed Answer: C

Explanation: The given job information will result in the Operator's annual dose rising to 4495 mRem (1850 mRem + 20 mRem + .75 hours * 3500 mRem/hour). This is above the normal annual dose control level without an extension (2000 mRem), above the maximum allowable dose with an extension (4000 mRem), but below the federal dose limit (5000 mRem).

A. Incorrect – The Operator's annual dose will be 4495 mRem, which is above the normal annual dose control level without an extension (2000 mRem).
B. Incorrect – The Operator's annual dose will be 4495 mRem, which is above the maximum allowable dose with an extension (4000 mRem).
D. Incorrect – The Operator's annual dose will be 4495 mRem, which is below the federal dose limit (5000 mRem).

Technical Reference(s): NDAP-QA-0625

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – LOC24 NRC #68

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.14
	Importance Rating	3.4

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

Proposed Question: #71

Unit 1 is in MODE 5 with the following:

- The Drywell is open for maintenance.
- Traversing Incore Probe (TIP) operation is required.

Which one of the following identifies the areas the Health Physics department must restrict access to prior to commencing TIP operation, in accordance with OP-178-001, TIP System?

In addition to the TIP room, access must be restricted to...

A. the CIG mezzanine, only.

B. the Drywell and north HCU area, only.

C. the CIG mezzanine and the Drywell, only.

D. the CIG mezzanine, the Drywell, and the north HCU area.

Proposed Answer: C

Explanation: Prior to TIP operation, OP-178-001 requires restricting access to the TIP room, the CIG mezzanine, and the Drywell.

A. Incorrect – Access must also be restricted to the Drywell. Plausible if candidate does not understand effect of TIP operation on Drywell dose rates.
B. Incorrect – The north HCU area does not need to be restricted. Plausible because this is in the area of the TIP room. The CIG mezzanine must be restricted. Plausible if candidate does not understand effect of TIP operation on CIG mezzanine dose rates.
D. Incorrect – The north HCU area does not need to be restricted. Plausible because this is in the area of the TIP room.

Technical Reference(s): OP-178-001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC23 NRC #71

Question History: LOC23 NRC #71

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.37
	Importance Rating	3.0

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

Proposed Question: #72

An emergency has been declared with the following:

- All Emergency Response Facilities have been activated.
- An In-Plant (India) Team is required to be dispatched to the field to isolate a leak.

Which one of the following identifies which facility is responsible for assembling, briefing, and dispatching the In-Plant (India) Team?

A. Main Control Room

B. Technical Support Center (TSC)

C. Operations Support Center (OSC)

D. Emergency Operations Facility (EOF)

Proposed Answer: C

Explanation: The Operation Support Center is responsible for assembling, briefing, and dispatching In-Plant (India) Teams during a declared emergency.

A. Incorrect – During non-emergency conditions, the Control Room normally is responsible for dispatching personnel for off-normal response in the plant. However, during a declared emergency with all Emergency Response Facilities activated, this responsibility moves to the Operations Support Center.
B. Incorrect – The Technical Support Center is involved in determining priorities for emergency response actions, but is not directly responsible for the actual assembling, briefing, and dispatching of In-Plant (India) Teams.
D. Incorrect – The Emergency Operations Facility maintains overall oversight of the emergency response, but is not directly responsible for the actual assembling, briefing, and dispatching of In-Plant (India) Teams.

Technical Reference(s): EP-PS-132

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC27 NRC #75

Question History: LOC27 NRC #75

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.17
	Importance Rating	3.9

Knowledge of EOP terms and definitions.

Proposed Question: #73

Which one of the following is the Cold Shutdown Boron Weight, as defined by the Emergency Operating Procedures?

A. 350 gallons

B. 850 gallons

C. 1150 gallons

D. 1650 gallons

Proposed Answer: D

Explanation: The EOPs define the Cold Shutdown Boron Weight to be 1650 gallons.

A. Incorrect – The EOPs define the Cold Shutdown Boron Weight to be 1650 gallons. 350 gallons is based on the remaining tank volume when Cold Shutdown Boron Weight is injected from a full tank.
B. Incorrect – The EOPs define the Cold Shutdown Boron Weight to be 1650 gallons. 850 gallons is the Hot Shutdown Boron Weight.
C. Incorrect – The EOPs define the Cold Shutdown Boron Weight to be 1650 gallons. 1150 gallons is based on the remaining tank volume when Hot Shutdown Boron Weight is injected from a full tank.

Technical Reference(s): EO-100-113 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-053 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.0

Knowledge of the process used to track inoperable alarms.

Proposed Question: #74

A Temporary Engineering Change is to be initiated on the Instrument Air system 'A' that will render the following Control Room Annunciator inoperable:

- AR-124-D01, INSTRUMENT AIR PANEL 1C140A, B SYSTEM TROUBLE

Which one of the following lists the location where the TMOD sticker is required to be placed, in accordance with NDAP-QA-1218, Temporary Changes?

A. On Panel 1C140A, only

B. On the AR-124-D01 annunciator window, only

C. On the alarm response procedure for AR-124-D01, only

D. On the AR-124-D01 annunciator window and on the alarm response procedure for AR-124-D01

Proposed Answer: C

Explanation: NDAP-QA-1218 requires a TMOD sticker be placed on the affected alarm response procedure only.

A. Incorrect – The TMOD sticker is required to be placed on the affected alarm response procedure, not the local panel.
B. Incorrect – The TMOD sticker is required to be placed on the affected alarm response procedure, not the annunciator window.
D. Incorrect – The TMOD sticker is required to be placed on the affected alarm response procedure, but not the annunciator window.

Technical Reference(s): NDAP-QA-1218

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC26R NRC #68

Question History: LOC26R NRC #68

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.5
	Importance Rating	2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #75

Unit 1 is operating at 100% power with the following:

- HPCI spuriously initiates.
- Annunciator AR-101-B05, RX BLDG AREA PANEL 1C605 HI RADIATION, alarms.
- The alarm is determined to be caused by high radiation in the HPCI Equipment Room.
- HPCI is secured.
- The radiation level in the HPCI Equipment Room returns to normal.

Which one of the following describes the resulting operation of the HPCI Equipment Room area radiation monitor trip unit?

The trip unit...

A. automatically resets.

B. must be manually reset in the Relay Room.

C. must be manually reset in the Main Control Room.

D. must be manually reset locally in the Reactor Building.

Proposed Answer: B

Explanation: When an ARM alarms and then the high radiation condition clears, the trip unit must be manually reset. The trip units are located on panel 1C605 in the Upper Relay Room.

A. Incorrect – The trip unit must be manually reset. Plausible because many alarm in the plant automatically clear once the offending condition goes away.
C. Incorrect – The manual reset is done in the Upper Relay Room, not the Main Control Room. Plausible because the annunciator comes into the Main Control Room and at some plants the trip units are also in the Main Control Room.
D. Incorrect – The manual reset is done in the Upper Relay Room, not locally in the Reactor Building. Plausible because there is a local auxiliary unit that provides a local alarm in the Reactor Building, but the trip unit in the Upper Relay Room contains the reset for both.

Technical Reference(s): AR-101-B05

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-79B RBO-4

Question Source: Modified Bank – LOC25 NRC #70

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295030 EA2.02
	Importance Rating	3.9

Low Suppression Pool Water Level

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

Proposed Question: #76

Unit 1 was operating at 100% power when a seismic event resulted in the following:

- Reactor power is 6%.
- Reactor water level has been lowered in accordance with EO-100-113, Level/Power Control.
- Reactor pressure is 650 psig, up slow.
- Reactor pressure is being controlled in a band between 600 and 800 psig with SRVs.
- Suppression Pool water temperature indications are:
  - Average SPOTMOS: 175°F, up slow.
  - Bottom SPOTMOS: 190°F, up slow.
- EO-100-103, Primary Containment Control, has been entered.
- Suppression Pool makeup is in service.
- Suppression Pool water level is 19', down slow.

Note: Heat Capacity Temperature Limit is provided on the following page.

Which one of the following describes the required control of Reactor pressure, in accordance with the EOPs?

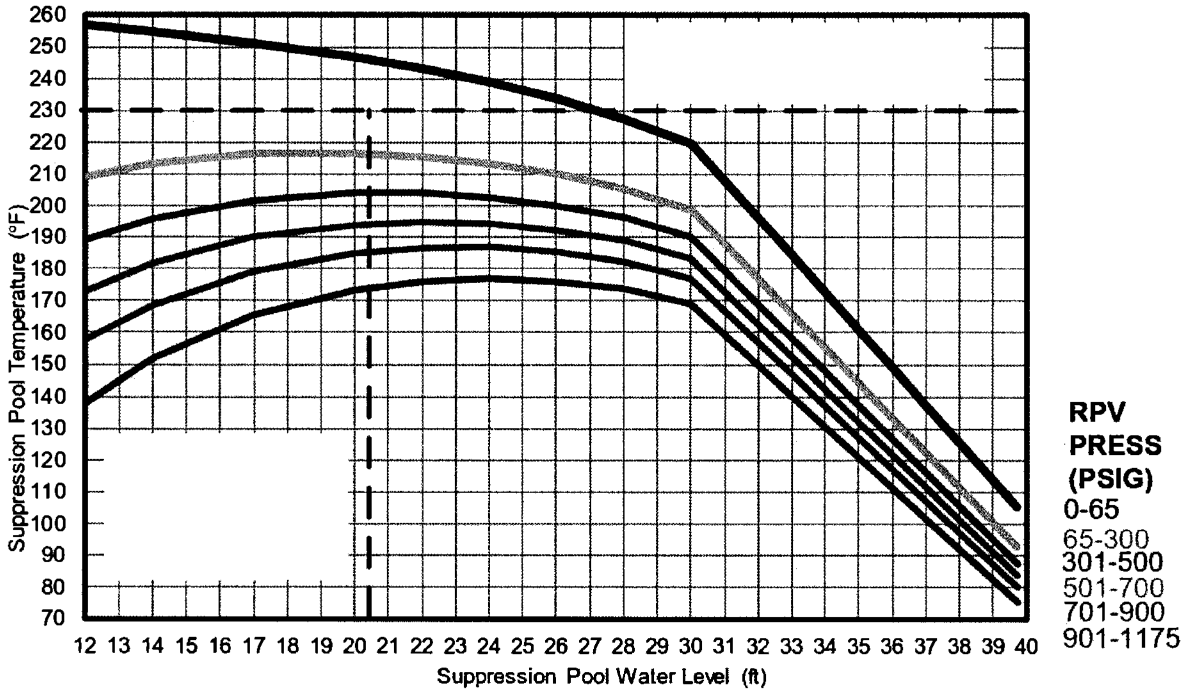
A. An Emergency RPV Depressurization is required.

B. Continued use of the entire current Reactor pressure band is acceptable.

C. Reactor pressure must be maintained at or below approximately 700 psig.

D. Reactor pressure must be maintained at or above approximately 700 psig.

FIG 2 HCTL
HEAT CAPACITY TEMPERATURE LIMIT



Proposed Answer: C

Explanation: With Suppression Pool water level below 20.5', the Average SPOTMOS indication is no longer valid and the Bottom SPOTMOS indication must be used to determine Suppression Pool water temperature for use in EO-100-103. With Suppression Pool water level at 19' and Bottom SPOTMOS indication at 190°F, the 501-700 psig HCTL curve is satisfied, but the 701-900 psig HCTL curve is violated. Therefore, current Reactor pressure of 650 psig is acceptable, but Reactor pressure must be maintained at or below 700 psig. Current plant conditions are approximately 2-3°F below the 501-700 psig HCTL curve, therefore Emergency RPV Depressurization is not required.

A. Incorrect – Current plant conditions are approximately 2-3°F below the applicable (501-700 psig) HCTL curve, therefore Emergency RPV Depressurization is not required.
B. Incorrect – Current Suppression Pool conditions are within the 501-700 psig HCTL curve, but would violate the 701-900 psig HCTL curve. Therefore, the upper ~100 psig of the current Reactor pressure band cannot be used.
D. Incorrect – Reactor pressure must be maintained below 700 psig, not above, to satisfy HCTL.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – NMP1 2015 Cert #76

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295006 AA2.01
	Importance Rating	4.6

SCRAM

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

Proposed Question: #77

Unit 1 is operating at 25% power with the following:

- A Feedwater Level Control malfunction occurs.
- An Operator places the Reactor Mode Switch in SHUTDOWN.
- 30 control rods remain withdrawn at position 48.
- All other control rods fully insert.
- Reactor water level reaches a low of -10" and then begins rising.
- APRMs indicate 3%, steady.
- EO-100-102, RPV Control, is entered.

Which one of the following describes the required procedure execution, in accordance with the Emergency Operating Procedures and OP-AD-300, Administration of Operations?

A. Remain in EO-100-102 and restore Reactor water level to a band of 13" to 54". EO-100-113, Level/Power Control, entry is NOT required.

B. Exit EO-100-102 and enter EO-100-113, Level/Power Control, based on control rod position. Remain in EO-100-113 and Reactor water level must be lowered to at least -60".

C. Exit EO-100-102 and enter EO-100-113, Level/Power Control, based on control rod position. Remain in EO-100-113 and Reactor water level may be stabilized at the current value.

D. Exit EO-100-102 and enter EO-100-113, Level/Power Control, based on control rod position. Then return to EO-100-102 based on Reactor power and restore Reactor water level to a band of 13" to 54".

Proposed Answer: C

Explanation: Upon entry into EO-100-102, a diagnostic step is executed to determine the need for entry into EO-100-113, Level/Power Control. With multiple control rods withdrawn past position 00, this step requires entry into EO-100-113. Once in this procedure, it is not exited until the Reactor is determined to be shutdown under all conditions without boron. This determination is based on control rod pattern, not actual Reactor power. Reactor power being less than or equal to 5% on APRMs affects the decision on Reactor water level control, but not the decision to exit EO-100-113. With Reactor power less than or equal to 5%, Reactor water level is not required to be intentionally lowered to at least -60". OP-AD-300 directs stabilizing Reactor water level near the current value.

A. Incorrect – Although Reactor power is less than 5%, entry into EO-100-113 is still required based on the given control rod pattern. 5% Reactor power is significant as both an EO-100-102 entry condition and a decision point for Reactor water level control in EO-100-113.
B. Incorrect – With Reactor power less than or equal to 5%, Reactor water level is not required to be intentionally lowered to at least -60". It is preferred to stabilize Reactor water level near the current value.
D. Incorrect – EO-100-113 must be entered based on the given control rod pattern. Reactor power less than 5% allows maintaining Reactor water level in the normal band, but does not allow exiting EO-100-113.

Technical Reference(s): EO-000-102, EO-100-113, OP-AD-300

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.01
	Importance Rating	3.6

Partial or Complete Loss of DC Power

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Cause of partial or complete loss of D.C. power

Proposed Question: #78

Both Units are operating at 100% power with the following:

- Diesel Generator (DG) B is out of service for overhaul.
- DG E is substituted for DG B.
- The following conditions for DG E Battery Bank 0D595 are reported to the Control Room:
    - o Float voltage of one connected cell is 2.05 VDC.
    - o Float voltage of one pilot cell is 2.09 VDC.
    - o Float current is 2.2 amps.
    - o All other cells tested were within allowable values.

Assuming that the above battery conditions CANNOT be corrected, which one of the following describes the latest that the plant can transition to Mode 3, in accordance with Technical Specifications?

A. 84 hours

B. 86 hours

C. 98 hours

D. 110 hours

Proposed Answer: B

Explanation: Technical Specification 3.8.6 conditions A, B, and F are entered based upon the initial conditions (connected cell voltage <2.07 VDC and float current >2 amps). Condition F is more restrictive and encompasses both initial conditions of low battery voltage and high float current. This requires declaring the battery inoperable immediately vice having time to correct the issues as in Conditions A and B. The battery is immediately declared INOPERABLE per F.1. Per TS 3.8.4 Condition F, once the battery is declared INOPERABLE, 2 hours is allowed to declare the EDG INOPERABLE. Once the EDG is declared INOPERABLE in TS 3.8.1 Condition B, 72 hours is allowed to restore the EDG to OPERABLE status, or be in Mode 3 in 12 hours. This allows a total of 86 hours before needing to be in Mode 3 (2+72+12=86 hours).

A. Incorrect – A total of 86 hours is allowed. 84 hours would be the answer if the 2 hours of TS 3.8.4 Condition E were not factored in.
C. Incorrect – A total of 86 hours is allowed. 98 hours would be the answer if only float current were out of spec (>2 amps). However, the addition of a cell <2.07 VDC requires a more restrictive shutdown time.
D. Incorrect – A total of 86 hours is allowed. 110 hours would be the answer if only cell voltage was out of spec (>2.07 VDC). However, the addition of float current >2 amps requires a more restrictive shutdown time.

Technical Reference(s): TS 3.8.1, 3.8.4, 3.8.6

Proposed references to be provided to applicants during examination: Technical Specifications 3.8.1, 3.8.4, 3.8.6

Learning Objective: TM-OP-002 RBO-8

Question Source: Bank – LOC24 NRC #77

Question History: LOC24 NRC #77

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 2.2.44
	Importance Rating	4.4

High Off-site Release Rate

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.

Proposed Question: #79

Unit 1 is operating at 100% power with the following:

- At 1530, an un-isolable steam leak occurs in the Turbine Building.
- Operators manually scram the Reactor and enter EO-100-102, RPV Control.
- The blowout panels open and OSCAR is evaluating the release.
- At 1550, the combined Noble Gas SPING data is indicated to be 2.3 E8 µCi/min, steady.
- Reactor conditions are as follows:
    - Containment Rad reading is 5 R/hr, steady.
    - "A" MSL CANNOT be isolated.
    - Reactor pressure is 900 psig, down slow.
    - Reactor water level is 40", steady.
    - RCIC is injecting.

Which one of the following describes proper control of Reactor pressure while waiting for Offsite Dose Calculations, in accordance with the Emergency Operating Procedures?

A. Maintain pressure between 800 and 1050 psig.

B. Open SRVs to cooldown at a rate less than 100°F/hr.

C. Open Turbine Bypass Valves irrespective of the cooldown rate.

D. Open all ADS valves to perform an Emergency RPV Depressurization.

Proposed Answer: B

Explanation: Entry is required into EO-100-105, Radioactivity Release Control, due to Noble Gas SPING > the Alert level in Table 11 (2.0 E8 μCi/min). With Noble Gas SPING stable well below the GE level (6.2 E9 μCi/min), Emergency RPV Depressurization is neither currently required nor anticipated. EO-100-102 requires Reactor pressure to be lowered, but the cooldown rate maintained below 100°F/hr. SRVs are preferred over Turbine Bypass Valves in this situation to minimize radioactivity release.

A. Incorrect – Upon initial entry into EO-100-102, the direction is to stabilize Reactor pressure <1087 psig and the normal control band initially given is 800-1050 psig. However, further direction in EO-100-102 directs lowering Reactor pressure with a cooldown rate of <100°F/hr. Maintaining Reactor pressure in a higher band is specifically undesirable in this situation since it raises the radioactivity release rate from an un-isolable steam leak outside of primary and secondary containments.
C. Incorrect – With Noble Gas SPING stable well below the GE level (6.2 E9 μCi/min), Emergency RPV Depressurization is neither currently required nor anticipated. EO-100-102 requires Reactor pressure to be lowered, but the cooldown rate maintained below 100°F/hr. If release rate was near or approaching the GE level, then pressure could be lowered irrespective of the cooldown rate.
D. Incorrect – With Noble Gas SPING stable well below the GE level (6.2 E9 μCi/min), Emergency RPV Depressurization is not currently required. SRVs are a proper method of pressure control, but not all ADS valves can be opened without violating cooldown rate.

Technical Reference(s): EO-100-105, EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC24 NRC #82

Question History: LOC24 NRC #82

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295025 2.4.9
	Importance Rating	4.2

High Reactor Pressure

Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: #80

Unit 1 is operating at 5% power with the following:

- A loss of coolant accident occurs.
- Operators manually scram the Reactor.
- Following the scram, all offsite power is lost.
- RHR and Core Spray pumps automatically start.
- HPCI starts and then trips.
- RCIC starts and is injecting.
- ADS has failed to initiate.
- Reactor water level is -178", down slow.
- Reactor pressure is 700 psig, down slow.

Which one of the following describes the appropriate action to be taken, in accordance with the Emergency Operating Procedures?

A. Enter EO-100-112, Emergency RPV Depressurization.

B. Enter the Steam Cooling section of EO-100-102, RPV Control.

C. Reset the Main Generator lockout and inject with Condensate pumps.

D. Rapidly depressurize the Reactor using Turbine Bypass Valves per EO-100-102, RPV Control.

Proposed Answer: A

Explanation: With Reactor water level below -161" and almost to -179" with injection subsystems lined up (RHR and Core Spray), EO-100-102 step RC/L-17 requires entering EO-100-112, Emergency RPV Depressurization.

Note: The question meets the K/A because Reactor pressure is high with respect to the capacity of available injection systems and the LOCA mitigation strategy in a low power / shutdown situation is tested.

B. Incorrect – Steam Cooling would only be entered if no injection source was available. While Reactor pressure is currently too high for RHR and Core Spray to inject, they are still available to inject following an Emergency RPV Depressurization and therefore counted towards the number of available injection systems.
C. Incorrect – If offsite power were available, then resetting the Main Generator lockout and injecting with Condensate would be a viable option for restoring and maintaining Reactor water level above -161". However, since all offsite power has been lost, Condensate pumps are unavailable for injection.
D. Incorrect – Rapidly depressurizing the Reactor with Turbine Bypass Valves to get within the capacity of low pressure injection systems would have been a viable option earlier in the accident. However, the conditions have degraded to the point where an Emergency RPV Depressurization is required, not just anticipated. Additionally, MSIVs are currently closed due to plant conditions.

Technical Reference(s): EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2013 NRC #85

Question History: NMP1 2013 NRC #85

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295026 2.2.22
	Importance Rating	4.7

Suppression Pool High Water Temperature

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #81

Unit 1 is operating at 80 percent power with the following:

- SO-150-002, Quarterly RCIC Flow Verification Test, in progress with the following timeline of events:

Time	Condition(s)
2200	RCIC is started.
2230	Suppression Pool average water temperature is 90°F, up slow.
2300	RCIC is secured. Heat addition to the Suppression Pool is terminated. Suppression Pool average water temperature is 95°F, stable.

Which one of the following describes the required action, in accordance with Technical Specifications?

Suppression Pool average water temperature must be restored to less than or equal to 90°F by the latest time of...

A. 1030 the next day.

B. 1100 the next day.

C. 2230 the next day.

D. 2300 the next day.

Proposed Answer: D

Explanation: Technical Specification 3.6.2.1 Condition entry is not required while RCIC is operating (adding heat to the Suppression Pool) since Suppression Pool average water temperature remains below limits of 105°F, 110°F, and 120°F. However, once heat addition to the Suppression Pool is terminated at 2300, Technical Specification 3.6.2.1 Condition A must be entered since Suppression Pool average water temperature is above 90°F. Required Action A.2 allows a maximum of 24 hours from this time to restore Suppression Pool average water temperature to less than or equal to 90°F (2300 the next day).

A. Incorrect – The latest time allowed is 2300 the next day. 1030 the next day would be the answer if the completion time for Technical Specification 3.6.2.1 Condition B were applied from the time Suppression Pool average water temperature first reached 90°F.
B. Incorrect – The latest time allowed is 2300 the next day. 1100 the next day would be the answer if the completion time for Technical Specification 3.6.2.1 Condition B were applied from the time heat addition to the Suppression Pool was terminated.
C. Incorrect – The latest time allowed is 2300 the next day. 2230 the next day would be the answer if the correct completion time were applied from the time Suppression Pool average water temperature first reached 90°F.

Technical Reference(s): TS 3.6.2.1

Proposed references to be provided to applicants during examination: Technical Specification 3.6.2.1

Learning Objective: TM-OP-059 RBO-8

Question Source: Bank – LOC24 Cert #82

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295018 2.4.47
Importance Rating	4.2

Partial or Complete Loss of CCW

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

Proposed Question: #82

Both Units are operating at 100% power with the following:

- High outside air temperatures and degraded spray capability are causing Spray Pond temperature to rise.
- Average Spray Pond temperature is 81.1°F and rising 0.2°F/hr.

Note: Assume the Spray Pond temperature trend remains constant.

Which one of the following describes the approximate earliest time at which Technical Specification Condition entry will be required due to high Spray Pond temperature?

A. 10 hours

B. 20 hours

C. 30 hours

D. 35 hours

Proposed Answer: B

Explanation: Technical Specification Surveillance Requirement 3.7.1.2 requires average water temperature of the UHS (Spray Pond) to be ≤85°F with both Units operating in Mode 1. The current trend would put Spray Pond average water temperature at 85.1°F in 20 hours (81.1°F + 20 hrs * 0.2°F/hr = 85.1°F).

A. Incorrect – At 10 hours, average water temperature would be 83.1°F, which is the approximate setpoint for annunciator AR-016-A12, EWS SPRAY POND HI TEMP. However, this is below the temperature requiring TS 3.7.1 Condition entry.
C. Incorrect – TS 3.7.1 Condition entry is first required at approximately 20 hours. 30 hours is the time at which Condition entry would be required if one Unit were in Mode 3 for 12-24 hours.
D. Incorrect – TS 3.7.1 Condition entry is first required at approximately 20 hours. 35 hours is the time at which Condition entry would be required if one Unit were in Mode 3 for at least 24 hours.

Technical Reference(s): TS 3.7.1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-54 RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	500000 EA2.04
	Importance Rating	3.3

High Containment Hydrogen Concentration

Ability to determine and/or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for wetwell

Proposed Question: #83

An accident on Unit 1 has resulted in the following conditions:

- Reactor water level is -150", up slow.
- Reactor pressure is 475 psig, down slow.
- Drywell pressure is 9 psig, up slow.
- Drywell temperature is 280°F, up slow.
- Suppression Chamber pressure is 8 psig, up slow.
- Suppression Pool average water temperature is 130°F, up slow.
- Suppression Pool water level is 24 feet, steady
- Drywell hydrogen concentration is 0.8%, up slow.
- Drywell oxygen concentration is 0.3%, steady.
- Suppression Chamber hydrogen concentration is 1.5%, up slow.
- Suppression Chamber oxygen concentration is 0.3%, steady.
- Dose projections indicate that the offsite release rate will remain below Technical Specification limits during Primary Containment venting/purging operations.

Which of the following describes the required control of Primary Containment venting and purging, in accordance with EO-100-103, Primary Containment Control?

A. Venting is required, only.

B. Purging is required, only.

C. Both venting and purging are required.

D. NEITHER venting NOR purging are required.

Proposed Answer: C

Explanation: The given Primary Containment gas concentrations indicate that hydrogen generation is in progress. Current concentrations are below the combustible limits of 5% oxygen and 6% hydrogen, however EO-100-103 pre-emptively directs venting the Primary Containment when hydrogen concentration exceeds a lower limit of 1%. Purging is also directed since venting can be accomplished. No conditions indicate that venting cannot be performed.

A. Incorrect – Venting is required by EO-100-103 step PC/G-4. Since venting can be performed, then next step (PC/G-5) also requires purging after venting is established.
B. Incorrect – Both venting and purging are required. No conditions indicate that venting cannot be performed.

D. Incorrect – Current concentrations are below the combustible limits of 5% oxygen and 6% hydrogen, however EO-100-103 pre-emptively directs venting the Primary Containment when hydrogen concentration exceeds a lower limit of 1%.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – LOC25 Cert #83

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295002 2.1.7
	Importance Rating	4.7

Loss of Main Condenser Vacuum

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

Proposed Question: #84

Unit 1 is operating at 40% power with the following:

- AR-131-A06, OFFGAS UNIT 1 TRAIN A FLOW HI, alarms.
- Off-gas flow is 80 scfm, up slow.
- Main Turbine exhaust pressure is 4.5" HgA, up slow.
- Condensate temperature is 130°F, stable.
- Main Generator load is 410 MWe, stable.

__Note:__ A portion of ON-143-001, Main Condenser Vacuum and Offgas System Off-Normal Operation, Attachment E is provided on this page.

Which one of the following describes the required control of the Reactor and/or Main Turbine, in accordance with ON-143-001?

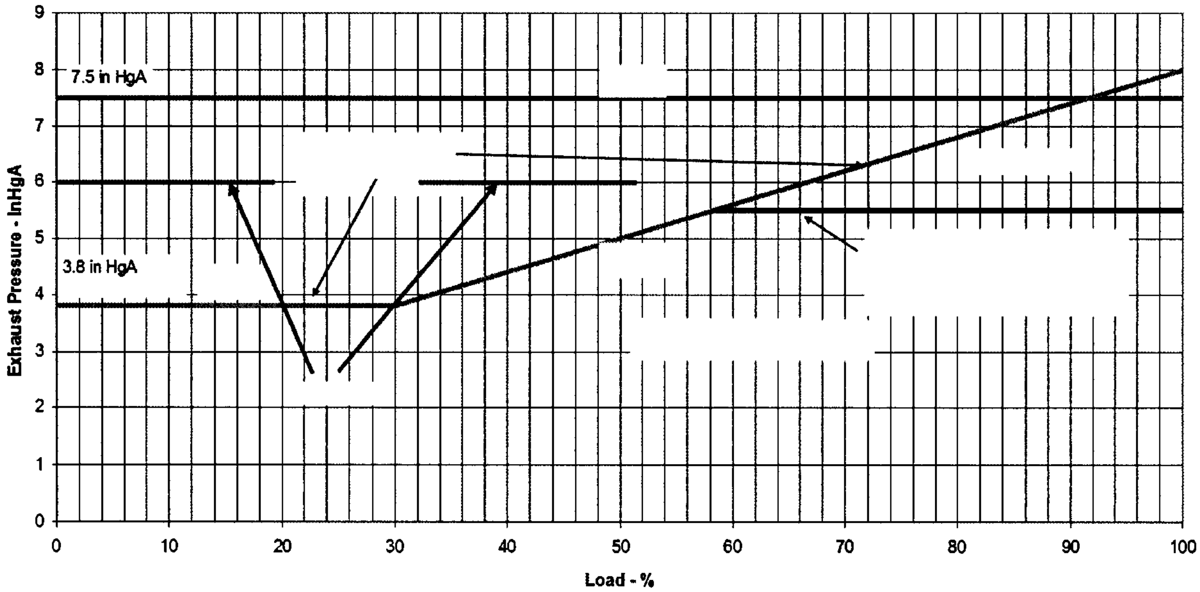
A. The Reactor and Main Turbine may continue to operate at the current power level.

B. Reactor power must be reduced in accordance with GO-100-012, Power Maneuvers.

C. The Reactor must be manually scrammed in accordance with ON-SCRAM-101, Reactor Scram.

D. Reactor power must be rapidly reduced and the Main Turbine must be manually tripped in accordance with ON-193-002, Main Turbine Trip, but a Reactor scram is NOT required.

SSES Turbine Exhaust Pressure Alarm and Trip Level



Proposed Answer: B

Explanation: ON-143-001 entry is required based on both degrading Main Condenser vacuum (rising Turbine back pressure) and abnormally rising Offgas flow. Due to the degrading Main Condenser vacuum, ON-143-001 requires a Reactor power reduction per GO-100-012 to maintain Turbine exhaust pressure exceeds the limits of Attachment E. With a Main Generator load of approximately 36% and exhaust pressure of 4.5" HgA, operation is outside the alarm curve (orange) of Attachment E. Since operation is only above the orange curve, neither a Reactor scram nor a Turbine trip are required. The operator has 5 minutes in this region before they must trip the main turbine

A. Incorrect – Since Condenser vacuum is continuing to degrade, ON-143-001 requires lowering Reactor power.
C. Incorrect – Since operation is currently within the limits of ON-143-001 Attachment E, a Reactor scram is not yet required.
D. Incorrect – Since operation is currently within the limits of ON-143-001 Attachment E, continued Reactor and Turbine operation are allowed for 5 minutes . It is plausible at low power levels to be able to rapidly lower Reactor power such that the Turbine can be tripped without scramming the Reactor.

Technical Reference(s): ON-143-001 Att E

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295035 EA2.02
	Importance Rating	4.1

Secondary Containment High Differential Pressure

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Off-site release rate: Plant-Specific

Proposed Question: #85

Unit 1 is operating at 100% power with the following:

- A RCIC steam leak develops in the Reactor Building.
- The leak cannot be isolated.
- Area Radiation Monitor (ARM) #2, RCIC PP * TURB ROOM, is in alarm at 50 mr/hr, stable.
- RCIC equipment area temperature is in alarm at 135°F, stable.
- Normal Reactor Building Ventilation is in service.
- Standby Gas Treatment (SGTS) is in standby.
- Reactor Building Zone 1 differential pressure is +0.05" WG.
- The Shift Manager is completing the Event Notification Report (ENR).

Note: EO-100-104, Secondary Containment Control, Tables 8 and 9 are provided on the following pages.

Which one of the following describes the required completion for ENR Form Block 3, EMERGENCY CLASSIFICATION, and Block 5, RELEASE IN PROGRESS, in accordance with the Emergency Plan?

	Block 3	Block 5
A.	ALERT	NO
B.	ALERT	AIRBORNE
C.	SITE AREA EMERGENCY	NO
D.	SITE AREA EMERGENCY	AIRBORNE

TABLE 8

TABLE 8 REACTOR BUILDING TEMPERATURE

RB AREA EL (FT)		MAX NORMAL ΔTEMP (°F)	MAX NORMAL TEMP (°F)	MAX SAFE TEMP (°F)	RECORDER POINTS Δ T, T	RB TEMP (°F)	
818	GENERAL AREA	N/A	110	120		______	______
779	GENERAL AREA	N/A	110	120		______	______
749	GENERAL AREA	N/A	110	120		______	______
	RWCU-PUMP ROOM	45	120	147	18,10	______	______
	RWCU-HEAT EXCH ROOM	45	120	147	19,11	______	______
	RWCU-PENETRATION ROOM	45	120	131	20,12	______	______
719	GENERAL AREA	N/A	110	120		______	______
	MAIN STEAM LINE TUN	60	157	177	7, 1	______	______
683	GENERAL AREA	N/A	110	120		______	______
	HPCI PIPE ROUTING AREA	45	120	167	17, 6	______	______
	RCIC PIPE ROUTING AREA	45	120	167	9*, 3*	______	______
670	GENERAL AREA	N/A	110	120		______	______
645	HPCI-EQUIP AREA	45	120	167	15,4	______	______
	HPCI-EMERG AREA COOLER	N/A	120	167	5	______	______
645	RCIC-EMERG AREA COOLER	N/A	120	167	2*	______	______
	RCIC-EQUIP AREA	45	120	167	7*,1*	______	______
645	RHR EQUIP AREA 1	45	110	142	8,2	______	______
645	RHR EQUIP AREA 2	45	110	142	9,3	______	______
645	CS PUMP ROOM A	N/A	110	142		______	______
645	CS PUMP ROOM B	N/A	110	142		______	______
645	RB SUMP ROOM	N/A	110	125		______	______

TABLE 9

TABLE 9 REACTOR BUILDING RADIATION

RB AREA EL (FT)	ARM NUMBER LO RANGE	ARM NUMBER HIGH RANGE	ARM CHANNEL DESCRIPTION	MAX NORMAL RADIATION FIELD	MAX SAFE RADIATION FIELD (MR/HR)	MAX SAFE RADIATION EO104 (R/HR)	RB RAD (R/HR)
818	35+	N/A	CASK STOR AREA	HI ALARM	10^4	10	____ ____
	14+	N/A	SPENT FUEL CRIT MON				____ ____
	15*	N/A	REFUEL FLOOR SOUTH				____ ____
	42+	N/A	REFUEL FLOOR WEST				____ ____
	47*	N/A	SPENT FUEL CRIT MON				____ ____
	N/A	49	REFUEL FLOOR AREA				____ ____
749	8+	52	RWCU RECIRC PP ACC	HI ALARM	10^4	10	____ ____
	10*	54	FUEL POOL PP AREA				____ ____
	11+	N/A	RX BLD SAMPLE ST				____ ____
719	5*	50	CRD NORTH	HI ALARM	10^4	10	____ ____
	6*	51	CRD SOUTH				____ ____
670	16+	53	REM SHDN ROOM ACC	HI ALARM	10^4	10	____ ____
645	3+	48	HPCI PP * TURB ROOM	HI ALARM	10^4	10	____ ____
645	2+	57	RCIC PP * TURB ROOM	HI ALARM	10^4	10	____ ____
645	25*	55	RHR A * C PP ROOM	HI ALARM	10^4	10	____ ____
645	1*	56	RHR B * D PP ROOM	HI ALARM	10^4	10	____ ____
645	4*	N/A	RB/RW SUMP ROOM	HI ALARM	10^4	10	____ ____

RANGE: + 0.01 - 10^2 MR/HR * 0.1 - 10^3 MR/HR

Proposed Answer: D

Explanation: The given conditions result in an Emergency Classification of Site Area Emergency (FS1 – Loss or Potential Loss of Any Two Barriers; loss of Primary Containment barrier and potential loss of RCS barrier based on high area temperature and rad level with un-isolable RCIC leak). The elevated radiation levels in Reactor Building Zone 1 with positive differential pressure means an unmonitored airborne release is occurring from the Reactor Building due to the event.

Note: Table F-1 Row E in the provided reference to the applicants will be blocked out to prevent Question #93 from being a direct lookup.

A. Incorrect – The potential loss of the RCS barrier by itself would result in an Alert, however the additional loss of the Primary Containment barrier requires a Site Area Emergency. A positive Reactor Building differential pressure with elevated radiation levels inside the building results in an unmonitored airborne release due to the event.
B. Incorrect – The potential loss of the RCS barrier by itself would result in an Alert, however the additional loss of the Primary Containment barrier requires a Site Area Emergency.
C. Incorrect – A positive Reactor Building differential pressure with elevated radiation levels inside the building results in an unmonitored airborne release due to the event.

Technical Reference(s): ENR Form, EP-RM-004 table F

Proposed references to be provided to applicants during examination: EAL Matrices (Table F-1 Row E blocked out

Learning Objective:

Question Source: Modified Bank – NMP1 2015 NRC #76

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	300000 A2.01
Importance Rating	2.8

Instrument Air

Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Air dryer and filter malfunctions

Proposed Question: #86

Unit 1 is operating at 100% power with the following:

- A malfunction with Instrument Air Dryer C is causing high differential pressure across the dryer.
- Attempts to shift to Instrument Air Dryers A and B have been unsuccessful.
- Instrument Air pressure is 100 psig, steady, on PI-12511A.
- Instrument Air header pressure is 88 psig, down slow, on PI-12564.
- Annunciator AR-107-G01, SCRAM PILOT VALVE AIR HEADER LO PRESS, is in alarm.
- Scram Air header pressure is 64 psig, down slow.

Which one of the following describes the cross-tie that is required to restore Instrument Air header pressure with the given malfunction and the need to direct a Reactor scram?

Cross-tie of...

A. Service Air to Instrument Air will restore Instrument Air header pressure. Directing a manual Reactor scram is required now.

B. Service Air to Instrument Air will restore Instrument Air header pressure. Directing a manual Reactor scram is NOT required now.

C. Unit 2 Instrument Air to Unit 1 Instrument Air will restore Instrument Air header pressure. Directing a manual Reactor scram is required now.

D. Unit 2 Instrument Air to Unit 1 Instrument Air will restore Instrument Air header pressure. Directing a manual Reactor scram is NOT required now.

Proposed Answer: C

Explanation: ON-118-001 contains steps regarding both cross-tie of Service Air to Instrument Air and Unit 2 Instrument Air to Unit 1 Instrument Air. A significant difference between these lineups is that Service Air inputs upstream of the Instrument Air Dryers, while Unit 2 Instrument Air inputs downstream of the Instrument Air Dryers. With the Air Dryers being the source of the issue, cross-tie of Service Air will NOT be able to restore pressure, while cross-tie of Unit 2 Instrument Air will be able to restore pressure. The conditions requiring a Reactor scram in ON-118-001 are NOT yet met, however AR-107-G01 does require a manual Reactor scram to be directed due to low Scram Air Header pressure.

A. Incorrect – Service Air inputs upstream of the Instrument Air Dryers. Therefore with the Air Dryer malfunction, Service Air will NOT restore pressure.
B. Incorrect – Service Air inputs upstream of the Instrument Air Dryers. Therefore with the Air Dryer malfunction, Service Air will NOT restore pressure. The conditions requiring a Reactor scram in ON-118-001 are NOT yet met, however AR-107-G01 does require a manual Reactor scram to be directed due to low Scram Air Header pressure.
D. Incorrect – The conditions requiring a Reactor scram in ON-118-001 are NOT yet met, however AR-107-G01 does require a manual Reactor scram to be directed due to low Scram Air Header pressure.

Technical Reference(s): ON-118-001, AR-107-G01

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-18 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	209001 A2.09
	Importance Rating	3.3

LPCS

Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low suppression pool level

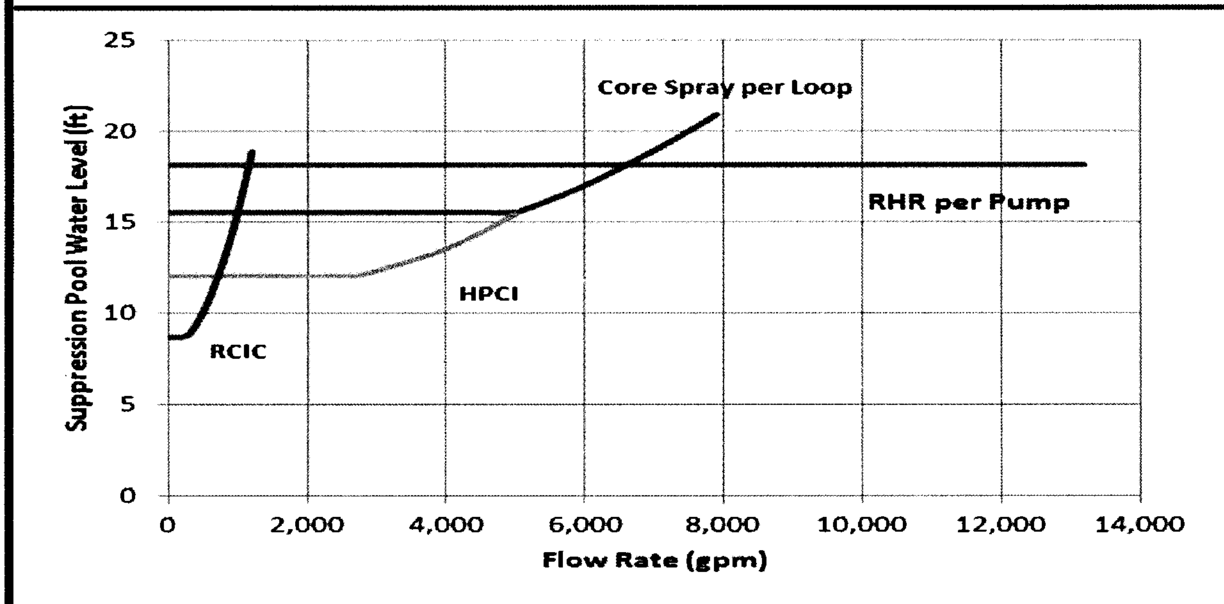
Proposed Question: #87

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is -170", steady.
- Core Spray Loop A is injecting 7500 gpm with flow maximized.
- No other injection sources are available.
- Suppression Pool water level is 16.5', down slow.

VL

FIG. 7 VL
RHR, CS, HPCI & RCIC VORTEX LIMIT



Which one of the following describes the status of the Core Spray Vortex Limit and the required control of Core Spray Loop A flow, in accordance with the Emergency Operating Procedures?

	The Core Spray Vortex Limit is...	Core Spray Loop A flow...
A.	being exceeded.	must be lowered.
B.	being exceeded.	may be maintained at the current value.
C.	NOT being exceeded.	must be lowered.
D.	NOT being exceeded.	may be maintained at the current value.

Proposed Answer: B

Explanation: With Suppression Pool water level at 16.5', the Core Spray Vortex Limit is approximately 5800 gpm. Therefore, the current Core Spray flow of 7500 gpm is exceeding the Core Spray Vortex Limit. However, EO-100-102 does not require lowering Core Spray flow. The Vortex Limit should be observed when possible, however the need for Core Spray injection to maintain adequate core cooling is of greater concern since Reactor water level is below the top of active fuel (-161") and stable with no other injection systems available.

A. Incorrect – EO-100-102 does not require lowering Core Spray flow. The Vortex Limit should be observed when possible, however the need for Core Spray injection to maintain adequate core cooling is of greater concern since Reactor water level is below the top of active fuel (-161") and stable with no other injection systems available.
C. Incorrect – With Suppression Pool water level at 16.5', the Core Spray Vortex Limit is approximately 5800 gpm. Therefore, the current Core Spray flow of 7500 gpm is exceeding the Core Spray Vortex Limit. EO-100-102 does not require lowering Core Spray flow. The Vortex Limit should be observed when possible, however the need for Core Spray injection to maintain adequate core cooling is of greater concern since Reactor water level is below the top of active fuel (-161") and stable with no other injection systems available.
D. Incorrect – With Suppression Pool water level at 16.5', the Core Spray Vortex Limit is approximately 5800 gpm. Therefore, the current Core Spray flow of 7500 gpm is exceeding the Core Spray Vortex Limit.

Technical Reference(s): EO-100-102, EO-000-100

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - LOC26R NRC #80

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 2.4.6
	Importance Rating	4.7

SLC

Knowledge of EOP mitigation strategies.

Proposed Question: #88

Unit 1 has experienced a failure to scram with the following:

- Initial ATWS power was >5%.
- Reactor water level has been intentionally lowered.
- Reactor water level is being controlled between -110" and -60".
- Reactor pressure is being controlled between 800 and 1050 psig.
- Based on the rate of control rod insertion, it is expected that EO-100-113, Level/Power Control, will be exited in two hours.
- The Unit Supervisor has directed boron injection using Standby Liquid Control (SBLC) in accordance with OP-153-001, Standby Liquid Control System.
- Initial SBLC tank volume is 1800 gallons.

Which one of the following describes the approximate time after SBLC initiation at which enough boron will be injected such that it is allowed to commence Reactor depressurization, in accordance with EO-100-113 and OP-153-001?

A. ~24 minutes

B. ~34 minutes

C. ~42 minutes

D. ~45 minutes

Proposed Answer: C

Explanation: If any boron is injected, EO-100-113 requires injection of the Cold Shutdown Boron Weight (1650 gallons) before allowing depressurization of the Reactor. OP-153-001 section 2.2 and Attachment A provide the direction for injecting boron with the SBLC system. These procedures direct injecting with one SBLC pump. This provides approximately 40 gpm of flow. Therefore, it will take approximately 42 minutes to inject 1650 gallons of boron solution into the Reactor.

A. Incorrect – It will take approximately 24 minutes to inject Hot Shutdown Boron Weight into the Reactor (932 gallons).
B. Incorrect – It will take approximately 34 minutes to inject 1350 gallons into the Reactor. This is the Tech Spec limit for 12% concentration and amount to be operable.
D. Incorrect – It will take approximately 45 minutes to inject the entir contents of the SBLC tank volume which is the requirement to secure SBLC pumps in EO-100-113.

Technical Reference(s): EO-100-113, SO-153-004

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-053 RBO-7

Question Source: Modified Bank – LOC26R NRC #22

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	217000 2.1.25
	Importance Rating	4.2

RCIC

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: #89

Unit 1 is operating at 100% power with the following:

- A large leak occurs on the RCIC pump suction piping inside the RCIC room.
- All attempts to isolate the leak have failed.
- Suppression Pool makeup has been initiated.
- Suppression Pool level is 22', down fast.

Note: A portion of EO-100-103, Primary Containment Control, is provided on this page.

Given the following possible actions:

(1) Scram the Reactor.
(2) Isolate HPCI.
(3) Perform an Emergency RPV Depressurization.

Which one of the following identifies which of these given actions, if any, will be required, in accordance with the Emergency Operating Procedures?

A. (1) only

B. (1) and (2) only

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

D. NEITHER (1), (2), NOR (3)

TABLE 18
SUPP POOL EQUALIZATION LEVELS

LEAK LOCATION	EXPECTED POOL LVL (ft.)
HPCI	18
RCIC	19.5
RHR A	16
RHR B	17
CS A	16
CS B	19

Proposed Answer: D

Explanation: With the given leak, entry will be required into EO-100-103, Primary Containment Control, due to low Suppression Pool water level, and EO-100-104, Secondary Containment Control, due to high Reactor Building Area water level. EO-100-103 Table 18, Supp Pool Equalization Levels, identifies the expected value at which Suppression Pool water level will stabilize given a leak from various systems. With an un-isolable leak from RCIC, Suppression Pool water level will be maintain at or above 19.5'. Since Suppression Pool water level remains above 17', EO-100-103 will require neither a Reactor scram nor isolation of HPCI. Since Suppression Pool water level remains above 12', EO-100-103 will not require an Emergency RPV Depressurization. Only one Reactor Building Area water level will exceed the Max Safe level and the leak is coming from a non-primary system. Therefore EO-100-104 will require neither a Reactor scram nor Emergency RPV Depressurization.

A. Incorrect – A Reactor scram is not required by EO-100-103 because Suppression Pool water level will remain above 17'. A Reactor scram is not required by EO-100-104 because the leak is not from a primary system.
B. Incorrect – A Reactor scram is not required by EO-100-103 because Suppression Pool water level will remain above 17'. A Reactor scram is not required by EO-100-104 because the leak is not from a primary system. Isolating HPCI is not required by EO-100-103 because Suppression Pool water level will remain above 17'.
C. Incorrect – A Reactor scram is not required by EO-100-103 because Suppression Pool water level will remain above 17'. A Reactor scram is not required by EO-100-104 because the leak is not from a primary system. Isolating HPCI is not required by EO-100-103 because Suppression Pool water level will remain above 17'. Emergency RPV Depressurization is not required by EO-100-103 because Suppression Pool water level will remain above 12'. Emergency RPV Depressurization is not required by EO-100-104 because the leak is not from a primary system and only one Reactor Building Area water level will exceed the Max Safe level.

Technical Reference(s): EO-100-103, EO-100-104

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	206000 A2.09
	Importance Rating	3.7

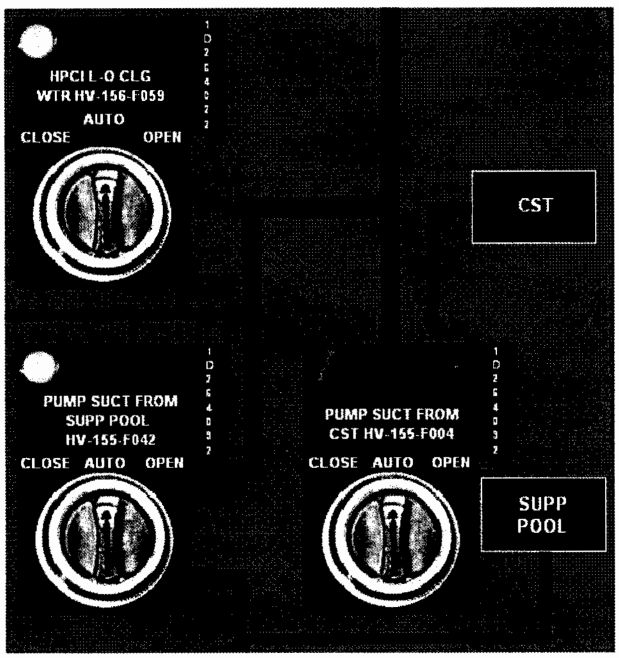
HPCI

Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low condensate storage tank level: BWR-2,3,4

Proposed Question: #90

Unit 1 is operating at 100% power with the following:

- Annunciator AR-114-E01, CONDENSATE STORAGE TANK LO WATER LEVEL, alarms.
- Condensate Storage Tank level is 40”, down slow, due to a leak.
- HPCI is aligned as follows:



Which one of the following describes the need for Condition entry in Technical Specifications (TS) 3.3.5.1, Emergency Core Cooling System (ECCS) Instrumentation, and 3.5.1, ECCS – Operating?

	TS 3.3.5.1 Condition Entry	TS 3.5.1 Condition Entry
A.	Required	Required
B.	Required	NOT required
C.	NOT required	Required
D.	NOT required	NOT required

Proposed Answer: B

Explanation: CST level is below the value (40.5") in TS Table 3.3.5.1-1 function 3.d and the HPCI pump suction has failed to swap from the CST to the Suppression Pool. This requires entry into TS 3.3.5.1 Condition D. Condition D allows up to one hour before declaring HPCI inoperable, which would then require TS 3.5.1 Condition entry. Manually swapping the HPCI suction from the CST to the Suppression Pool in this hour would eliminate any future need to declare HPCI inoperable and TS 3.5.1 Condition entry. Additionally, TS 3.5.1 bases state, "The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the condensate storage tank (CST), it is capable of providing a source of water for the HPCI and CS systems." Therefore, direct TS 3.5.1 Condition entry is not required based on low CST water level.

A. Incorrect – TS 3.5.1 bases do not require TS 3.5.1 Condition entry on low CST water level and TS 3.3.5.1 Condition D allows more time before declaring HPCI inoperable, therefore TS 3.5.1 Condition entry is not required.
C. Incorrect – CST level is below the value (40.5") in TS Table 3.3.5.1-1 function 3.d and the HPCI pump suction has failed to swap from the CST to the Suppression Pool. This requires entry into TS 3.3.5.1 Condition D. TS 3.5.1 bases do not require TS 3.5.1 Condition entry on low CST water level and TS 3.3.5.1 Condition D allows more time before declaring HPCI inoperable, therefore TS 3.5.1 Condition entry is not required.
D. Incorrect – CST level is below the value (40.5") in TS Table 3.3.5.1-1 function 3.d and the HPCI pump suction has failed to swap from the CST to the Suppression Pool. This requires entry into TS 3.3.5.1 Condition D.

Technical Reference(s): AR-114-E01, TS 3.5.1 and bases, TS 3.3.5.1

Proposed references to be provided to applicants during examination: TS 3.5.1 and 3.3.5.1 (w/ allowable values removed from TS table 3.3.5.1-1)

Learning Objective: TM-OP-52 RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	215002 A2.03
	Importance Rating	3.3

RBM

Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of associated reference APRM channel: BWR-3,4,5

Proposed Question: #91

Unit 1 is operating at 98% power with the following:

- APRM 1 drifts low to 20%.
- MCPR is 1.62.

Which one of the following describes the operability and applicability of the Rod Block Monitor (RBM) A upscale rod block function under current plant conditions, in accordance with Technical Specifications?

The RBM A upscale rod block function is currently...

A. operable and required to be operable.

B. inoperable and required to be operable.

C. operable and NOT required to be operable.

D. inoperable and NOT required to be operable.

Proposed Answer: B

Explanation: APRM 1 is the primary reference for RBM A. With APRM 1 indicating only 20%, RBM A automatically bypasses, and is therefore inoperable. The RBM upscale rod block function is currently required to be operable by Technical Specifications and the Core Operating Limits Report (COLR) since Reactor power is above 95% with MCPR below 1.70.

Note: The Technical Specification applicability of the RBM upscale rod block function is SRO level knowledge because it requires use of Technical Specification Table 3.3.2.1-1 and the Core Operating Limits Report (COLR), not just "above-the-line" information in Technical Specification 3.3.2.1.

A. Incorrect – With APRM 1 indicating only 20%, RBM A automatically bypasses, and is therefore inoperable.
C. Incorrect – With APRM 1 indicating only 20%, RBM A automatically bypasses, and is therefore inoperable. The RBM upscale rod block function is currently required to be operable by Technical Specifications and the Core Operating Limits Report (COLR) since Reactor power is above 95% with MCPR below 1.70.
D. Incorrect – The RBM upscale rod block function is currently required to be operable by Technical Specifications and the Core Operating Limits Report (COLR) since Reactor power is above 95% with MCPR below 1.70.

Technical Reference(s): TS 3.3.2.1, COLR

Proposed references to be provided to applicants during examination: Technical Specification 3.3.2.1, COLR

Learning Objective: TM-OP-78K RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	288000 2.4.8
	Importance Rating	4.5

Plant Ventilation

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question: #92

Unit 1 is operating at 100% power with the following:

- A malfunction has occurred with the Reactor Building Zone 1 differential pressure controller.
- ON-RBHVAC-101, High/Low Reactor Building Differential Pressure, is being performed.
- Entry into EO-100-104, Secondary Containment Control, is also now required.

Which one of the following describes the correct procedure implementation?

A. Enter EO-100-104. Continue performing ON-RBHVAC-101. In the event of a conflict between ON-RBHVAC-101 and EO-100-104, EO-100-104 is the overriding procedure.

B. Enter EO-100-104. Continue performing ON-RBHVAC-101. In the event of a conflict between ON-RBHVAC-101 and EO-100-104, ON-RBHVAC-101 is the overriding procedure.

C. Exit ON-RBHVAC-101 and enter EO-100-104. ON-RBHVAC-101 is re-entered at the step in progress after exiting EO-100-104.

D. Exit ON-RBHVAC-101 and enter EO-100-104. ON-RBHVAC-101 entry conditions are re-evaluated after exiting EO-100-104.

Proposed Answer: A

Explanation: EO-100-104 is entered based on Zone 1 differential pressure less than 0.25" WG vacuum for 4 hours. There is no requirement to exit ONs when EOPs are entered. In fact, both procedures are executed concurrently. The EOPs are higher-tiered documents than the ONs, therefore in the event of a conflict, the EOP must be followed.

B. Incorrect – The ON does contain specific guidance to assist with this problem, whereas the EOP contains more general guidance. However, EOPs are higher-tiered documents than ONs.
C. Incorrect – The EOP is the higher-tiered document, but the ON is still used in parallel.
D. Incorrect – The EOP is the higher-tiered document, but the ON is still used in parallel.

Technical Reference(s): AD-10 Att B

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank - NMP1 2015 NRC #88

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	272000 2.2.38
	Importance Rating	4.5

Radiation Monitoring

Knowledge of conditions and limitations in the facility license.

Proposed Question: #93

Which one of the following identifies a radiation monitor and associated threshold reading used to define Loss of the Fuel Clad Barrier, in accordance with the Emergency Plan?

A. Containment high range radiation; 7 R/hr

B. Containment high range radiation; 3,000 R/hr

C. Offgas pretreatment radiation; hi alarm setpoint

D. Offgas pretreatment radiation; hi-hi alarm setpoint

Proposed Answer: B

Explanation: Form EP-RM-004-F Table F-1, Fission Product Barrier Matrix, identifies Containment high range radiation >3000 R/hr as a Loss of the Fuel Clad Barrier.

Note: The question meets the generic K/A by testing knowledge of the Emergency Plan, which is a condition of the facility license (10 CFR 50.47, 50.54(q), Appendix E).

Note: The provided reference for Question #85 is to be edited to avoid making this question a direct lookup.

A. Incorrect – 3000 R/hr is the threshold value, NOT 7 R/hr. 7 R/hr is the threshold value for Loss of the Reactor Coolant System Barrier.
C. Incorrect – Offgas pretreatment radiation monitoring is used to define Unusual Event SU4.1. This indicates fuel clad degradation, but does not define loss of the barrier.
D. Incorrect – Offgas pretreatment radiation monitoring is used to define Unusual Event SU4.1. This indicates fuel clad degradation, but does not define loss of the barrier.

Technical Reference(s): Form EP-RM-004-F

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – JAF 4/14 NRC #99

Question History: JAF 4/14 NRC #99

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.42
	Importance Rating	3.4

Knowledge of new and spent fuel movement procedures.

Proposed Question: #94

Unit 1 is shutdown for a refueling outage with the following:

- Core shuffle is in progress.
- An irradiated fuel bundle is about to be moved from the Reactor core to the Spent Fuel Pool.
- Then, the grapple camera becomes inoperable.

Which one of the following describes the required control of fuel moves, in accordance with OP-0RF-008, Fuel and Blade Guide Handling Activities?

Fuel moves...

A. must be stopped until the grapple camera is returned to service.

B. must be stopped until the grapple camera is returned to service or a portable camera is installed.

C. may continue with the grapple camera inoperable, but only if Shift Manager approval is obtained.

D. may continue with the grapple camera inoperable, but only if an additional Spotter is stationed on the Refuel Platform to independently verify all moves.

Proposed Answer: C

Explanation: OP-0RF-008 step 7.1.12.b allows fuel moves without the grapple camera operable, but requires Shift Manager permission to move fuel with the grapple camera inoperable.

A. Incorrect – OP-0RF-008 step 7.1.12.b allows fuel moves without the grapple camera operable.
B. Incorrect – OP-0RF-008 step 7.1.12.b allows fuel moves without the grapple camera operable.
D. Incorrect – OP-0RF-008 step 7.1.12.b requires Shift Manager permission for fuel moves with the grapple camera inoperable, not stationing an additional Spotter on the Refuel Platform.

Technical Reference(s): OP-0RF-008

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-081 RBO-7

Question Source: Modified Bank – Vision SYSID 741

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.20
	Importance Rating	3.8

Knowledge of the process for managing troubleshooting activities.

Proposed Question: #95

Both Units are operating at 100% power with the following:

- Diesel Generators (DGs) A and C are inoperable and are undergoing troubleshooting.
- DGs B and D are protected, in accordance with NDAP-QA-0340, Protected Equipment Program.
- The DG system engineer contacts the Unit Supervisor to request for I&C to take a voltage reading at the local control panel for DG B in support of the troubleshooting.

Which one of the following identifies the allowable response to this troubleshooting request, in accordance with NDAP-QA-0340?

I&C may...

A. perform the measurement as long as they are continuously escorted in the protected equipment area by an Operator. Formal approval is NOT required for this activity.

B. perform the measurement once formal approval has been obtained. The highest level of required approval is the Shift Manager.

C. perform the measurement once formal approval has been obtained. The highest level of required approval is the Operations Manager.

D. NOT perform the measurement until DG B is NO longer protected.

Proposed Answer: C

Explanation: NDAP-QA-0340 Attachment F controls work on protected equipment. The requested voltage measurement will require work directly on the protected equipment, therefore if is classified as High Impact. High Impact work requires formal approval from the Shift Manager, Assistant Operations Manager, and the Operations Manager.

A. Incorrect – Formal approval is required for work on protected equipment per NDAP-QA-0340 Attachment F.
B. Incorrect – Since the measurement will require work directly on protected equipment, it is classified as High Impact. High Impact work requires formal approval from the Shift Manager, Assistant Operations Manager, and the Operations Manager. Low Impact work would only require formal approval from the Shift Manager.
D. Incorrect – NDAP-QA-0340 allows work to be performed on protected equipment using Attachment F, Protected Equipment Work Approval Form.

Technical Reference(s): NDAP-QA-0340

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – Vision SYSID 287

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.11
	Importance Rating	4.3

Ability to control radiation releases.

Proposed Question: #96

Unit 1 is operating at 100% power with the following:

- A liquid effluent discharge of a set of Sample Tanks is in progress.
- Annunciator AR-107-E06, RADWASTE EFFLUENT MON HI RADIATION, alarms.
- Abnormally high radiation trends are observed on the Effluent Radiation Recorder, RR-06433, on the Liquid Radwaste and Chemical Process Panel OC301.
- Combined Cooling Tower blowdown flow is 7000 gpm.

Which one of the following describes the required action to be directed, in accordance with ON-069-001, Abnormal Radiation Release Liquid?

A. Ensure high radiation setpoint countrate calculations are correct and properly entered into RITS-06433, Liquid Radwaste Radiation.

B. Start an additional River Water Makeup Pump and open the blowdown flow control valve to lower the concentration through dilution.

C. Open the blowdown flow control valve to lower the concentration through additional dilution, only.

D. Ensure the LRW Sample Tank Pump for the tanks being released are secured.

Proposed Answer: A

Explanation: The LRW Effluent Discharge Isolation valves will automatically isolate upon high radiation signal, and the operators are directed to verify high radiation setpoint countrate calculations and determine if they have been appropriately entered into RITS-06433 Liquid Radwaste radiation.

B. Incorrect – This would only be required if the conditions were caused by combined cooling tower blowdown low flow, which would require increasing flow greater than 6000 gpm by starting additional pump or opening flow control valves. No specific indication is given of blowdown low flow.
C. Incorrect – This would only be required if the conditions were caused by combined cooling tower blowdown low flow, which would require increasing flow greater than 6000 gpm by starting additional pump or opening flow control valves. No specific indication is given of blowdown low flow.
D. Incorrect – Although the LRW Effluent Discharge isolation valves automatically close on high radiation, the pump is not required to be secured. The operating pump recirculation lines are opened per ON-069-001.

Technical Reference(s): ON-069-001, AR-107-001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - LOC24 NRC #98

Question History: LOC24 NRC #98

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.16
	Importance Rating	4.4

Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, AOPs and SAMGs.

Proposed Question: #97

Unit 1 has experienced a loss of coolant accident with the following:

- EO-100-102, RPV Control, and EO-100-103, Primary Containment Control are being implemented.
- Emergency RPV Depressurization has been performed due to low Reactor water level.
- All available Reactor injection systems have been maximized.
- Core Spray flow is 15,000 gpm, stable.
- Reactor pressure is 50 psig, stable.
- Reactor water level is -220", down slow.

Which one of the following describes the required procedure implementation?

A. Remain in EO-100-102 and EO-100-103.

B. Exit EO-100-102 and EO-100-103. Enter SAG-1, RPV, Containment, and Radioactivity Control. Entry into SAG-2, RPV and Primary Containment Flooding, is NOT required.

C. Exit EO-100-102 and EO-100-103. Enter SAG-2, RPV and Primary Containment Flooding. Entry into SAG-1, RPV, Containment, and Radioactivity Control, is NOT required.

D. Exit EO-100-102 and EO-100-103. Enter both SAG-1, RPV, Containment, and Radioactivity Control, and SAG-2, RPV and Primary Containment Flooding.

Proposed Answer: D

Explanation: Since the Reactor has already been depressurized, injection sources have been maximized, and Reactor water level cannot be determined to be above -210", EO-100-102 step RC/L-20 requires exiting all EOPs and entering all SAGs.

A. Incorrect – Adequate Core Spray flow exists to remain in the EOPs, however Reactor water level is too low (<-210"). Therefore, both EO-100-102 and EO-100-103 must be exited.
B. Incorrect – When the EOPs are exited, both SAG-1 and SAG-2 must be entered.
C. Incorrect – When the EOPs are exited, both SAG-1 and SAG-2 must be entered.

Technical Reference(s): EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – Limerick 2012 NRC #99

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.6
	Importance Rating	3.6

Knowledge of the process for making changes to procedures.

Proposed Question: #98

A change is being processed for OP-144-001, Condensate and Feedwater System, to alter technical steps in the section for returning a Condensate pump to service following maintenance. The procedure section does NOT introduce any temporary plant configuration changes.

Which one of the following describes required reviews for this procedure change, in accordance with NDAP-QA-0004, Procedure Change Process?

Plant Operations Review Committee (PORC) review is __(1)__.

A 10 CFR 50.59 review is __(2)__.

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

Proposed Answer: C

Explanation: NDAP-QA-0004 requires a 10 CFR 50.59 review be performed since this procedure does NOT have a previous 10 CFR 50.59 exemption. NDAP-QA-0004 Attachment H contains guidance on which changes must receive PORC review. OP-144-001 is not included in the list of procedures requiring PORC review.

A. Incorrect – PORC review is not required.
B. Incorrect – PORC review is not required. 50.59 review is required.
D. Incorrect – 50.59 review is required.

Technical Reference(s): NDAP-QA-0004

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2013 NRC #96

Question History: NMP1 2013 NRC #96

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.13
	Importance Rating	3.2

Knowledge of facility requirements for controlling vital / controlled access.

Proposed Question: #99

Which one of the following identifies the ***lowest*** level of emergency declaration that requires initiating a Site Evacuation, in accordance with EP-PS-100, Emergency Director, Control Room, Emergency Plan – Position Specific Instruction?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: C

Explanation: Evacuation of personnel at the Unusual Event and Alert emergency levels is optional. Once a Site Area Emergency is declared, EP-PS-100 requires a Site Evacuation.

A. Incorrect – Evacuation of personnel at the Unusual Event emergency levels is optional, but not required.
B. Incorrect – Evacuation of personnel at the Alert emergency levels is optional, but not required.
D. Incorrect – Site Evacuation is required at the General Emergency level, but it is also required at the lower Site Area Emergency level

Technical Reference(s): EP-PS-100

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – SSES 2002 NRC SRO #11

Question History: SSES 2002 NRC SRO #11

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.14
	Importance Rating	3.8

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

Proposed Question: #100

Unit 1 is operating at 100% power with the following:

- Steam is coming out of a doorway leading to the Feedwater heaters.
- An NPO is being sent to investigate.
- An RWP has been prepared for entry into the area.
- A Health Physics Technician has been assigned to make the entry with the NPO.
- It is desired for the entry to be made without a reduction in Reactor power or hydrogen injection flow.

Given the following individuals:

(1) Plant Manager
(2) Radiation Protection Manager
(3) Operations Manager
(4) Shift Manager
(5) Unit Supervisor

Which one of the following identifies the individuals authorized to direct this entry, in accordance with NDAP-QA-1191, ALARA Program and Policy?

A. (1) and (2) only

B. (2) and (3) only

C. (3) and (4) only

D. (4) and (5) only

Proposed Answer: D

Explanation: NDAP-QA-1191 specifically requires direction from either the Unit Supervisor or Shift Manager to enter into a steam affected area without a reduction in Reactor power or hydrogen injection flow.

A. Incorrect – NDAP-QA-1191 specifically requires direction from either the Unit Supervisor or Shift Manager to enter into a steam affected area without a reduction in Reactor power or hydrogen injection flow.
B. Incorrect – NDAP-QA-1191 specifically requires direction from either the Unit Supervisor or Shift Manager to enter into a steam affected area without a reduction in Reactor power or hydrogen injection flow.
C. Incorrect – NDAP-QA-1191 specifically requires direction from either the Unit Supervisor or Shift Manager to enter into a steam affected area without a reduction in Reactor power or hydrogen injection flow.

Technical Reference(s): NDAP-QA-1191

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC26R NRC #98

Question History: LOC26R NRC #98

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments: